1 NUCLEAR REACTOR DESIGN

100 GENERAL

The nuclear reactor design part of this report is a description of the Yankee Atomic Electric Company reactor and the reactor safeguard considerations associated with it. The reactor description is based on a reference design by Westinghouse Electric Corporation and is subject to change in detail before manufacture.

The proposed pressurized light water reactor plant is designed to produce ultimately 492 mw of heat and 134 mw of net electrical generation at full power. The initial core to be operated in the reactor, however, is designed to produce 392 mw of heat at full power which provides approximately 110 mw of net electrical generation.

The reactor is fueled with slightly enriched uranium dioxide UO_2, in the form of pressed and sintered 0.3 in. diameter cylindrical compacts stacked in 0.336 in. OD full length stainless steel tubes in a core 7.5 ft in height. The reactor is cooled and moderated by light water. Pending results of the Research and Development Program, it is planned that the fuel compacts will be snugly fitted, and no bond will be used to improve heat transfer to the 0.015 in. thick cladding wall. The basic tubular fuel rod of stacked compacts is bundled into fuel assemblies of convenient size, containing approximately 300 tubes. These assemblies are loaded vertically upon the core support plate to form a uniformly enriched, uniformly spaced, cylindrical rod lattice core of 76 fuel assemblies. The total UO_2 mass is approximately 24,000 kg, or 53,000 lb. The initial fuel enrichment in the U-235 isotope is approximately 2.6 per cent to provide an estimated core life between refuelings of 10,000 hr. The lattice spacing selected for the reference design of the initial core is 0.420 in. center to center. The volumetric composition of the reference core design is 37.5 per cent fuel (UO_2), 50 per cent water, 8 per cent stainless steel, 3.2 per cent zirconium (control rod followers), and the balance, 1.3 per cent, voids. The average heat flux in the reference core design is 86,800 Btu per sq ft-hr; the calculated maximum-to-average value for heat flux is 5.17.

Water in the reactor and in the main coolant loop is maintained at the system pressure of 2,000 psia. At full power, the inlet water temperature to the 392 mw heat core is 491 F, and the outlet temperature is 522 F. Full power conditions yield 520 psia saturated steam at the outlet of the steam generator. The maximum fuel cladding surface temperature is 642 F, causing some local boiling of the subcooled liquid within the coolant channel; no bulk boiling occurs at the outlet of any coolant channel under the nominal loop pressure of 2,000 psia.

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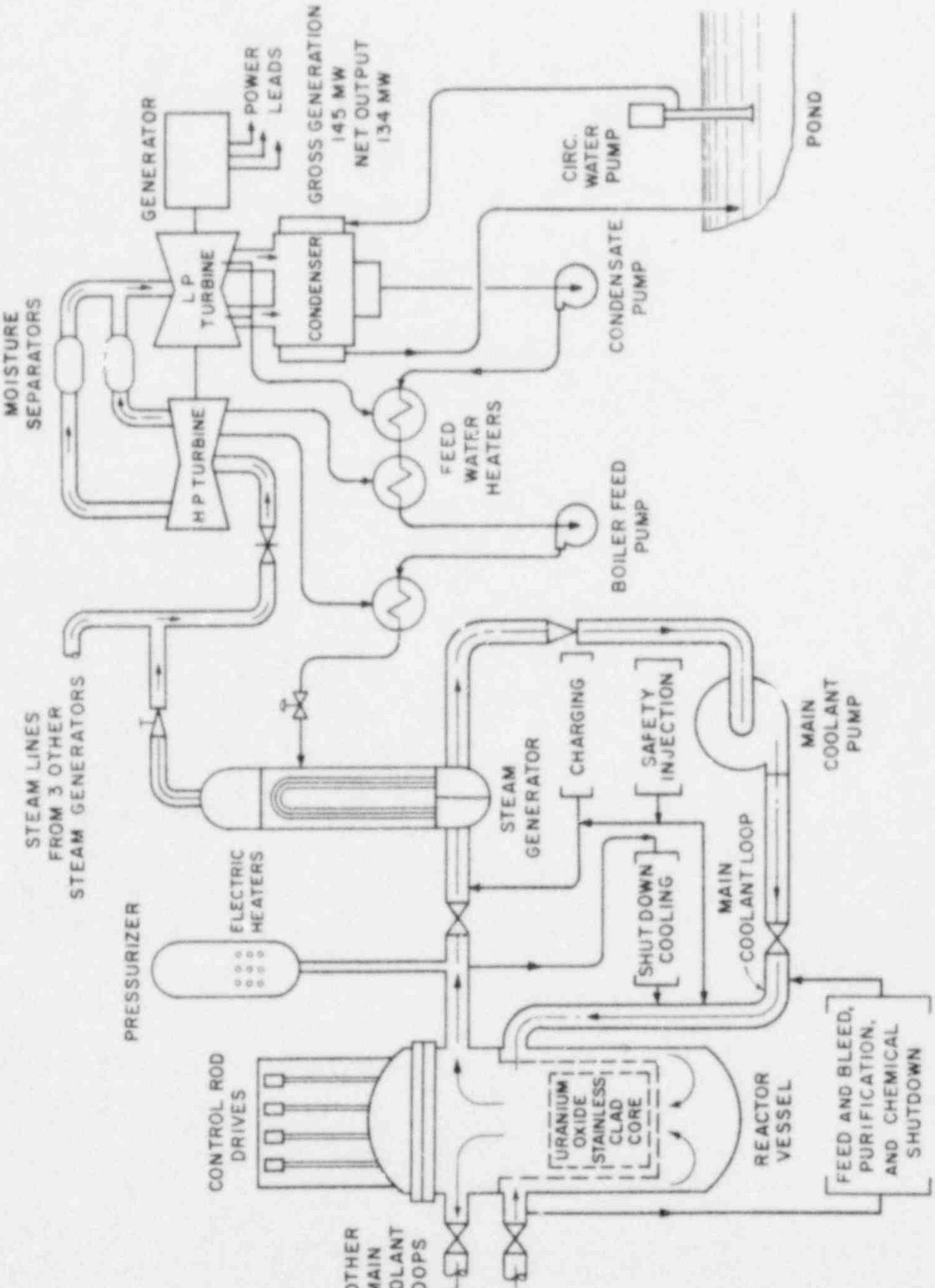
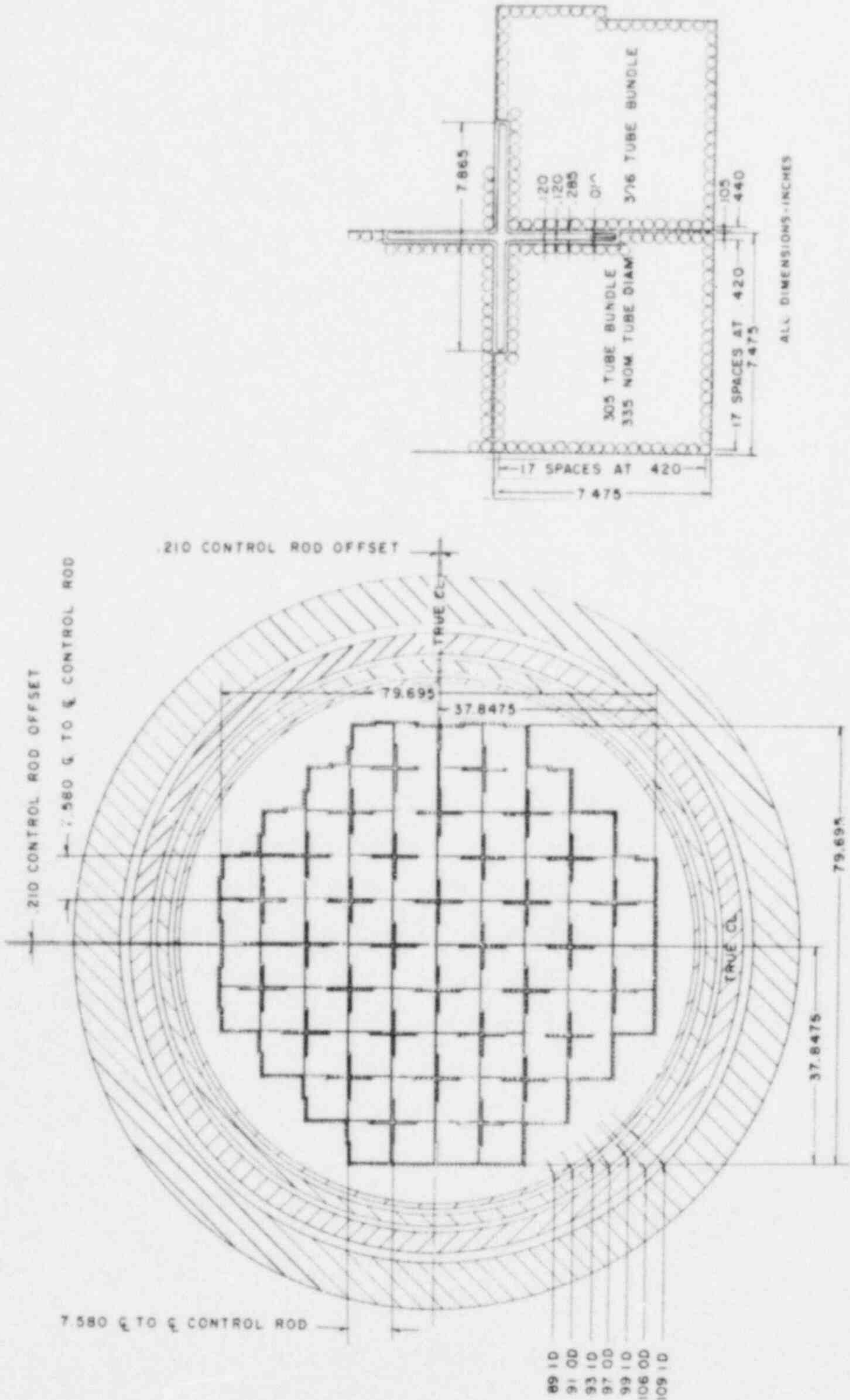


SIMPLIFIED FLOW DIAGRAM

PRIMARY AND SECONDARY SYSTEMS



REACTOR CORE CROSS SECTION

101 CORE DESIGN

Mechanical Design of the Core

General

The 392 mw heat reactor core approximates the shape of a right circular cylinder 74.4 in. in diameter and 90 in. high, giving a length-to-diameter ratio of 1.2. The core consists of four substantially identical quadrants containing 19 fuel assemblies each, providing a total of 76 fuel assemblies in the complete core. The assemblies are square in cross section and are assembled in a close-packed square lattice. Figure 3 is a cross section of the core.

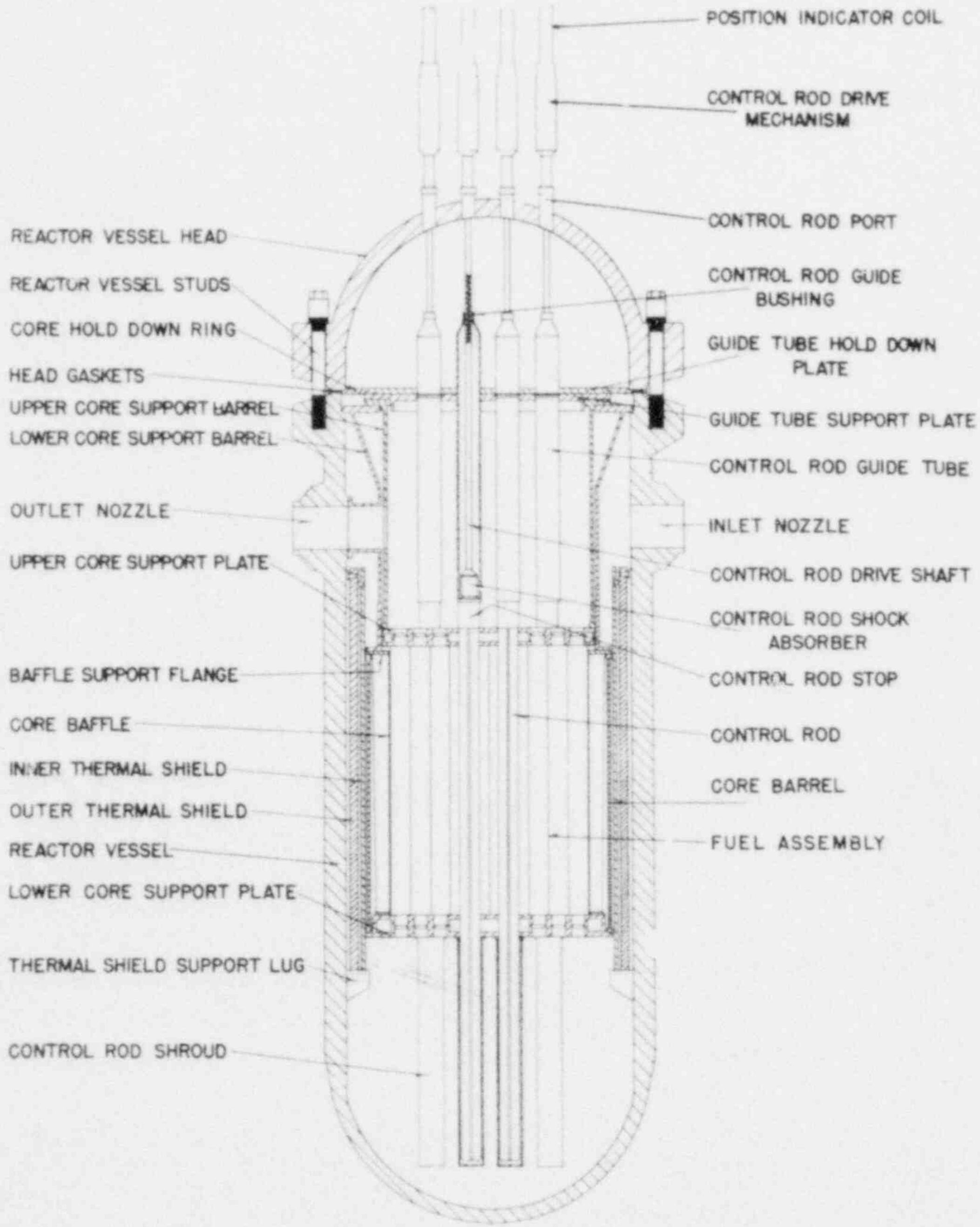
The 76 individual, replaceable fuel assemblies are held in the core between the lower support and the upper core support plate. Holes are provided in both support plates for the coolant inlet and discharge nozzles of the separate assemblies. These support plates are provided with 32 cross shaped slots to allow passage of the 24 cruciform control rods and the 8 cruciform shim elements. The axis of the control rods is parallel to the vertical axis of the core; the control rods are each actuated by a separate mechanism above the core. Reactivity of the core is increased by lifting the control rods out of the core in a vertical direction. For refueling purposes, the control rod drive mechanisms are disengaged from the control rods, leaving the control rods fully inserted in the core. After removal of the upper core support plate, the fuel assemblies are removed individually.

The reactor core is surrounded by a form-fitting baffle which confines the flow within the fuel bearing zone. The baffle is contained within the core barrel, the structural member running between the top and bottom support plates. Water flowing between the form-fitting baffle and the barrel acts as a reflector. Figure 4 is a section through the reactor showing the general assembly.

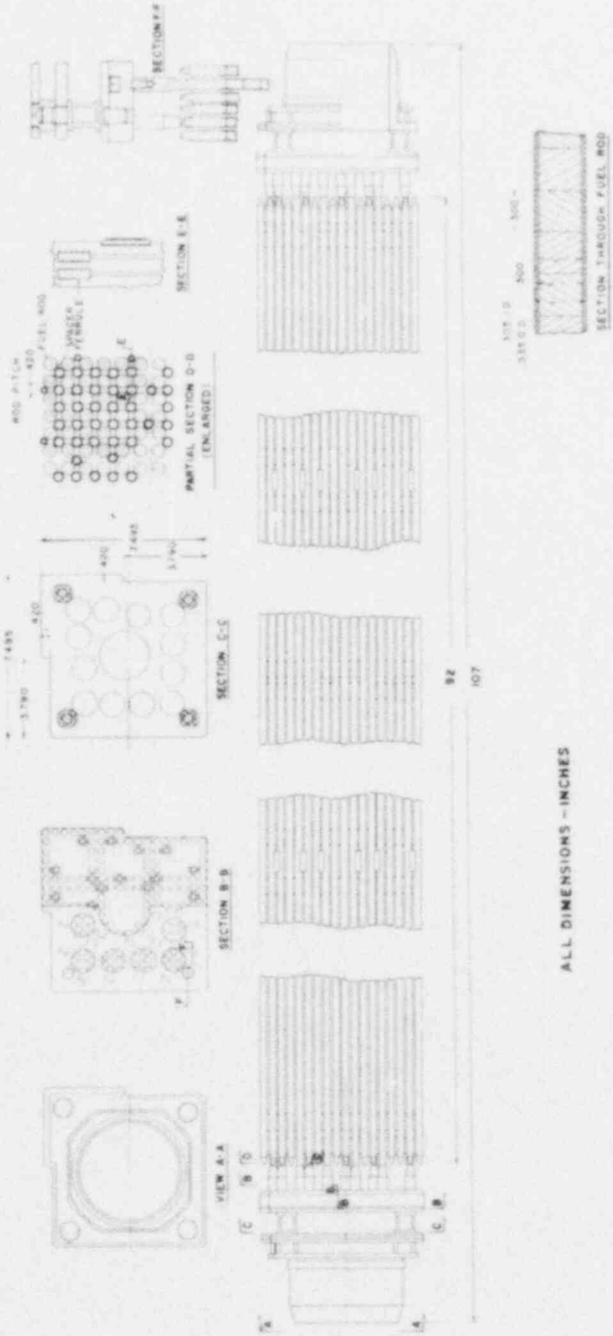
Fuel Assemblies

The details of construction of the fuel assemblies are shown in Figure 5. Each fuel assembly has a total length of approximately 107 in. and a core, or "active", length of 90 in. The assemblies are roughly square in cross section, approximately 7 1/2 by 7 1/2 in.

The basic element in a fuel assembly is the fuel pellet. The pellet is 0.30 in. in diameter and 0.30 in. high. Current experiments indicate that it may be feasible to increase the height of the pellet to 0.60 in. The pellet is manufactured by sintering a powder-compact of enriched UO_2. After sintering, the individual pellets are centerless ground to obtain the required dimensional tolerances. The uranium is enriched in the fissionable isotope U-235 to 2.6 atom per cent. Three hundred pellets are assembled



REACTOR SECTION



FUEL ASSEMBLY - SECTIONS AND DETAILS

in a stainless steel tube to make a fuel rod having an overall length of 92 in. and an active fuel length of 90 in. Stainless steel plugs are welded in the ends of each tube.

The fuel rods are assembled on a square lattice with a center-to-center distance of 0.42 in. to make a fuel assembly. Only two fuel assembly designs are used, having 305 and 306 fuel rods, respectively. The fuel rods are assembled in a square 18 x 18. Rods are omitted from this pattern as required to provide slots for the passage of the blades of the cruciform control rods, thereby reducing the number of rods from 324, the number in a full 18 x 18 arrangement, to 305 and 306.

The fuel rods are assembled into fuel assemblies by brazing or welding to provide the proper spacing between rods and to furnish the required structural rigidity to the assembly. Tubular spacers or ferrules, approximately 3/4 in. long, are placed between the fuel rods at intervals of 14 in. along the length of the bundle. The ferrules in adjacent water channels are staggered to minimize the obstruction to the flow of coolant along the fuel rods. After the rods and ferrules are brazed into a rigid structure, the bundle is joined to the upper and lower nozzle assemblies to form a complete fuel assembly. An end plate is mechanically joined to 16 of the fuel rods by means of machine screws. These fuel rods have special end pieces which extend beyond the other rods and are drilled and tapped to receive the machine screws. The remainder, and by far the larger number, of the fuel rods are pointed to minimize disturbance to coolant flow.

Control Rods

The reactor control rods are cruciform in shape and 24 in number. The control rods are of the neutron absorbing type with extensions of low neutron absorption cross section material which act as guides and prevent the formation of a "water hole" in the core when a control rod section is withdrawn. The neutron absorbing material for the control rods is specified, nominally as silver-cadmium-indium alloy. This material is "black" to thermal neutrons. The material for the control rod extensions is Zircaloy-2.

In the reactor core, the number of types of fuel assemblies is limited to two in order to provide maximum flexibility in the interchange of assemblies within the core during reloading. This simplification results in a core design having 32 cruciform slots of which only 24 are occupied by control rods. The remaining 8 cruciform slots may be used for shim elements. The shim elements may be made up of neutron absorbing, fuel, or inert material. If no reactivity effect from the shim elements is desired, they may be constructed of Zircaloy-2. In such a case, the nuclear effect of the shim elements is to eliminate the undesirable neutron flux peaking which would occur if the water is not excluded from the slot. The shim elements also restrict the bypass coolant flow.

Core Structure

The reactor core structure consists, in general, of a lower core support plate, a baffle structure, a two-section core support barrel, and an upper core support plate as shown in Figure 4. The functions of the core structure are:

To support the weight of the fuel assemblies and maintain orientation

To support and secure the position of the control rod extension shrouds

To absorb the impact of the control rods on the upper support plate during a scram

The entire structure is fabricated of Type 304 stainless steel to minimize corrosion.

The lower core support plate is a rigid assembly of two perforated plates approximately 1 3/4 in. and 1 1/4 in. thick joined by welding to 76 sleeves into which the nozzles at the lower end of the fuel assembly fit. The sleeves act to stiffen the two plates.

The overall height of the plate assembly is approximately 8 in. This design minimizes thermal stresses by providing access for coolant within the support plate. In addition to supporting the fuel assemblies, the support plate positions the control rod extension shrouds.

The baffle structure separates the cooling water flowing downward outside of the core from that flowing upward through the core. Reinforcing ribs, running axially through the midpoints of the larger baffle walls, strengthen the structure to withstand the hydraulic pressure differential.

The lower section of the core support barrel which is 1 in. thick is between the baffle structure and the thermal shielding. The lower support plate is bolted securely to the lower rim of this barrel. The barrel and the baffle structure are bolted firmly to the upper section of the core support barrel.

The upper section of the core support barrel is held down on a ledge in the pressure vessel wall by the vessel head acting through a core hold down ring. Four flanged nozzles in the side of the barrel position the entire structure in a lateral direction by their contact with the pressure vessel outlet nozzles.

The upper core support plate is identical to the lower support plate. It maintains the orientation of the fuel assemblies and absorbs the impact of the control rods during a scram. This plate and barrel are removable for easy access to the core.

Control Rod Drive Mechanism

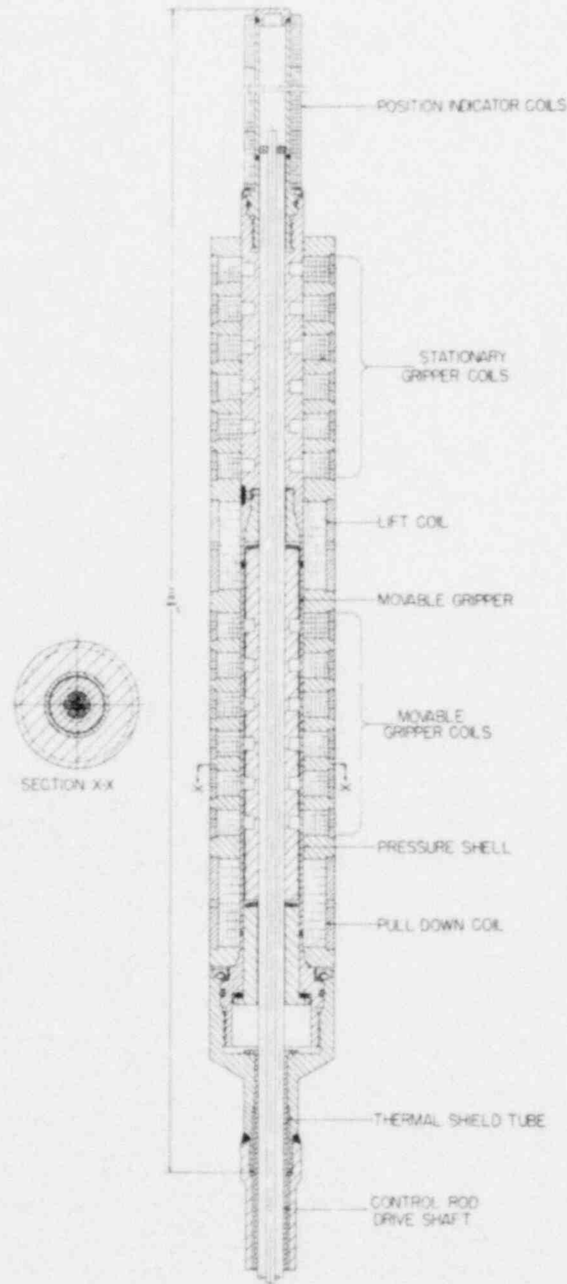
Each of the 24 control rods is moved by its own individual mechanism. It is not planned, at present, that the eight additional shim elements will be movable; however, if it later seems desirable, the shim elements might be moved by attaching each to one of the 24 mechanisms already present. A magnetic, jack-type control rod drive mechanism is planned. The magnetic, jack-type mechanism was originally proposed by Mr. J. W. Young of the Argonne National Laboratory and has been developed at that laboratory. A modification of this basic design is now under development by Westinghouse Commercial Atomic Power. The general arrangement of the mechanism is shown in Figure 6.

With the magnetic, jack-type mechanism, the only components which operate in the high-pressure main coolant system are the lifting tubes, the movable gripper, and the extension shaft which couples the lifting tubes to the control rod. The lifting tubes consist of six tubes arranged around a center tube of nonmagnetic material. The movable gripper surrounds the seven lifting tubes. The seven tubes and gripper are contained within a pressure shell which is attached to the head of the reactor vessel.

The electromagnet coils which actuate the mechanism are external to the pressure tube and surround it. There are four sets of coils designated as the "stationary gripper coils" or holding coils, "movable gripper coils", "lift coil", and "pulldown coil". The control rod is locked in a stationary position by energizing the holding coils. Under the force of the magnetic field, the magnetic lifting tubes deflect causing friction contact between the tubes and the pressure shell. Incremental movement of the control rod is obtained by energizing the movable gripper coils and deenergizing the holding coils. This action locks the ferromagnetic tubes to the gripper, which may be then moved up or down by energizing either the lift coil or the pulldown coil. After the motion has been completed, the holding coils are reenergized, and the gripper returned to its original position.

The cycle is programmed through a motor-driven controller. The gear train and the inherent speed limitation in the driving motor limit the speed of travel of the control rod to a value which is safe in terms of the resulting rate of change of reactivity.

The position of the control rod is determined at all times by a magnetically operated pickup mounted above the control rod drive mechanism.



CONTROL ROD DRIVE MECHANISM

The motion of control rods is limited so that reactivity can not be added more rapidly than 1.03 $10^{-4} \Delta$ k/k per sec. This requires a minimum of 74 sec to go from delayed to prompt critical.

Another important safety feature of this type of control rod drive mechanism is the ability to scram the control rod at any portion of the cycle by deenergizing all coils, allowing the friction contact to release, and permitting the control rod to fall by gravity. Experiments performed on similar rods dropped in water at Argonne National Laboratory and Bettis Field have resulted in an acceleration of approximately .8 of gravity. The lack of latches, gears, or other mechanical devices which might become jammed also increases the reliability of this type of mechanism. Another feature insuring reliability is that each operating cycle, in effect, tests the ability of the mechanism to scram when all coils are deenergized.

The design of a magnetic, jack-type control rod mechanism applicable to the reactor design has been completed. A prototype of the final mechanism will be constructed and tested exhaustively to determine its feasibility and reliability under operating conditions.

Thermal and Hydraulic Design of the Core

General

The thermal and hydraulic design of the reactor core is developed on the basis of the following assumptions:

Steam conditions at full load at the outlet of the steam generator are 520 psia, 471 F; the log mean temperature difference in the steam generator is 33.2 F at full load

The maximum heat transfer flux in the core does not exceed 50 per cent of the burnout heat flux as predicted by the Jens-Lottes or Bettis correlations. (ANL-4627; BPA-A1W (IM)-3)

Local boiling, or surface boiling of the subcooled liquid, is permissible within the core

Bulk boiling is not permitted within the core

The hot channel factors account for variations in dimensions, flow distribution, and neutron flux. The hot channel factors also take into account perturbations in the neutron flux due to the presence of control rods and shim elements

Thermal Design

The following engineering and nuclear hot channel factors are used in developing the core design:

$F_{\Delta T} = 3.36$ $F_{\theta} = 7.36$ $F_Q = 5.17$

$F_{\Delta T}$ is the number by which the average coolant temperature rise through the core is multiplied to get the coolant temperature rise in the hottest channel. This factor is most important in core design because it determines whether bulk boiling occurs in the coolant at the outlet of the hot channel.

The average coolant temperature rise in the core is 34.4 F, based on a heat transfer flow through the core of 33.3 million pounds per hour out of a total coolant flow of 37 million pounds per hour. When 34.4 is multiplied by 3.36 it gives a temperature rise in the hot channel of 115.6 F. Since the coolant enters the reactor at 491 F, the water at the exit of the hot channel is 607 F, correcting for the change in specific heat with temperature, this value reduces to 599 F. Since water boils at 636 F when it is under 2,000 psia pressure, there is a margin of approximately 37 F in the core design before bulk boiling occurs. Uncertainties in instrument readings may reduce this margin. This limitation is one of the basic heat design criteria for the reactor.

F_{θ} is the factor by which the average temperature drop from the overall metal surface to the bulk water, or film drop, is multiplied to get the maximum film drop. This hot channel factor is important in that it determines whether local boiling occurs in the core by specifying whether the maximum metal surface temperature is above the boiling point of the water in the reactor. In the reactor design the average film drop is 14.3 F. Multiplying this figure by an F_{θ} of 7.36, the maximum film drop becomes 105 F. Based on an average metal surface temperature of 568 F, the maximum metal surface temperature would then be 673 F; however, local boiling occurs, which reduces the surface temperature to 642 F. This is 6 F above the saturation temperature of water at 2,000 psia, which is 636 F. Local boiling is accepted within the core. Reactors which have been operated at power levels where local boiling occurred have demonstrated no observable changes in characteristics as a result of the transition to this condition. It is not apparent from reading normal operating instruments on a reactor that local boiling has begun.

The hot channel factor F_Q has significance relative to burnout. In the reactor design the average operating heat flux is 86,800 Btu per sq ft-hr. Applying the hot channel factor F_Q of 5.17, a maximum heat flux of 449,000 Btu per sq ft-hr is obtained. This is compared with a calculated burnout flux of 1.55 million Btu per sq ft-hr. It is apparent from this, that F_Q could double without the maximum heat transfer flux rising to a point close enough to the burnout heat flux to be considered important.

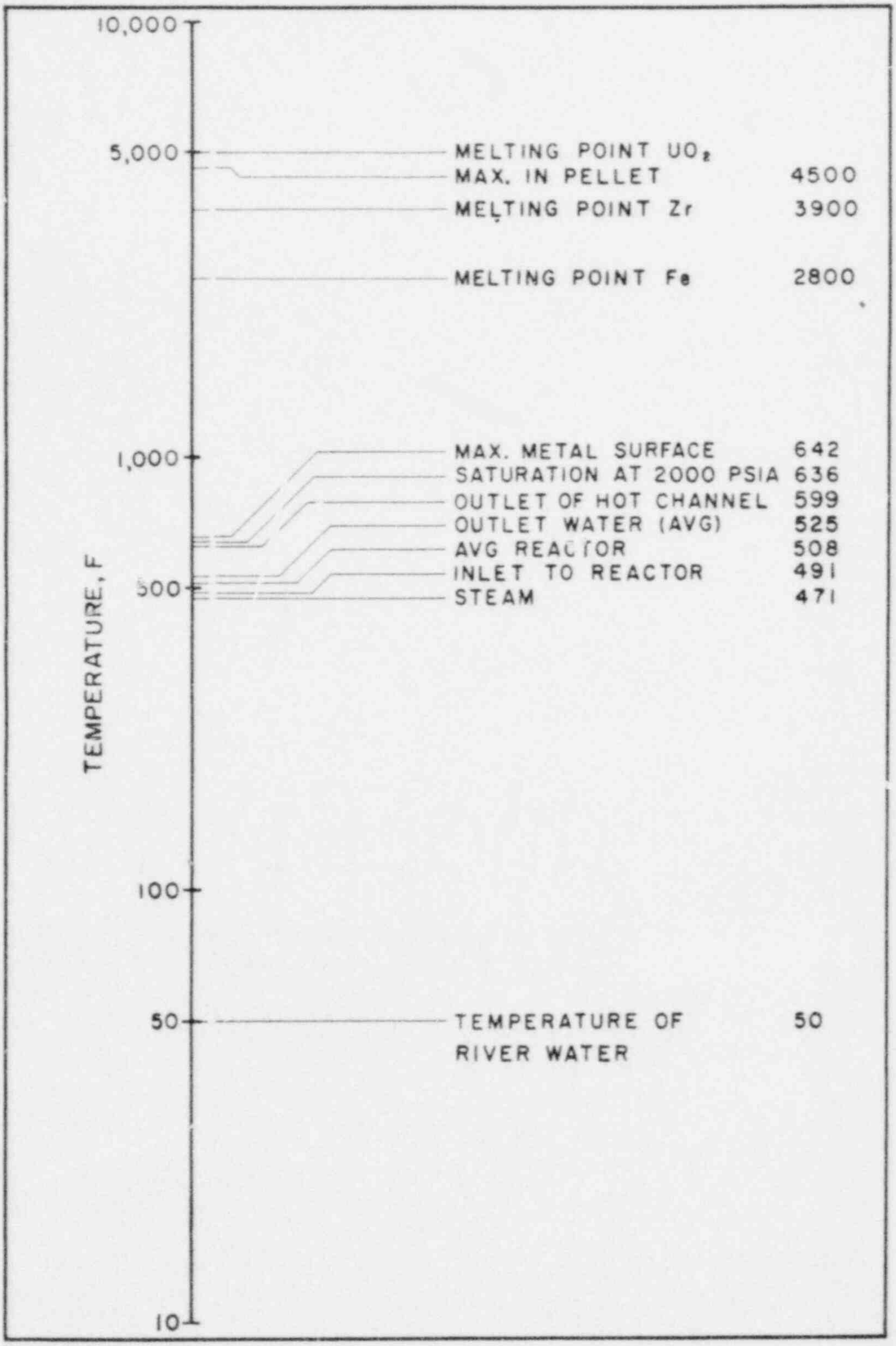
The hot channel factor F_Q also enters into the calculation of the maximum center temperature of the hottest fuel pellet in the reactor core. Based on the maximum heat flux, the center temperature of the hottest pellet is 4,500 F. The melting point of UO_2 is reported at 5,000 F. The significant temperatures of the reactor plant are given in Figure 7.

Hydraulic Design

From the standpoint of hydraulics, the reactor is a single-pass, upward-flow core. Coolant enters the reactor vessel from the four main coolant system piping nozzles at the top of the vessel and is deflected to flow downward through the annuli between the thermal shields at the periphery of the vessel. The direction of coolant flow is reversed at the bottom of the vessel. The coolant flows up through the core and is finally returned to the main coolant system piping through a second set of four nozzles, located at the upper end of the vessel.

Of the design coolant flow in the main coolant system, 90 per cent is estimated to be available for heat transfer purposes. The remaining 10 per cent by-passes the heat transfer surfaces through various passages, such as those about the control rods. As a compromise of mechanical, thermal, and nuclear considerations, a water velocity of 14 fps is selected, giving a pressure drop of 14 psi across the core. The 14 psi includes a local boiling correction of 2 psi and is based on the occurrence of local boiling over 1/3 the length of each fuel rod, which results in a doubling of the pressure drop. These factors represent the most unfavorable conditions of calculation and experiment. The total pressure drop from reactor vessel inlet nozzle to discharge nozzle is estimated to be 28 psi. The thermal and hydraulic characteristics of the reactor core are shown in Table 1.

In the cold condition, the reactor core design allows 1/4 in. between the shoulder on the upper nozzle of each fuel assembly and the lower side of the upper support plate. This clearance provides for differential thermal expansion between the fuel assemblies and the core barrel. In the cold condition, therefore, one or more assemblies could move by as much as 1/4 in. with respect to the remainder of the core, resulting in a reactivity change.



REACTOR TEMPERATURES-INITIAL CORE

Table 1

MECHANICAL DESIGN DATA - INITIAL CORE

Total heat output, Btu/hr	$1,338 \times 10^6$ (392 mw)
Coolant Flow	
Total rate, lb/hr	37.0×10^6
Heat transfer rate, lb/hr	33.3×10^6
Area in fuel rod cross section, sq ft	14.8
Velocity along fuel rods, fps	14.1
Pressure Drop	
Total across vessel, psi	28
Across core, psi	14
Temperatures, F	
Average coolant in core	508
Average coolant rise in core	34.4
Average coolant rise in vessel	31
Average film drop	14.3
Maximum surface	642
Maximum center of fuel	4,500
Outlet of hot channel	599
ΔT_m at exchanger	33.2
Steam temperature (520 psia)	471
Inlet to vessel	491
Heat Transfer	
"Active" surface area, sq ft	15,400
Average flux, Btu/sq ft-hr	8.68×10^4
Maximum flux, Btu/sq ft-hr	4.49×10^5
Average film coefficient, Btu/sq ft-hr	6,060
Burnout flux	
Jens & Lottes correlation	1.55×10^6
Bettis correlation	1.28×10^6
Hot Channel Factors	
Heat flux	5.17
Film drop	7.36
Coolant rise	3.36
Fuel Rod	
Outside diameter, in.	0.336
Gap thickness, in.	0.0017
Tube wall thickness, in.	0.015
Total number of fuel rods	23,218
Fuel length per rod, ft	7.5
Rod lattice, in.	0.420
Equivalent diameter of unit cell, ft	0.0276
Rods per assembly	305 and 306
Total number of fuel assemblies	76
General	
Total core area, sq ft	30.1
Equivalent core diameter, ft	6.2
Length to diameter ratio of core	1.2
Length to diameter ratio of a flow channel	271
Weights	
Fuel, at 10.07 gm/cu cm, lb	53,500
Clad, at 8.03 gm/cu cm, lb	9,000

The effect of coolant flow on the position of the fuel assemblies and, at the same time, the lifting effect on control rods by the coolant have been investigated. The net vertical hydraulic force acting on a fuel assembly or a control rod is the difference between the weight of the assembly or rod and the buoyancy, pressure difference across its length, and skin friction drag. Calculations indicate that the net vertical force on a fuel assembly is downward and its minimum value at room temperature is greater than 300 lb. Calculations for the control rods indicate that the minimum downward force is 200 lb. It is concluded, therefore, that the position of a fuel assembly and the position of a control rod are stable against normal hydraulic forces in the reactor core.

In any case, the maximum reactivity change available from a simultaneous displacement of all control rods by 1/4 in. is less than .00075 $\frac{\Delta k}{k}$.

Heat Output as a Function of Time After Shutdown, Decay Heat

The heat produced by beta and gamma radiation after reactor shutdown and following an infinite period of operation at constant power, is given in Figure 35. Curve A shows the instantaneous rate of heat generation as a function of time following shutdown. The heat generation rate is expressed in terms of per cent of the constant power level preceding shutdown. The important range of the curve is at the level of 2-3 per cent for most of the time between 100 and 1,000 sec and 1-2 per cent between 1,000 and 10,000 sec.

Curve B in Figure 35 shows the integrated heat output as a function of time after shutdown. This curve gives data which may be converted to pounds of water evaporated. For example, from shutdown to 1 hr after shutdown, the weight of water evaporated would be approximately 25,000 lb, 400 cu ft.

These curves have been calculated for an assumed infinite period of operation prior to shutdown. The effect of shorter periods of operation is significant, and would reduce the decay heat production at any given time. For example, if there had been only 10 hr of operation prior to shutdown, the decay rate at 1,000 sec following shutdown would be 0.9 per cent of initial power, or about half that shown in Figure 35. The infinite operation decay curve is selected for convenience and to represent a "worst case" situation.

The actual power level at which the reactor is assumed to have operated can be any value up to the maximum capability of the reactor. The nominal design rating of the reactor with the initial core is 392 mw of heat or 1,338 x 10^6 Btu per hr. The maximum reactor main coolant loop system capability is designed for 492 mw of heat or 1,680 x 10^6 Btu per hr. All of these values are based on four loop operation of the plant.

Nuclear Design of the Reactor Core

General

The reference design was selected from a preliminary nuclear parameter study of the reactor core. The nuclear design data are summarized in Table 2. The study involved a determination of the reactivity lifetime of various core configurations in which water-to-metal ratio and stainless steel clad thickness were independent variables. These calculations were carried out using the IBM-704 electronic digital computer. The computer code used for the calculations is referred to as the CAP-1 Code. It is based on a uniform burnout of fuel within the core and takes account of resonance absorption in Pu^{239} and Pu^{240}. In this study, the heat transfer area and the UO_2 loading were held constant.

Core lifetime of 13,300 hr is predicted by the CAP-1 Code calculation using an initial enrichment figure of 2.6 atom per cent. Since the assumption of uniform burnout gives a longer than actual lifetime and since previous calculations have indicated that the true lifetime is approximately 30 per cent lower than that calculated on the basis of uniform burnout, it is reasonable to state that a fuel enrichment of 2.6 per cent gives the required, stated lifetime of 10,000 hr. These figures are based on reloading the entire core at one time.

The selection of 0.015 in. cladding for the reference design is based on a compromise of nuclear considerations and structural limitations. The 0.42 in. pitch between rod centers is a compromise between excessive coolant flow and increased loading requirements. This pitch yields a net fuel cost which is close to the minimum value for a wide range of values for plutonium credit. The equivalent water-to-uranium volume ratio corresponding to this pitch is 2.8 where the equivalent metallic uranium volume is based on a density of 18.7 g per cu cm.

Criticality calculations are made using a modified one-group method with an equivalent Fermi age for the thermalization process. Cross section data were obtained from BNL-325. The fast effect, ϵ , is determined by Hellens equation, and empirical relationship based on a heterogeneous core.

$$\epsilon - 1 = \frac{0.1565}{1 + 0.875\, \rho_w \frac{V_w}{V_u} + 0.288 \frac{V_s}{V_u}}$$

Where:

ρw = density of water

Vw = volume of water

Vu = volume of uranium

Vs = volume of structural material

Table 2

NUCLEAR DESIGN DATA - INITIAL CORE

Power as heat, mw	392
Lifetime, hr	10,000
Burnup, mw days/metric ton uranium, avg	7,600
Pressure, psia	2,000
Coolant average temperature, F	508
Core average diameter, in.	74.4
Core active height, in.	90
Reactor buckling, B^2, cm^{-2}	.000722*
Active Volumes	
Fuel, cu in.	147,000
Water, cu in.	195,000
Zirconium, cu in.	12,600
Stainless steel, cu in.	31,300
Voids, cu in.	4,940
Total, cu in.	391,000
Weights	
Fuel, lb	53,500**
Stainless steel, lb	9,000
Zircaloy-2, lb	2,980
Total, lb	65,500
Volume ratio, H_2O/UO_2	1.33**
Volume ratio, H_2O/U	2.80***
Initial enrichment, atom per cent	2.6
Final enrichment, atom per cent	1.9
Atom ratio, H/U-238 (hot)	3.14
Atom ratio, H/U-235 (hot)	118.0
Typical Performance Data	
Initial conversion ratio	0.733
Cumulative conversion ratio	0.636
k_{eff}	
Cold, clean and max	1.186
Hot, clean	1.113
Hot, new, equilibrium Xe and Sm	1.067
Final, equilibrium Xe and Sm, full power	1.000
Pu produced, kg	104.0
Pu-240 content, per cent	9.39
Fraction of energy from:	
U-235	0.704
U-238	0.068
Pu	0.228
Nuclear Parameters	
p, resonance escape probability	0.727
ϵ, fast fission factor	1.04
τ, neutron age, sq cm	55.5

*Assumes reflector savings of 7.5 cm
**Based on uranium dioxide density of 10.07 g/cu cm
***Based on uranium density of 18.7 g/cu cm

The thermal utilization, f, however, is calculated on the assumption of a homogeneous core. This simplification in analysis overestimates f only by 0.3 per cent. The usual representation for resonance escape, p, is modified by the Dancoff correction for rod shielding.

The worth of a central "black" control rod in the hot reference core has been calculated to be approximately 1 per cent $\frac{\Delta k}{k}$. If no interaction effects are assumed and the worth of eccentric rods is assumed to vary as $J_o^2(B_T r)$ where $J_o(B_T r)$ is the primary mode of the thermal neutron flux distribution, the worth of the 24 control rods is between 10 and 12 per cent $\frac{\Delta k}{k}$. Eight "black" neutron absorbing shim elements around the outside of the core have a reactivity value of 0.8 per cent.

A chemical shutdown system is provided which controls reactivity as indicated in Figure 13.

Temperature Coefficient of Reactivity

The temperature coefficient of reactivity has been calculated for the hot and cold reactor core. The reactivity changes fall into two categories:

Moderator temperature coefficient which results from change in absorption, resonance escape, leakage and fast fission effect due to a change in water density and a change in effective cross sections as a result of a change in the neutron temperature

The Doppler coefficient which results from changes in resonance absorption due to the Doppler broadening of neutron capture resonances in U-238

The moderator temperature coefficient of reactivity is sometimes referred to as the "slow" coefficient, since there is some time lag before the water becomes heated, or cooled, and its density changes. The neutron temperature which affects the coefficient through the cross sections for the various reactor materials suffers a similar lag, as it is largely determined by the moderator temperature. The procedure used to determine the moderator temperature coefficient is to differentiate the 4-factor formula for k infinity and obtain the temperature coefficient due to each of the factors. The Hellens formula is used to calculate ϵ at temperatures above and below room temperature and operating temperature. The temperature coefficient for the fast fission effect is obtained from a plot of these data. The contribution to the temperature coefficient by changes in resonance escape probability is determined by calculating p with the homogeneous resonance escape formula at several temperatures above room temperature and the operating temperature. The value determined from the data in the operating temperature range is checked using the heterogeneous resonance escape probability formulation.

The contribution to the temperature coefficient arising from neutron leakage is calculated using the equation,

$$\frac{1}{k}\left(\frac{\partial k}{\partial T}\right)_{leakage} = \frac{B^2}{1 + M^2B^2} \quad \frac{\partial M^2}{\partial T}$$

where the partial derivative of the migration area M^2 is determined from a plot of migration area vs temperature in the room temperature and operating temperature ranges. In calculating the contribution due to changes in thermal utilization with temperature, the only nuclear density change which is large enough to be appreciable is that of water. The cross section variations which show up in thermal utilization are those which result from changes in "non-1/v factors" since, practically speaking, the 1/v variation of most neutron absorption cross sections are cancelled out.

The result of these calculations are given in Table 3.

Table 3

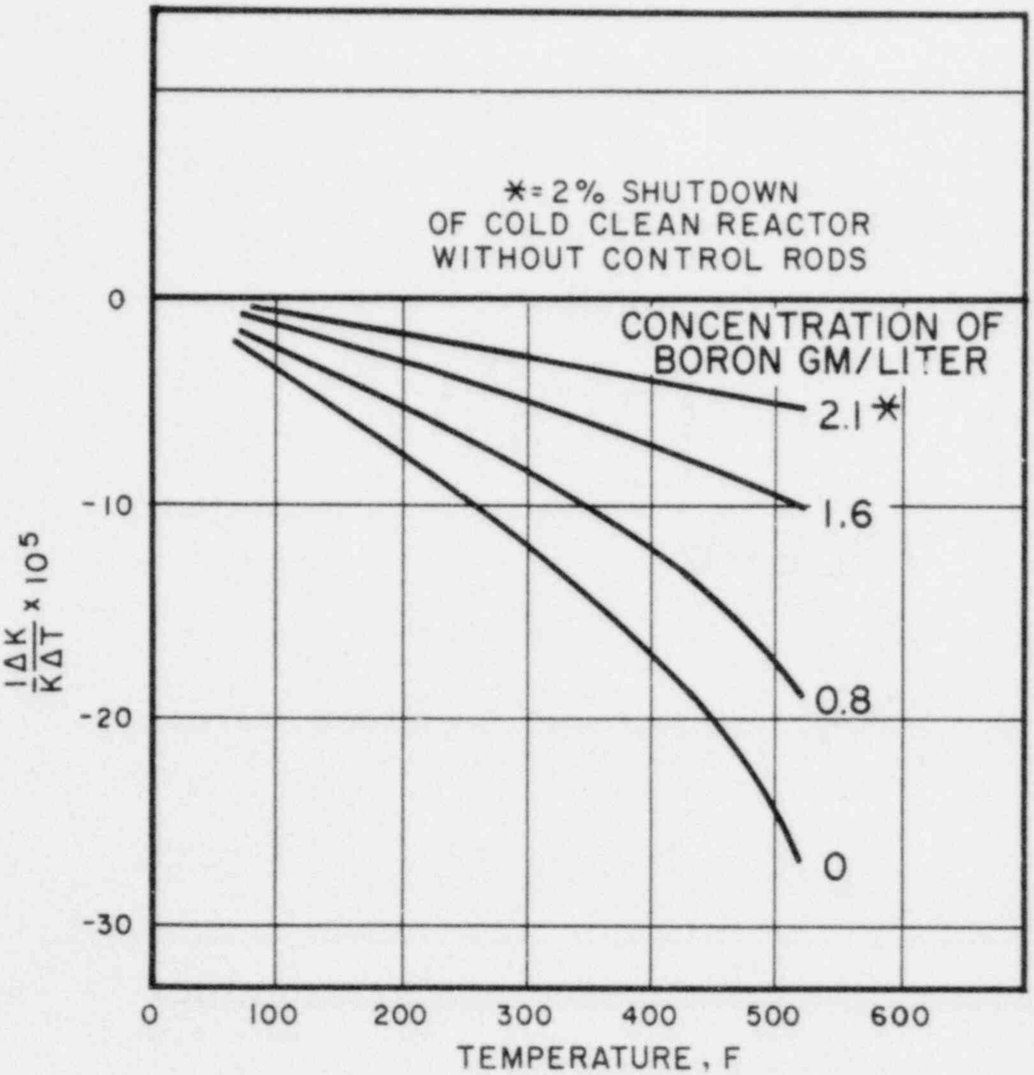
Temperature Coefficient of Reactivity

Water Temperature	68 F	508 F
Contribution from:		
Fast effect	+.5 x 10^{-5}/F	+4 x 10^{-5}/F
Resonance escape	-2.5	-33
Thermal Utilization		
Without chemical neutron absorber	-.3	+6
Leakage	-.4	-4
Total, without chemical neutron absorber	-2.7 x 10^{-5}/F	-27 x 10^{-5}/F

The effect of chemical neutron absorber in the coolant moderator during plant warmup is such as to decrease the negative temperature coefficient. However, it is anticipated that there will never be enough chemical poison present to cause the temperature coefficient to go positive. The concentration required for a zero coefficient is 2.6 g of boron per liter of coolant in the hot reactor, or 2.3 g per liter when the reactor is cold. However, 2.1 g per liter provides 2 per cent shutdown of the cold clean core without any control rods. Figure 8 shows the temperature coefficient of reactivity vs temperature at several boron concentrations. Figure 8A shows the temperature coefficient of reactivity vs boron concentration at three different coolant temperatures. The reactor is always subcritical at boron concentrations which could result in positive temperature coefficients, even with all control rods withdrawn.

Pressure Coefficient of Reactivity

In the primary plant, reactor plus main coolant system, the nominal system pressure is 2,000 psia. Since the temperature controls for the pressurizer work over a finite range, and since there are surges in the system due to changing flow of the coolant, the actual operating pressure may fluctuate above and below the nominal system pressure of 2,000 psia. The pressure swings are calculated to be ± 150 psi. With changes in system pressure, the density of the moderator in the reactor changes, giving rise to an increase or decrease in reactivity. The effect may be described as a pressure coefficient of reactivity. Since this factor is a function, among other things, of the total neutron absorption in



TEMPERATURE COEFFICIENT OF REACTIVITY vs TEMPERATURE

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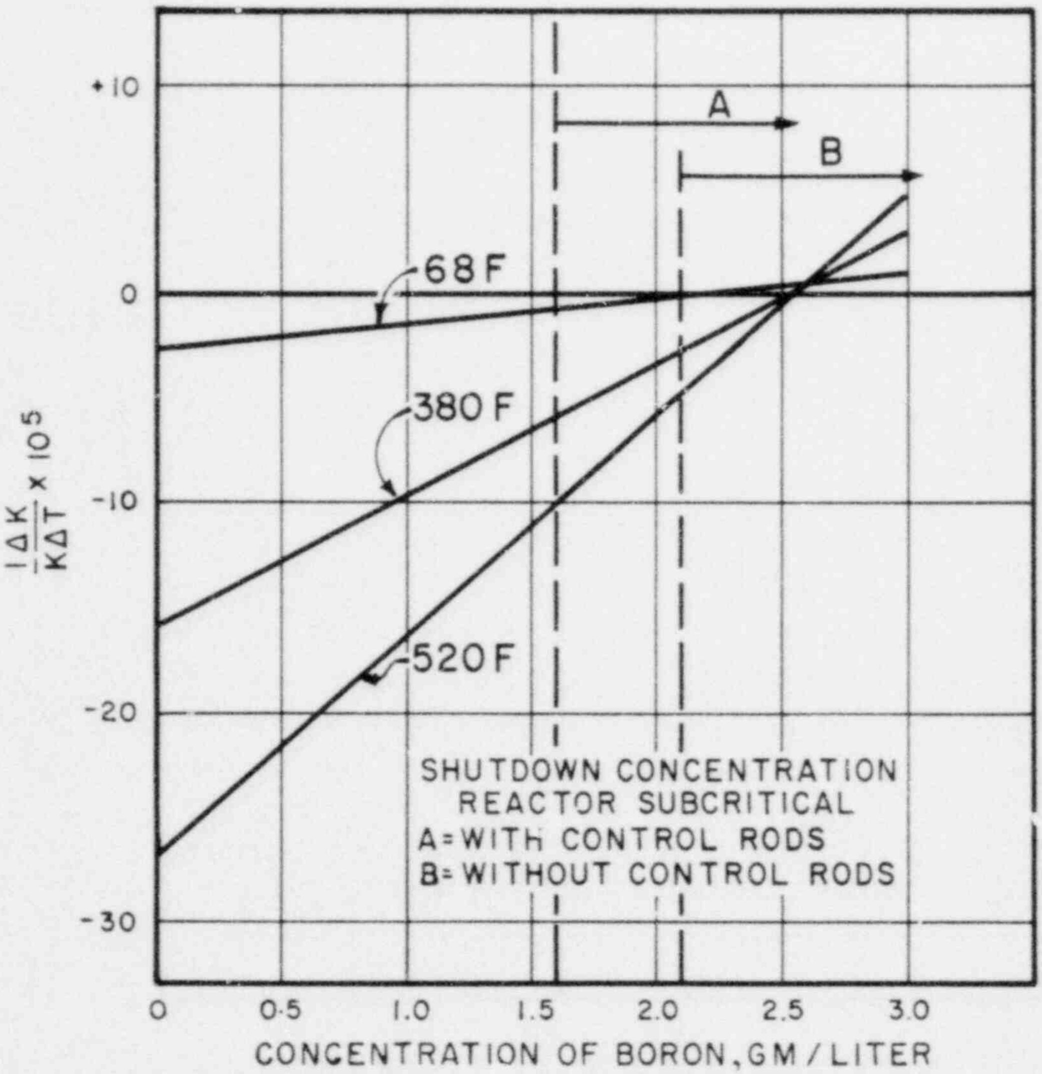

TEMPERATURE COEFFICIENT OF REACTIVITY vs BORON CONCENTRATION

the moderator, two values have been computed, one without chemical neutron absorber in the coolant and one with chemical neutron absorber. The data are shown in Table 4.

Table 4

Pressure Coefficient of Reactivity

Water temperature 508 F	System pressure 2,000 psia
Without chemical neutron absorber	$+2.8 \times 10^{-6}$ per psi
With chemical neutron absorber (1.6 g boron per liter)	$+1.0 \times 10^{-6}$ per psi

The pressure coefficient of reactivity of the reactor is positive. During plant transients, this coefficient opposes the temperature coefficient, since positive pressure surges occur simultaneously with positive temperature surges. The pressure coefficient, being smaller, never overrides the temperature coefficient, but reduces somewhat its effectiveness.

The pressure changes in the primary system due to changes in the temperature within the pressurizer, which result from the on-off type of control, are smaller than those associated with plant transients and, in general, take place over a relatively long period. It is difficult to see how any hazard could be associated with pressure changes.

Doppler Coefficient of Reactivity

The reactor fuel is 2.6 per cent enriched uranium dioxide. Since this is a homogeneous fuel, that is, the U-235 and U-238 are intimately mixed, the temperature of the fissionable (U-235) and fertile (U-238) materials are the same. As a result, the broadening of the neutron absorption resonance peaks in U-238 with increasing temperature is a rapid effect and results in a "prompt" negative temperature coefficient of reactivity.

The Doppler effect is caused by the spread in relative velocities between neutrons of a given vector velocity and uranium nuclei with various vector velocities in such a manner that the effective widths of absorption resonances are increased, thus decreasing the self-shielding of uranium nuclei.

The U-238 resonance integral has been measured to have a temperature coefficient of + 1 x 10^{-3}/C (Nucleonics Vol. 10, No. 5, 64, 1952). Differentiating the expression for resonance escape with respect to temperature, the following expression is obtained:

$$\frac{1}{p}\frac{\partial p}{\partial T} = -\left(\frac{N_0}{\xi\Sigma_s}\right)\frac{\partial\,\text{Resonance Integral}}{\partial T}$$

This expression is evaluated and the data are shown in Table 5.

Table 5

The Doppler Coefficients

-.7 x 10^{-5} per deg F, at 68 F

-.8 x 10^{-5} per deg F, at 508 F

An additional "prompt" coefficient due to uranium dioxide density change with temperature is estimated to be an order of magnitude smaller than the Doppler effect and is, therefore, neglected.

Void Coefficient of Reactivity

Two of the basic assumptions in the design of the reactor core are that local boiling, surface boiling of the subcooled liquid, is permissible within the core but that bulk boiling is not allowed. The presence of local boiling does not alter the reactivity of the reactor provided it is restricted to a small region of the core.

The reactor core is designed so that, in normal operation, bulk boiling does not occur even in the hottest channel. Under accident conditions, however, it is conceivable that bulk boiling may occur. The reactivity of the reactor then may be expected to be altered by the presence of steam voids. The effect of steam voids on reactivity is evaluated quantitatively and expressed as a coefficient of reactivity. In the operating range where the average temperature of the coolant is 508 F, the void coefficient of reactivity is negative with a value of -0.3% $\Delta k/k$ per % void. The effect of voids on the k_{eff} of the core without chemical poison is shown in Figure 9, in which curves are plotted for three mean core temperatures. As the temperature of the reactor is lowered, k_{eff} increases and more reactivity becomes available; therefore, a larger per cent void is required to shut down the reactor.

A change in system pressure may be expected to have an effect on the void volume. Given a set of initial conditions, if the system pressure were to increase as a result of reactor instrumentation calling for heat to be added to the pressurizer, the voids would be reduced in volume. The time required by the pressurizer to go from the bottom of the dead band of 1,850 psia to 2,000 psia is 16.5 minutes. If a maximum 10 per cent void is assumed, which is an extreme estimate of the voids due to local boiling, and if -0.3% $\Delta k/k$ per % void is used for the void coefficient of reactivity, the rate of reactivity addition will be 3.0×10^{-5} Δk per sec. This is approximately one third the maximum rate of reactivity change associated with operation of the control rods.

Although operation of the reactor is predicated on little or no boron in the coolant under power operating conditions, the effect of boron on the uniform void coefficient has been investigated. Figure 9A shows the effect of boron concentration on uniform void coefficient at three different temperatures.

Effects of Plutonium Buildup on Core Characteristics

At the end of the core life, it is anticipated that approximately one third of all fissions will take place in the plutonium-239 which has built up following capture of neutrons by U-238. At this point, the average plutonium "enrichment" is approximately 0.5 per cent compared to the initial U-235 enrichment of 2.6 per cent. The half-life of the delayed neutron precursors, as well as the delayed neutron fraction for Pu-239, is different from that for U-235. Since the plutonium is built up in the reactor core in regions of high statistical weight, the effects of plutonium on the characteristics of the reactor have been analyzed in some detail.

At the time of this writing, no experimental information is known to be available on the effects of plutonium on the transient characteristics of light water moderated, low enrichment power reactors. Most of the experimental information available comes from Hanford and applies to graphite moderated reactors. Since the graphite moderated reactor has a different neutron spectrum than a light water moderated reactor, extrapolation of this information to the Yankee core is difficult. Nevertheless, the Hanford information indicates that the plutonium effects on moderator temperature coefficient are small compared to the temperature coefficient produced by water density changes in a light water moderated reactor. Some Pu-239 Doppler coefficient measurements have been made in conjunction with the fast breeder reactor program at Argonne National Laboratory. Again, this information can at best be used only qualitatively on the Yankee reactor because of the different neutron spectrum.

While confirmation of this particular phenomenon by experimental and operational information is lacking, calculations which appear to be generally well corroborated by experiment indicate that plutonium buildup will have little effect on the kinetic characteristics of the Yankee core. The following paragraphs summarize the results of this analytical work.

Steady State Effects on Reactivity and Flux Distributions: Figure 9B is a plot of available k_{eff} versus core life showing the effect of buildup of plutonium, depletion of U-235 and U-238, and buildup of U-236 and fission products. This was obtained from the CANDLE* four group pointwise burnout code which accounts for nonuniform burnout of fuel. Control was assumed to be in the form of a homogeneously distributed poison. The calculations indicate that, while the rate of loss of reactivity is small during the initial part of core life, the available reactivity decreases throughout core life. Thus, no harmful reactivity changes are anticipated as a result of plutonium buildup.

Figure 9C is a plot of the thermal neutron flux as a function of radius, after having been averaged axially, at the beginning and end of core life. This was obtained from the same CANDLE calculation and it also assumes a uniform control poison. The effect of control rods would be to flatten somewhat the distribution at the beginning of core life and to raise the peak at the end of life, depending on the rod program.

Moderator Temperature Coefficient: Plutonium affects the moderator, or slow temperature coefficient because temperature rises in the moderator cause the neutron spectrum to move up in median energy. Because of the large Pu-239 resonance at 0.3 ev it would be expected that this spectrum shift would result in a larger percentage of captures by that isotope. The increase in neutron absorption by plutonium does not necessarily cause a large rise in reactivity per degree rise in moderator temperature, however, because the ratio of radiative capture to fission (α) also rises in the resonance region. The total reactivity change due to this effect could still be important with a large change in moderator temperature in a core which does not have an overwhelmingly large compensating moderator density change. Large plutonium effects might, therefore, be expected in reactors with graphite or beryllium moderation. The pressurized water reactor, however, can have only a small total rise in moderator temperature (the relief valves in the primary system limit the pressure and, hence, temperature rise) and has such a large negative temperature coefficient due to density changes at operating temperature that the plutonium effect is quite small in comparison. Calculations have shown

*WAPD-TM-53, "Candle - A One Dimensional Few-Group Depletion Code for the IBM-704", May, 1957 that the total change in the moderator coefficient at 516 F with a plutonium buildup to 0.55 per cent of total fuel during the period of core life results in only a 3 per cent decrease in absolute magnitude of the negative coefficient.

At lower temperatures the effect of plutonium is more pronounced simply because the water density change is smaller. This should not be a serious problem in the Yankee core, however, as the start-up procedures will not allow criticality to be attained at a temperature below 250 F. At this temperature the negative moderator coefficient suffers a 14 per cent decrease in absolute magnitude over the core life, which is not excessive from the standpoint of safety.

The problem involved in determining the changes in the temperature coefficient of the reactor as a function of burnup will be outlined briefly below. An independent check of the calculations on the effect of plutonium buildup on the temperature coefficient of the Yankee reactor has been made by NDA. The work is reported in NDA 2072-1 "Effects of Plutonium Buildup on Temperature Coefficient", W. L. Brooks, H. Soodak, April 1, 1958, which is attached as Appendix I to Part B of the license application. This study, which consisted of both computations and an information survey, corroborated the results and conclusions outlined here. It should be pointed out that the core studied here is the latest modification and is slightly different from the one considered in other parts of this Hazards Report. Thus, the calculations give answers differing in detail from those reported elsewhere in this document.

The formula used for the effective multiplication k of the reactor is the standard one:

$$k = \frac{\eta f \epsilon p}{(1 + \tau B^2)(1 + L^2 B^2)}$$

The temperature coefficient may be expressed as:

$$\frac{1}{k}\frac{\partial k}{\partial T} = \frac{1}{\eta f}\frac{\partial(\eta f)}{\partial T} + \frac{1}{\epsilon}\frac{\partial \epsilon}{\partial T} + \frac{1}{p}\frac{\partial p}{\partial T} - \frac{1}{1 + L^2 B^2}\frac{\partial(1 + L^2 B^2)}{\partial T} - \frac{1}{(1 + \tau B^2)}\frac{\partial}{\partial T}(1 + \tau B^2)$$

Calculations have been carried out on these partial coefficients, for the cold-clean, the hot-clean, the cold-end of life, and the hot-end of life reactors. A brief summary of the results is shown in Table 5a.

Table 5a

Yankee Core Temperature Coefficients at Beginning and End of Life

Partial Coefficient	Cold-Clean	Hot-Clean	Cold End of Life	Hot End of Life
	50 - 86 F	498 - 534 F	50 - 86 F	498 - 534 F
$\frac{1}{\eta f} \frac{\partial \eta f}{\partial T}$	$+0.34 \times 10^{-5}$	$+3.76 \times 10^{-5}$	$+3.42 \times 10^{-5}$	$+4.18 \times 10^{-5}$
$\frac{1}{\epsilon} \frac{\partial \epsilon}{\partial T}$	$+0.19 \times 10^{-5}$	$+2.40 \times 10^{-5}$	$+0.19 \times 10^{-5}$	$+2.40 \times 10^{-5}$
$\frac{1}{p} \frac{\partial p}{\partial T}$	-2.64×10^{-5}	-32.0×10^{-5}	-2.64×10^{-5}	-31.7×10^{-5}
Leakage Terms	-0.43×10^{-5}	-7.03×10^{-5}	-0.49×10^{-5}	-6.92×10^{-5}
$\frac{1}{k} \frac{\partial k}{\partial T}$	-2.5×10^{-5}	-33×10^{-5}	$+0.5 \times 10^{-5}$	-32×10^{-5}

The details of these calculations are contained in Westinghouse Report YAEC-73 to be issued in July 1958 and which will be filed as Appendix II to Part B of the license application.

A brief study of these results leads to several important conclusions. First, the hot reactor at temperature will clearly have a strong negative coefficient. This is due to the effect of water expansion on p, the resonance escape probability. Second, the cold-clean reactor should have a small, but negative, temperature coefficient. The largest single term in making the cold-clean temperature coefficient negative is the effect of resonance escape probability. Third, in the cold, end of life reactor the magnitude and even the sign of the temperature coefficient is uncertain. This is due for the most part to an increase in the effect of the ηf contribution. This increase is attributable to the non $\frac{1}{v}$ behavior of the plutonium formed during the reactor life, as mentioned above. The important point is that the overall result of $+0.5 \times 10^{-5}$ is made up of several large positive and negative contributions. These contributions in the case of ηf and p are five to six times the magnitude of the final result. It is clear that no possible calculation which can be done at present can hope to establish the cold end of life temperature coefficient accurately enough to be of value.

each isotope by the fraction of fissions occurring in that isotope. The table below shows the values obtained. The drop in the delayed fraction from .00699 to .00556 is not large enough to make the presently specified control rod withdrawal rates unsafe under the most pessimistic conditions.

Table 5b

Variation of Delayed Neutron Fraction and Prompt Neutron Lifetime During Core Life

Time	Beginning 0	Middle 5,000 Hr	End 10,000 Hr
Delayed Fraction	.00699	.00609	.00556
Prompt Life, 68 F	2.4×10^{-5} sec	-	2.1×10^{-5} sec
Prompt Life, 516 F	2.5×10^{-5} sec	-	2.1×10^{-5} sec

The prompt neutron lifetime is of importance in determining the rapidity of transients occurring if prompt critical is reached. If the reactor is below prompt critical, the period is of course related to the much longer delayed neutron lifetimes. The largest contribution to the prompt neutron lifetime is the diffusion time of thermal neutrons, given approximately by $1/v_s \Sigma_a$. Plutonium has an effect on this quantity only because it has a larger cross section than the atoms it replaces. This increases both v_s because of spectrum hardening, and Σ_a, but the change in prompt neutron lifetime is not large as can be seen from the values given in the table, above. A good part of the change, which occurs is due to the buildup of fission product poisons, increasing Σ_a and, therefore, v_s, and is independent of plutonium buildup.

Effects of Nonuniform Plutonium Distribution: Whatever changes occur in the core as a function of burnup will be more pronounced at the center where the flux has been higher than average. Thus, there should be more plutonium buildup near the core center than at the edge. Since the statistical weight of the center region is higher than average, the effects of plutonium should thus be larger than would be obtained by the calculations discussed above which assumed a uniform distribution of plutonium throughout the core. It is expected that the maximum plutonium concentration in the center of the core will be .82 per cent of the fuel, or 1.5 times greater than the average value of .55 per cent.

In order to be certain of the safety of the system, the temperature coefficient, void coefficient, delayed neutron fraction, and prompt neutron lifetime were recalculated for the same reactor with an assumption of three times the expected information because the neutron spectrum seen by the sample would not be the same as that which exists in a full-sized core at elevated temperatures containing appreciable quantities of plutonium. The reason for this is that large quantities of plutonium will have a marked effect on the neutron spectrum because of the plutonium resonance structure. This fact is corroborated by experience at the Westinghouse Bettis Plant in which small samples of control rod materials indicated considerably different rod worth values than were obtained from full-scale control rod worth measurements. Even if a large number of plutonium-bearing rods were made up and inserted in the critical experiment, the effect of spatial distribution and neutron leakage could not be simulated because of the great difference in reactor size. The critical experiment contains roughly 5,000 lb of UO_2, whereas the loading for the first core of the Yankee reactor is over 50,000 lb.

A more practical program to determine the effects of plutonium buildup on the kinetic characteristics of the reactor would be to make measurements in the operating reactor itself. Since there will be no plutonium in the core at the beginning of life, and the fuel distribution will be uniform, analytical information coupled with experimental data from the part core critical experiment can satisfactorily predict the kinetic characteristics of the full-scale reactor before there is a buildup of plutonium. This information can be verified during the initial checkout of the core at the reactor site. As small amounts of plutonium build up in the reactor, measurement of temperature coefficient and other quantities affecting the operation of the reactor can be made in the actual plant, and a history obtained for the various quantities of interest. The information obtained from such a program is directly applicable, obviously, no extrapolation to the operation of the reactor being required.

Measurements will be made initially and after 2,000 hr of core life. Subsequent measurement schedules will be based on the findings of the first measurements. Any adverse change in the kinetic characteristics of the core will then be discovered before it can introduce an appreciable effect on the operation of the reactor. An experimental program of this nature will not only ensure the safe operation of the reactor, but also provide valuable information which may be used in the design and operation of other pressurized water power reactors.

<u>Doppler Temperature Coefficient</u>: Plutonium has a wealth of resonance structure which affects the Doppler coefficient. The fission resonances in U-235 have a similar effect. Information from fast reactor projects has indicated that both U-235 and Pu-239 give a positive contribution to the Doppler coefficient; however, Pu-239 gives a smaller contribution than U-235. U-238 also has a Doppler coefficient due to resonance absorption of neutrons. It has been well substantiated that this is a negative effect. Also, the Doppler coefficient of a mixture of U-238 and U-235 does not go to zero until the proportions are in the region of 2 to 1 or 1 to 1; thus, the effect of U-235 and Pu-239 concentration changes on the Doppler coefficient should be small in the Yankee reactor where their combined concentrations will be less than 4 per cent of the U-238 present. Therefore, it is concluded that, throughout the core life, there is no significant change in the overall Doppler coefficient due to buildup of plutonium. The presence of higher plutonium isotopes should not change this conclusion significantly because any positive contribution from fissionable Pu-241 should be nullified by the pure absorption resonance in Pu-240 at 1 ev.

<u>Void Coefficient</u>: Plutonium is expected to have much less effect on void coefficients than on temperature coefficients. Plutonium enters into the moderator temperature coefficient through variations in neutron temperature and the accompanying change in its effective absorption cross section. However, the process of void formation at constant temperature involves little neutron spectrum change. Calculations indicate that the negative void coefficient of reactivity actually increases in absolute magnitude by 12 per cent of its initial value during the core life.

<u>Delayed Neutron Fraction and Neutron Lifetime</u>: Keepin's recent data on delayed neutron fractions* shows the fraction of delayed neutrons (β) from thermal fission of Pu-239 to be .0021. This compares with values of .0064 for U-235 and .0157 for fast fission of U-238. Thus, one effect of plutonium buildup will be to lower the average fraction of fission neutrons which are delayed. Prompt critical is, therefore, reached at a somewhat lower value of excess reactivity.

The effective delayed neutron fractions at beginning, middle, and end of life have been calculated by weighing β for

*G. R. Keepin, T. F. Wimett, and R. K. Zeigler, "Delayed Neutrons from Fissionable Isotopes of Uranium, Plutonium and Thorium", Journal of Nuclear Energy <u>6</u>, p. 1, (1957).

average final plutonium concentration. The results obtained with such an extreme assumption have given added assurance that plutonium gives rise to no untoward effects. Table 5c shows the values obtained in this study and compares them with the values found for the clean new core. The temperature and void coefficients are actually improved, while the delayed fraction and prompt life show some adverse effects which are not of major importance.

Table 5c

Kinetics Parameters for End of Life Core
Assuming Triple Pu Concentration

	Clean New Core	End of Life With $3 \times N^{Pu}$
$\frac{1}{k}\frac{\partial k}{\partial T}$, 300 F	-13×10^{-5}	-15×10^{-5}
$\frac{1}{k}\frac{\partial k}{\partial T}$, 516 F	-33×10^{-5}	-36×10^{-5}
$\frac{\Delta k}{k}$ per % void, 516 F	-.3%	-.4%
Delayed Neutron Fraction	.0070	.0047
Prompt Neutron Life, sec, 516 F	2.5×10^{-5}	1.4×10^{-5}

Conclusions: It appears from the analytical work described above that transient problems in the Yankee reactor could exist only at low temperatures, that is, below, say, 150 F. Since the magnitude and the sign of the moderator temperature coefficient at room temperature are in doubt, even if effects of plutonium are not considered, the reactor temperature will be raised by heat from an external non-nuclear source to at least 250 F, where the moderator coefficient is -10×10^{-5}/F, before it is brought to critical. The negative effect of the component of the moderator temperature coefficient due to water expansion will, therefore, always be greater than any effects introduced by plutonium buildup.

Because of the lack of experimental information available on the kinetic effects of plutonium, some thought was given to performing experimental work at the WREC in conjunction with the Yankee part-core critical experiments. However, detailed consideration of such a program indicated that it would not provide information pertinent to the operation of the Yankee reactor. The use of plutonium samples in the core, for example, for danger coefficient measurements, would not yield valid

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Cold Reactor at Beginning of Life

The uranium-235 enrichment required to give this reactor a 10,000 hour life at 392 mw is 2.6 per cent. As shown in reactivity Table 6, the k_{eff} of the cold, clean reactor is 1.186. This is controlled by a combination of control rods and a chemical neutron absorber, boron, dissolved in the coolant. As the reactor is brought to the average operating temperature of 508 F, the k_{eff} drops to 1.113. The objective is that the hot clean core be controlled by the rods alone. The k_{eff} decreases to 1.067 after a few days of reactor operation when the xenon and samarium neutron absorbers have reached equilibrium values. From this value, it decreases steadily throughout the full-power life, which ends when k_{eff} reaches unity. Effective multiplication factor, k_{eff} vs temperature is plotted in Figure 10.

Table 6

Reactivity
(Control Rods Withdrawn)

	k_{eff}
Cold, clean	1.186
Hot, clean	1.113
Hot, new, equilibrium Xe and Sm	1.067
10,000 hr, equilibrium Xe and Sm	1.000
10,000 hr, no Xe	1.027

Hot Reactor With Chemical Neutron Absorber

The hot, clean reactor has a $k_{eff} = 1.113$. If 1.6 g per liter of natural boron, the quantity of neutron absorber necessary to control the cold core with rods, is left in the main coolant loop, the k_{eff} of the hot core at 508 F would be 0.943 without the rods. If the rods are inserted, this decreases to 0.855. These numbers all decrease as the energy withdrawal causes fission product build-up and formation of plutonium. Xenon poisoning in its various stages also causes these numbers to decrease.

Xenon Transients

Of all the various xenon transients, the one of most interest from the hazards standpoint is the one which causes the greatest drop in k_{eff} per unit time. This case occurs when the reactor is brought to full power during a maximum xenon override.

After shutdown from full power, the Xe-135 concentration rises to a peak at 8.1 hr. If the reactor is started during the peak xenon concentration, the xenon burns out rapidly to below its equilibrium value, because no iodine precursor was formed during shutdown. When the reactor is started up, the iodine is formed again, and gradually comes back up to equilibrium. During the initial stage, burnout is rapid. It must be established that available control rates can handle the maximum rate of decrease in neutron absorption cross section and the consequent increase in reactivity.

Figure 11 shows this situation for the case in which a new reactor core is run to equilibrium at full power, shutdown, and then started at the time of maximum xenon. This maximum xenon gives a Σ_a = .0071 cm^{-1} at 8.1 hr after shutdown, compared to .00415 cm^{-1} at equilibrium. Upon starting up again, Σ_a^{Xe} drops to about .0032 cm^{-1} after 8.9 hr and then rises towards equilibrium. The rate of change Σ_a^{Xe} is plotted and converted into dk/dt. The rate of change of k_{eff} with time is maximum at start-up with a value of $+3.5 \times 10^{-6}\% \frac{\Delta k}{k}$ per sec. This can be handled easily by the control rod system and thus presents no possibility of a runaway.

The xenon instability problem, xenon tilt, as described in CRRP-657 by A. G. Ward of Canada, has been investigated.

For the Yankee core, the migration area is approximately 56 square centimeters; the length of the core is 7 1/2 ft; and the core diameter is 6.2 ft. The square of the core height in feet divided by the migration area equals 1, and the square of the core diameter in feet divided by the migration area equals 0.68. Based on these ratios, and according to Dr. A. Henry, a xenon oscillation in the Yankee reactor is possible; however, it is not probable.

The oscillation has a 30 hr time constant. The oscillation becomes appreciable at 10^{13} neutrons per square centimeter per second and increases with higher flux.

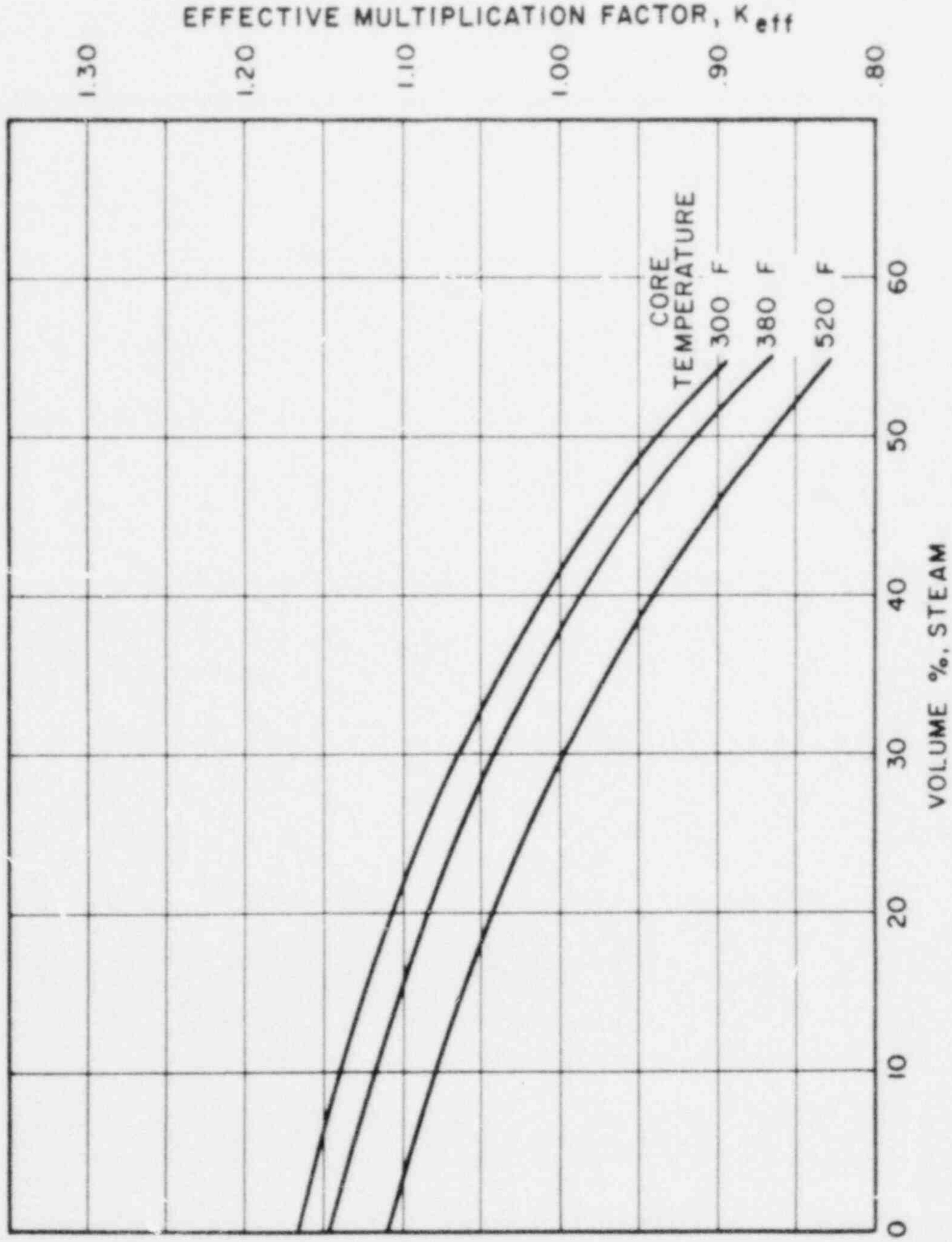
It is possible to consider that an oscillation will occur if enough reactivity is tied up by the xenon to equal the reactivity requirements of a second mode of the neutron flux. The problem is not an academic one, since it has been observed in large reactors. This problem will be investigated for the Yankee reactor more thoroughly and instrumentation will be provided so that it can be both detected and, with the available control rods, controlled.

The xenon instability provides a design condition mitigating against a reactor with only chemical control and no control rods, since the control rods are needed to modify the distortions of a neutron flux which may result from the xenon oscillations. The cause for concern with the xenon oscillations is the fact that the flux may be so perturbed that design hot channel factors are exceeded and thermal damage occurs to the core.

Handling of Fuel

Figure 12 shows the k_{eff} of groupings of various numbers of fuel assemblies, and the Δk contributed by the last one added to the periphery of the group. A cylindrical geometry is assumed so that these values of k represent maximum values; furthermore, immersion in cold water is assumed. Seven assemblies are required to achieve criticality. This limit applies only to assemblies being placed in a cluster; an infinite string stacked side by side would remain subcritical.

In the Yankee plant, new fuel assemblies are stored dry in individual compartments which would be subcritical even with total water immersion. Spent assemblies are pulled up out of the core and, one by one, sent down a chute into a spent fuel pit located below the level of the reactor. In this storage pit, they are stored under water on 15 in. centers so that no critical configuration can arise. This conclusion has been verified experimentally by Dr. D. Callahan at Oak Ridge National Laboratory.



REACTIVITY EFFECT OF VOIDS IN REACTOR CORE
(NO CHEMICAL POISON IN COOLANT)

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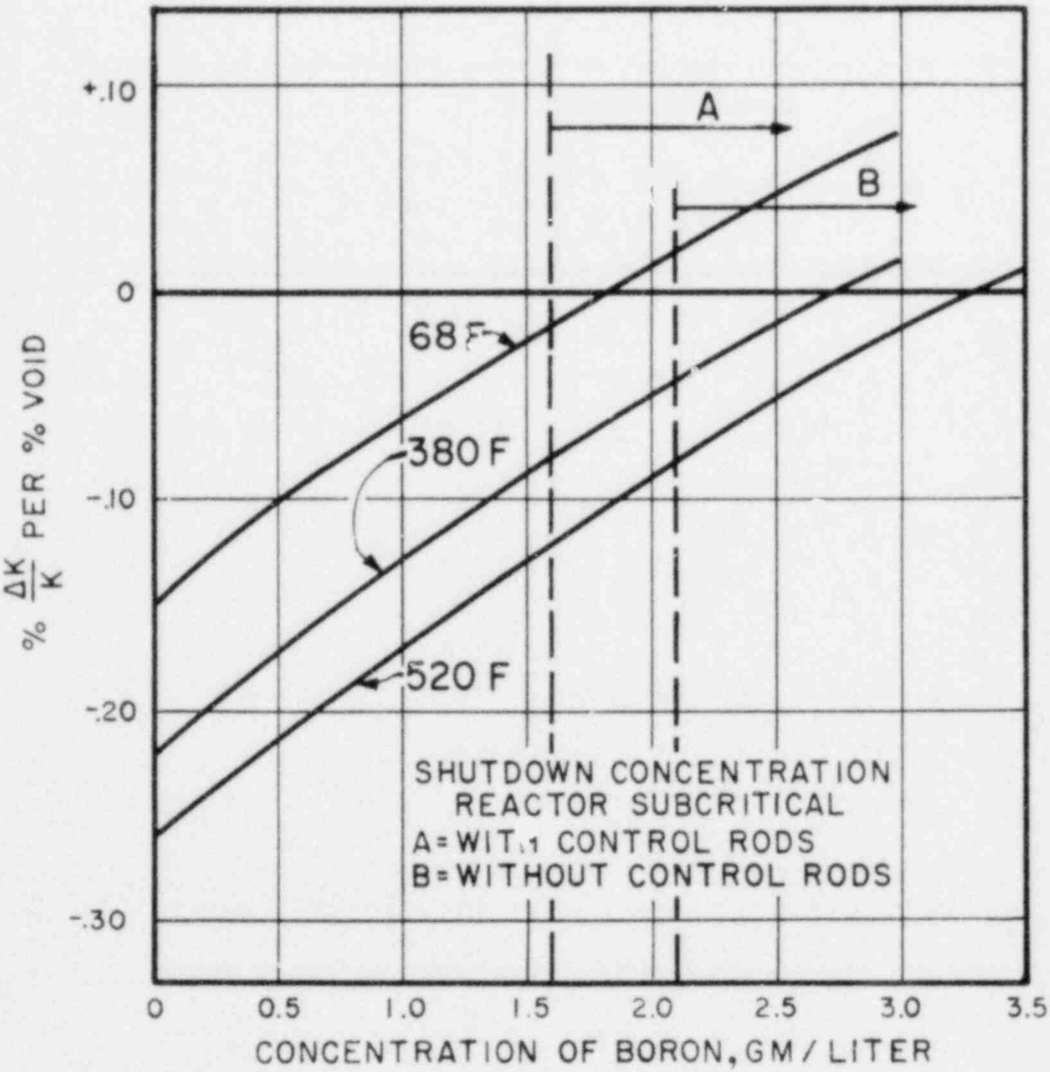

UNIFORM VOID COEFFICIENT
vs BORON CONCENTRATION

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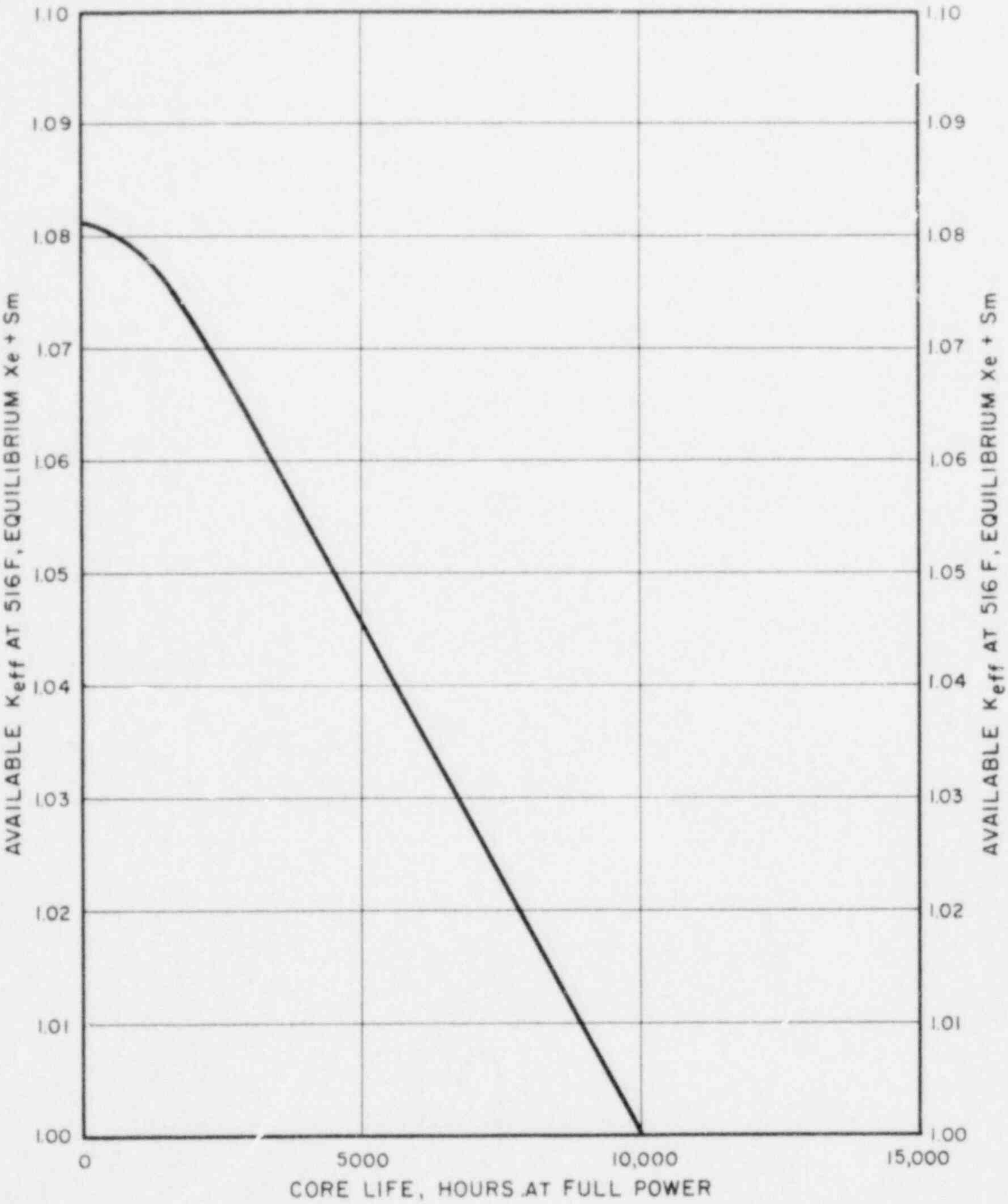
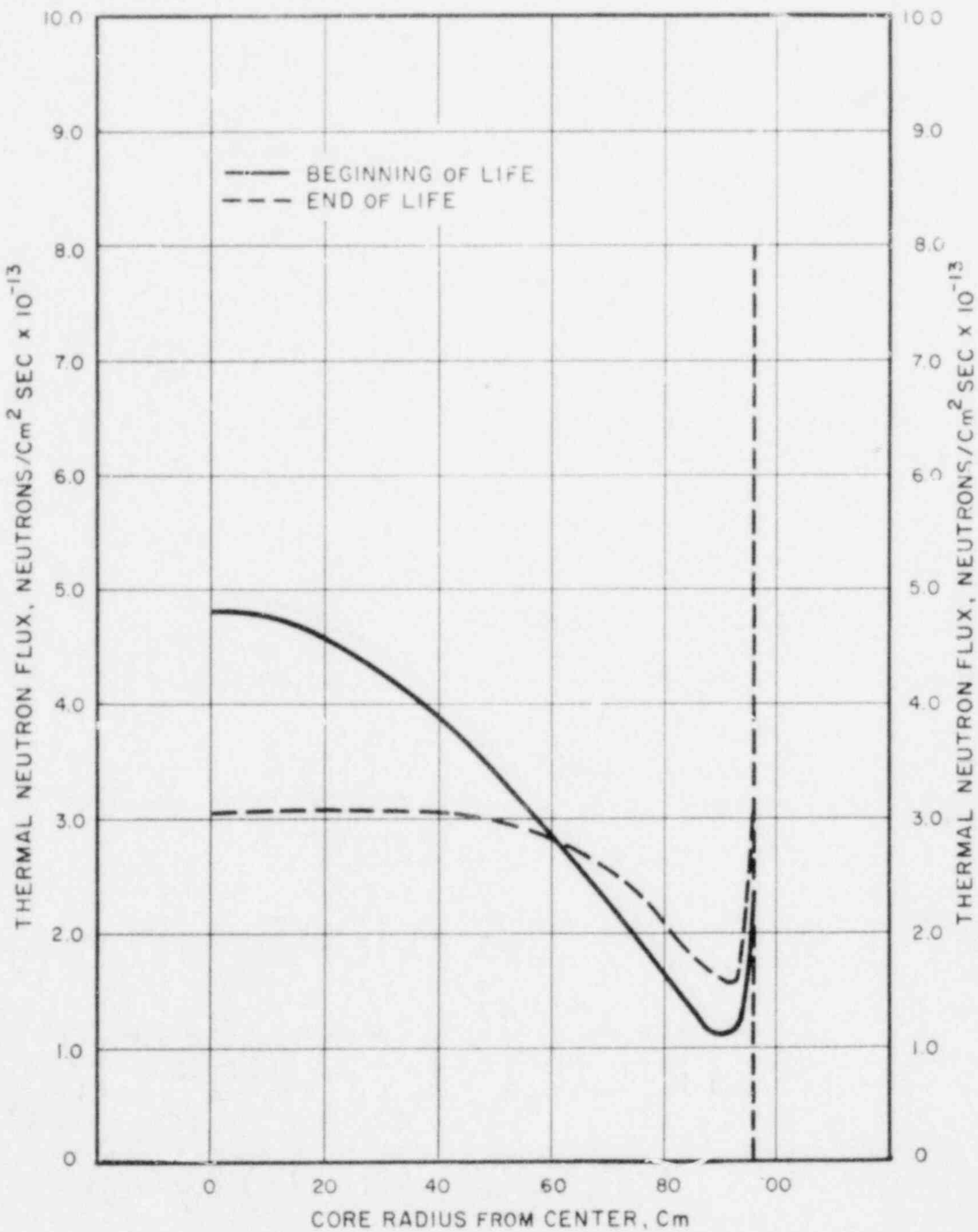
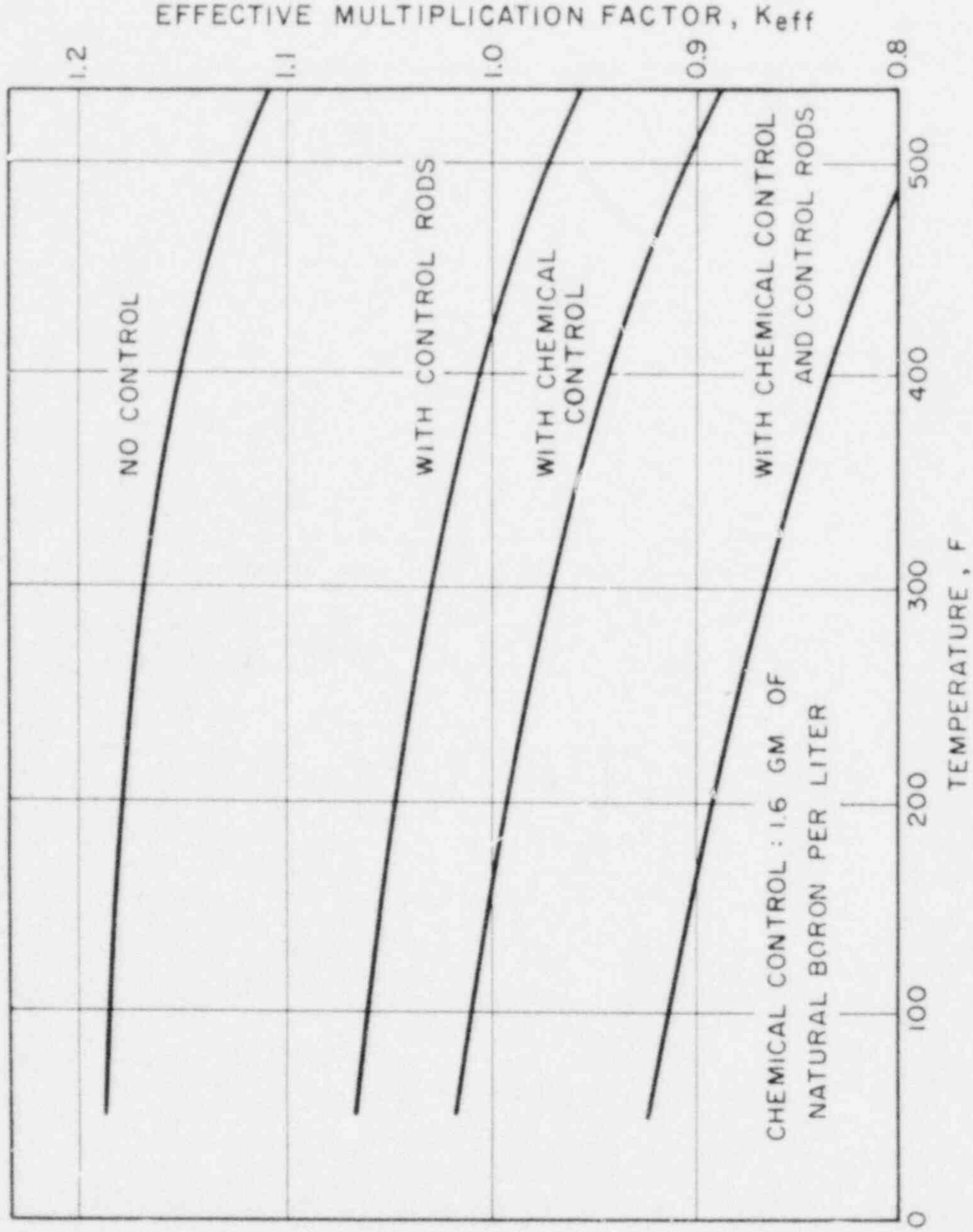
FIG. 9B



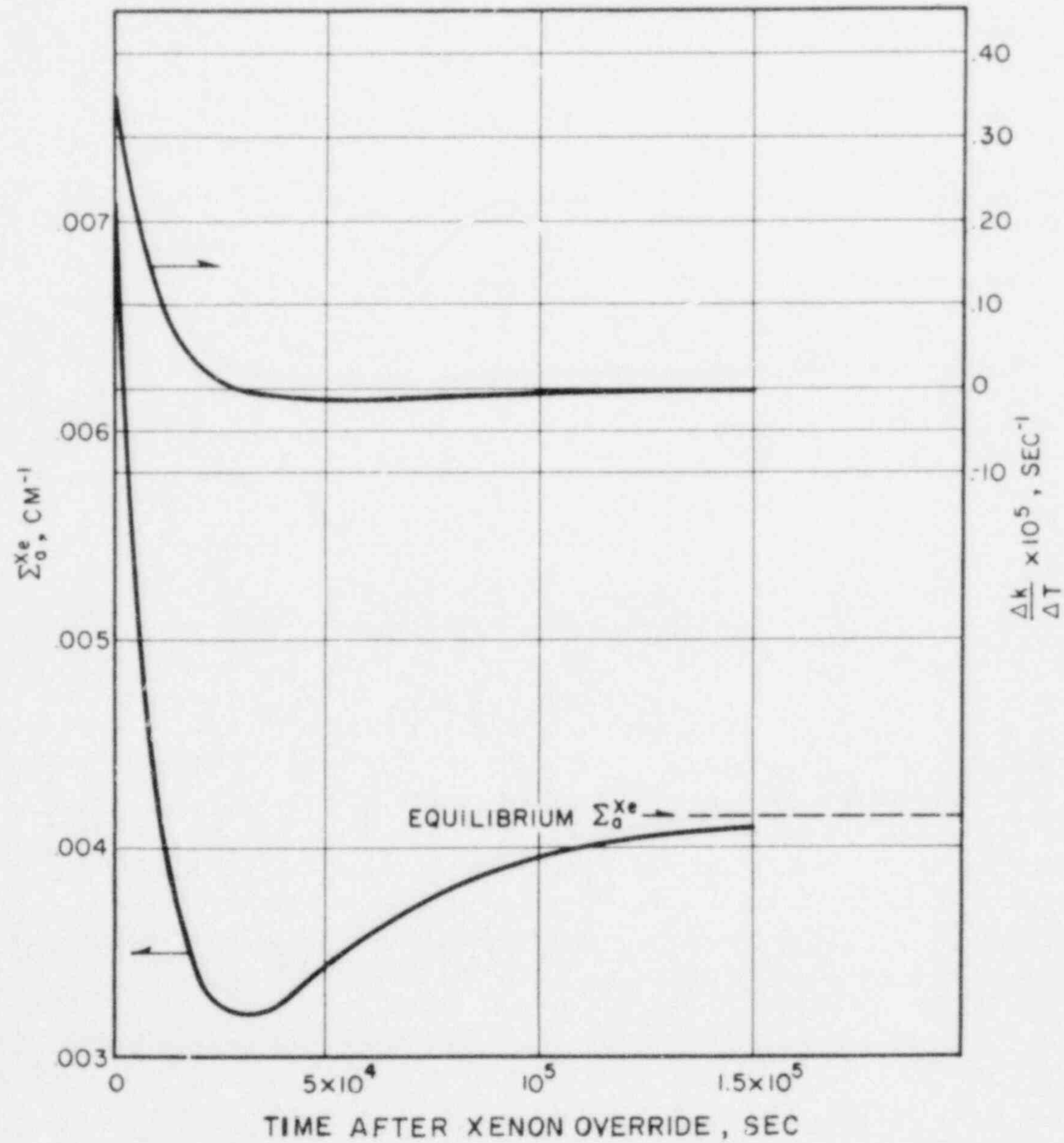
AVAILABLE EXCESS REACTIVITY vs CORE LIFE



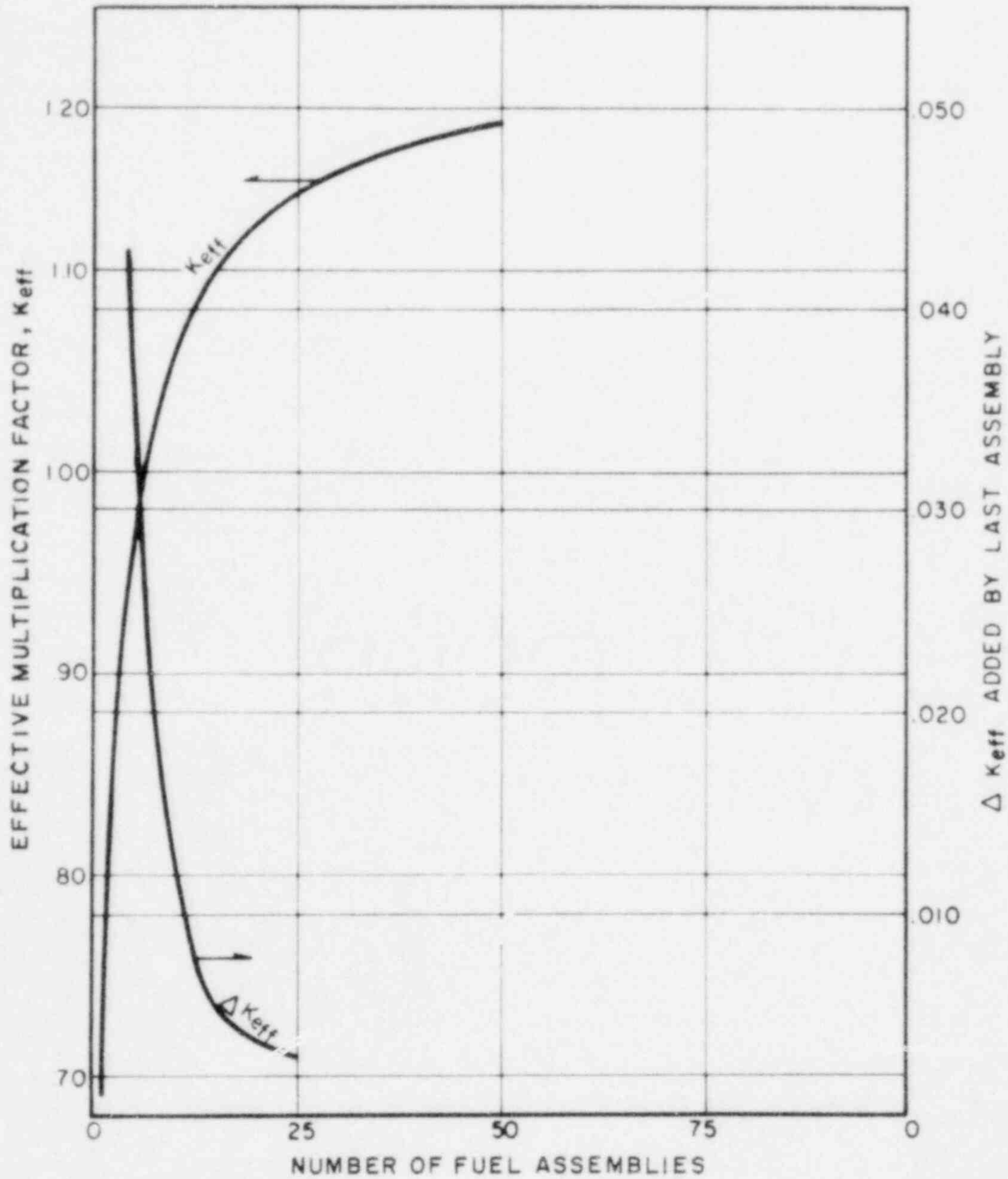
THERMAL NEUTRON FLUX vs CORE RADIUS
AT BEGINNING AND END OF CORE LIFE



EFFECTIVE MULTIPLICATION FACTOR vs TEMPERATURE



XENON POISON AND RATE OF CHANGE OF k FOR STARTUP AFTER MAXIMUM XENON OVERRIDE



REACTIVITY vs NUMBER OF FUEL ASSEMBLIES
(NO CONTROL RODS)

103 CRITICAL EXPERIMENTS

General

The cold critical experiments for the reactor will be performed in the middle of 1960. They will be performed in the reactor vessel at Rowe, Massachusetts and with the instrumentation which will be used in operating the reactor. The experiments will be performed by a group which will include Westinghouse personnel who have worked on the low power critical experiments at the Westinghouse Reactor Evaluation Center. A thorough check-out of all instrumentation and scram devices will precede any critical experiments at Rowe.

The general procedure proposed is to make use of all the experimental information obtained at the Westinghouse Reactor Evaluation Center and bring the reactor up to criticality very slowly. During the loading of fuel, neutron multiplication measurements will be made using a neutron source. The reactor vessel closure will be bolted in place. The reactor will be brought up to cold criticality. Experiments will be performed with the reactor in this state, and the worth of control rods and chemical neutron absorber will be determined. The next experiments will consist of pumping coolant containing boron through the reactor and observing results of the action on reactivity.

Temperature coefficient experiments will be conducted at various water temperatures obtainable by allowing the main coolant pumps to heat up the water. After the maximum temperature is obtained by this means, partial power operation will proceed by bringing the reactor up to 1 or 2 per cent of full power. Flow tests will be continued, and control rod worths will be constantly measured as the chemical neutron absorber dissolved in the water is reduced with rising temperature in the reactor.

Partial power operation will be used in an experimental sense, at first, until complete control stability of the reactor and complete reliability of the associated circuits are assured, and all system difficulties are resolved.

After this time, the reactor will be raised to approximately 10 per cent of full power and then to approximately 25 per cent of full power for some power output runs of short duration to test the overall plant. With the tests complete, power will be increased in small steps up to approximately 50 per cent of full power, and some long runs will be taken at 50 per cent of full power. After operation at 50 per cent of full power becomes routine, power will be increased to a level which conservative calculations indicate at that time is completely safe with no nucleate boiling.

The reactor power plant will be run at this power level for some time to get operating experience and to burn up some of the fuel. After a lengthy run at this power level, a thorough review of the entire plant will be made to determine whether it is desirable to open the reactor pressure vessel and observe the fuel and remove a few fuel assemblies for examination.

If the radioactivity of the main coolant loop and the crud is at a reasonable level, further power increases will be made until 100 per cent rating of the core is attained. If main coolant loop water radioactivity continues to be low after running at 100 per cent rating, the entire plant will again be thoroughly reviewed, and a decision will be made as to whether to increase the reactor core output to a level which is considered safe by realistic calculations available at that time. These calculations will be as realistic as possible in the sense that no melting at the center of the fuel will be permitted, nor bulk boiling at the outlet of the hot channel permitted. Any incremental increases in power level above 392 mw will be submitted to the AEC for review. If the reactor operates satisfactorily at the new power level, a run will be made which will take the fuel to the maximum burn-up. Experiments will be performed at that time to assure, or to help in obtaining, more power output from the second core.

A similar procedure will be followed for later cores, except that the time scale will be condensed until it is routine to put a core into the reactor and operate at full power almost immediately.

Fuel handling into and out of the reactor will also be studied for criticality hazard. Rearrangement of fuel in the partially burned up reactor will be the subject of experimentation.

Part Core Critical Experiments

The part core critical experiments consist of obtaining reactivity data from a 7,000 lb uranium dioxide reactor designed to simulate on a reduced scale the nuclear aspects of the Yankee reactor. The fuel rods will consist of approximately 6,000 stainless steel tubes, 4 ft long, containing 0.3 in. diam uranium dioxide pellets similar to those in the final reactor. It will be controlled by means of nine control rods. Reactivity experiments will be run at three different volume ratios of water-to-uranium.

The experimental information to be obtained during the Yankee Research and Development criticality program will be applicable to the first critical experiments to be performed at the Rowe site. The data will include steady state reactivity temperature coefficient, rod worths, flux plots, and kinetic behavior of the reactor as indicated by a transfer function which will be obtained if it is feasible and pertinent to the final reactor core.

It is planned that the experiments will be performed at the Westinghouse Reactor Evaluation Center, WREC, and will establish information about critical mass and the maximum number of fuel assemblies which can be loaded into the reactor before criticality must be considered. The worth of control rods will be established by measuring the worth of the prototype control rods in the experiments at WREC. The effect of the source of neutrons on the reactor will be studied in a calculation of the radiation level to be expected at the instruments. This will be helpful in positioning the nuclear instruments at the Rowe site for the first full size cold critical experiments.

The control rods in the critical facility will be used to evaluate the effect of a chemical neutron absorber in the reactor, the interaction of control rods one with another, and their relative worth as a function of radius in the reactor. Flux plots will be obtained in order to give some indication of nuclear hot channel factors which may exist in the final reactor. The effect of voids in the reactor as a function of radius also will be fully explored in the WREC critical experiments. These experiments will establish the effect of boiling in the final reactor to a greater degree of certainty than is possible analytically. The effect of water at partial core height will also be studied at WREC. Since the effect of water as a function of height may be large, experiments will not be performed when a small rise in height of the water could make the reactor prompt critical.

One of the significant functions which the WREC criticality experiments will serve is that of training a crew which is familiar with the reactor performance and criticality considerations so as to minimize the possibility of accident at the Rowe site. Yankee will assign personnel for training at the Westinghouse Reactor Evaluation Center to gain experience and to become familiar with the nuclear behavior of the reactor. Emphasis will be given to the kinetic response of the reactor as affected by temperature, control rods, and chemical neutron absorber dissolved in the water, as well as mechanical motion of various components of the reactor.

Initial Loading of the Reactor

The initial loading of the reactor at the Rowe site will be with borated cold water in the vessel and will be monitored with instruments which have been checked out with the neutron source. Close monitoring will be made of the flux level during loading of the reactor. The multiplication of the neutrons at all times will be observed to approximate effective worth of the fuel elements added to the reactor. The multiplication will be maintained far below unity and the partially loaded core will not become critical. A centrally located control rod will be available for scram purposes at all times during loading. It will be scrammed by means of a control rod drive mechanism installed temporarily for this purpose. The procedures to be used are as follows:

Borated water will be pumped into the reactor vessel and main coolant loop, and it will be checked for boron concentration.

All instruments will be checked out with sources.

The centrally located control rod will be raised for safety purposes on its temporary suspension and tested so as to be available for instrument or manual scram purposes. This control rod is worth more in reactivity than single addition of a fuel assembly.

The fuel assemblies will be added one by one, starting at the center of the reactor with the fuel assembly containing the neutron source first.

Control rods will be added as the adjacent fuel assemblies are loaded because of the hinge in the control rod.

The multiplication of neutrons will be measured after each fuel assembly is added. If substantial multiplication is observed, more boron will be added to the water until the multiplication decreases.

Fuel assemblies and control rods will be added until the reactor is completely loaded.

The suspended control rod will be lowered into the core.

The holddown plate and other internal equipment will be loaded into the reactor.

The closure will be set in place and bolted.

Reactor Start-up

The first reactor start-up will be performed in the bolted reactor vessel at room temperature with some pressurization to prevent pump cavitation under flow conditions. Flow will be provided by one pump to maintain uniform chemical concentration. All water in any loop which may be valved off or any connecting auxiliary system which may be opened to the main system will be tested to be sure the water it contains is borated to the highest degree expected in the reactor. The procedures are as follows:

Instruments will be checked with sources and so positioned during the original start-up that they can record the neutron level from the source with boron in the water.

Control rods will be checked to establish that each one will scram and then 12 control rods will be raised so that they can be scrammed for safety purposes if necessary.

After it has been established either that there has been no multiplication or that the control rods will control a given amount of reactivity and will scram, some of the chemical neutron absorber will be removed. During neutron absorber removal, 12 control rods will be in and 12 out, so as to provide safety and a means for increasing reactivity for test purposes.

After each incremental removal of chemical neutron absorber, the control rods will be calibrated to establish how much control there is available in the reactor with all rods in and to determine how close the reactor is to criticality, with all rods removed.

When substantial multiplication occurs, a measurement of control rod worth will be performed and chemical neutron absorbers will be removed from the water to account for a fraction of that controllability.

The reactor will eventually go critical by means of control rod motion. Since it is planned that the control rods will have less than the 20 per cent effectiveness needed to control the cold, clean reactor, there will still be some boron in the water when the reactor eventually goes critical.

The reactor will be operated at criticality with varying amounts of boron in the water and with varying control rod positions to calibrate the chemical neutron absorber against the control rods and gain experience in use of the chemical control system.

The effect of water temperature on the criticality of the reactor will be observed as the reactor is brought up to temperature by the pumps. Essentially, zero power operation will be used. Only one pump will be used to provide the initial flow and temperature rise.

Other pumps and loops will be started until all loops are pumping and reactor operation is smooth.

The effect of valving off various loops will be studied.

After all pumps have been checked out and the reactor operates satisfactorily at zero power, the temperature of the water will be raised with the pumps until the desired temperature is established and reactor stability and operation is again checked.

When it is established that the reactor can be controlled, scrammed and the chemical control system and control rods operate smoothly, the reactor will be brought to low power operation and to normal operating temperature.

A thorough step-by-step procedure will be written out and approved before any reactor start-up experiments are performed. All experiments will be performed under the authority of one person, whose written approval must be obtained before any specific experiment is initiated. The senior man present during the experiment must approve the steps of the procedure as they are implemented before the next step may be taken.

Rearrangement of Fuel Within the Reactor Core

After the fuel in the first loading is burned out the next fuel loading will be added. It is being considered that this will be approximately 50 per cent of the fuel in the core.

The center fuel from the reactor will be removed. The fuel around the periphery of the reactor will be moved to the center of the reactor, the new fuel being added to the outside of the reactor. During the rearrangement of fuel within the reactor core, neutron measurements will be taken at all times. However, no attempt at achieving criticality will be attempted during the rearrangement and, in fact, the boron concentration will be maintained in the water at such a high level that very small multiplication of neutrons will be observed.

All core rearrangements will be made with critical instrumentation operating and at least one central control rod available for scram. The control element used will have a worth greater than any single possible core shift or addition. Thus, if the reactor should go critical, the control rod will automatically scram the reactor. In addition, all core reloadings will be conducted under critical experiment conditions.

Theoretical work will be continued in order to improve the design for future cores. Some core developments which provide definite advantages from a theoretical basis will be checked experimentally by putting them in the reactor core. In this way, the power level will be raised so that on subsequent cores, the full 492 mw of heat will be released.

104 CONTROL

General

A prime objective of the Yankee project is to achieve the lowest possible nuclear fuel cost consistent with safety and reliability. In order to accomplish this, every element of the nuclear fuel cycle must be examined for cost reduction possibilities. Fuel fabrication, processing and inventory charges are all important items and their final contribution to the fuel cost per kilowatt-hour is dependent on the length of time a core can remain in the reactor.

The core is presently being designed for 10,000 hours at full power. Core lives of this duration have not yet been achieved in any operating reactor using slightly enriched uranium fuel. Experimental results, however, indicate that from the standpoint of irradiation damage to the fuel and structural materials and from the standpoint of corrosion and thermal cycling, this result can quite possibly be attained.

A core life of the order of 10,000 hours raises difficult problems of control, particularly in a pressurized water reactor with its large negative temperature coefficient. Approximately 19% excess reactivity must be provided in the clean cold core in order to remain critical at operating temperatures and with equilibrium poisons present to the end of the 10,000 hours' life. This excess reactivity requirement may be broken down as follows:

Cold to hot	7%
Fuel burnup	7%
Equilibrium xenon and samarium	5%
Total	19%

In order to assure that the reactor can be rendered subcritical at room temperature with a new, clean core, a margin of 5% above this figure must be provided, and the total control range needed is, therefore, 24%.

In any reactor of this type, it becomes a problem to provide this amount of control entirely by means of mechanical control rods. When stainless steel is used as a cladding and structural material, the fuel enrichment must be increased slightly and the problem becomes somewhat more acute since the worth of a given control rod material and configuration is significantly less than when low cross-section materials, such as zirconium or aluminum, are used for cladding. In this reactor, a centrally located mechanical control rod made of a material that is black to thermal neutrons only has a worth of approximately 2%. Additional control rods located at points away from the center have diminishing worths until those located near the periphery of the core have a value of only 0.1%. Accordingly, a very large number of control rods, possibly 75, would be necessary in order to cover the desired range of 24%. If such a number of control rods were to be used--each with individual drive mechanism--it would mean 75 precision mechanisms, 75 penetrations through the vessel head, and would further mean sub-dividing the core itself which now consists of 76 assemblies into perhaps four times that number of smaller units. The large number of penetrations through the vessel head would seriously complicate fabrication and raise formidable questions of structural integrity. Some of these objections could be avoided by ganging a number of control rods to a single mechanism, but this suggestion has always met with limited enthusiasm because of the mechanical difficulties that arise and also because of the inability with ganged control rods to regulate various regions of the core through individual rod programming.

An additional disadvantage of a large number of control rods is the fact that to accommodate them, about 6% of the fuel rods would have to be omitted, thereby decreasing heat transfer area by the same amount.

Further heat transfer loss is encountered because of bypassing more coolant around the heat producing surfaces through the many control rod channels. With the same general core configuration, the dimensions of the core would have to be increased to remain at the same average and maximum heat flux levels.

Because of these difficulties a control scheme, using a combination of mechanical control rods and a chemical neutron absorber dissolved in the coolant-moderator, is proposed for this reactor. Reliance is placed on the natural stability inherent in a pressurized light water reactor to handle short-term transients. Twenty-four mechanical control rods are used to control reactivity at operating temperatures. Space is provided for eight additional shim rods near the periphery of the core which are to be used if necessary to adjust initial reactivity of the core. The control rods themselves can be programmed to attain favorable flux patterns during operation and, in addition, can be used under manual control to counteract any tendencies toward xenon tilt or instability. A homogeneous chemical neutron absorber is added to the coolant-moderator for cold shutdown and to hold the reactor subcritical in a clean condition at room temperature.

While initially it is not intended to operate the reactor at power using the homogeneous chemical neutron absorber as a shim control, the ultimate possibility of such operation is believed to offer many advantages. Chief of these is the fact that if the excess reactivity can be counteracted by the dissolved chemical neutron absorber, it would permit operation at full power with all but one or two mechanical control rods fully withdrawn and, therefore, available as safety rods. Operating in this manner would increase the thermal and nuclear performance of the core while measurably reducing the duty on the expendable mechanical control rods and the wear and tear on their associated drives. The natural stability of a pressurized water reactor lends itself to the slow reactivity changes provided by injection and dilution of the liquid neutron absorber. In addition, the use of a homogeneous shim offers the possibility of employing the entire volume of the core for heat production, thus realizing maximum heat transfer capability and minimizing the possibility of local hot spots and fuel burnout.

Borax III and EBWR have been operated successfully at power for limited periods of time using a dissolved boron compound as a homogeneous shim. This experimental evidence is encouraging, but it is recognized that there are still many problems associated with operating a reactor in this manner and that these problems are not at this time well understood. The Research and Development Program now underway includes an extensive investigation of the behavior of boron compounds in solution with both in-pile and out-of-pile dynamic loop experiments planned. The results of this program, together with operating experience in the actual Yankee plant, may point the way to methods for safely using chemical neutron absorbers in the primary coolant during full power reactor operation.

Control Rods

Mechanical control is provided by 24 cruciform control rods located in four concentric rings around the center of the reactor. Provision is also made for eight additional control or shim elements in the outer region of the core. By placing at these locations fixed elements of a neutron absorbing material, inert material, or fuel, the initial reactivity of the core may be adjusted to the desired level.

A design objective for the first core is to provide sufficient control rod worth to render the reactor 3% subcritical with a clean core at operating temperature. To bring the reactor from this point to 5% subcritical at room temperature, a chemical neutron absorber will be added to the main coolant water.

The total control available from the present design using 24 silver-cadmium-indium control rods according to conservative calculations based on absorption of thermal neutrons only lies between 10% and 12%. Experiments with such rods in critical assemblies, however, indicate control rod worths higher by 30% than rods black to thermal neutrons only. This effect is thought to be due to additional absorptions at energies above the thermal range. Since the k_{eff} of the hot clean reactor is 1.113 and total control rod effectiveness is calculated at 10% to 12% on the basis of thermal neutron absorptions only, the control rods are not adequate to meet design objectives holding the hot clean core 3% subcritical. The Research and Development Program will reduce the uncertainty in these values. If experimental evidence shows that the control rods are inadequate, five possible procedures will be investigated for obtaining more control, as follows:

In accordance with technical discussions between Yankee and Westinghouse, it has been agreed that a two-region core is a reasonable alternative design for the reactor. Since a two-region core has advantages associated with heat transfer, burnup and control, a considerable effort will be expended on this design so that it may be used for the first core. If two different enrichments are used for the first loading in the reactor vessel, the k_{eff} of the hot clean core will be approximately 1.08. This reduction in k_{eff} would allow control rods of presently calculated worth to hold the core 3% subcritical in the hot clean condition.

The possibility of leaving the chemical neutron absorbing compound in the main coolant during power operation is desirable from a nuclear design standpoint and would provide any additional control required. The undesirable aspects of using chemical control during power operation are those associated with the chemistry of the main coolant. If the use of chemical control during power operation is adopted, it would probably require redesign of some of the plant systems, such as the waste disposal and purification systems.

Additional control amounting to approximately 2½% can be gained by adding highly enriched uranium fuel to the control rod followers with an equivalent reduction in the enrichment of the fuel in the core. This added control would probably meet design objectives.

The eight outer control slots in the present core design, which it is contemplated might be used for fixed elements, could be provided with rods connected to mechanisms, and 0.8% additional control could be achieved in this manner. The incremental cost associated with such a small increase in control makes this change unattractive.

If all other methods prove to be impracticable, additional control rods could be added to the reactor by redesigning the core and the reactor vessel head. This method does not appear to be desirable at the present time because of mechanical complications, structural difficulties and increased costs.

The control rods are scrammed into the core under the following conditions:

Excess neutron level

Short period during reactor start-up

Low main coolant loop flow

High or low main coolant loop pressure

Manual scram

When the reactor is at power, automatic run-in of the control rods is initiated by high temperature in any one of the four main coolant loops.

Alarms are provided for:

High reactor outlet temperature

Loss of turbine generator load

Reactor period less than 20 seconds

An interlock is provided that does not allow a loop to discharge into the system when the water temperature in that loop differs by more than 50°F from the water temperature in the active loops at the reactor vessel inlet. This is accomplished by a permissive circuit coupled to the motorized valve. An alarm for this condition is also provided.

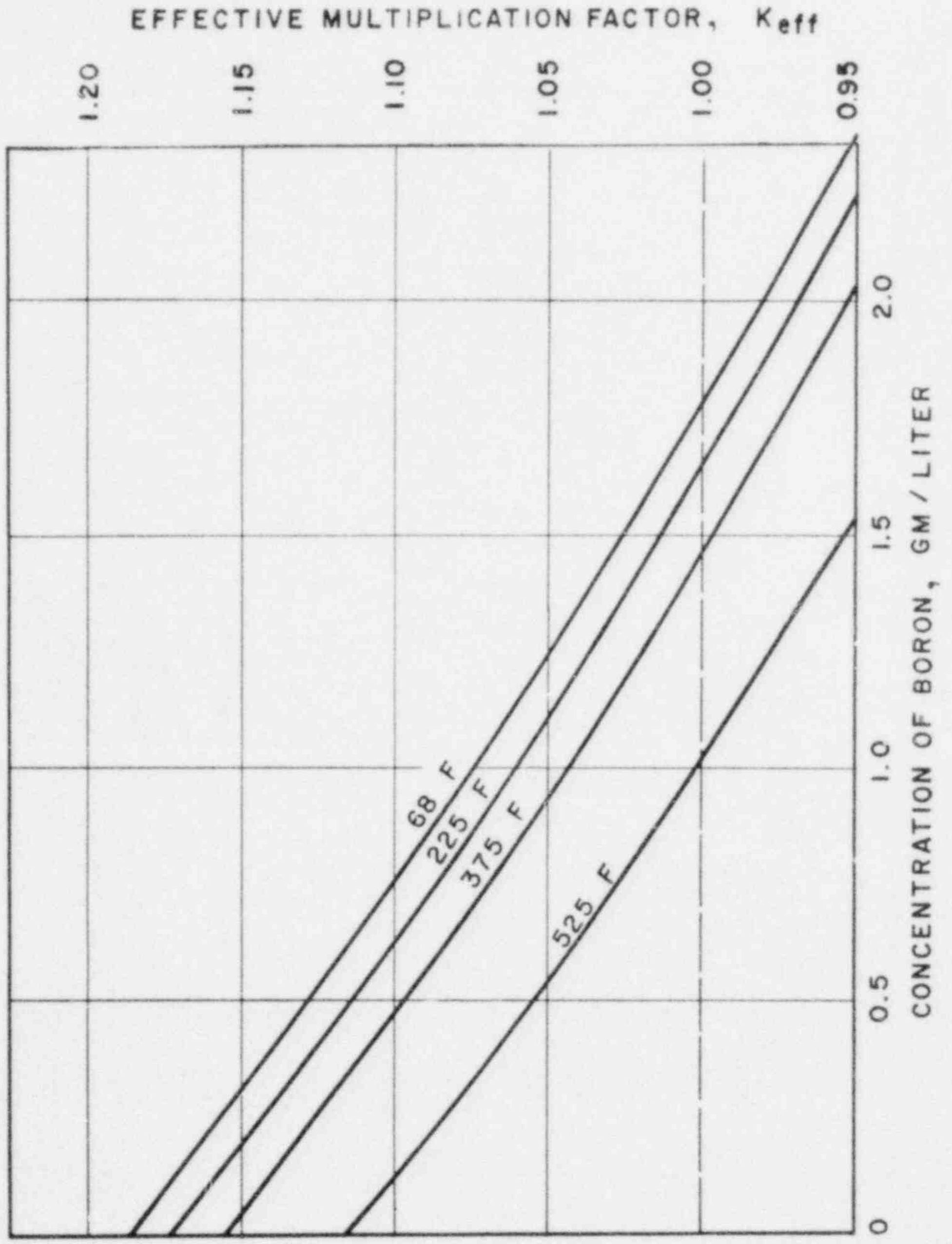
Drop time of control rods at 0.8 the acceleration of gravity is 0.6 second which for a total rod worth of 10% provides a reactivity decrease of 16% per second.

Chemical Control

The chemical control system is designed to shut down the cold clean reactor by a margin of approximately 5% in Δk. with all the control rods inserted. This margin allows one or more centrally located control rods to be fully withdrawn for safety purposes and still have the reactor 2% to 3% subcritical. In the present design, this would require about 1.6 g of natural boron per liter of main coolant.

The chemical compound which will be used in the chemical control system has not been finally selected, although boric acid and ammonium pentaborate are possibilities as indicated by the results of a development program at Bettis. Boron compounds have good thermal stability and have adequate solubility in the cold reactor. The solubility of boric acid at room temperature is 50 g per liter of water, which means that approximately 8 g of boron per liter can be retained in the coolant. The solubility is thus more than five times greater than required. The effective multiplication factor for the present core design as a function of boron concentration in the main coolant is shown in Figure 13. A concentration of 1.6 g of natural boron per liter of water is sufficient to reduce k_{eff} to unity at any temperature above 225°F. even though all control rods are withdrawn.

The present design makes use of a bleed-feed system to change the concentration of the chemical neutron absorber. The present maximum rate of water injection for this system is about 100 gpm. Since the mechanical control rods can handle xenon and samarium transients, there is no need for faster action by the chemical control system. At a bleed-feed rate of 100 gpm in a 3,000 cu ft system, the maximum [illegible] change in reactivity is 0.0005% $\Delta k/k$ per second, which is well [illegible] limits.



EFFECTIVE MULTIPLICATION FACTOR
VS
NEUTRON ABSORBER CONCENTRATION

Questions that have been raised in connection with reactor control through use of a homogeneous chemical neutron absorber in the main coolant water include the question of thermal stability of the chemical solution at operating temperature and pressure, and possible interaction between the chemical control agent and other additives present in the water. Considerable work has been done in this field at Bettis for the PWR project. The following conclusions have been stated.

1. The boric acid is stable in solutions at high temperature and pressure.
2. Ammonium borate solutions are likewise satisfactorily stable under these conditions.
3. The use of lithium hydroxide in combination with boric acid is probably satisfactory with very low quantities of lithium hydroxide. However, if the concentration of lithium hydroxide is comparable with the boric acid concentration used, the combination may be unsuitable.

The conclusions are based on experiments in autoclaves and loops. A possible adverse effect which could occur in a reactor is the precipitation of anhydrous lithium metaborate ($LiBO_2$). Experiments indicated a precipitation out of lithium metaborate at the interface between the water and vapor phases. This is known as the drying-up phenomena.

Another question which has been raised is the possibility of inverse solubility with temperature of boron compounds which might possibly be used in the chemical control system. Lithium borate is the only compound which has been found to have this property. The solubility of boric acid increases rapidly with temperature. Solutions of lithium hydroxide and boric acid are sufficiently soluble at temperatures up to about 500°F. At higher temperatures the so-called drying-up phenomena can occur as described above. The solubility of lithium borate decreases from .3 mole per liter at 500°F. to approximately .2 mole per liter at 600°F.

Solubilities as a function of temperature do not affect the present plan to operate the reactor from hot to cold with the chemical control system. Since concentration corresponding to the solubility of lithium borate at 700°F. is still adequate to control the cold clean reactor, it is more than sufficient to maintain subcriticality at temperatures from 500°F. to 700°F.

The Research and Development Program for the plant, supported by the AEC under Contract No. AT(30-3)-222, includes four major projects which pertain to the problems of chemical control. Project 2.0 is concerned with calculations of the nuclear physics problems and effects of chemical control on reactivity coefficients. Project 3.0 is concerned with autoclave and dynamic loop out-of-pile studies of two reference water combinations with a chemical neutron absorber. Corrosion effects on materials as well as deposition and absorber injection and dilution problems are being studied. Project 3.0 also includes Van-de-Graaff irradiations of chemical neutron absorber solutions. Project 10.0 is the performance of a critical experiment which will experimentally check the nuclear calculations on chemical absorbers made under Project 2.0. Project 11.0 consists of in-pile pressurized water loop tests in the MTR, some of which will use the reference chemical absorber selected from the out-of-pile experiments and other information available. At the conclusion of the Project 11.0 experiment, the characteristics of the chemical absorber (nuclear, corrosion, precipitation, etc.) should be well established.

105 REACTOR CORE EVOLUTION

It must be recognized that there are many uncertainties in predicting the performance of power reactors and that there is as good a chance in actual operation of exceeding design performance as there is of falling short. In view of this, it is planned to begin with an initial core that is designed on a conservative basis for outputs in the 100 to 120 mw range. Possibly, such a core could deliver the rated plant output of 134 mw for which the steam-electric equipment and other plant components will be sized. If, however, experience shows that plant operation is limited by core performance, it will be possible to make modifications in the design of subsequent cores which will allow the plant to produce the rated 134 mw.

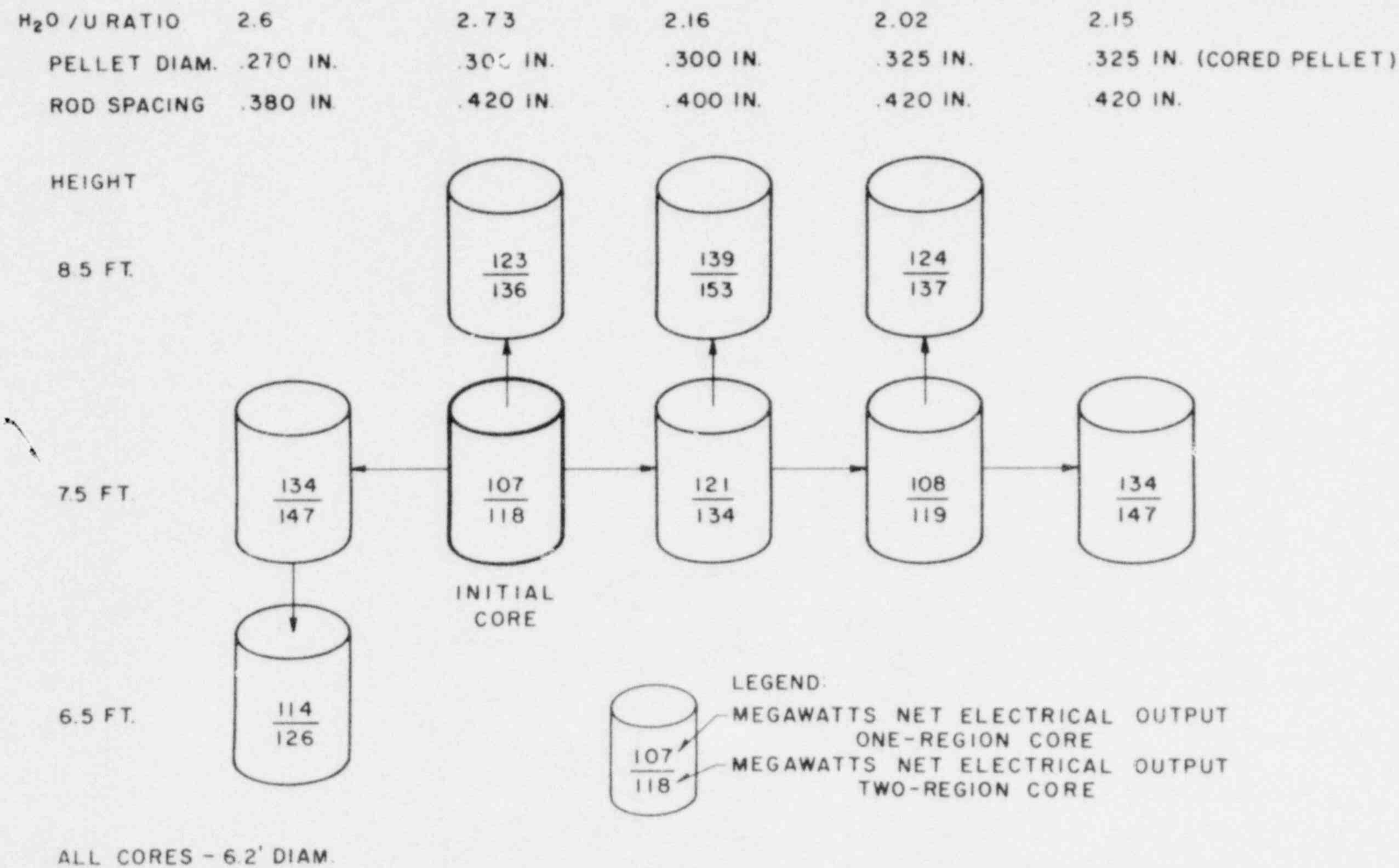
After studying a number of variations of core design covering a range of fuel rod diameters, lattice spacings and core heights, Westinghouse and Yankee have agreed to settle on a nominal design for the initial core with a pellet diameter of 0.3 in., a square pitch lattice spacing of 0.42 in., a core height of 7.5 ft and a core diameter of 6.2 ft. Present calculations indicate that such a core will deliver 107 mw operating with single region loading and 118 mw with two region loading. These ratings are at the beginning of core life and may be expected to increase because of flux flattening as the core is burned out. In the case of single region loading, this may amount to as much as 10 per cent. There is also the possibility that higher ratings may be attained even initially because of more favorable heat transfer conditions than are now assumed.

The pressure vessel is designed to accommodate a somewhat larger core, 8.5 ft in height, in the event that such a move later turns out to be necessary to reach the 134 mw rating. Figure 14 shows the modifications that could be made in cores subsequent to the initial core to reach the desired output. It is the intent to settle on a fuel assembly module such that any of the pellet diameters and lattice spacings may be used later without extensive modifications of the reactor plant.

Considerable operation of the Yankee plant is anticipated before the desirability of increasing the initial core output to 134 mw net electrical can be evaluated. During this period, studies will be made of latest available test data on materials, and the Yankee plant performance record. An amendment to the license application will be submitted covering the changes and evaluations necessary for the increased power level.

FIG. 14

REACTOR CORE EVOLUTION



2 PLANT DESIGN

200 GENERAL

The description of plant design is based on studies prepared by Westinghouse Electric Corporation and Stone & Webster Engineering Corporation, and represents a 492 mw heat, 134 mw net electric, design for all fluid systems and plant equipment.

The primary reactor system includes reactor vessel, steam generators, pumps, valves and piping, which make up the high pressure heat transfer loop. A composite diagram is shown on drawing 646-J-420. The material in contact with the main coolant is principally Type 304 stainless steel with a surface finish of 125 microinch rms. The apparatus and piping employed in the main coolant system is cleaned and prepared according to best commercial practice. All main coolant system apparatus which falls within the jurisdiction of the ASME Boiler and Pressure Vessel Code is fabricated to meet the code requirements. In general, the ASME Boiler and Pressure Vessel Code Interpretations Case No. 1224, Special Ruling, applies. The main coolant system and the auxiliary primary systems are constructed using, principally, welded joints, but can not be classified as hermetically sealed, since gaskets and high pressure fittings are used where safe and applicable. The coolant contained in the system is pure light water with temperatures in the range of 450 to 650 F at approximately 2,000 psia, except that during reactor plant cold shutdown, a nuclear chemical neutron absorber is dissolved in the coolant.

The reactor vessel is a cylindrical container of carbon steel, the inner surfaces clad with stainless steel, with a hemispherical bottom and closed at the top with a removable head, roughly hemispherical in shape. The removable head incorporates a bolted ring, gasket and seal weld; and also mounts the control rod drive mechanisms for top penetration of the core. Main coolant water enters the top of the vessel from the four coolant pumps, flows downward through the thermal shield annuli, and makes a single pass upward through the core channels. Support and hold-down mechanisms are provided for the core to take care of pressure differentials and weight loads. There are no penetrations of the reactor vessel below the top of the core.

Main coolant pressure control is accomplished as follows:

An electrically heated pressurizer, is designed to maintain the normal operating pressure of 2,000 psia, and pressure transients are handled by the pressurizer surge chamber, the electrical heaters and a variable flow spray line in the pressurizer steam dome. Positive pressure surges which exceed the capability of

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☒ FILMED ON APERTURE CARD NO 8101070140-01 the pressurizer are handled by a pressure control and relief system which actuates relief valves when loop pressures exceed 2,400 psia.

High water purity for the light water coolant-moderator is maintained by the materials of construction and the purification system. The all stainless steel construction of the primary system insures that the corrosion rate is low, and a hydrogen gas corrosion inhibitor is utilized to reduce structural and clad corrosion rates further. The use of oxide fuel and operation with a limited number of clad failures results in small releases of fission products and corrosion products from the uranium dioxide core material. The purification system is designed to maintain loop water purity and remove activated particles. Purification is accomplished by demineralizers in a low pressure system using reinjection of the purified water by the normal charging pumps.

The reactor plant is provided with four identical heat transfer loops, including the piping, steam generators, pumps and valves necessary for functioning. Pipe size is 20 in. OD, Schedule 160, and the coolant pumps are of the hermetically sealed, canned rotor type. The main coolant system stop valves are of the packed gland type with leakoff lines to the vent and drain system. Auxiliary piping components and systems are adequate to cover all the functions of plant start-up, operation, shutdown, and maintenance. Drain, fill, decontamination, safety injection, blowdown, and sampling systems are provided. A shutdown cooling system for removal of reactor decay heat is provided and includes components for forced circulation, in addition to the natural circulation of the main loops.

Protective equipment systems, and facilities are provided at the plant to safeguard physical property, plant personnel, and the public at large. Equipment insuring the safety of the public includes the vapor container, radiation shielding, radiation monitoring, controlled access fencing, and the waste disposal system providing for on-site retention of radioactive waste.

The vapor container is designed to contain the pressure build-up resulting from a major loss of water accident rupture of one 20 in. line which releases to equilibrium pressure and temperature within the vapor container, the entire main coolant system, and one steam generator secondary side. Radiation shielding is adequate to protect plant personnel and the public. During normal operation, the shielding design is such as to limit plant personnel dosages to less than 1/10 of the AEC standard of 300 mr per week. The radiation monitoring system provides visible and audible alarms to indicate levels which may result in danger to personnel. Plant monitoring includes air-borne particle detectors, steam generator leak detectors, neutron counters, gamma detectors, and comprehensive site area monitoring stations. Portable instruments are also provided for use by industrial hygiene personnel.

Further protective equipment associated with the nuclear plant includes the overall instrumentation and control system, the fuel handling system, the safety injection system, and the waste disposal system. The instrumentation and control system safeguards the entire plant and personnel in addition to the inherent stability features of this reactor type. Fuel handling is accomplished under proper conditions of safe shielding and storage of clad fuel. The safety injection system is provided as additional protection in order to limit damage in the event of a major loss of water accident. The waste disposal system is designed to accomplish holdup, purification, and preparation of solids, liquids and gases, for ultimate safe disposal.

Steam generated from the heat exchangers in the main coolant loops operates a conventional turbine generator. Condensate returned from this is pumped through closed feed water heaters back to the steam generators. Electric power is stepped up to the voltage of the local transmission system. Circulating water for the condenser is provided from nearby Sherman Pond, which is also the source of cooling water and the demineralized plant make-up.

201 REACTOR PRESSURE VESSEL

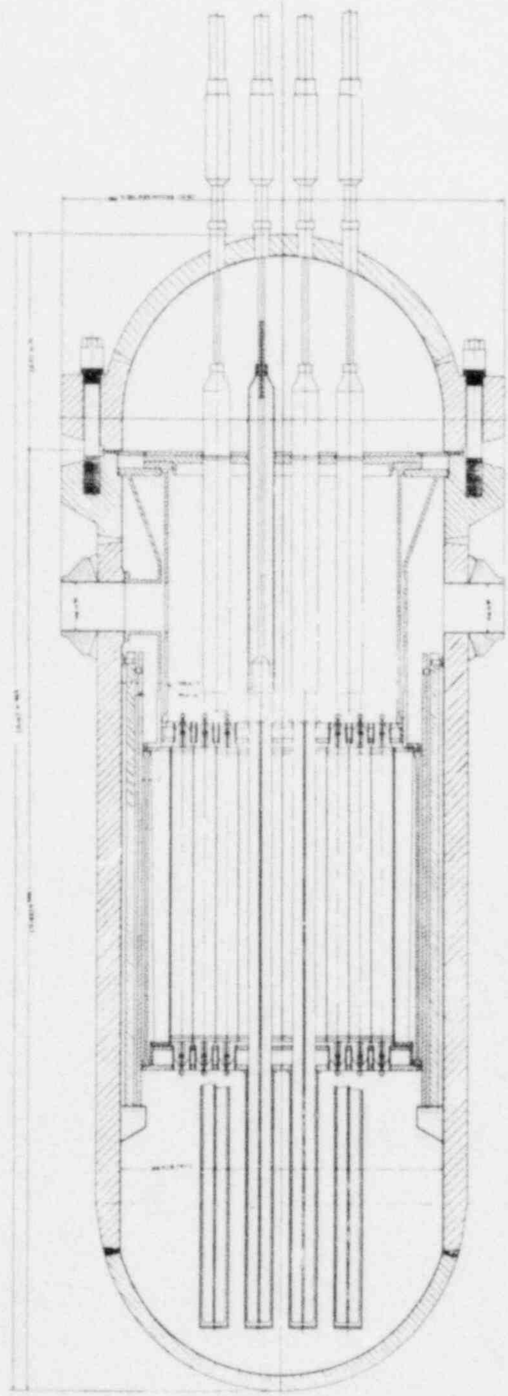
The reactor vessel is cylindrical in shape, with a hemispherical bottom head and a removable closure head. It is approximately 31.5 ft overall height by 109 in. internal diameter, as shown in Figure 15. The cylindrical portion of the vessel is made of carbon steel plate approximately 8 in. thick; the bottom head is 4 in. thick; and the reactor vessel head is approximately 6 3/4 in. thick. All internal surfaces of the vessel in contact with coolant water are clad with Type 304 stainless steel.

The vessel is designed in accordance with ASME Boiler and Pressure Vessel Code, Section VIII, (Unfired Pressure Vessels). The design pressure is 2,500 psia and the design temperature is 650 F.

Main coolant water enters the vessel through four inlet nozzles near the top, flows down through the thermal shield annuli, up through the core, and leaves the vessel through four outlet nozzles located at the same level as the inlet nozzles.

The concentric, cylindrical, stainless steel thermal shields rest on local supports near the bottom of the vessel. Their purpose is to limit thermal stress in the reactor vessel shell during full power operation by absorbing radiation emanating from the core.

All of the reactor vessel internal supporting structure is Type 304 stainless steel. The two thin stainless steel barrels that support and hold down the core are supported on a ledge near



SECTIONAL VIEW-REACTOR VESSEL AND CORE

the vessel top flange. All of the internals are held in place by the reactor vessel head which presses against the core hold-down ring-top plate combination.

The reactor vessel head is approximately hemispherical in shape with a heavy flange for bolting to the reactor vessel flange. Both the closure studs and nuts are applied and removed with an impact wrench. Special, dial-indicating, elongation gages are used to limit the tension in each stud. Leak tightness is secured from gaskets with provision for a backup seal weld. Operating experience will show whether seal welding of the reactor vessel head is required.

The reactor control rod drive mechanisms are welded to the reactor vessel head and are handled as an integral part of the head.

The fast neutron flux at the inside wall (attentuation of approximately 10 through wall) of the pressure vessel integrated over 30 years of reactor operation is calculated to be 10^{20} neutrons per square centimeter. Experimental data exist at Oak Ridge which state that changes in the properties of steel which has been exposed to 2×10^{18} neutrons per square centimeter are measurable. These effects have to do with increase in hardness and decrease in ductility of the material. However, it has also been found that these effects can be annealed out in approximately 30 minutes at 600 F. Since the steel enclosing the main coolant loop for the Yankee reactor will be approximately 500 F, a diffusion calculation has been made using the experimental point at 600 F as a check. This calculation indicates that the irradiation effects will be annealed out as they occur and that there will be no serious effect to any portion of the primary coolant-moderator container.

202 MAIN COOLANT SYSTEM

Function

The function of the main coolant system is to transfer the heat generated in the reactor core to the steam generators to produce steam for the steam-electric plant.

General Description

The main coolant system consists of four closed piping loops connected in parallel to the single reactor vessel. The main coolant is circulated through these closed loops from the reactor vessel to the steam generators and back to the reactor vessel. The principal equipment in each of the loops are two gate type stop valves, a steam generating heat exchanger, canned-motor type circulating pump, check valve, relief valve, and thermal insulation. Each main coolant loop also includes a warm up crossover line with stop valve which connects the hot piping leg to the cold leg.

Pressure and temperature instrument piping connections are also provided. The main coolant system is shown on drawing 646-J-421.

Basis for Design

The main coolant system is designed to transfer 1.7 billion Btu per hr from the reactor core. It converts this heat by evaporating water in the steam generators and supplying 1.9 million lb per hr of dry and saturated steam for the steam-electric plant. Main coolant enters the reactor vessel at 486 F and leaves at 524 F. It is circulated through the reactor vessel at a rate of 37 million lb per hr. The total volume of coolant in the main coolant system is approximately 3,000 cu ft. The steam leaving the steam generator is at a pressure of approximately 500 psia saturated. At rated load, the moisture in the steam is 1/4 of 1 per cent. The feed water returning to the steam generator is at a temperature of approximately 340 F.

The main coolant system is designed to permit a normal increasing and decreasing rate of load change in the steam-electric plant of 10 per cent per minute. In emergencies, however, the steam-electric plant may be unloaded nearly instantaneously.

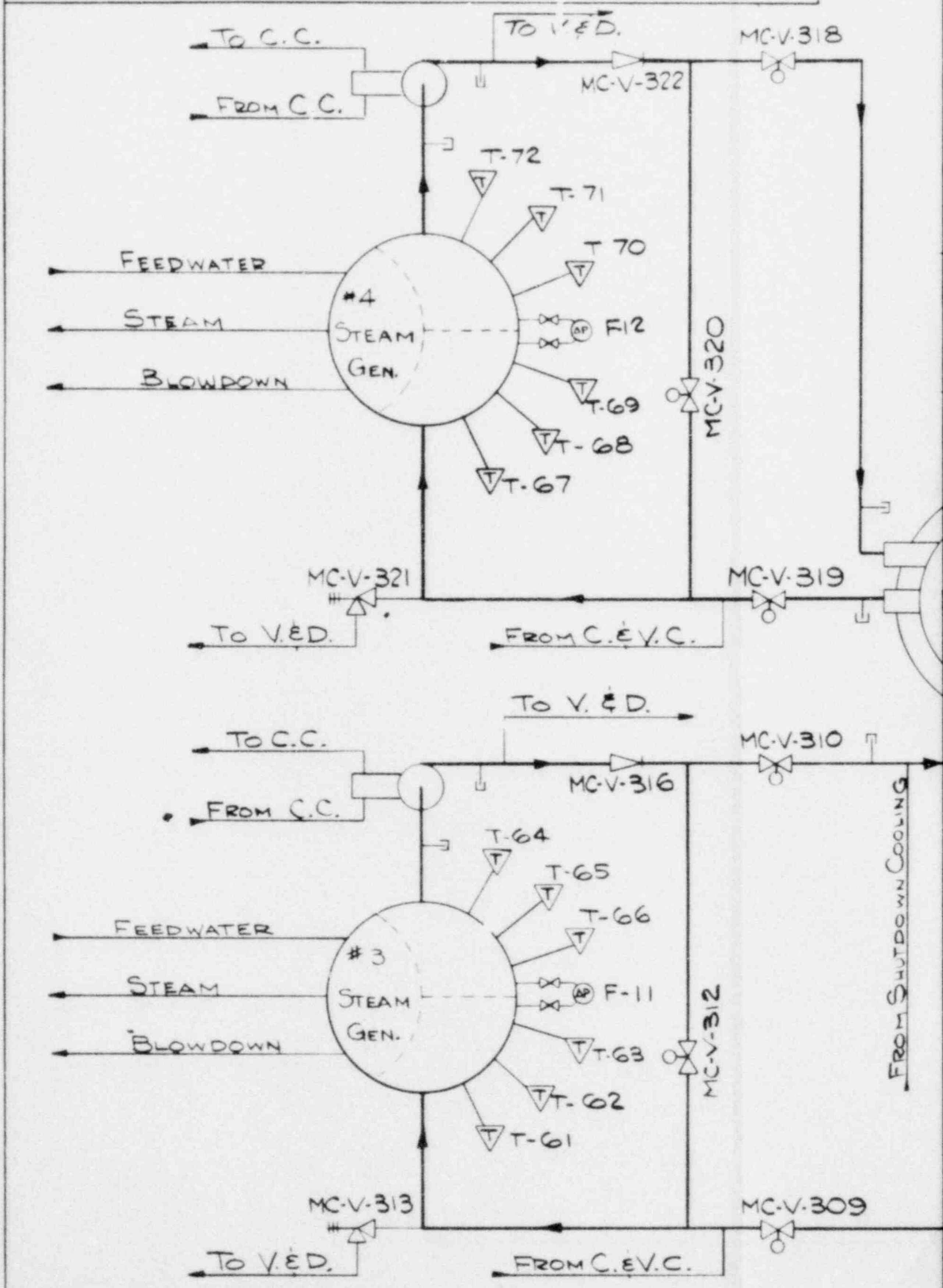
In order to maintain to an acceptable level, thermal stresses in the equipment handling the main coolant system, the heating and cooling rates are limited to 100 F per hr.

The operating pressure in the main coolant system is maintained within the range of 1,850 to 2,500 psia by the apparatus in the pressure control and relief system. The normal

LEGEND

P.C. & R. = PRESSURE CONTROL & RELIEF   C.C. = COMPONENT COOLING

C.&V.C. = CHARGING & VOLUME CONTROL   V.& D. = VENTS & DRAINS



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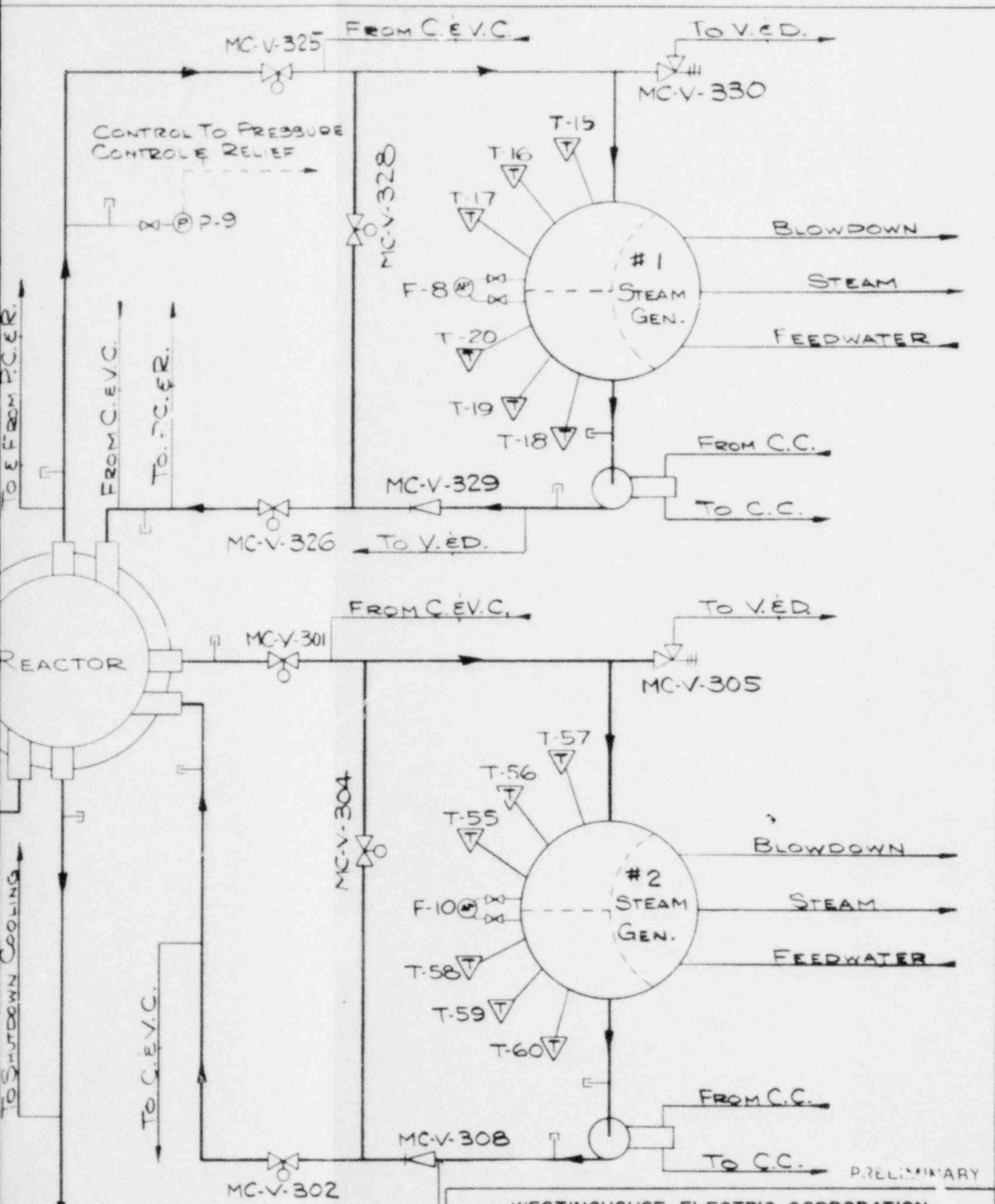


FROM C. & V.C.
TO V. & D.
MC-V-325
MC-V-330
CONTROL TO PRESSURE CONTROL & RELIEF
P-9
MC-V-328
T-15
T-16
T-17
#1 STEAM GEN.
BLOWDOWN
STEAM
FEEDWATER
F-8
T-20
T-19
T-18
FROM C.C.
TO C.C.
MC-V-329
MC-V-326
TO V. & D.
TO & FROM P.C.&R.
FROM C.&V.C.
TO P.C.&R.
REACTOR
FROM C.&V.C.
TO V.&D.
MC-V-301
MC-V-305
T-57
T-56
T-55
MC-V-304
#2 STEAM GEN.
BLOWDOWN
STEAM
FEEDWATER
F-10
T-58
T-59
T-60
FROM C.C.
TO C.C.
MC-V-308
MC-V-302
TO V. & D.
TO C.&V.C.
TO SHUTDOWN COOLING
PRELIMINARY
WESTINGHOUSE ELECTRIC CORPORATION
TITLE YANKEE ATOMIC ELECTRIC COMPANY
MAIN COOLANT SYSTEM
DRTM. SHAFFER
APPD. L.G. STODDART
646J421
DIV. C.A.P.A.
PITTSBURGH, PA.

operating pressure is 2,000 psia with pressure swings of plus or minus 150 psi. Automatically operated relief valves, actuated by pressure signal, limit abnormal pressure surges to 2,400 psia. These relief valves are backed up by full capacity code safety valves on the pressurizer which open at 2,500 psia. If the pressure drops below 1,850 psia, the reactor is shut down. The main coolant system is designed so that no leakage normally escapes to the vapor container atmosphere. Any leakage that does occur is controlled by high pressure packing leakoff to low pressure piped drains.

The main coolant system is designed for a maximum allowable working pressure of 2,500 psia. As required by the ASA Code for Pressure Piping, the system is hydrostatically tested to 3,750 psia. As an exception to this code, all hydrostatic testing is performed with the system at a minimum temperature of 100 F.

All main coolant apparatus which falls within the jurisdiction of the ASME Boiler and Pressure Vessel Code, Section VIII, is fabricated to meet the code requirements. In general, the ASME Boiler and Pressure Vessel Code Interpretations, Case No. 1224, (Special Ruling) applies. This ruling states:

"1 - It is the opinion of the Committee that vessels that are an integral part of nuclear installations and built in accordance with the requirements of the ASME Boiler and Pressure Vessel Code as modified or defined in this and subsequent cases, meet the intent of the Code, and each vessel shall be marked as required by the section to which it is built including the appropriate Code Symbol. In addition the words, "Case No. " shall appear on the Data Report.

2 - All vessels that are an integral part of nuclear installations shall be constructed in accordance either with the requirements of Section I or with the requirements of Section VIII for vessels that are to contain lethal substances.

3 - It is intended that jurisdiction over piping external to vessels shall terminate at: (1) the first circumferential joint for welding end connections; or (2) the face of the first flange in bolted flange connections; or, (3) the first threaded joint in that type of connection."

The steam generators are constructed in accordance with ASME Boiler and Pressure Vessel Code, Section VIII. The tube side of the steam generators is designed to a maximum allowable working pressure of 2,500 psia; the shell side to approximately 925 psi gage. The design temperature for the steam generators is 650 F. All metal surfaces in contact with the main coolant are of Type 304 stainless steel or equivalent. The main coolant circulating pumps are the canned-motor type. They are single speed centrifugal pumps designed for continuous operation at any temperature from ambient to 650 F, and at any pressure from 200 psia to 2,500 psia. These pumps are so designed that there is no leakage to the atmosphere. While they are not code equipment the pressure containing parts are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, where applicable.

The piping in the main coolant system is 20 in. Schedule 160, and of Type 304 stainless steel material. It is designed and fabricated in accordance with the ASA B31.1-1955, Code for Pressure Piping, Sections 1 and 6. The design temperature is 650 F, and the maximum allowable working pressure is 2,500 psia. The velocity of the main coolant through the piping is 38 ft per sec. Although the anticipated corrosion in the system is negligible, the code corrosion allowance of 0.065 in. is included in the specified pipe wall thickness. The piping is of welded construction throughout. Motor operated gate valves are employed in the main coolant system to permit isolation of a loop from the reactor vessel. These valves are designed to open and close against a 500 psi pressure differential and are capable of withstanding a 2,500 psi pressure differential in either direction.

The valve stem leakage is drained to a closed drain system and none escapes to the atmosphere.

A small relief valve is provided in each loop to prevent overpressurization of a main coolant loop when isolated from the reactor vessel and the pressurizer vessel. These valves are set to relieve at a pressure of 3,000 psi gage.

A check valve is employed in each main coolant loop to restrict reverse flow in the event of pump failure in a particular loop. These valves are of the conventional swing type check, except that they have a small hole in the disc. This hole permits a small coolant flow, through the loop in which a pump has failed, to maintain equalized temperatures.

203 INSTRUMENTATION AND CONTROL

Function

The function of the instrumentation and control is twofold. First, it provides the operator with continuous information on the plant performance during its operation. Second, it warns the operator of any unusual condition which might occur; automatically regulates small disturbances around the steady state; automatically shuts the reactor plant down for disturbances which exceed a predetermined value; and provides entire fail-safe operation.

General Description

The nonnuclear instrumentation and control of the reactor plant is conventional in principle. However, certain special features are necessary for those instruments in direct contact with the radioactive coolant, because of their inaccessibility during plant operation. The principal system quantities to be measured and controlled are pressure, temperature, liquid level and flow. Information from the detecting instruments, which are in continuous contact with the radioactive process, is transmitted, through adequate amplification circuits, to the control room where it is indicated or recorded. In addition, sufficient alarms, both audible and visual, are mounted on the control board, and operate for predetermined preset values of the process.

The nuclear instrumentation and control equipment consists of neutron and associated primary plant information channels for supervisory monitoring, safety shutdown and automatic nuclear level control in the power range. Multiple channels are provided, as necessary, to insure instrumentation and shutdown control under all conditions of operation and routine maintenance. The reliability of the equipment is high to insure uninterrupted power output from the plant without sacrifice of safety.

The start-up equipment consists of four channels, which compute and indicate the neutron level and rate of change of the neutron level from the source flux to the power range, thus providing information to the operator for manual start-up of the reactor from the control board. The start-up instrumentation channels also provide a stop rod signal to the control rod programming panel upon the existence of a rate of change of flux greater than a preset value. The start-up instrumentation channels are divided into two ranges, source range and intermediate range, with each range consisting of two duplicate channels, A and B. These channels are provided as the source ranges, the input being supplied by highly sensitive BF_3 proportional counters. The intermediate range channels are supplied with a direct current proportional to neutron flux level by compensated ion chambers. The information supplied by the start-up instrumentation equipment is made available by indication on meters or strip chart recorders.

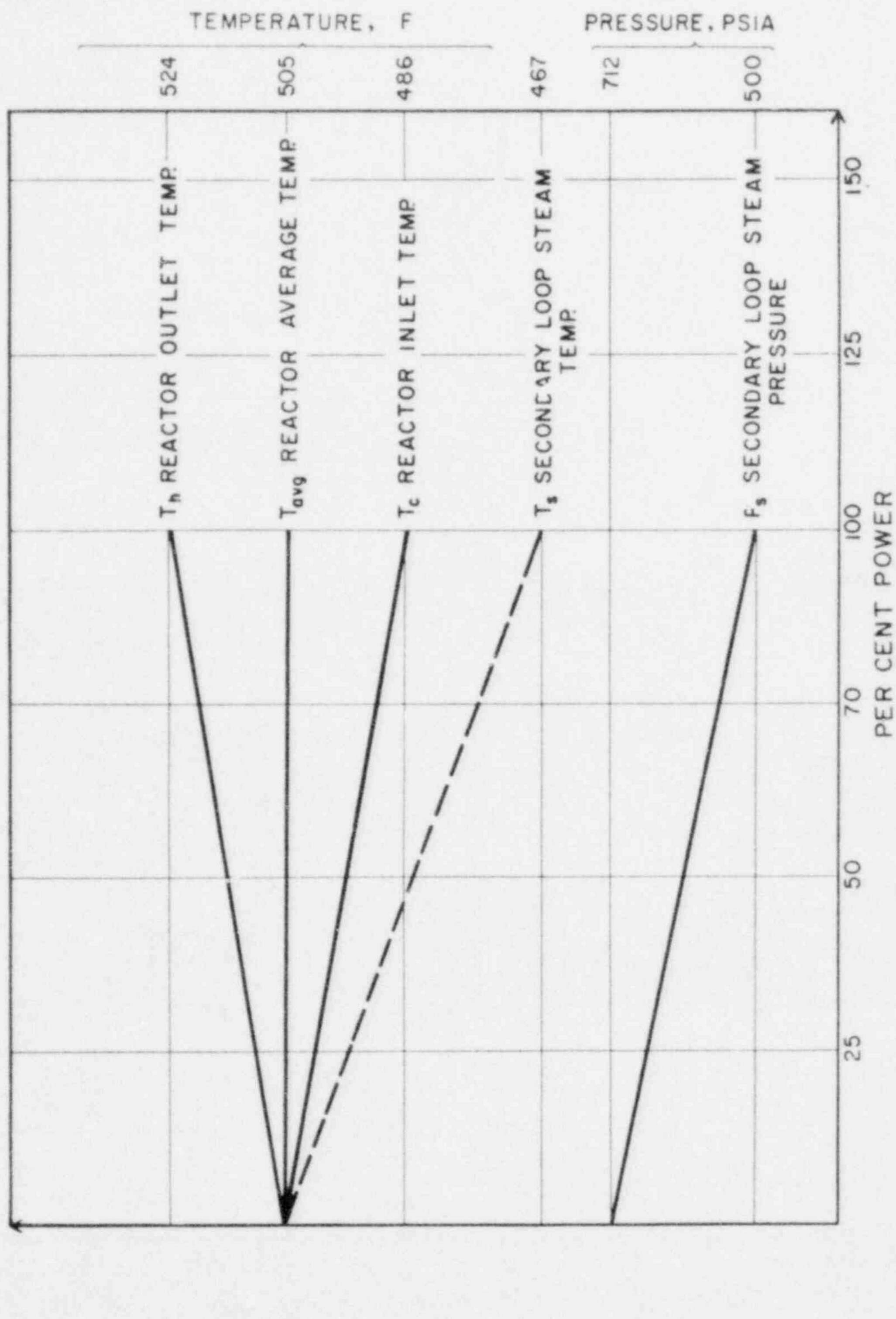
The power range equipment consists of three ionization chamber channels. These channels indicate the neutron level from the intermediate range through the power range. In addition, each channel supplies a shutdown signal upon the existence of a neutron level greater than a preset value. The simultaneous occurrence of any two of these shutdown signals results in tripping the shutdown signal amplifier and consequent insertion of the control rods. This coincidence feature eliminates the possibility of reactor power interruption due to a transient to one neutron detection channel, and still meets all safety requirements to prevent overpower operation.

The level information supplied by the power range equipment is made available for operation observance on the control board and on strip chart recorders. Incorporated with the power range equipment is the annunciator system with the alarm and shutdown panel. This panel gives a visible and audible indication of unsafe conditions in the nuclear or primary plant system and the source of the signal.

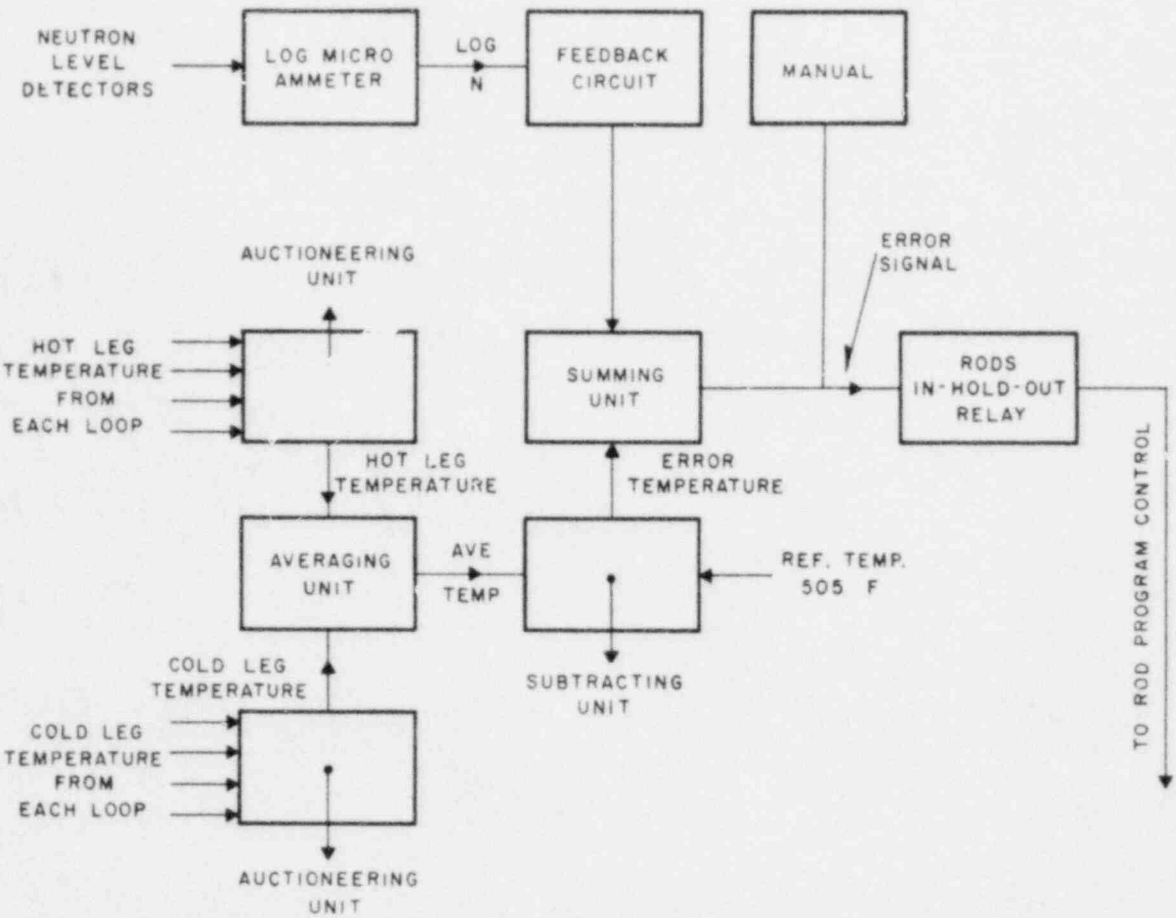
The reactor control is obtained by means of 24 neutron absorbing control rods which are operated in groups, either manually or automatically. At steady state operation, the reactor is automatically controlled by means of a group of preselected control rods. The reactor is controlled automatically to maintain the average temperature, T_{avg}, constant in the main coolant circuit.

Figure 16 shows curves of the reactor outlet temperature, reactor inlet temperature, reactor average temperature, secondary loop steam pressure, and secondary loop steam temperature as a function of reactor power.

Figure 17 is a block diagram which illustrates the general scheme which is used with the constant T_{avg} reactor control program. The signals from the temperature detectors located in the hot and cold legs of each one of the main coolant loops are fed into auctioneering units, where they are summed. The summed information from each one of the auctioneering units is averaged in an averaging unit and yields a signal proportional to the average temperature of the primary plant. This T_{avg} signal is compared with a reference temperature signal, in a subtracting unit, and any difference between the two signals results in an error temperature signal. A stabilizing feed-back loop from the reactor gives a signal proportional to the logarithm of the neutron level in the reactor, for disturbances around the steady state. A summing unit measures both the error temperature and the neutron level signals and, for predetermined excessive variation around



CONSTANT T_{avg} PROGRAM



REACTOR CONTROL SYSTEM
CONSTANT T_{avg} PROGRAM

the steady state average temperature, energizes a relay which closes the circuit to the drive mechanisms of the control rods. The control rods are thus driven in the direction to eliminate the error signal.

Each control rod is provided with its own position indicating scheme, using a variable impedance coil mounted on the control rod drive mechanism and a bridge circuit. Complete supervision of the reactor is obtained at the control board. Twenty-four alarm lights are mounted on the board, and these indicate when individual rods have reached the extreme of travel. They notify the operator that the automatic system has reached the limit of its control, that individual rods have been inadvertently dropped and the number of rods which have dropped in case of faulty operation of the shutdown system.

The control rods are driven in or out of the reactor by means of "jacking" type mechanism. A system of distributed coils energized with a direct current source provides the magnetic force necessary to hold the rods in position, if the specific set of coils is energized, and to move the rods in the desired direction and for a desired distance, when the "moving" and "holding" coils are energized sequentially. Loss of power to the mechanism coils results in a fail-safe operation and causes the rods to drop by gravity and scram the reactor.

Safety signals are kept to a minimum consistent with overall plant safety. The following is a set of signals which initiate shutdown in addition to manual scram:

<u>High Neutron Level.</u> A three-channel system in the power range which requires two channels to trip in coincidence decreases the probability of a false shutdown. The choice of a three-channel system with coincidence, therefore, represents a practical compromise between optimum false shutdown protection and optimum safety reliability.

<u>Short Reactor Period in the Start-up Range.</u> Reactor period indication is used to enable the operator to ascertain criticality and then to make the reactor supercritical on the desired exponential rise. If the period becomes short, trouble is imminent since minimum time is available to stop the rise of power before ratings are exceeded. For this reason, a reactor period safety control signal to the control rod drive mechanism is used for stopping too rapid a rise in power.

<u>Low Main Coolant Loop Flow.</u> Loss of main coolant pumps causes the total flow to decrease, and, unless measures are taken to scram the reactor before the flow has reached too low a value, a prohibitive

temperature rise takes place in the fuel assemblies. A coincidence circuit requires that at least three pumps fail to initiate a scram signal. Partial loss of electrical power supply, equipment mechanical failure, or a combination of the two which results in the loss of more than two main coolant pumps, therefore, initiate a scram signal.

High or Low Main Coolant Loop Pressure. High or low loop pressure in the main coolant loop which results in prohibitive temperature conditions for the fuel assemblies initiate a scram signal. Excessive pressure rise in the main coolant loop also initiates a scram signal.

Audible and visible alarms are provided and actuated under the following conditions:

High reactor outlet temperature
Loss of turbine generator load
Reactor period less than 20 sec

Suitable interlock circuits make it impossible to raise the rods following a safety scram signal. In other words, the safety signal can not be overridden by any action of the operator. However, unless a scram signal has been initiated, the operator is able to override the automatic circuits and decrease the power of the reactor or shut it down completely, by manual action, in any circumstances. When the reactor is at power, automatic run-in of the control rods will be initiated by excessive temperature in any one of the four main coolant loops.

Interlocks are provided in the main coolant system which prevent opening any of the main stop valves when a difference of more than 50 F exists between the water temperature in the loop isolated by the valve and the reactor vessel.

Basis for Design

The plant shielding design dictates that personnel are not permitted inside the vapor container when the reactor is critical. Therefore, information from detectors mounted inside the vapor container is transmitted to indicators or recorders mounted on the control board which is located in the control room external to the vapor container. The control board thus serves to centralize all the instrumentation and control functions of the plant which are necessary for both safe and efficient operation.

Individual circuit and overall system design, as well as component selection, are based on achieving maximum safety and reliability without complexity. Each circuit design is considered from the point of view that the plant output must not be interrupted, nor its safety jeopardized by any component failure. This philosophy has led to features such as complete magnetic-amplifier control and instrumentation whenever a scram signal must follow a disturbance which can not be tolerated, parallel safety systems and dual start-up channels. Electronic components are used for the instrumentation when it is only necessary to indicate or record a given measured quantity.

For reactor plant components outside the vapor container, the instrumentation is conventional, and local indication or recording is provided.

204 PURIFICATION SYSTEM

Function

The function of the purification system is to remove impurities from the main coolant. The purification system receives the entire spectrum of elements in various chemical forms; however, the important impurities to be removed are: Br_2, Rb, I_2, Xe, Cs, Kr, Ba, Fe, Ni, S, P, Si, Mn, U, Ta, Cu, Mo, Sn, B, NH_3, N_2, H_2, C, Np, Pu, Na, and Sr. These impurities appear as solids in suspension, solids in solution, gases in solution, and gases out of solution. Certain gases such as argon, xenon, krypton, radon, helium, and neon, are not handled directly by the purification system, but are piped to the waste disposal system.

General Description

The purification system, shown on drawing 646-J-423, is a low pressure arrangement consisting of two demineralizers, two circulating pumps, resin fill tank and transport equipment, and necessary piping, valves and fittings.

Basis for Design

The purification system is designed to remove corrosion products formed at the rate of about 10 mg/sq dm/mo plus the removal of up to 5 lb of uranium dioxide and associated fission products per year. It is assumed that the rates of corrosion and release chemicals leached from the fuel are constant throughout the year. Soluble impurities are removed by ion exchange and insoluble impurities are removed by mechanical filtering.

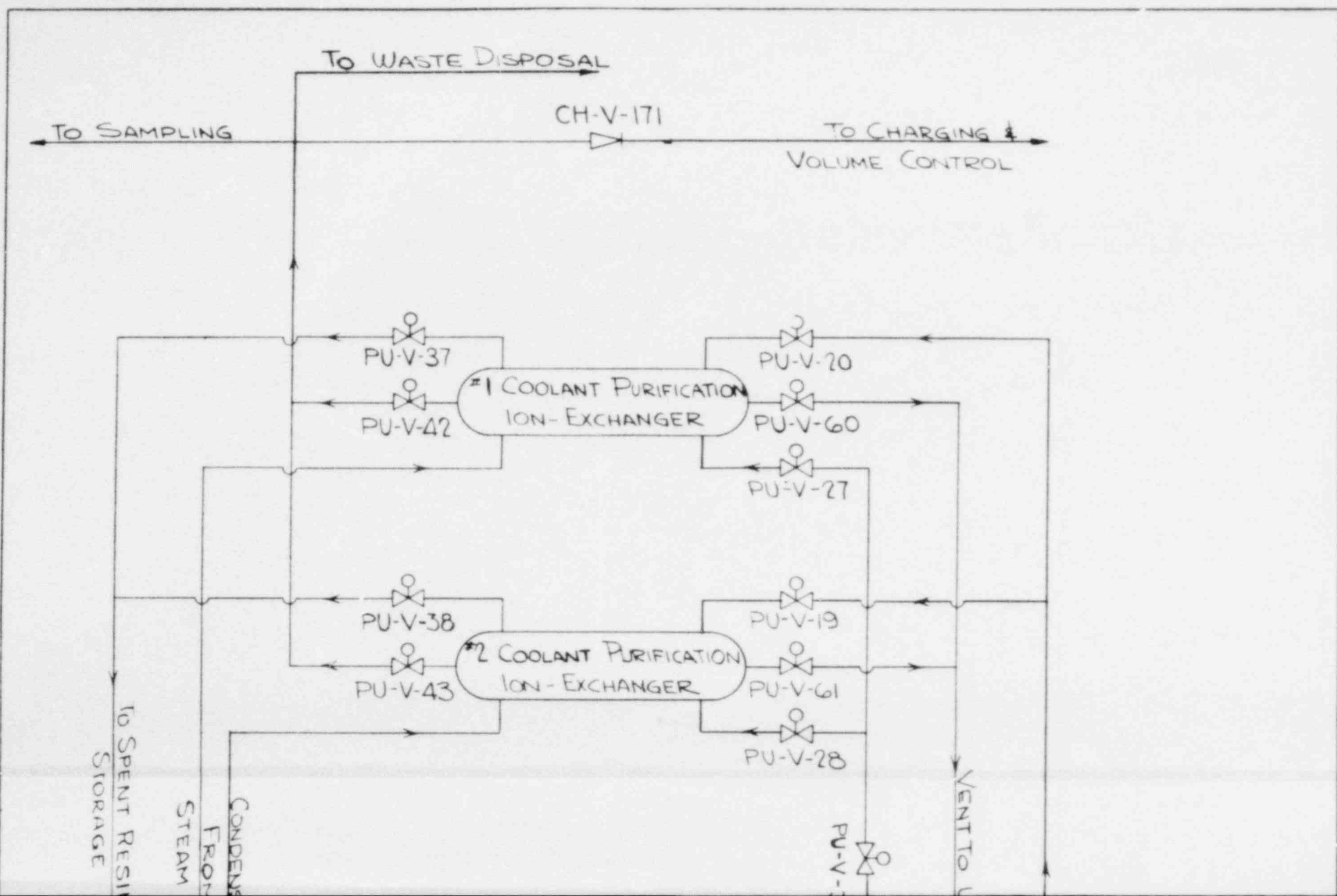
The main coolant flow circulating through the purification system is adjustable from 10 to 100 gpm. Continuous operation of the purification system is not vital to plant operation. It can be isolated and the resin replaced without interfering with the operation of the overall plant.

A mixed bed resin is used. The temperature of the resin is limited to 140 F and the resin is not regenerated.

The system design pressure is 125 psi gage and design temperature is 140 F.

The major portion of the system is located external to the vapor container in a shielded and limited access area under the vapor container.

Filtering is accomplished by the resin bed itself and by integral mechanical filters in the demineralizer units.


To Waste Disposal
To Sampling
CH-V-171
To Charging &
Volume Control
PU-V-37
PU-V-42
#1 Coolant Purification
Ion-Exchanger
PU-V-20
PU-V-60
PU-V-27
PU-V-38
PU-V-43
#2 Coolant Purification
Ion-Exchanger
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PU-V-61
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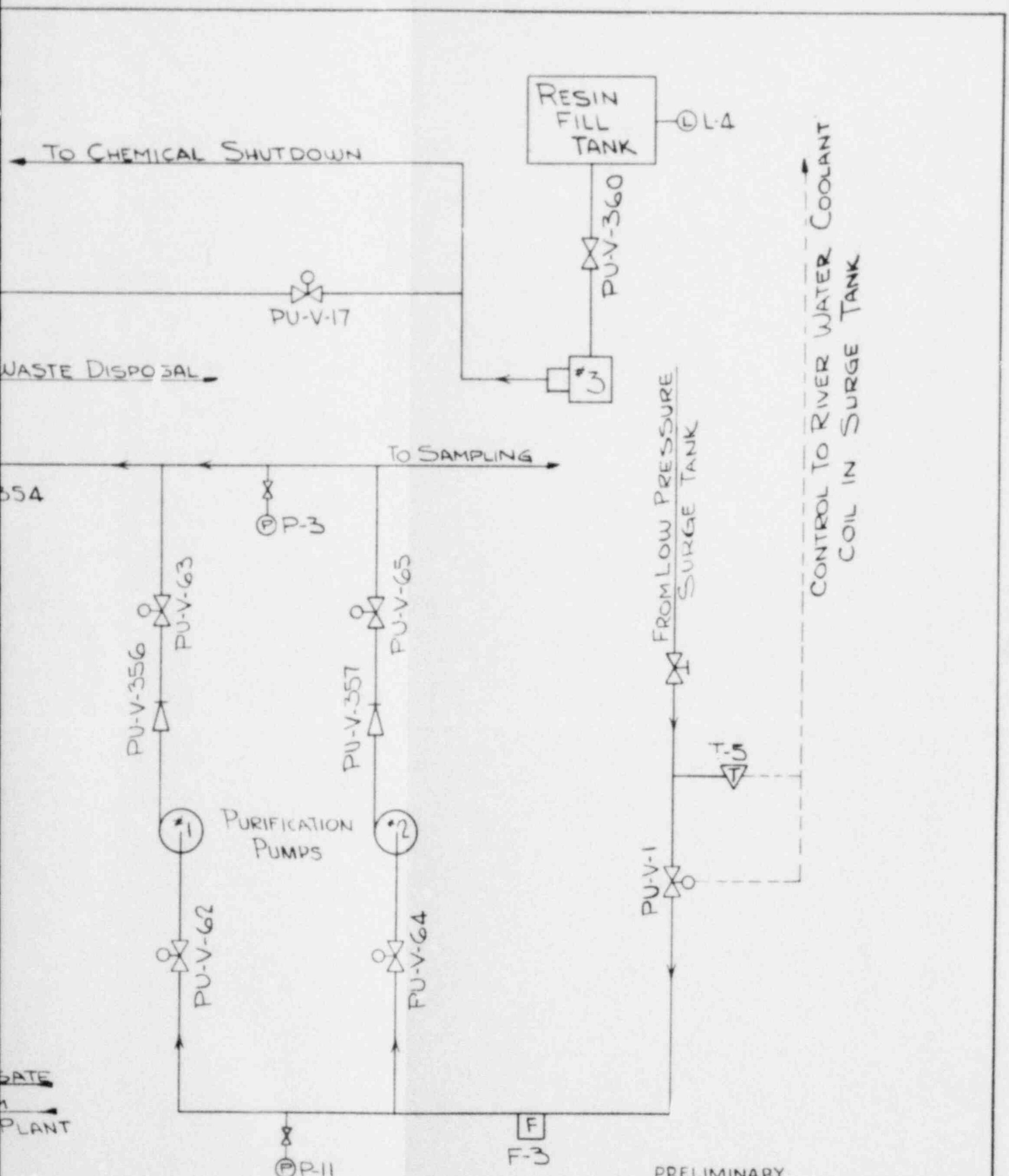

PRELIMINARY

WESTINGHOUSE ELECTRIC CORPORATION

TITLE YANKEE ATOMIC ELECTRIC COMPANY

PURIFICATION SYSTEM

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DIV. C.A.P.A.		PITTSBURGH, PA.		

205 CHARGING AND VOLUME CONTROL SYSTEM

Function

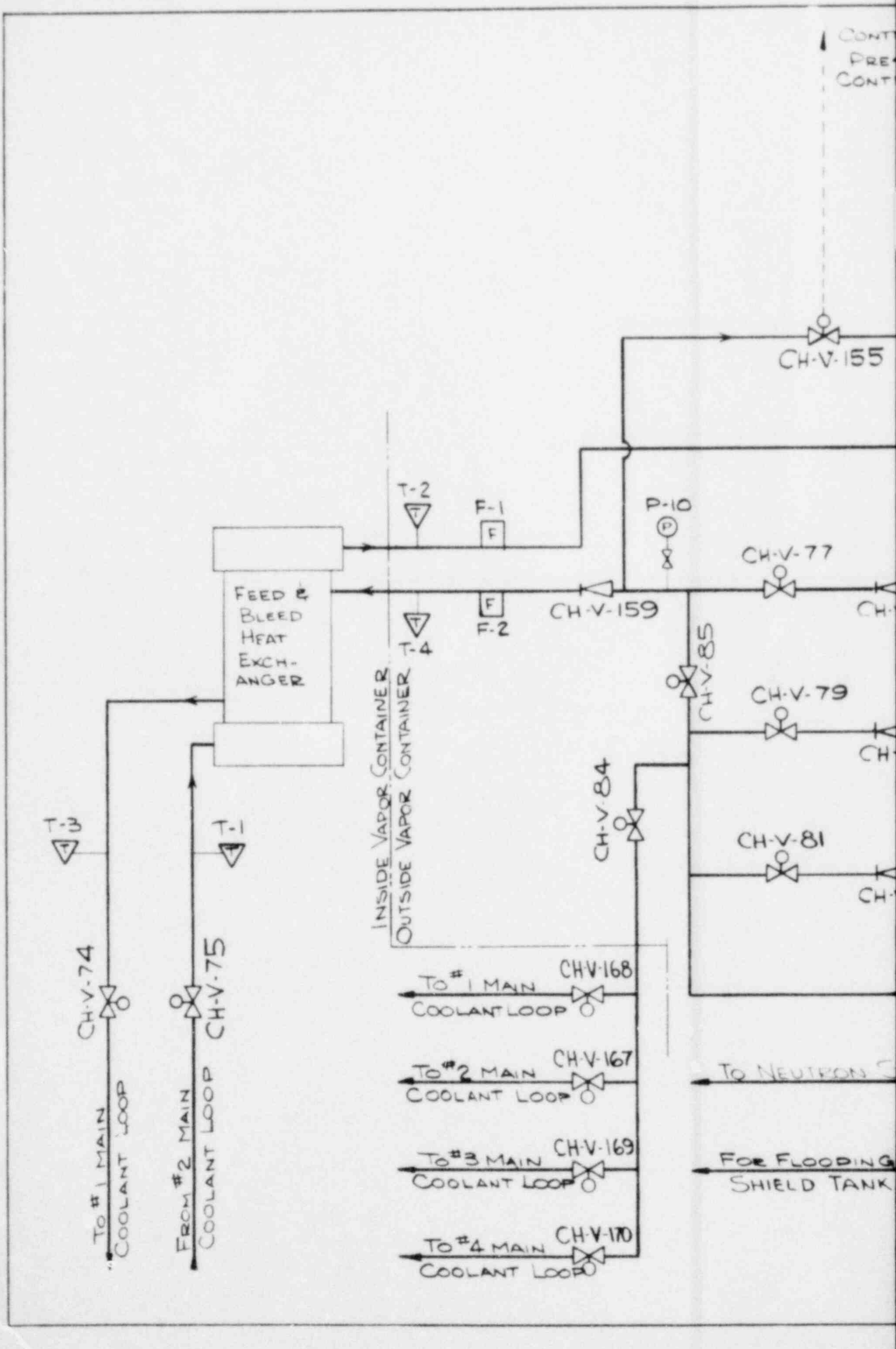
The functions of the charging and volume control system are: to fill the main coolant system or an isolated loop with demineralized water; to provide make-up water for main coolant system leakage during normal plant operation; to maintain the desired level in pressurizer by bleeding off water from the main coolant system due to a main coolant water expansion during normal plant operation; to maintain desired level in pressurizer vessel by charging water into the main coolant system to compensate for primary water contractions during normal plant shutdown; to provide a means of charging borated water, or other suitable chemical neutron absorber into the main coolant system during scheduled plant shutdown; to provide cooled low pressure main coolant for the purification system during normal plant operation; to quench discharge from safety and relief valves in the pressure control system; and to provide facilities for hydrostatic testing main coolant system and high pressure auxiliaries.

Although not a direct function of the charging and volume control system, certain elements of the system are used to remove noncondensable gases from the main coolant water and to maintain a corrosion inhibitor in the main coolant water.

General Description

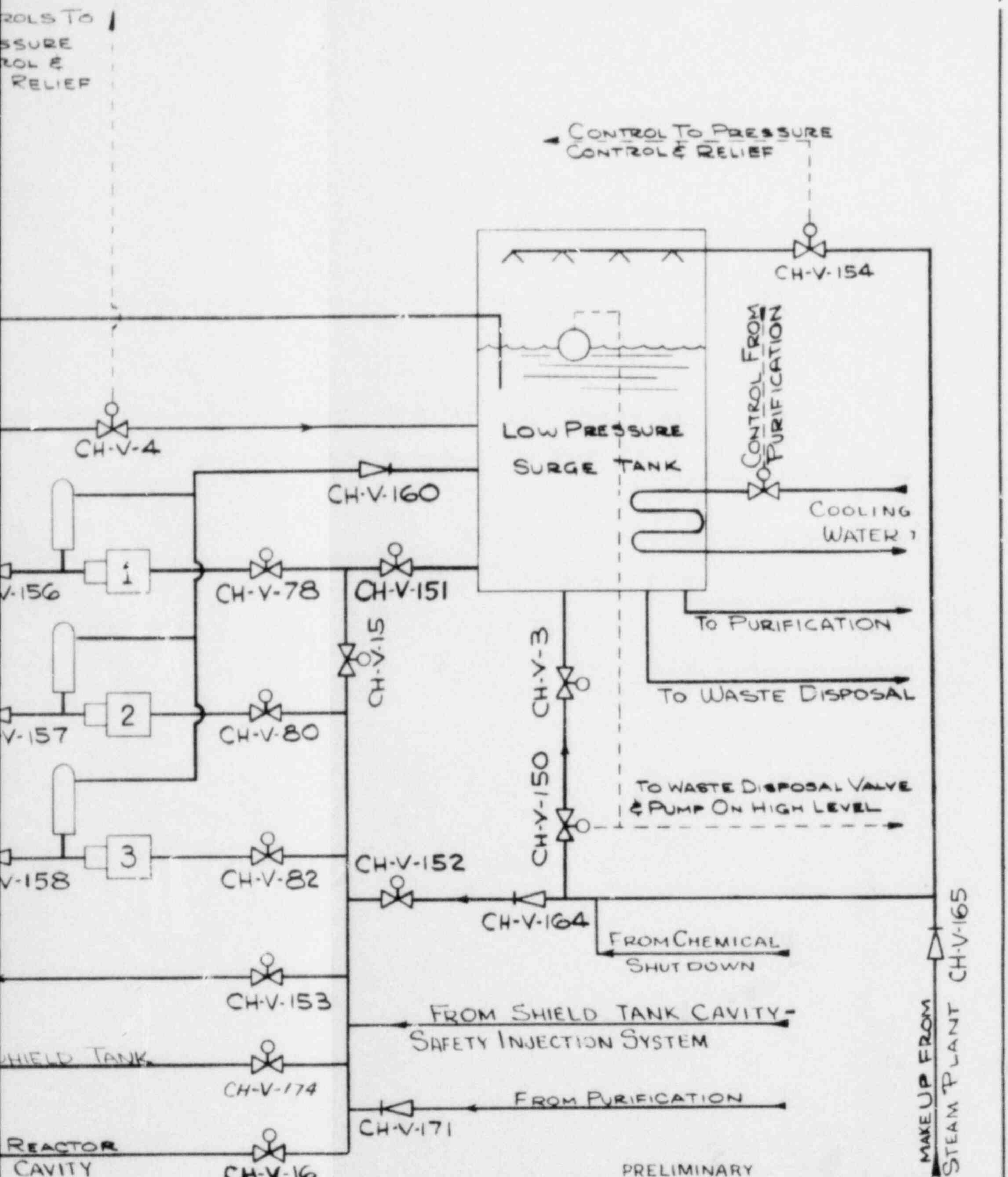
The charging and volume control system is shown on drawing 646-J-430. During normal steady state plant operation, water is bled from the main coolant system at approximately 2,000 psia and 500 F. The water passes through a regenerative heat exchanger where the temperature is reduced to 200 F. The water then flows through a control valve, which reduces the pressure, and into the low pressure surge tank. The water in the tank is maintained at a temperature of 120 to 130 F by a cooling coil. The surge tank contains hydrogen at a pressure of 20 to 30 psia. The water in the low pressure surge tank is circulated through the purification system and the purified water is fed back into main coolant loop No. 1 by a constant flow reciprocating pump. The return charging water flows through the regenerative heat exchanger and back into a cold leg in the main loop, re-entering at approximately 400 F.

A by-pass line makes it possible to unload the charging pumps into the surge tank. The by-pass line, sized for about 50 gpm, has a control valve which opens on high water level in the pressurizer. The control valve fails safe, by closing, in case of loss of power or loss of control. Under emergency conditions, all three pumps operate in parallel at the discretion of the operator, maintaining a flow of 100 gpm.



CH-V-155
T-2
F-1
P-10
CH-V-77
FEED & BLEED HEAT EXCHANGER
CH-V-159
F-2
T-4
CH-V-85
CH-V-79
INSIDE VAPOR CONTAINER
OUTSIDE VAPOR CONTAINER
CH-V-84
CH-V-81
T-3
T-1
CH-V-74
CH-V-75
TO #1 MAIN COOLANT LOOP
FROM #2 MAIN COOLANT LOOP
CH-V-168
TO #1 MAIN COOLANT LOOP
CH-V-167
TO #2 MAIN COOLANT LOOP
CH-V-169
TO #3 MAIN COOLANT LOOP
CH-V-170
TO #4 MAIN COOLANT LOOP
TO NEUTRON
FOR FLOODING
SHIELD TANK

113-119



PRELIMINARY

WESTINGHOUSE ELECTRIC CORPORATION				
TITLE YANKEE ATOMIC ELECTRIC COMPANY				
CHARGING & VOLUME CONTROL SYSTEM				
DFTM. SHAFFER	1-7 56	APPD. T.G. STUSCART	1/9/57	646J430
DIV. C.A.P.A.		PITTSBURGH, PA.		

Depending upon the operation of the charging and volume control system, the control valve in the bleed line regulates the flow from 10 to 150 gpm. The flow during normal plant operation is approximately 33 gpm. In case of loss of power or loss of control, the system fails safe.

In addition to maintaining a continuous feed, the pumps are used to charge borated water into the main loop and to test hydrostatically the main coolant system and its service systems.

The low pressure surge tank quenches the discharge from the pressurizer safety and relief valves. A quenching spray is used to reduce a pressure increase in the surge tank in the event that a safety or relief valve operates in the pressure control system. The surge tank contains approximately 500 cu ft of water at 120 to 130 F during normal plant operation.

Basis for Design

The diaphragm control valve in the bleed line regulates the flow from 10 to 150 gpm. While the main coolant system is at a pressure of 2,000 psia, the flow during steady state operation is approximately 33 gpm.

The feed and bleed lines are used to maintain the desired water level range in the pressurizer vessel, although this is not a controlling factor in sizing the pressurizer for positive and negative surges in the vessel.

The following operations take place during a positive surge in the pressurizer:

As the water rises in the pressurizer, it reaches a point where a level sensing device actuates the control valve in the bleed line, causing the flow to increase through the valve.

As the water level continues to rise, the valve opens to its maximum position until the flow through the bleed line is approximately 150 gpm.

If the control valve in the bleed line is wide open and if the water level in the pressurizer continues to rise, the control valve at the charging pump discharge line opens and the feed flow, normally back into the main coolant system, is dumped into the low pressure surge tank.

The following operations take place during a negative surge in the pressurizer:

When the water level drops in the pressurizer, it reaches a point where the control valve in the bleed line is actuated and the flow through the valve decreases until the valve is completely closed.

If the control valve in the bleed line is closed and if the water level in the pressurizer continues to decrease, a low level alarm sounds. If this occurs, pump No. 2 and pump No. 3 can be started by the operator. With three pumps operating, approximately 100 gpm are charged into the main coolant system

The feed and bleed line accommodates the water level changes in the pressurizer during plant start-up and shutdown. If the main coolant is heated at a rate of 100 F per hr, water must be bled from the main loop at a maximum rate of approximately 50 gpm. Since the bleed line is sized for a maximum flow of 150 gpm, this flow can easily be handled. A decrease in the water level in the pressurizer due to normal leakage from the main coolant system is also compensated for by the feed lines.

Three reciprocating pumps are employed in the charging and volume control system. All pumps have a flow of 33 gpm at a working pressure of 2,300 psi. Any one of the three pumps is a spare for either of the others. Each pump is provided with a relief valve which discharges into a common header and back into the low pressure surge tank and each can be isolated from the line.

The low pressure surge tank is sized to quench the maximum discharge envisaged from the safety or relief valves in the pressure control system. The cylindrical surge tank has a volume of 1,000 cu ft and contains 500 cu ft of water at 120 to 130 F during normal plant operation. The cumulative positive surge at the end of the maximum positive surge expected from the main coolant relief system causes the pressure to rise to approximately 150 psi gage and the corresponding temperature to 330 F. If a pressure of 150 psi gage is exceeded, the safety valves on the surge tank open. The pressure build-up in the surge tank, due to the piston action of the rising water is reduced by a spray which begins to function when the pressure reaches 100 psi gage in the tank. The flow rate of the spray is 5 to 10 gpm.

A cooling coil is used to cool the water in the surge tank from 220 to 120 F. The flow of the cooling water in the coil is controlled by a temperature regulating valve. The normal flow through the coil is approximately 60 gpm, and the maximum flow is approximately 300 gpm. The maximum allowable working pressure of the surge tank is 150 psi gage.

The water bled from the main coolant system enters the tube side of the regenerative heat exchanger at approximately 500 F and discharges from the tube side at about 220 F. The maximum flow through the tube side is 150 gpm. The water from the charging pumps enters the shell side of the regenerative heat exchanger at about 130 F and discharges from the shell side at approximately 400 F. The maximum flow through the shell side is 100 gpm based on the capacity of the charging pumps.

The maximum allowable working pressure of both the shell and tube sides is 2,500 psia.

206 PRESSURE CONTROL AND RELIEF SYSTEM

Function

The functions of the pressure control and relief system are to maintain the required main coolant pressure at the reactor outlet during steady state operation; to limit to an allowable range the pressure changes caused by main coolant thermal expansion and contraction during normal power plant load transients; and to prevent the pressure in the main coolant system and auxiliaries from exceeding the design pressure.

General Description

The pressure control and relief system consists of a pressurizer vessel containing a 2-phase mixture of steam and water, replaceable immersion heaters, code safety valves, remotely operated relief valves, spray system, interconnecting piping, valves, and instrumentation connections, as shown on drawing 646-J-422. The electrical heaters, located in the lower section of the pressurizer vessel, accomplish pressurization of the main coolant system by maintaining the water and steam at saturation temperature. The heaters are capable of pressurizing and raising the temperature of the pressurizer and contents at the desired rate during start-up of the plant. The heaters are turned on in successive steps when system pressure decreases below operating pressure.

The pressurizer vessel is designed to accommodate positive and negative surges of the main coolant system caused by normal power plant load transients. During a positive surge, the spray system condenses steam in the pressurizer to limit the pressure increase to a value which can not actuate the remotely operated relief valves. During a negative surge, flashing of steam in the pressurizer keeps pressure above a minimum value fixed by the reactor core heat transfer design and safety requirements.

Code safety valves and remotely operated relief valves are provided to accommodate large pressure surges which are beyond the capacity of the pressurizer. The code safety valves are capable of preventing system pressure from exceeding a design pressure of 2,500 psia based on ASME Boiler Code, Section I. Pressure switch operated relief valves operate at a pressure of 2,400 psia to minimize the operating frequency of the code safety valves.

Basis for Design

During power plant load changes, the heat input from the reactor and the heat removed by the steam generators become unequal and a coolant volume change occurs. The pressurizer is designed to prevent excessive pressure changes caused by the volume change during a normal power plant load transient. Power changes are limited to 10 per cent of full reactor power per minute.

The quantity of surge based on 10 per cent of full power per minute depends on the value of the reactor negative temperature coefficient. The surge quantity is calculated for two conditions, based on the following design characteristics for the main coolant system:

Operating normal pressure	2,000 psia
Average coolant temperature	505 F
Active flow volume	2,400 cu ft
Total system volume	3,000 cu ft

Condition 1 - The reactor negative temperature coefficient stops the surge 10 sec after a change in steam generator average temperature. The quantity of the surge, which is approximately the same for positive or negative, is slightly less than 1 cu ft. Therefore, 1 cu ft is the maximum surge expected during a normal power plant load change, with an effective negative temperature coefficient.

Condition 2 - The reactor negative temperature coefficient is nearly zero and, therefore, it does not correct the reactor output and the power change continues at the constant rate of 10 per cent per minute. Action is taken with reactor core control rods to correct the condition, and this action is taken within 30 sec. In this case, the quantity of the surge is 10 cu ft 30 sec after a change in steam generator average temperature.

The pressurizer maintains the main coolant system pressure at 2,000 psia during normal operation, and limits the pressure fluctuations due to load transients to 1,850 psia during a negative surge and 2,200 psia during a positive surge. The positive surge requires the largest pressurizer based on the isentropic compression of the steam volume. The spray system that condenses steam during the positive surge decreases the size requirement of the pressurizer.

The positive surge based on a 10 cu ft surge, as in Condition 2, requires a pressurizer volume of 120 cu ft with a spray system. The same negative surge requires only 70 cu ft of pressurizer volume.

Additional pressurizer volume is included to allow for level indicator error, heater unit volume, and insurance against uncovering heaters during a large negative surge. Provision is made for 30 cu ft of additional volume so that a pressurizer of 150 cu ft satisfies the requirements of the system.

Electrical immersion heaters, rated at 125 kw, are required in the 150 cu ft pressurizer during start-up to raise temperature at a rate of 100 F per hr.

A constant small flow of main coolant is sprayed into the pressurizer steam volume to serve as a degassifier and as a means of recirculating the water in the pressure control system. Since the recirculation water returns to the main coolant system through the surge pipe, the water in this pipe is close to the saturation temperature. Therefore, water entering the pressurizer on the positive surge does not radically decrease the water temperature and thus avoids a serious negative surge following the main coolant system positive transient. Steady state heater load to overcome heat loss to recirculation water and through insulation is less than 10 kw.

Large positive surges connected with electric generator loss require a rapid controlled reduction in power. A complete loss of flow requires scram. The pressure relief valves are designed to prevent overpressure during the maximum possible surge. Loss of main coolant by an accident causes depressurization of the main coolant system and causes the reactor to shut down.

A system for dumping steam from the secondary side of the steam generator to the condenser is provided as artificial load for the main coolant system.

207 DECONTAMINATION SYSTEM

Function

The functions of the decontamination system are to supply a decontamination solution to the main coolant system or to any auxiliary system in the primary plant for the purpose of removing radioactive fission products and corrosion products from the wetted internal surfaces of components, piping, and fittings; and to receive, neutralize, and transport radioactive decontamination solution to the waste disposal system.

General Description

Further research and development is required before a satisfactory decontamination system can be designed. This section of the Hazards Summary Report will be submitted at a later date.

208 WASTE DISPOSAL

Function

The waste disposal system receives, contains, adequately treats and safely disposes of all radioactive wastes in such a manner as to yield concentrated liquid wastes and ashes solidified in concrete. Low activity level liquid wastes are discharged to the environment after being diluted with the main turbine condenser effluent cooling water. Gaseous wastes expected in small quantity, after suitable dilution with air, are dispersed to the atmosphere under favorable meteorological conditions.

The potential sources of radioactive liquid and gaseous wastes to be processed by the waste disposal system are as follows:

Main Coolant System
Charging and Volume Control System
Purification System
Sampling System
Chemical Shutdown System
Vent and Drain System
Shutdown Cooling System
Vapor Container Drain Liquid
Safety Injection - Shield Tank Cavity System
Radioactive Laboratory, Decontamination Cubicle and Decontamination Pad Drain Liquids
Contaminated Laundry Drain Liquid
Contaminated Area Floor Drain Liquid
Steam Generator Drain Liquid

Radioactive wastes from the reactor plant, operating at steady state, appear as solids in suspension, solids in solution, gases in solution and gases out of solution. If no fuel rod cladding defects occur, only activated corrosion products and radiolitic gases are piped to the waste disposal system for safe disposal. Plant operation, however, will continue with some fuel cladding defects, depending upon the adequacy of the primary plant shielding and the capacity of the waste disposal system.

Intermittently, other liquid wastes containing radioactive materials are handled by the system. These wastes include liquids from the radioactive laboratory, the decontamination cubicle and pad, and the contaminated laundry in the Service Building; drain liquids from various primary auxiliary systems not radioactive normally but which may, under certain conditions, become radioactive; drain liquids from the shield tank cavity system and spent fuel pit which occur only infrequently; and activity dilution liquid and boron dilution liquid comprising relatively large volumes of liquid at each complete plant shut-down and start-up. When the plant is operated with some fuel cladding defects, the activity level in the main coolant must be reduced by bleed and feed dilution during plant shutdown. This is necessary so that the activity of the shield tank cavity water, after it mixes with the diluted main coolant remaining in the isolated reactor, will be sufficiently low to permit refueling operations. On the second and subsequent plant start-ups, the boron-containing water used in cold shut-down of the plant will be radioactive and must be disposed of safely.

Adequate monitoring of the waste disposal system is provided to assure safe operation, storage, drumming and controlled release of activity either to the environment or to approved waste disposal sites. Sufficient tank capacity is provided to permit suitable analysis of liquid wastes before a decision is made to treat and then reuse in the primary system or dilute and release to the environment.

General Description

The waste disposal system consists of liquid and gas storage tanks, gas stripper, evaporator, incinerator, wet gas scrubber, pumps, compressors, heat exchangers, fans, filters, instruments, piping and valves, all as shown on drawing No. 9699-RM-41F.

Basis for Design

The system is designed to handle the total quantity of radioactive liquid, gaseous and solid wastes originating in the primary plant, secondary plant and Service Building, based on primary plant operation 330 days per year, two complete shut-downs per year, and 15 days for inspection and maintenance during each shutdown. The total quantity of radioactivity entering the system is based on an equilibrium fission product activity in the main coolant of 37.64 microcurie per ml resulting from cladding defects in 1 per cent of the total number of fuel rods and a main coolant equilibrium corrosion product activity of 0.830 microcurie per ml. The total and isotopic activities given herein are the best data presently available, and they are subject to revision as better information is developed. The following

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☒ FILMED ON APERTURE CARD NO 8101070140-01 is a list of all radioactive isotopes having an equilibrium activity level in the main coolant greater than 0.01 microcurie per ml:

Nonvolatile Fission Products

Isotopes	Activity, Microcurie per ml
Rb-88	0.25
Rb-89	0.032
Sr-89	0.036
Mo-99	0.029
Te-101	0.033
I-131	1.6
Te-132	2.2
I-132	2.1
I-133	2.1
Te-134	0.15
I-134	0.31
I-135	0.94
Cs-137	0.088
Cs-138	0.14
Cs-139	0.026
Ba-139	0.025

Volatile Fission Products

Isotopes	Activity, Microcurie per ml
Kr-85 (4.4 hr)	0.096
Kr-85 (10.3 yr)	1.5
Kr-87	0.19
Kr-[illegible]	0.024
Xe-133	24.2
Xe-135	1.5
Xe-138	0.046

Corrosion Products

Isotopes	Activity, Microcurie per ml
Mn-56	0.52
Co-60 (5.3 yr)	0.062
Fe-59	0.071
Na-24	0.15
Cr-51	0.24

The design also provides for reducing the activity of the recovered liquid wastes sufficiently to permit their re-use as primary plant make-up water. This minimizes the quantity of water to be discarded and, hence, the total activity in the liquid wastes discharged from the plant. The total radioactivity in all liquids released to the environment annually will be approximately 0.007 curie, which is many times less than that permitted at existing AEC installations.

The waste liquids, irrespective of activity level, are divided into two classifications: reactor plant effluents defined as radioactive liquids containing dissolved hydrogen and fission product gases; and, radioactive liquids containing dissolved air. The gaseous wastes are subdivided into hydrogen, containing radioactive fission gases, and air with undetectable activity. Solid wastes are combustible and noncombustible.

Each classification and type of waste requires a somewhat different treatment process. The basic processes used in this system are: natural decay of radioactive isotopes, steam stripping of the liquid wastes to remove volatile fission products, evaporation to concentrate radioactive constituents in a small volume of liquid waste to be solidified in concrete and incineration to concentrate activity in a reduced volume of solid wastes.

Liquid Wastes

The volume and activity of liquid wastes entering the system that always require treatment, based on an assumption of cladding defects in 1 per cent of all fuel rods, are as follows:

	Average Volume, Cu Ft per 6 Months	Average Gross Activity, Microcurie per ml
Reactor Plant Effluents		
Main Coolant (No Boron)		
Normal operation	1,320	30.1
Shutdown (Activity dilution process)	9,210	10.77*
Main Coolant (With Boron)		
Shutdown (Drain loops and equipment)	2,165	0.580
Start-up (Boron dilution process and coolant expansion)	8,733	0.002
Radioactive Liquids Containing Air		
Incinerator Rotoclone drain liquid	120	0.03
Radioactive laboratory sink, decontamination cubicle and pad drain liquid from Service Building	4,140	0.029
Total Volume	25,688 or 4,280 cu ft per month average	

Total Average Activity = 5.45 microcurie per ml

*The activity dilution liquid is passed through one of the purification ion exchangers, giving an assumed decontamination factor of 10 for nonvolatile fission products and corrosion products.

The major volume of reactor plant effluents entering waste disposal consists of activity dilution liquid pumped from the low pressure surge tank to the activity dilution decay tank; boron dilution liquid pumped from the low pressure surge tank to the waste hold-up tank; and liquids from primary plant valve stem leak-offs and the main coolant loop drains which flow by gravity to the primary drain collecting tank. These liquids are pumped batchwise to the waste hold-up tank. Incorporated as part of the purification ion exchange system are provisions for passing boron-free waste liquids being discharged to waste disposal from the low pressure surge tank or the primary drain collecting tank through an ion exchange bed to remove nonvolatile fission and corrosion products on resin. This method of operation will be used to remove part of the nonvolatile activity, if the extra decontamination factor is required when discharging activity dilution liquid to waste disposal, or if it proves to be more economical than evaporation and drumming.

The waste hold-up tank serves as a surge tank, and, as soon as sufficient volume has accumulated for one evaporator batch, processing is started through the gas stripper and evaporator. Since the activity dilution liquid contains over 70 per cent of all nonvolatile radioactivity discharged to waste disposal, it is stored separately from other reactor effluents and decayed a total of 30 days to reduce the nonvolatile fission product activity by a factor of 31 before further treatment, which permits operating the evaporator at a higher concentration factor.

Liquid from either the waste hold-up tank or the activity dilution decay tank is charged continuously at 5 gpm to the top of the gas stripper. This liquid is stripped essentially free of all fission product gases and all inert gases by a countercurrent flow of steam on 20 bubble trays operating at slightly above atmospheric pressure. The stripper is designed to give a decontamination factor of about 10^7 for volatile radioactive isotopes.

Stripped liquid from the bottom of the gas stripper is charged to an electrode type evaporator which operates at 10 psi gage and utilizes the current passing through an electrolyte to generate heat. This type evaporator uses the same principle as employed for many years in large steam generators in Canada and Europe, and small steam generators in this country. The electrode evaporator was selected for this plant to eliminate the difficulties experienced with corrosion and scaling of heat transfer surfaces in more conventional steam heated or direct-fired evaporators when used for evaporating radioactive waste liquids.

The electrode evaporator is designed to operate with a highly conductive liquid, as it is expected that the conductivity of the waste liquids will vary over a wide range. If the liquid being evaporated is not sufficiently conductive, a suitable spiking chemical will be used. Pilot runs by the manufacturer on a test electrode evaporator, using a synthetic decontamination solution and other expected solutions, indicate spiking will be unnecessary and no difficulty will be encountered with scaling of the electrodes, even in saturated boric acid solutions. Although corrosion of the cast iron electrodes is very slow, provision is made in the design for easy replacement of the electrode tips.

The steam leaving the evaporator passes through a series of bubble trays, countercurrent to a stream of evaporator distillate, and then through a stainless steel mesh and glass wool packed deep bed filter to reduce carry-over of entrained radioactive aerosols and obtain a bottoms to distillate decontamination factor of about 10^7. A portion of the steam is returned and used for gas stripping. The remaining steam is condensed and the resulting evaporator distillate is pumped through a cooler to a test tank, where it is sampled for laboratory analysis. The gross nonvolatile activity in the distillate is less than a factor of 10 above the maximum permissible concentration given in the AEC Regulations for an unrestricted area. This liquid is transferred to the primary water storage tank for subsequent use as primary plant make-up and for the bleed and feed dilution operations. The test tanks and the primary water storage tank are provided with floating roofs with seals at the tank wall to minimize diffusion of oxygen into the stored liquid, in order to maintain an oxygen content of less than 0.2 ppm. Any excess primary make-up water will be discharged at a controlled rate to the main turbine condenser cooling water discharge line, resulting in a dilution factor many times greater than that required to satisfy AEC Regulation (10 CFR, Part 20).

The evaporator concentration factor is adjusted by monitoring the bottom of the evaporator to produce a bottoms liquid which, when mixed with cement and solidified in 55-gal steel drums and then stored to give a total of 60 days' decay, meets all AEC and ICC Regulations for common carrier shipment of radioactive materials. These drums are shipped away without additional shielding for ultimate disposal in an approved manner, either on land or at sea. The total number of drums of solidified bottoms liquid required to dispose of radioactive liquid wastes is estimated to be 524 per year.

Liquids containing dissolved air which are normally only slightly radioactive from nonvolatile constituents, such as laundry waste liquid from the Service Building and contaminated area floor drains, are held in the monitored waste tanks for sampling and laboratory analysis. If, after dilution with the main condenser effluent cooling water, the monitored liquid meets the standards of public authorities, it is discharged at a controlled rate to the condenser discharge line. Monitored liquids and all other waste liquids containing air, which are expected or known to have a nonvolatile activity level too high to obtain adequate dilution with the quantity of main condenser water available, are transferred to the gravity drain tank. All liquid pumped or drained to the gravity drain tank is processed the same as the reactor effluents, but in separate batches to prevent the mixing of air with fission gases and hydrogen which could result in the possible formation of an explosive hydrogen-air mixture.

The volume, activity level and total gross nonvolatile activity in the recovered primary make-up water and in the liquid wastes discharged from the plant are as follows:

Recovered Primary Grade Make-up Water

Returned to primary plant, cu ft per 6 months	21,140
Excess discharged from plant, cu ft per 6 months	3,600
Gross activity, microcurie per ml	1.44×10^{-7}
Total gross activity discharged from plant in excess water, curie per year	3.0×10^{-5}

Liquid Wastes Discharged Without Treatment

Volume discharged from plant, cu ft per 6 months	7,150
Total gross activity discharged from plant in contaminated laundry drain liquid, curie per yr	7.13×10^{-3}

Only liquid waste from the contaminated laundry is shown, since the total volume of floor drains and other liquids can not be determined.

Gaseous Wastes

The gaseous wastes consist almost entirely of hydrogen and radioactive fission product gases which are dissolved in the liquid discharged to waste disposal, or which continuously leak through or are released intermittently to the primary drain collecting tank by the pressure control valve on the low pressure surge tank. Fission gases and hydrogen are collected from the stripper condenser and from the vapor space of all reactor effluent liquid drain and hold-up tanks in a completely closed waste gas header system. This is compressed to a gas surge drum, which is bled back to the compressor suction to maintain a constant pressure on the waste gas header and a cushion to permit filling and emptying of tanks. Initially, this system will be filled with nitrogen and this atmosphere may be maintained indefinitely at the option of the operator..

The net gas make collects gradually in the waste gas surge drum and is removed once each month from the compressor discharge line and stored under pressure in one of three gas decay drums for about 60 days to reduce the activity. The decayed gas discharged from the drum is passed through a deep bed particulate filter and then released at a carefully controlled rate to the suction side of one of two air dilution fans. Interlocks are provided to shut off, automatically, the flow of gas if the fan stops. The decayed gas and dilution air are thoroughly mixed during passage through the fan, after which the mixture is discharged to the atmosphere through a short stack used only for wastes from the waste disposal system. The stack gases are continuously monitored. The gaseous waste equipment is designed for continuous discharge of decayed gas to the atmosphere 20 out of every 30 days of plant operation. This provides a 10-day interval without discharge in the event of a prolonged weather inversion.

Nearly all the activity in the air discharged from the stack is caused by 10.3 year krypton-85 isotope since, after 60 days' decay, the activity of the 5.3 day xenon-133 is very small. Neither Handbook 52 nor the AEC Regulation give the maximum permissible concentration of krypton-85 in air. Since krypton-85 is a noble gas, the MPC of this isotope in air should be determined by the same method used for establishing the MPC of xenon isotopes in air. Information and equations given in Handbook 52 indicate the MPC for noble gases is based on the radiation dose received during continuous exposure when completely surrounded by a radioactive gas. Using these equations, the calculated Handbook 52 MPC for krypton-85 was determined and, by similarity with xenon, the AEC Regulation MPC for this isotope in restricted and unrestricted areas was obtained. These concentrations, plus published data for xenon-133, are given in the following:

	Microcurie per ml	
	Krypton-85	Xenon-133
Handbook 52	3.36×10^{-6}	4×10^{-6}
AEC restricted area	1.0×10^{-5}	1.3×10^{-5}
AEC unrestricted area	3.36×10^{-7}	4×10^{-7}

When averaging concentrations over a period of one month or one year, the MPC calculated for continuous exposure in an unrestricted area may be increased by a factor of 1.5, since radioactive air is to be discharged only 20 out of every 30 days of plant operation. This gives a corrected MPC in the air discharged from the stack of 5.0×10^{-6} microcurie per ml for krypton-85, and 6×10^{-6} microcurie per ml for xenon-133.

Based on the intermittent discharge of a mixture of air and radioactive krypton-85 and xenon-133 at an arbitrarily selected gross activity of 2.5×10^{-6} microcurie per ml, the volumes and activity levels of gaseous waste, assuming cladding defects in 1 per cent of all fuel rods, are as follows:

Average volume of gaseous wastes, scf per month	206
Average gross activity of gaseous wastes:	
At zero decay, microcurie per ml	79.1
At 60 days decay, microcurie per ml	4.26
Discharge rate of decayed gaseous wastes, scf per hr*	0.43
Air dilution volume, cfm	12,200
Average gross activity of air discharged from the stack, microcurie per ml	2.5×10^{-6}
Total activity discharged to the atmosphere, curie per month	25.5
Total volume of krypton-85 released, ml per month	16

*Based on discharging decayed gas 20 days out of every 30 days of plant operation.

The air discharged from the stack during a 20-day period has a gross activity level one-quarter that permitted by the AEC Regulation for mixed identified isotopes in a restricted area. In order to satisfy the AEC Regulation MPC of 5.0×10^{-7} microcurie per ml for discharge two-thirds of the total time in an unrestricted area, an additional dilution factor of about 5 is required. Since the decayed gas will not be released during periods of weather inversion, a dilution factor many times greater than this can be obtained by atmospheric dilution on the basis of the Sutton equation, with the distance term taken as 600 ft from source point to the nearest property boundary.

A program of field work to measure air currents at the site has been undertaken as outlined in Section 301. The purpose of this work is to validate the atmospheric dilutions postulated above.

Air normally having undetectable activity from the stripper condenser, when stripping liquids containing air, is passed through a particulate filter and discharged to the waste disposal stack. This confines any unexpected volatile activity and permits monitoring this air as it is released.

Combustible Solid Waste

Combustible solid wastes, such as removable floor covering, cloths used for decontamination, shoe coverings and contaminated paper, are transported to the waste disposal building in combustible fiber drums. The drums will be re-used until they become contaminated. This waste is burned in a specially designed incinerator, based on a U.S. Bureau of Mines recommendation using free vortex flow of combustion air over the grate. Bottled gas is used as fuel to start and complete the combustion.

The flue gases from the incinerator are mixed with cool air and then passed through a combination wet gas scrubber and induced draft fan, in which essentially all particulate material 5 microns and larger and more than 70 per cent of those particles one micron and larger is removed, and the gases are further cooled. The gas is then filtered through a glass wool packed deep bed filter for final cleanup and discharged to the stack.

The incinerator is designed to burn about 80 lb per batch. After a complete charge is burned, the residue ash is dropped through the cone bottom into an open top 55-gal steel drum about one-half full of water. The drum is connected securely to the special head attached to the bottom of the cone, so that dust from the ash can not escape from the drum into the room. When the drum contains about 15 batches of ash, the two sprays inside the special head are operated to wet the ash thoroughly. Liquid is decanted from the settled ash, the drum lowered from the special head and a standard clamp-on head installed. The number of batches of ash and the volume of liquid remaining in the drum are adjusted so that, after mixing with cement, the mixture solidifies. Drums containing the solidified mixture are ready for shipment and disposal by burial. Liquid drained from the wet gas scrubber, in excess of that remaining with the settled ash in the drums, is pumped to the gravity drain tank for further treatment.

Noncombustible Solid Waste

The noncombustible solid waste consists of cartridges from small filters, glass wool from large filters and various items of contaminated plant equipment. The filter elements and small items of plant equipment are immobilized and shielded in concrete in 55-gal drums, with final disposal by burial. Large pieces of plant equipment are immobilized in concrete and buried at the plant site.

Equipment Capacities and Ratings

The capacities and ratings of equipment shown on Drawing No. 9699-RM-41F which follows page 208:2 are:

Tanks	Net Operating Capacity
1 - Primary building sump tank (TK-24), gal	484
1 - Gravity drain tank (TK-27), gal	4,700
2 - Monitored waste tanks (TK-29-1,2), gal each	1,370
1 - Primary drain collecting tank (TK-30), gal	7,500
1 - Waste hold-up tank (TK-31), gal	75,000
1 - Activity dilution decay tank (TK-32), gal	75,000
1 - Distillate accumulator (TK-33), gal	60*
2 - Test tanks (TK-34-1,2), gal each	8,040
1 - Comperssor K.O. drum (TK-35)	-
1 - Waste gas surge tank (TK-36), cu ft	4,160
3 - Gas decay drums (TK-37-1,2,3), cu ft each	60
1 - Primary water storage tank (TK-39), gal	135,000
1 - Ash dewatering sump (TK-43), gal	100

Pumps	Design Gpm, each
2 - Primary building sump tank pumps (P-24-1,2)	75
2 - Gravity tank transfer pumps (P-25-1,2)	10
2 - Monitored tank transfer pumps (P-26-1,2)	50
2 - Collecting tank transfer pumps (P-27-1,2)	50
2 - Waste liquid transfer pumps (P-28-1,2)	20
2 - Stripper bottoms pumps (P-29-1,2)	10
1 - Distillate pump (P-30)	10
1 - Test tank effluent pump (P-31)	20
1 - Ash dewatering sump pump (P-32)	10

Other Equipment	Design Rating, each
2 - Waste gas compressors (C-3-1,2), scfm	20
1 - Gas stripper (T-1), gpm	5
1 - Evaporator (EV-1), gpm	6
2 - Air dilution fans (FN-1-1,2), cfm	12,200
1 - Incinerator (M-1), lb per hr	40
1 - Rotoclone (M-2), lb per hr	480

*1/2 full of liquid

209 SHUTDOWN COOLING SYSTEM

Function

The functions of the shutdown cooling system are to remove heat from the main coolant after normal shutdown procedure has effected cooling and depressurization by the steam by-pass dump line; and to remove the decay heat, due to degeneration of fission products in the reactor core, during extended shutdown periods for maintenance and fuel replacement.

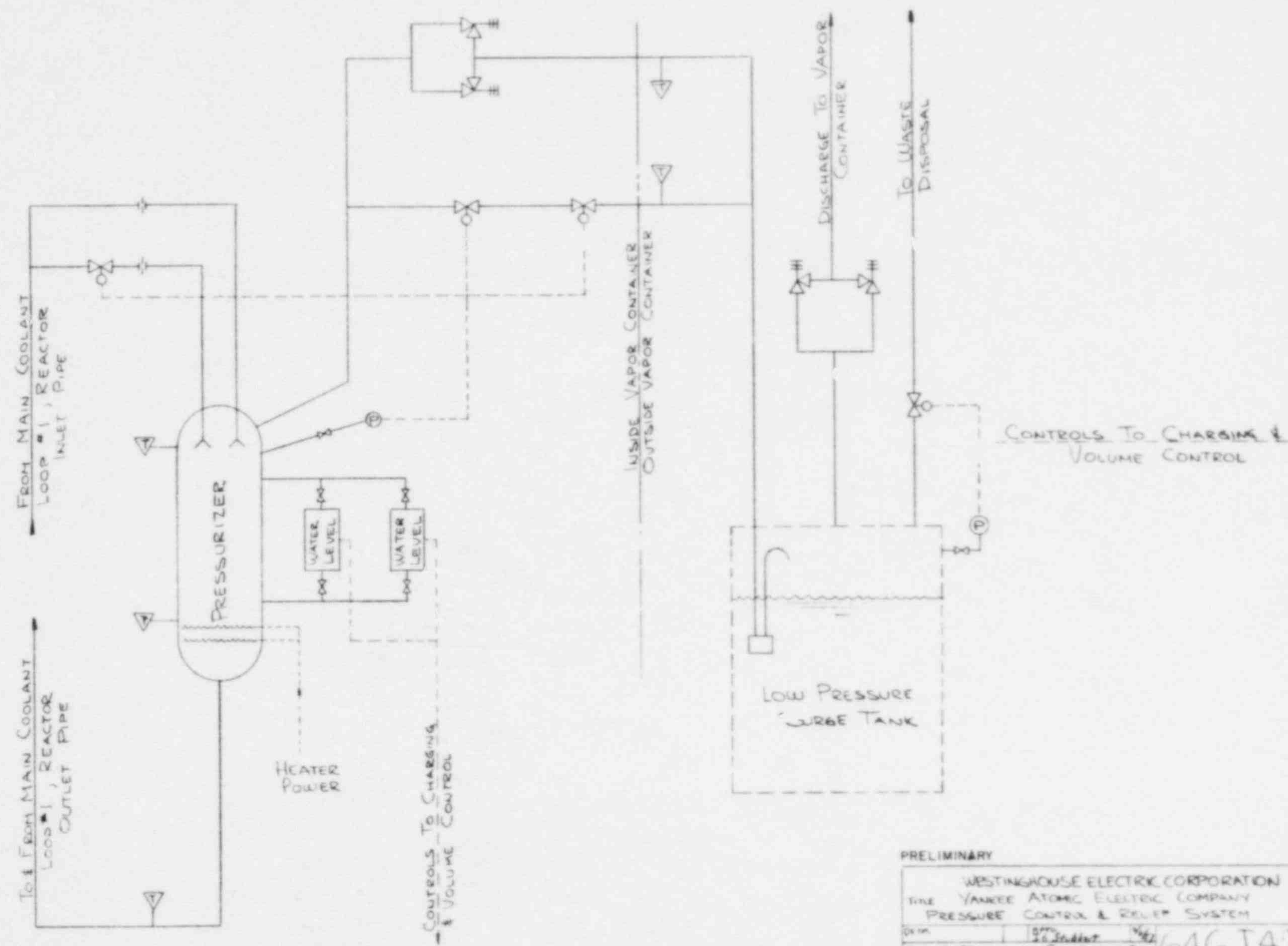
General Description

The shutdown cooling system consists of two heat exchangers, two circulating pumps, piping, and valves arranged in a loop parallel with the main coolant loops. This arrangement is shown on drawing 646-J-425. The shutdown cooling system circulates main coolant through the reactor by means of circulating pumps and transfers the heat load to the raw river water via heat exchangers. These heat exchangers employ double walled tubes in order to prevent contamination of the river water which might result from leakage of the main coolant water through the tubes or tube sheets.

Basis for Design

The shutdown cooling system is designed to remove 1 per cent of reactor full power when the reactor is shut down, depressurized, and cooled. The shutdown cooling system is put in operation 3 to 4 hr after the main coolant system has been cooled and depressurized at a maximum rate of 100 F per hr by using the steam generators to remove the decay and sensible heat. After this period, the decay heat produced by the reactor has decreased to approximately 1 per cent of full power depending upon previous operating history. One per cent of full power, 492 mw, is about 17,000,000 Btu per hr.

The main coolant water passes through the inside of the double walled tubes, and river water circulates around the outside of the tubes of a typical double wall tube design shell and tube heat exchanger. The annular space between the two fluids is open to a telltale tank at atmospheric pressure, so that any leakage into the space is indicated by the telltale. Because of arrangement in elevation, the two fluids are always at a higher pressure than atmospheric; thus, reverse flow leakage into the river water or main coolant water would not be possible.

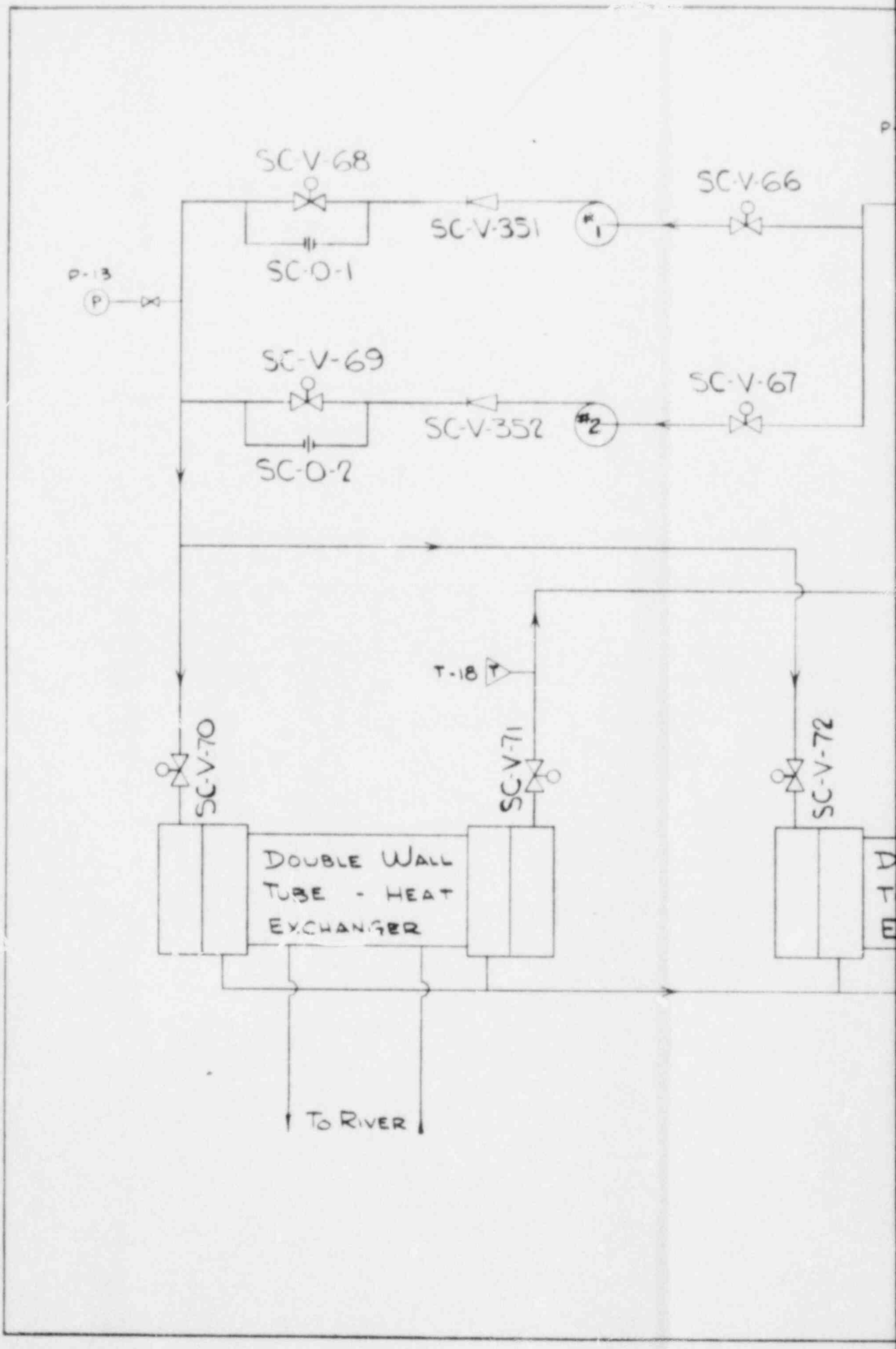


PRELIMINARY

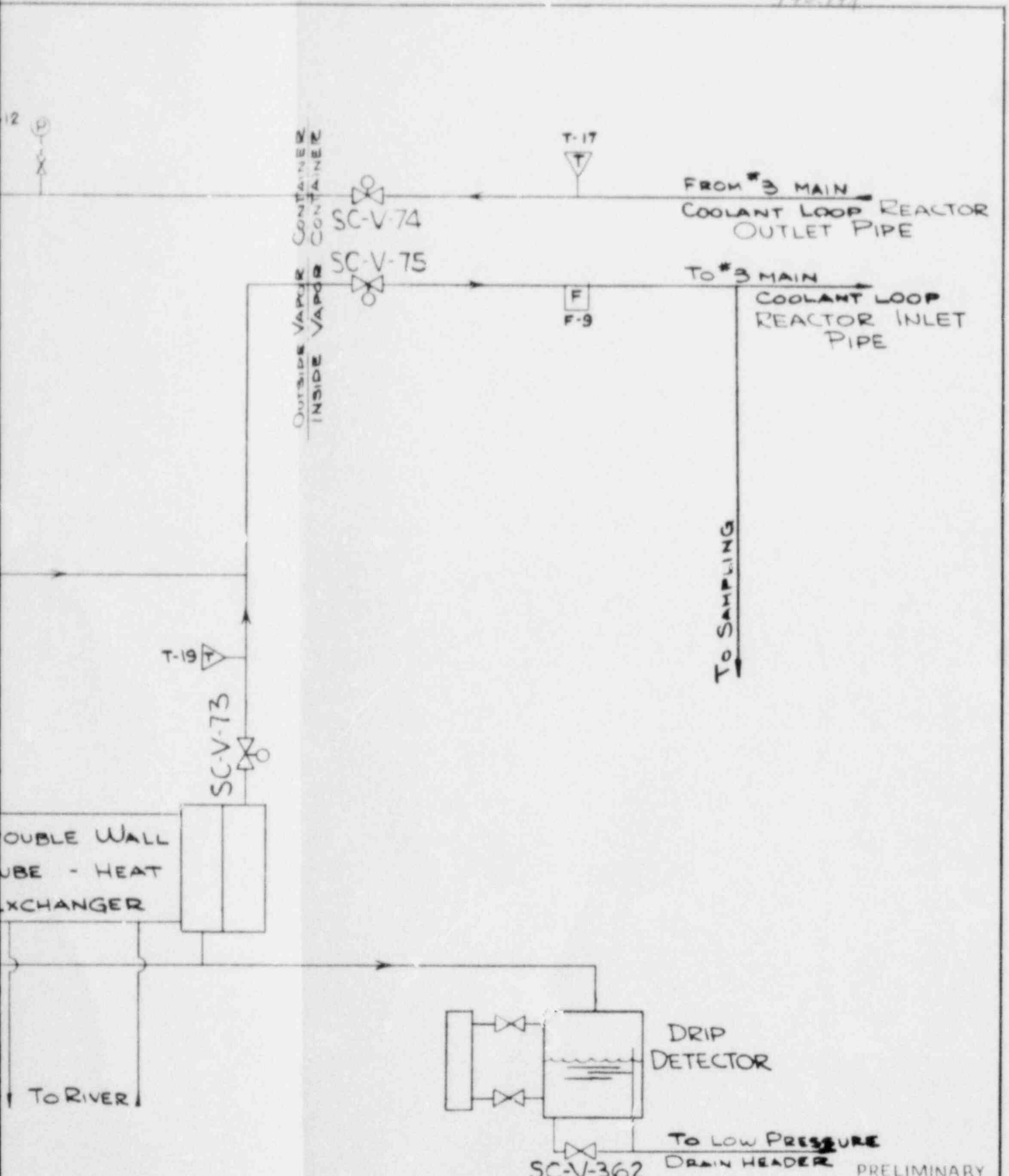
WESTINGHOUSE ELECTRIC CORPORATION

TITLE YANKEE ATOMIC ELECTRIC COMPANY
PRESSURE CONTROL & RELIEF SYSTEM

DIV. CAPA — PITTSBURGH, PA — 646J422



SC-V-68
SC-V-351
SC-O-1
SC-V-66
P-13
P
#1
SC-V-69
SC-V-352
SC-O-2
SC-V-67
#2
T-18
T
SC-V-70
SC-V-71
SC-V-72
DOUBLE WALL
TUBE - HEAT
EXCHANGER
TO RIVER



PRELIMINARY

WESTINGHOUSE ELECTRIC CORPORATION

TITLE YANKEE ATOMIC ELECTRIC COMPANY

SHUTDOWN COOLING SYSTEM

DFTM. SHAFFER	12-5-58	APPD.		646J425
DIV. C.A.P.A.		PITTSBURGH, PA.		

Careful operating procedure is used when placing this system in service to prevent excessive thermal shock which results if the heat exchangers are put into service too suddenly.

All units in this system are sealed against leakage to the atmosphere, or provided with controlled leakage glands and piping to convey the leakage to appropriate drain tanks.

210 MONITORING AND ALARM SYSTEM

Function

The function of the monitoring and alarm system is to detect, compute, and indicate the radiation level at selected locations inside and outside the plant. If these levels exceed predetermined values, alarms are actuated. Radiation monitoring serves a dual purpose; the first is to warn of any radiation health hazard which might occur; the second is to give early warning of plant malfunction which might result in a health hazard or plant damage.

General Description

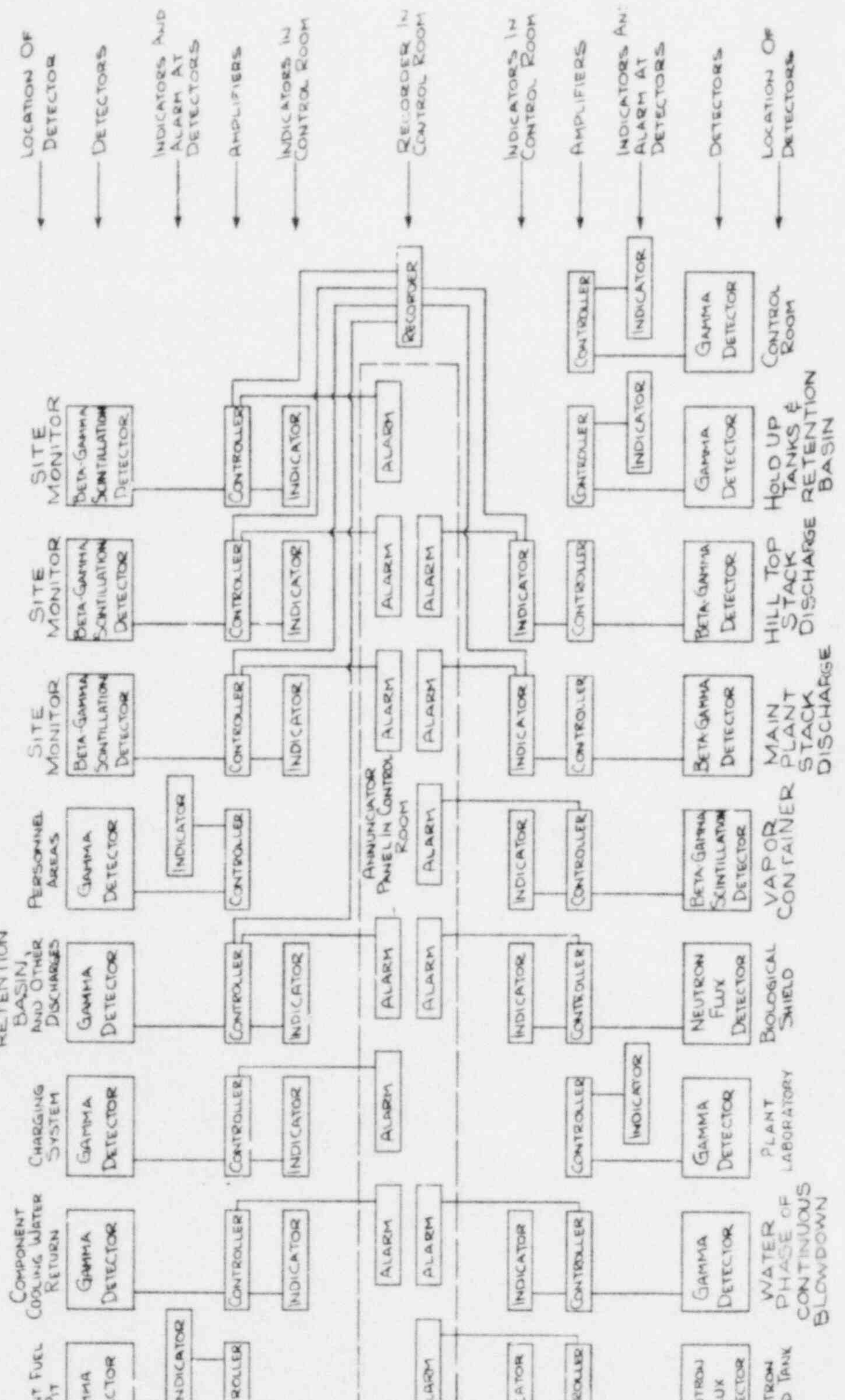
Figure 18 shows a block diagram of the instruments used in the monitoring and alarm system. Each detector has a controller which includes the amplification circuit. Each detector has an indicator and an alarm located in the control room, except those for the spent fuel pit, personnel areas, laboratory, waste storage tanks, and control room, which have local indicators. A multipoint recorder is used to record the activity of all plant discharges and other site locations. Monitored plant discharges include the plant stacks, laboratory and sanitary day tanks, vapor container and turbine plant drain tank, waste disposal retention basin discharge, and circulating water discharge.

A detector measures the neutron flux in the neutron shield tank, and provides continuous indication in the control room. Another detector measures the neutron flux in the biological shield, and actuates an alarm if the level exceeds a predetermined value.

The atmosphere within the vapor container is continuously monitored both as a means of control on the plant facilities which use the stack and of providing a permanent record of the activity released from the plant. A beta-gamma sensitive scintillation detector is used for this purpose, and an alarm is provided which operates when the radiation level reaches a prohibitive value.

A gamma sensitive detector is mounted in the continuous steam generator blowdown to detect leakage of main coolant into the secondary system. The steam generator blowdown, rather than the steam generator, is chosen for this measurement because it is sufficiently removed from the main coolant loop to make sensitive measurements possible, and a concentration in the liquid phase is realized. An alarm is provided which sounds when the radiation level reaches a predetermined value.

A portable gamma sensitive detector is placed on the bundle of main coolant sampling lines from the main coolant loops and serves to monitor the level of each sample before drawing.



MONITORING AND ALARM SYSTEM

A portable gamma sensitive detector is located in the control room and keeps the operator aware of his own safety in the instance of an alarm from some other part of the plant.

A portable gamma sensitive detector is used for monitoring the level in the neighborhood of the spent fuel pit when personnel is present in that area.

A portable gamma sensitive detector is located in all areas occupied by personnel.

A portable gamma sensitive detector is used to monitor the waste storage tanks and enables the personnel to predict and confirm proper radiation levels for safe discharge of liquid effluent.

A gamma sensitive detector monitors the cooling water return from the heat exchangers of the main coolant purification system, the pumps and valves. An alarm is actuated when the radiation level exceeds a predetermined value.

A gamma sensitive detector is used for monitoring the main coolant charging system, to detect backup leakage into clean water pipes. An alarm is actuated if this condition occurs.

A gamma sensitive detector is used to monitor the sanitary, laboratory, retention basin and other discharges to prevent discharge of high activity material to the environment. Continuous recording of this activity is a valuable legal record. An alarm is provided which operates when the radiation level reaches a prohibitive value.

Two beta-gamma sensitive detectors are used to monitor the activity of stack discharges. Continuous samples of the monitored atmosphere are passed through a small shielded container housing G-M tube detectors. An alarm is actuated when the radiation level reaches a prohibitive value.

Four site monitoring stations are located at selected points to serve as monitors of the air leaving the plant site. One of these stations is located in the vicinity of the hilltop stack.

211 RADIATION SHIELDING

Function

Radiation shielding is designed to provide biological protection wherever a potential health hazard exists. Radiation emanates from the reactor, the main coolant system, and other auxiliary systems. The shield design is divided into five categories according to function: the neutron shield, the primary shield, the secondary shield, the fuel handling shield, and the auxiliary shields.

General Description

The neutron shield is an annular, water-filled, steel plate tank surrounding the reactor vessel in the radial direction. It is designed to prevent neutron activation of the plant components within the vapor container and to prevent overheating or dehydration of the primary concrete shield immediately surrounding it. The large volumes of water above and below the core in the reactor vessel provide the necessary neutron protection in the axial direction. The details are shown on drawing 9699-FM-1C.

The primary shield is a reinforced concrete structure immediately adjacent to the exterior of the neutron shield tank which, together with the tank, serves to attenuate radiation from the reactor to the level of the radiation emanating from the main coolant system. The bottom portion of the shield is an integral part of the main structural concrete support for the reactor vessel. The upper portion of the shield, which is approximately cylindrical in shape, extends from the main structural support to the working floor above the reactor. That section of the primary shield between the top of the reactor vessel and the working floor also serves as a shield during refueling operations. Removable block shielding is provided above the reactor vessel to maintain skyshine compatible with radiation design levels during full power operation.

The secondary shield, which surrounds the entire reactor plant within the vapor container, reduces reactor and main coolant radiation to design levels. The bottom portion of the shield is an integral part of the main structural concrete support for the reactor plant. The cylindrical side portion of the shield extends upward from the main support structure to support the working floor above the reactor and the main crane tracks near the top of the vapor container. The working floor above the main coolant system compartments also serves as shielding to attenuate skyshine during full power operation.

Within the secondary shield, additional shielding is provided between each main coolant loop to protect maintenance personnel during shutdown operations. Shielded penetrations are provided in the primary and secondary shields for piping and instrumentation, and to vent vapors resulting from flashing main coolant if a major rupture in a main coolant loop should occur.

The fuel handling shield facilitates the removal and transfer of spent fuel assemblies and control rods from the reactor vessel to the spent fuel pit. It is designed to attenuate radiation from spent fuel, control rods, and reactor vessel internals to a level consistent with design criteria. The compartment above the reactor vessel, the shield tank cavity, is flooded with borated water during refueling operations to provide a temporary radiation shield, and a medium for removing decay heat from a single spent fuel assembly during transfer and from residual activity of the reactor vessel internals during temporary storage. The spent fuel assemblies and control rods are remotely lowered out of the vapor container through the spent fuel chute into the spent fuel pit. Concrete and lead completely shield the spent fuel chute. Lead is used where space limitations prohibit the use of concrete. The shielding is designed to prevent radiation streaming during the period a fuel assembly is passing through the vapor container and main concrete support. Since the spent fuel pit is above grade, its concrete walls are designed to shield personnel from radiation emanating radially from the spent fuel assemblies and control rods during storage. Water in the spent fuel pit protects personnel who work above the pit during refueling operations. After storage for decay, the spent fuel assemblies and control rods are transferred under water to lead coffins for shipment to reprocessing plants.

Auxiliary shielding is designed to protect personnel in the plant laboratory and in the vicinity of the waste disposal, purification, and chemical shutdown systems. Low activity emitters in the waste disposal system are shielded by fenced exclusion areas. Auxiliary shielding is provided around the counting room to reduce background.

Additional concrete is placed around the control room to provide a shield for key operating personnel in the event of a major rupture of the main coolant system, resulting in the release of volatile fission products into the vapor container.

Basis for Design

The following radiation levels are used as design criteria for specifying shielding:

	mr per hr
Working stations during full power operation	0.75
Ground level directly beneath vapor container during full power operation	7.5
Intermittently manned ground level areas during full power operation	2.0
Fuel handling areas during refueling operations	6.0

212 CHEMICAL SHUTDOWN SYSTEM

Function

The functions of the chemical shutdown system are to inject a neutron absorbing chemical into the main coolant system at shutdown to complement the neutron absorbing control rods, and to remove 95 per cent of the chemical in about 11 hr during plant start-up. The system is designed for normal shutdown and accomplishes it in those cases where the time factor is not critical.

General Description

The chemical shutdown system consists of two ion exchangers, a mixing tank, transfer pump, and miscellaneous piping, valves, and fittings, as shown on drawing 646-J-426.

Basis for Design

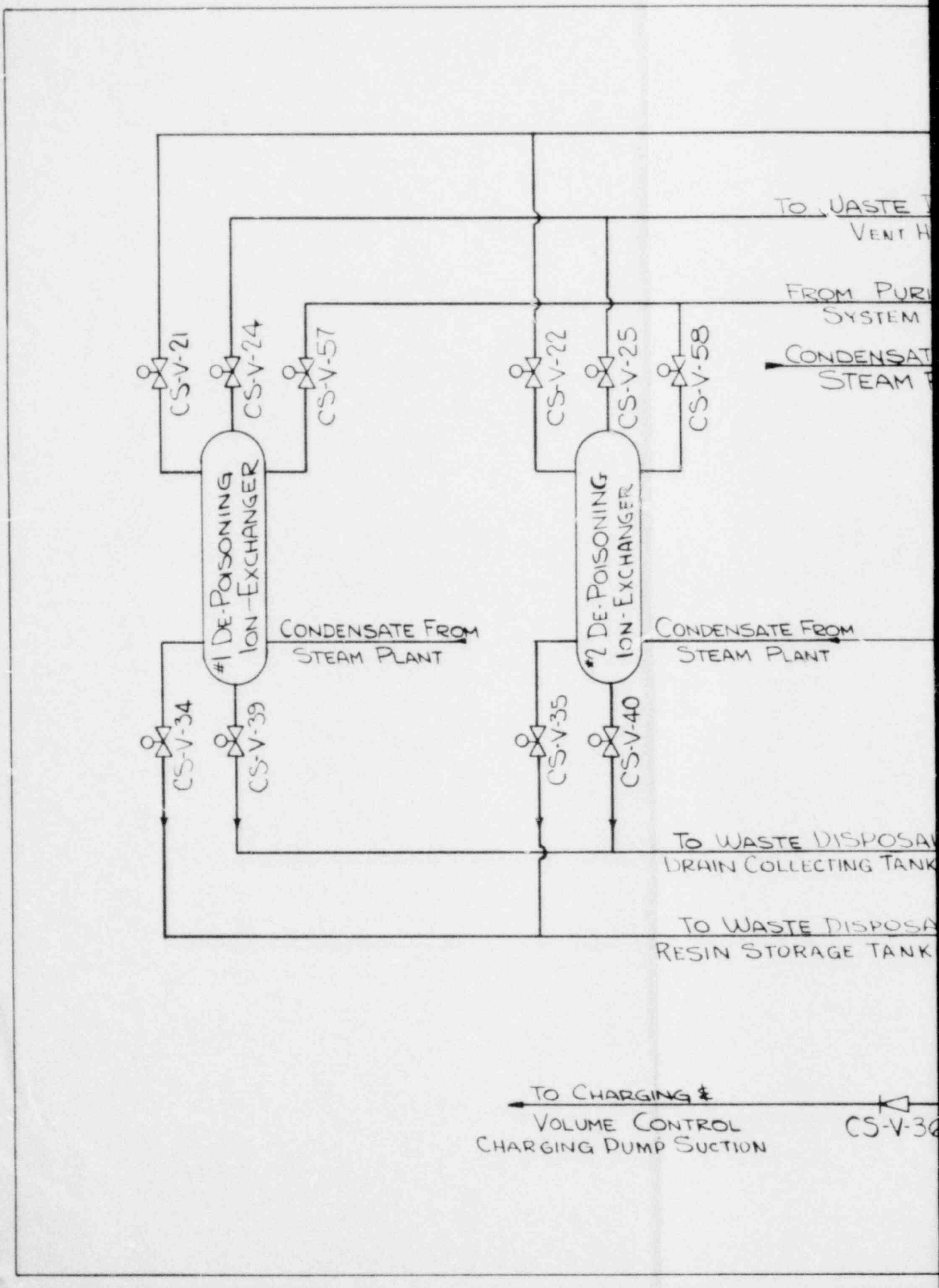
The system is sized for adding approximately 1.6 g of boron per liter of reactor coolant. This neutron absorbing chemical is added in the form of boric acid in a 15 wt % premixed solution and is injected while the reactor plant is still at operating temperature and pressure. When one loop is being drained, the chemical neutron absorbing solution is pumped into and distributed throughout the primary plant, eliminating the possibility of later changing the boron concentration.

Boric acid does not increase the system corrosion rate. Tests over long periods have shown the boric acid corrosion rate of Type 304 stainless steel to be negligible. Also, the boric acid is in contact with the stainless steel for only a short time.

The boric acid is added to the main coolant system in 30 min, corresponding to a pump flow rate of 100 gpm.

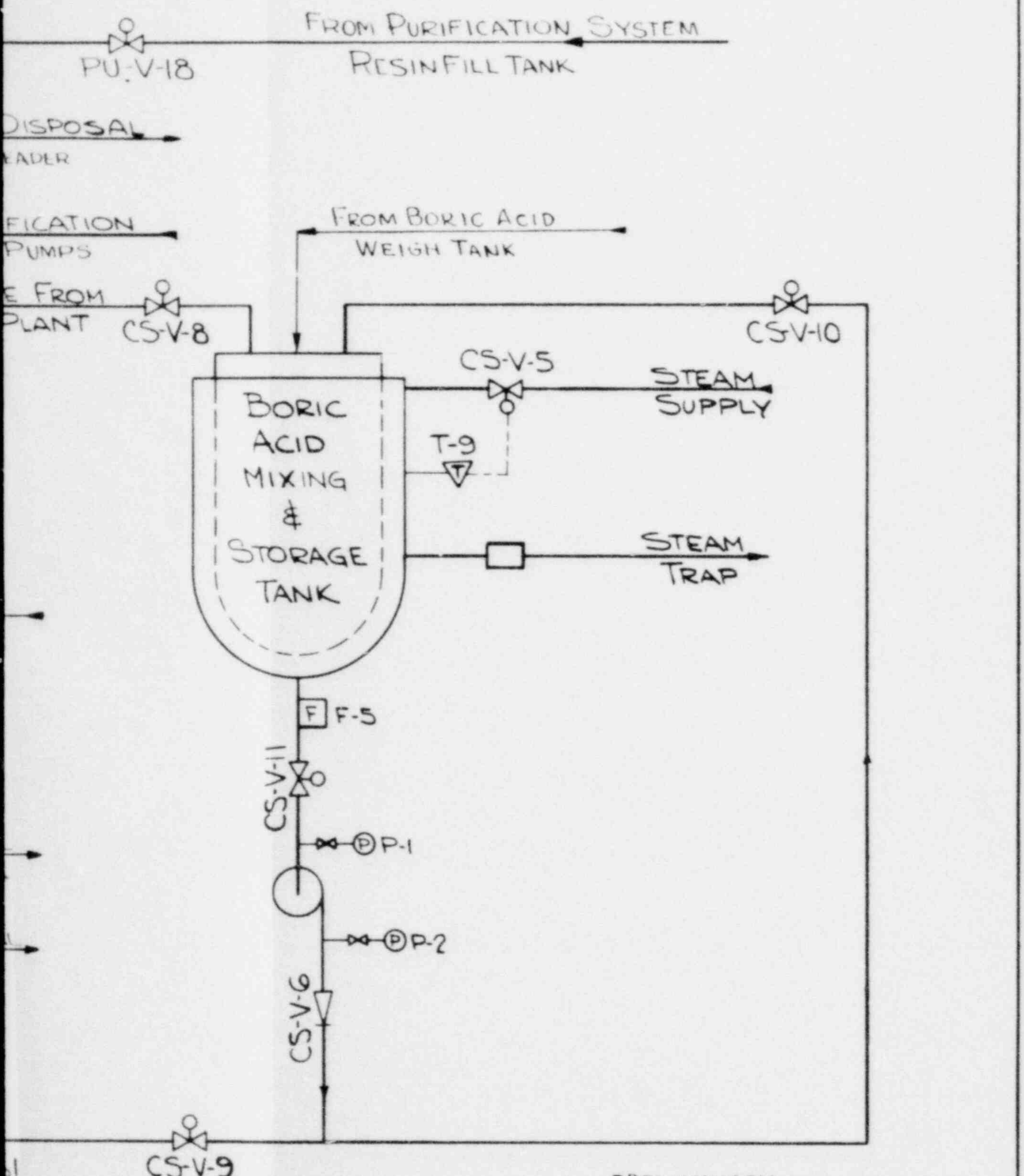
During the start-up operation, the boric acid solution is removed from the main coolant system. In the initial phase, the boric acid concentration is reduced by dilution and recirculation so that after 11.2 hr, only 5 per cent of the boric acid remains in the main coolant. The remaining boric acid is removed by ion exchange so that, after 22 hr, the main coolant system has no appreciable amount of boric acid remaining in solution.

The ion exchange resin temperature is limited to a maximum of 140 F. Remote means for replacing and disposing of the exhausted resin are provided.



TO WASTE
VENT H
FROM PURI
SYSTEM
CONDENSAT
STEAM P
CS-V-21
CS-V-24
CS-V-57
CS-V-22
CS-V-25
CS-V-58
#1 DE-POISONING ION-EXCHANGER
#2 DE-POISONING ION-EXCHANGER
CONDENSATE FROM STEAM PLANT
CONDENSATE FROM STEAM PLANT
CS-V-34
CS-V-39
CS-V-35
CS-V-40
TO WASTE DISPOSA
DRAIN COLLECTING TANK
TO WASTE DISPOSA
RESIN STORAGE TANK
TO CHARGING & VOLUME CONTROL
CHARGING PUMP SUCTION
CS-V-3

150-151



PRELIMINARY

WESTINGHOUSE ELECTRIC CORPORATION

TITLE YANKEE ATOMIC ELECTRIC COMPANY

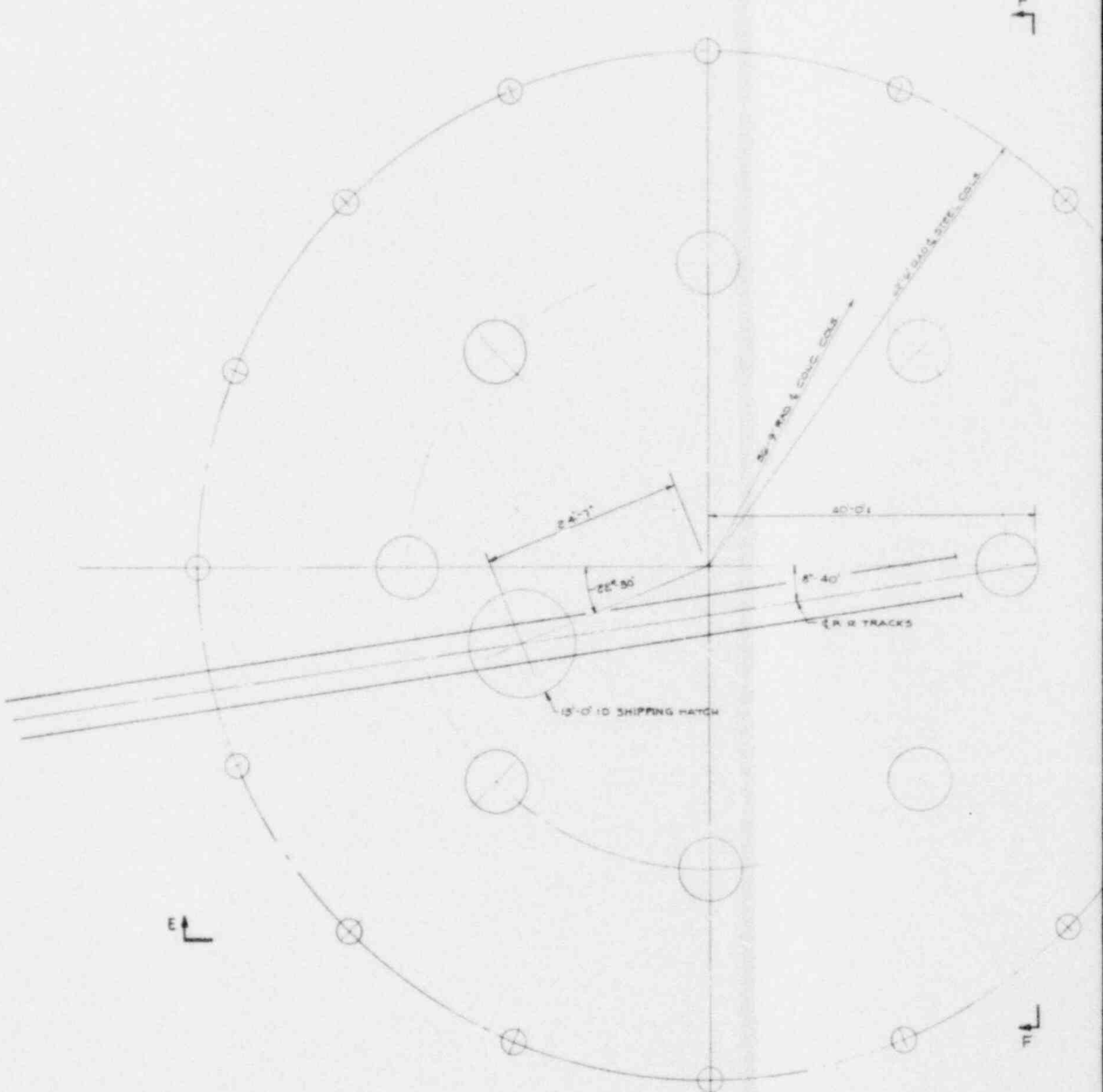
CHEMICAL SHUTDOWN SYSTEM

DR. [illegible] | [illegible]/57 | APPD. T. G. Stevens | 1/4/57

DIV. C.A.P.A. PITTSBURGH, PA. 646J426

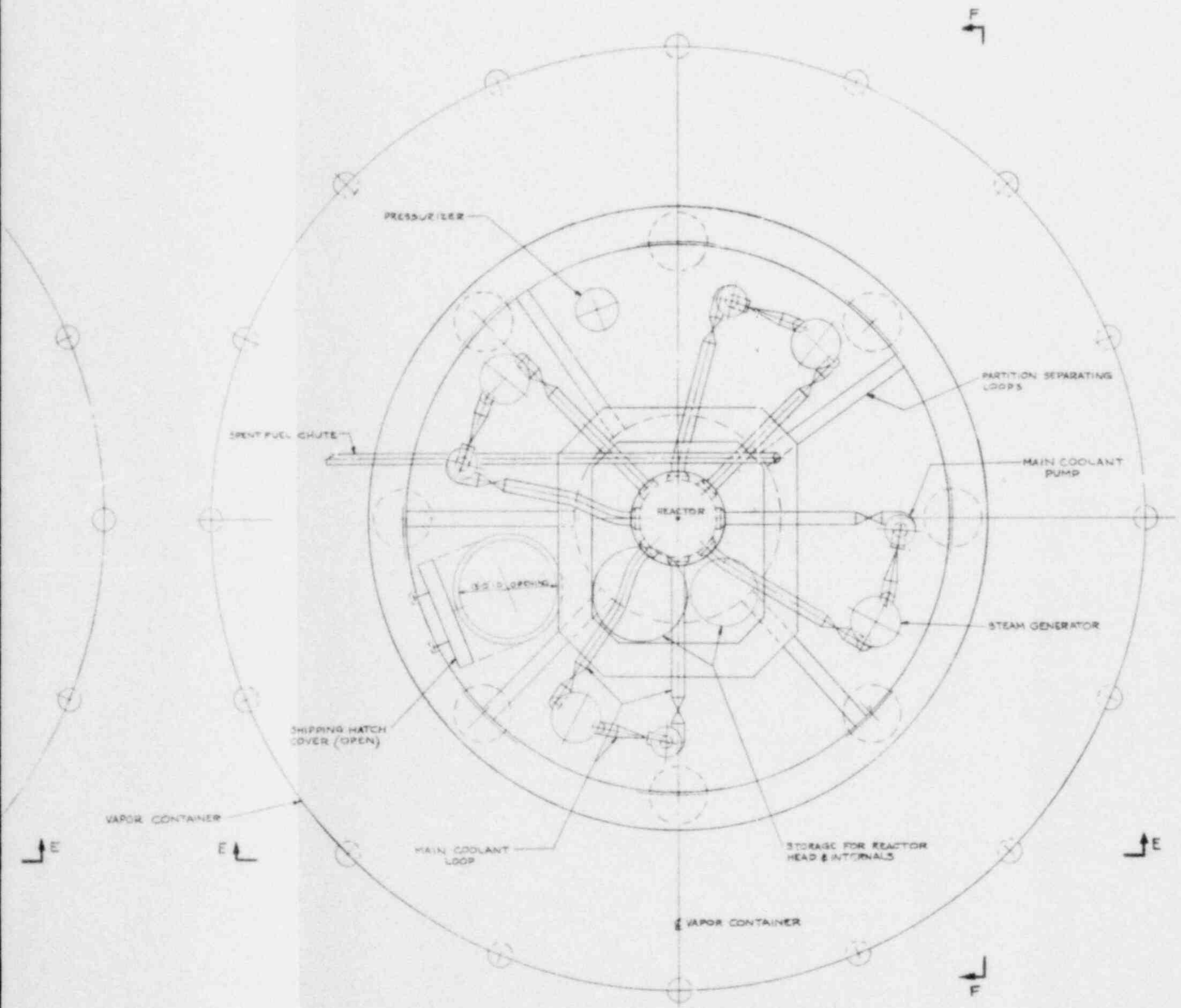
The boric acid mixing tank is kept at 150 F to attain a 15 per cent boric acid solution.

Isolation and operation of the radioactive portion of the system is performed by means of manually operated valves, with reach rods through the shielding depending upon access provided, or other dependable valve operators.



PLAN "A-A"

154-155



PLAN 'B-B'

NOTES
FOR LOCATION OF PLANS SEE DWG 9699-FM-1C

ISSUE	DESCRIPTION		CHKD.	INSP.	CORR.	APPR.
5						
4						
3						
2						
1	ORIGINAL ISSUE	6-12-??				

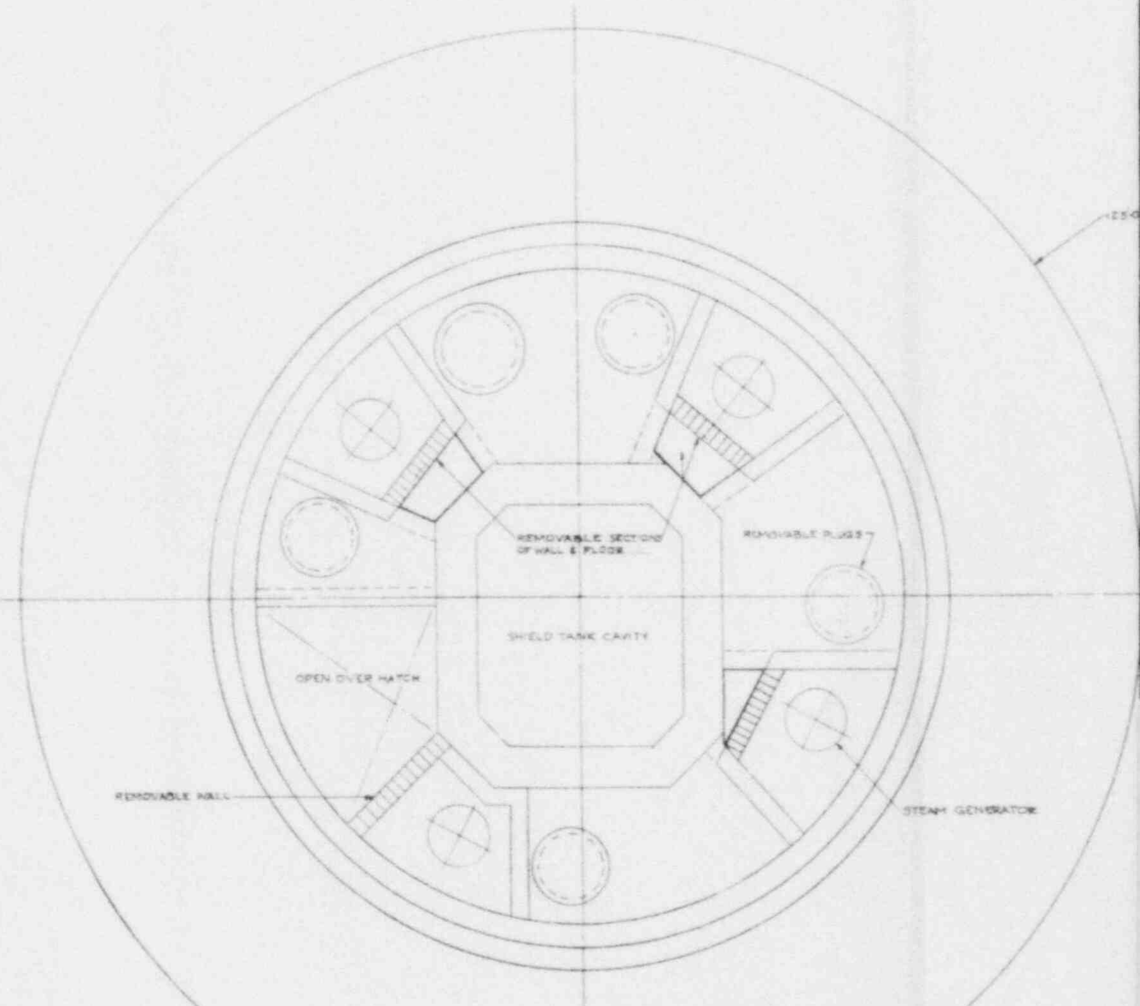
ARRANGEMENT VAPOR CONTAINER SH-1

YANKEE ATOMIC ELECTRIC COMPANY

STONE & WEBSTER ENGINEERING CORPORATION

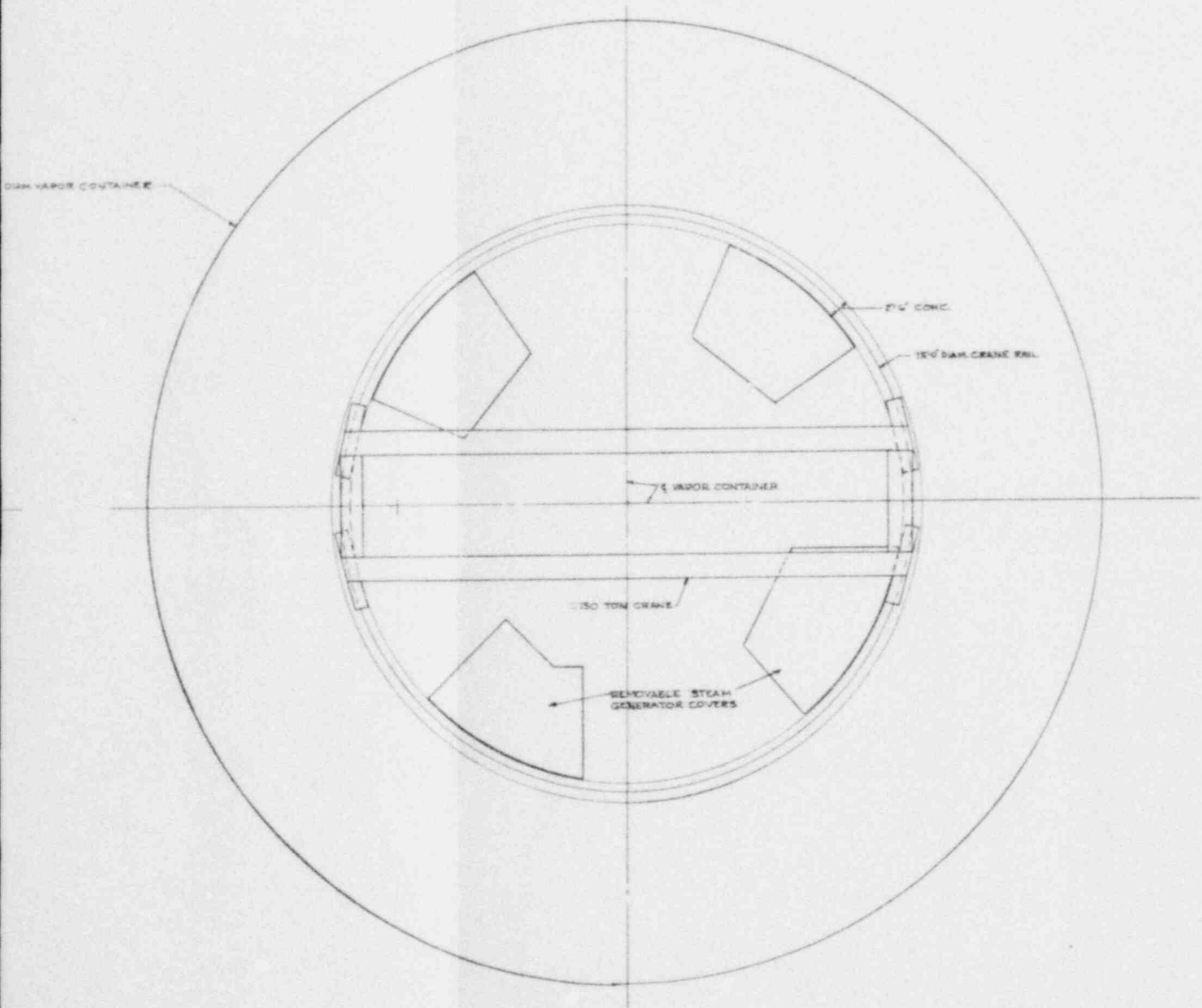
SCALE-FEET 0 5 10 15

9699-FM-1A


REMOVABLE SECTIONS OF WALL & FLOOR
REMOVABLE PLUGS
SHIELD TANK CAVITY
OPEN OVER HATCH
REMOVABLE WALL
STEAM GENERATOR

PLAN "C-C"

156-157



PLAN "D-D"

NOTE:
FOR LOCATION OF PLANS SEE DWG. 9699-FM-1C.

ISSUE	DESCRIPTION	CHKD.	INSP.	COORD.	APPR.
5					
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3					
2					
1	ORIGINAL ISSUE 6-18-57				

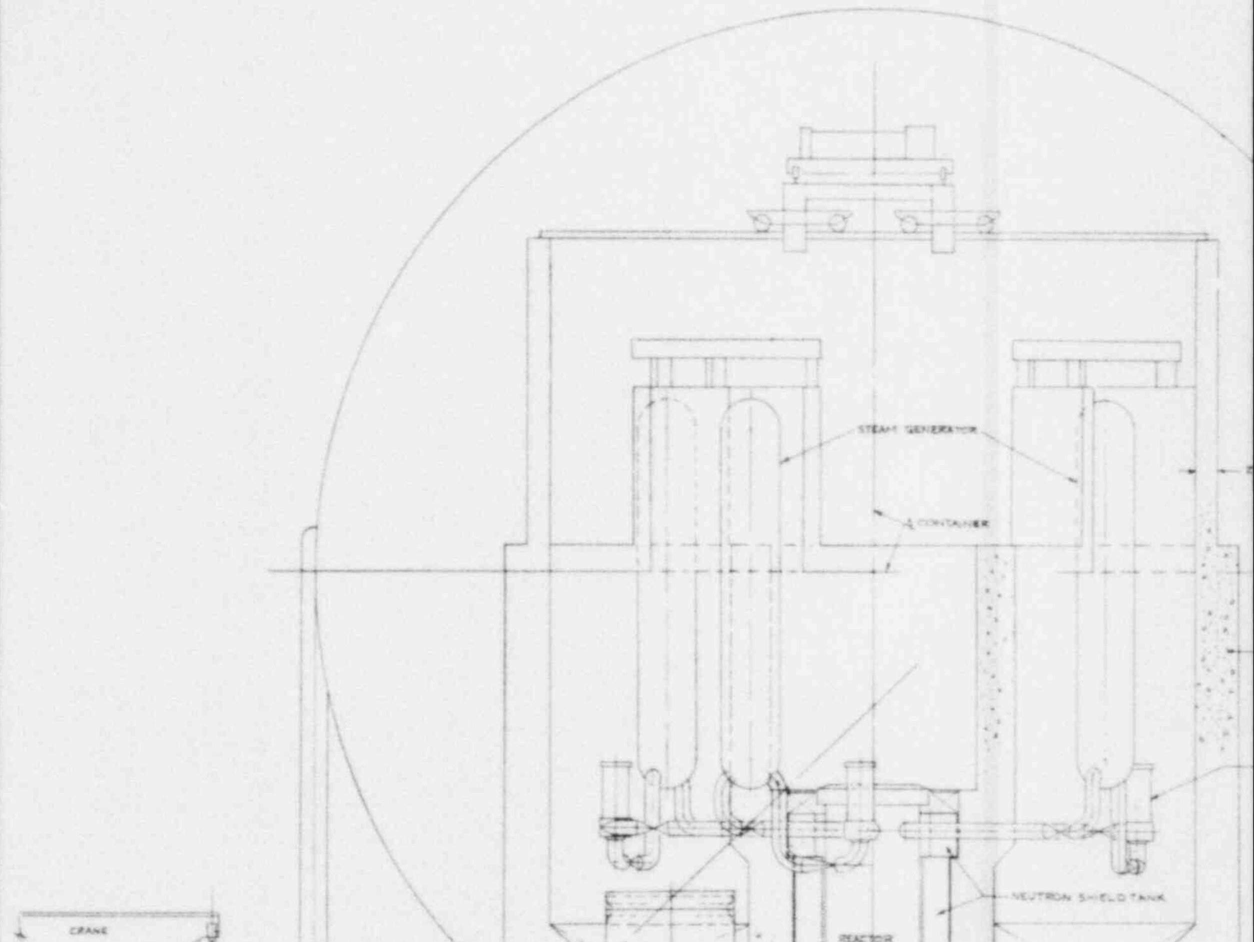
SCALE - FEET: 0 5 10 15

ARRANGEMENT VAPOR CONTAINER SH 2

YANKEE ATOMIC ELECTRIC COMPANY

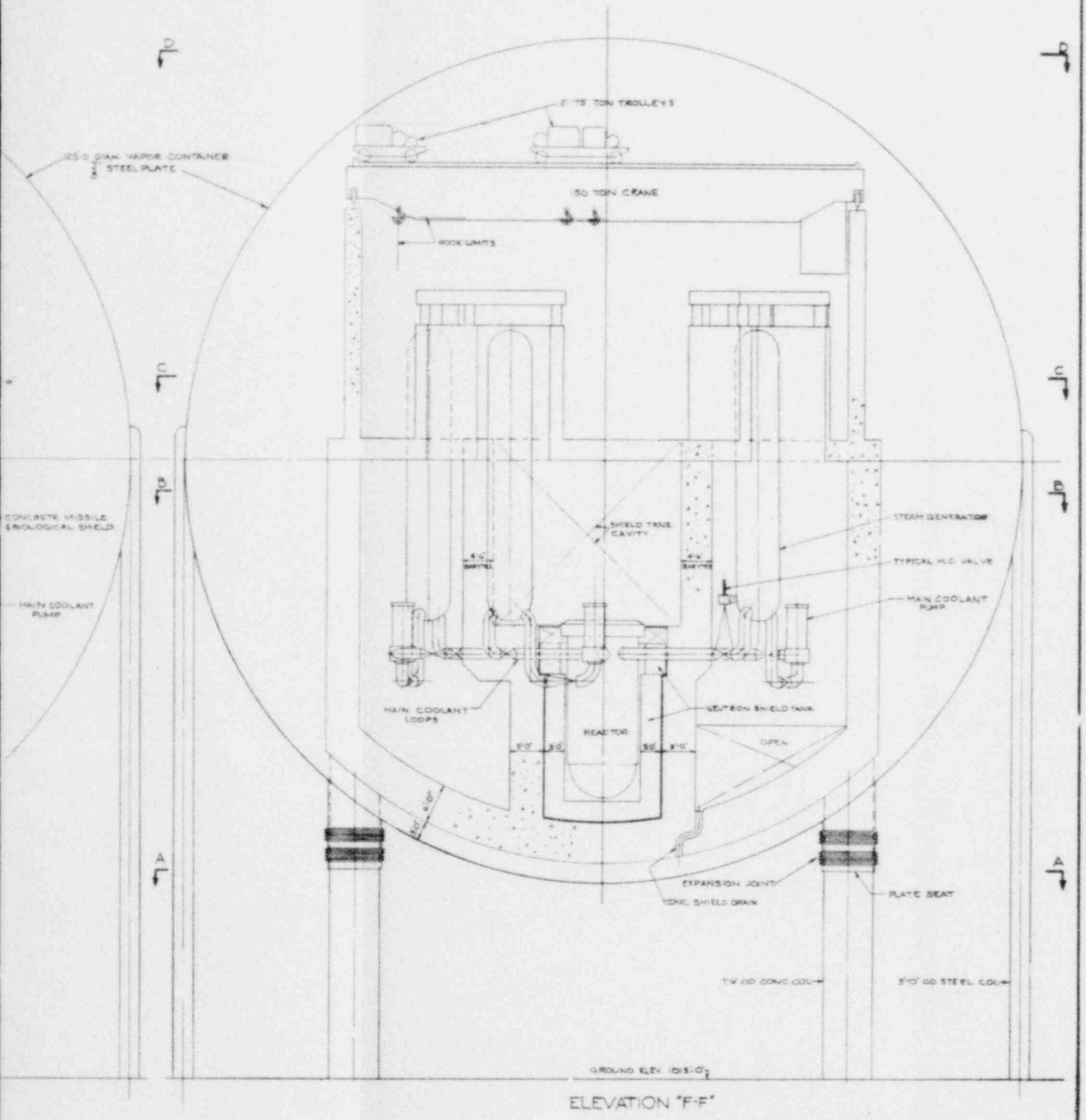
STONE & WEBSTER ENGINEERING CORPORATION

9699-FM-1B



ELEVATION "E-E"

158-159



ELEVATION "F-F"

NOTES

SHIELD WALL THICKNESSES NOT FINAL.
FOR PLAN VIEWS SEE 9699-FM-1A & FM-1B.
FOR DETAILS OF SPENT FUEL CHUTE SEE DWG 9699-FM-19A

REVIS	DESCRIPTION	CHKD	INSP	CORR	APPR
5					
4					
3					
2					
1	ORIGINAL ISSUE 2-12-57				

ARRANGEMENT VAPOR CONTAINER SH 3

YANKEE ATOMIC ELECTRIC COMPANY

STONE & WEBSTER ENGINEERING CORPORATION

0 5 10 15
SCALE-FEET

9699-FM-1C

213 VAPOR CONTAINMENT

Function

The vapor container is a steel envelope which surrounds the main coolant equipment loops and encloses all pressurized parts of the main coolant system. It prevents the release of radioactivity to the atmosphere in the unlikely event of an accident resulting from a rupture and release of fluid from the main coolant system within the containment vessel.

When the reactor is critical or when the main coolant system is pressurized with nuclear fuel in place, the vapor container is closed and pressure-tight. All access openings, vent connections, pipe lines not required for operation, and the spent fuel chute are kept closed with tight shutoff valves or gasketed doors.

The vapor container, when closed, is maintained at a pressure level slightly higher than atmospheric for continuous leakage indication, with allowance made for variations due to temperature change.

Associated with the outer steel vapor container is an inner reinforced concrete structure which supports the main coolant loop equipment, attenuates radiation from the main coolant loop to a tolerable level outside the vapor container, and acts as a stop for objects possessing kinetic energy. This concrete structure is not designed to contain pressure.

General Description

The layout of the vapor container is shown on drawings 9699-FM-1A, 1B and 1C.

The vapor container is a steel spherical shell, 125 ft in diameter and with a minimum wall thickness of 7/8 in. The spherical shape is selected since it uses a minimum of material for a given volume and internal pressure. The spherical shape permits the most accurate determination of secondary stress and facilitates the design of the necessary penetrations.

The plate material is ASTM Specification A-300, Class A-201 Grade B, firebox quality, a carbon-silicon steel of suitable quality for forming and welding in pressure vessel service. The tensile strength is 60,000-72,000 psi with a minimum yield point of 32,000 psi. The atmospheric temperature outside the uninsulated sphere occasionally approaches -25 F, so that the shell metal temperatures may be close to the freezing point during operation. Specification A-300 material is employed for its superior impact value at low temperature, equivalent to 15 ft-lb at -50F.

The vapor container is designed, built and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII (Unfired Pressure Vessels), and the code stamp is applied. The vapor container is not provided with a relief valve, in accordance with special ruling, Case No. 1235, which states:

> "It is the opinion of the Committee that, since it is intended that these vessels be designed and built to safely contain all the lethal radioactive substances that may be released in case of a maximum credible accident affecting the reactor vessel or primary coolant circuit or both, and because of the hazardous character of the materials, which might be released, pressure relief devices are not required."

The stress permitted by the Code in the specified plate is 15,000 psi. The Code further specifies that the design stress shall be reduced by a factor of 0.9 when employing welded seams with 100 per cent radiographic inspection. The resulting design stress is 13,500 psi.

The design pressure of the vapor container is 31.5 psi gage, corresponding to a membrane stress of 13,500 psi in a 125 ft diam sphere with a minimum plate thickness of 7/8 in.

The internal pressure of the vapor container in the event of a major loss of water accident is 34.5 psi gage. This pressure includes the 10 per cent overpressure permitted by the Code under paragraph UG-125(c), which states "All unfired pressure vessels other than unfired steam boilers shall be protected by pressure relieving devices that will prevent the pressure from rising more than 10 per cent above the maximum allowable working pressure, except when the excess pressure is caused by exposure to fire or other unexpected source of heat." A 10 per cent increase in the design pressure of 31.5 psi gage results in an allowable pressure of 34.5 psi gage which corresponds to the internal pressure developed in the major loss of water accident.

The spherical vessel is supported on steel columns.

The pressurized equipment within the vapor container is surrounded by a reinforced concrete cylinder, the bottom of which is a segment of a sphere. Concrete wall thickness is 4.5 to 7 ft. Ordinary concrete is employed having a density of 150 lb per cu ft, except in several areas in which space restrictions require high density concrete.

The concrete structure is supported on eight reinforced concrete piers which penetrate the spherical container. These penetrations are sealed with stainless steel expansion joints. The joints are welded to a steel plate which passes completely through each concrete pier below the expansion joint and which is also welded to the interior reinforcing rods, thus completing the metallic vapor seal of the container vessel. The support construction permits the steel and concrete structures to move freely and independently of each other, thereby eliminating temperature stresses resulting from restraint.

Pipe lines, not required for normal operation, which enter the vapor container, are provided with valves located outside the vessel wall and maintained in a closed position in order to maintain the integrity of the vapor container. Pipe lines, required for normal operation, which enter the containment vessel are each provided with two check valves, one inside and one outside the shell. Operating outgoing lines are each provided with a closure trip valve arranged to close automatically on pressure rise in the container.

Details of Vapor Container

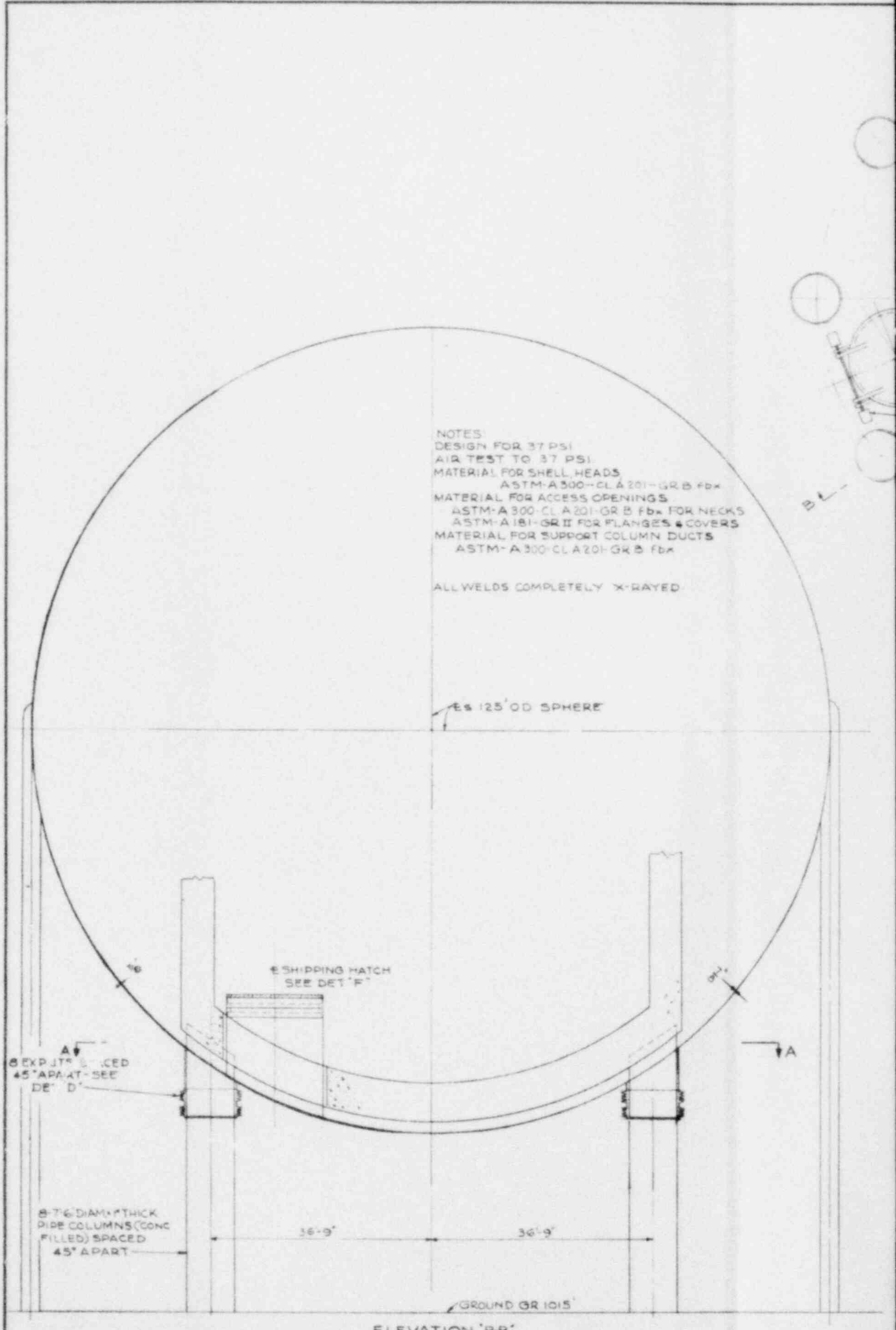
Typical details of the vapor container are shown on drawings 9699-FM-11A and 12A.

All penetrations of the sphere are reinforced to the full strength value of the metal removed. All shell seams are completely radiographed, as well as all welds in the penetrations wherever possible. All welds not amenable to radiographic examination are subjected to a magnetic particle inspection at every pass.

All high temperature piping entering or leaving the spherical shell is isolated from the shell by means of a convoluted expansion joint encased in a steel protective sleeve. These expansion joints eliminate the necessity of heavily reinforcing the spherical shell to contain the forces and moments resulting from pipe expansions.

Conduit fittings are welded in groups into the heads of special blisters which, in turn, are welded to the spherical shell. This design facilitates construction, testing and any required corrective reworking. The conductor is generally mineral insulated copper sheathed cable which is seal brazed to the conduit to ensure leak tightness.

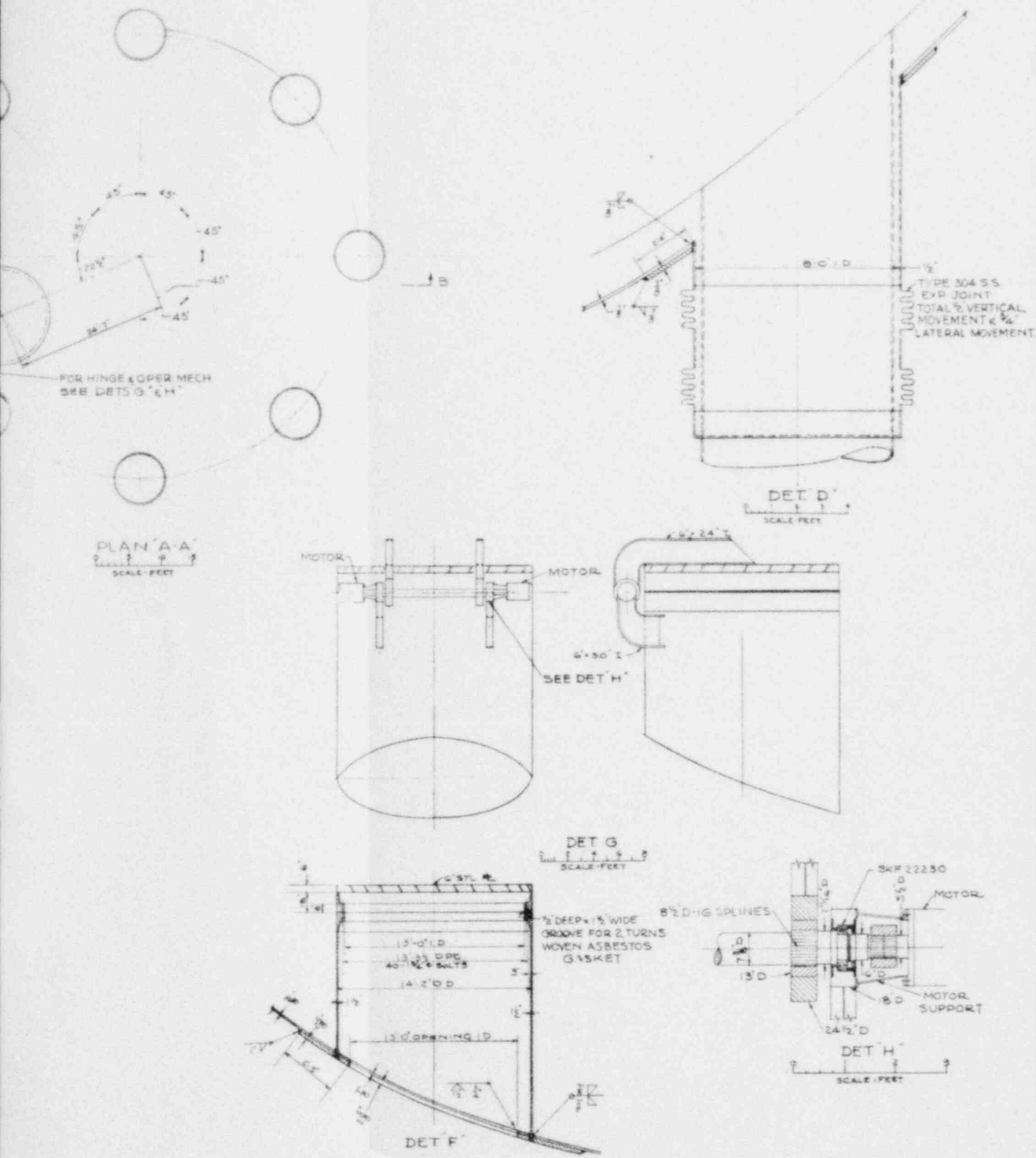
The internal concrete structure consists of two concentric cylinders of 3,000 psi compressive strength reinforced concrete. These cylinders are tied together with five reinforced concrete radial walls so located as to provide an isolation compartment for each main coolant loop and for an access way into the structure. The wall of the outer cylinder and the radial walls are perforated with ports sized to limit the differential pressure across the concrete walls to a value of 6 psi at the time of a major loss of water accident.



ELEVATION 'BB'

SCALE-FEET

162-163



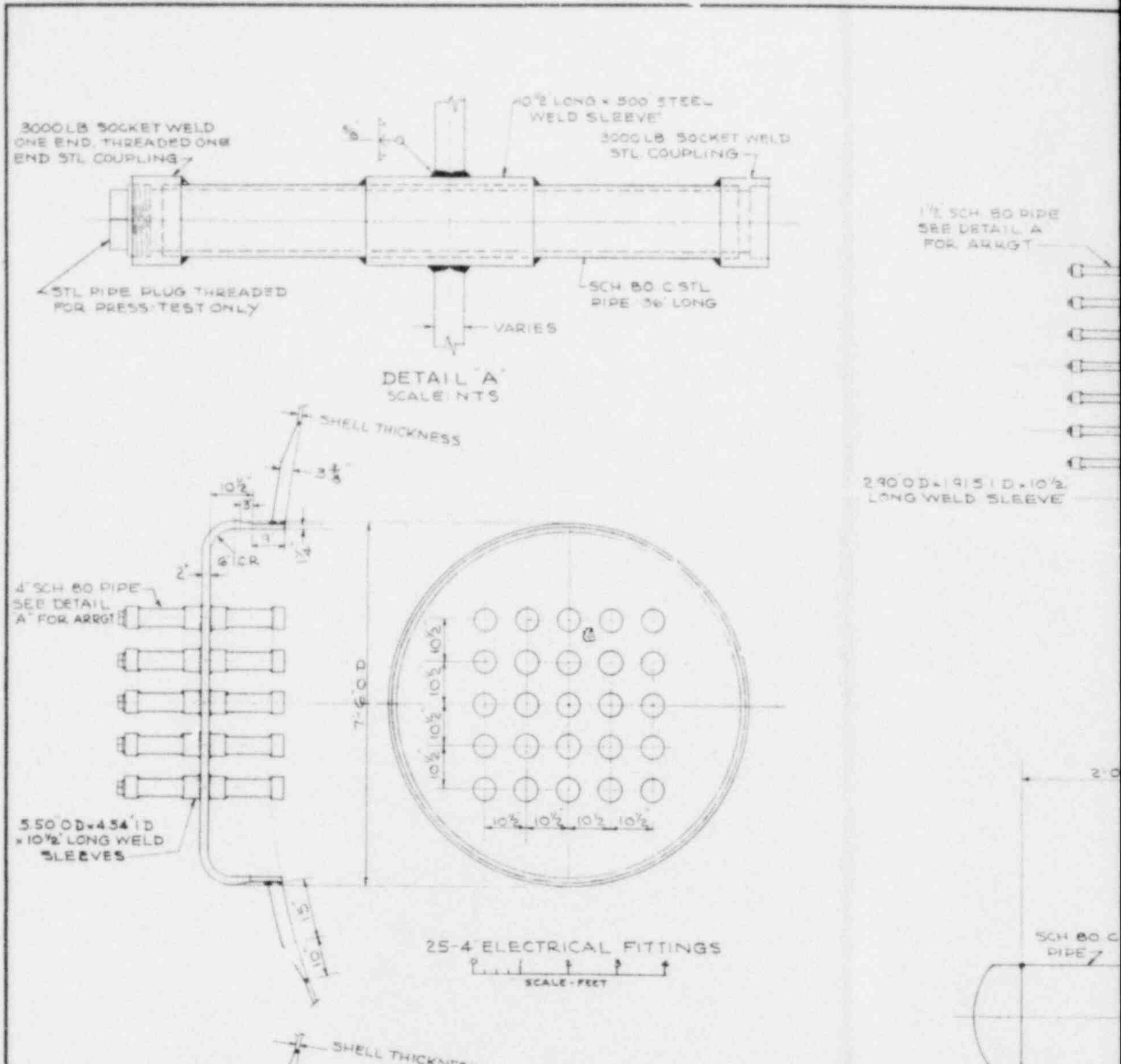
	ORIGINAL ISSUE 2-18-57				
MARK	DESCRIPTION	CHKD	INSP	CORR	APPR

VAPOR CONTAINER
SPHERICAL ARRANGEMENT

YANKEE ATOMIC ELECTRIC COMPANY

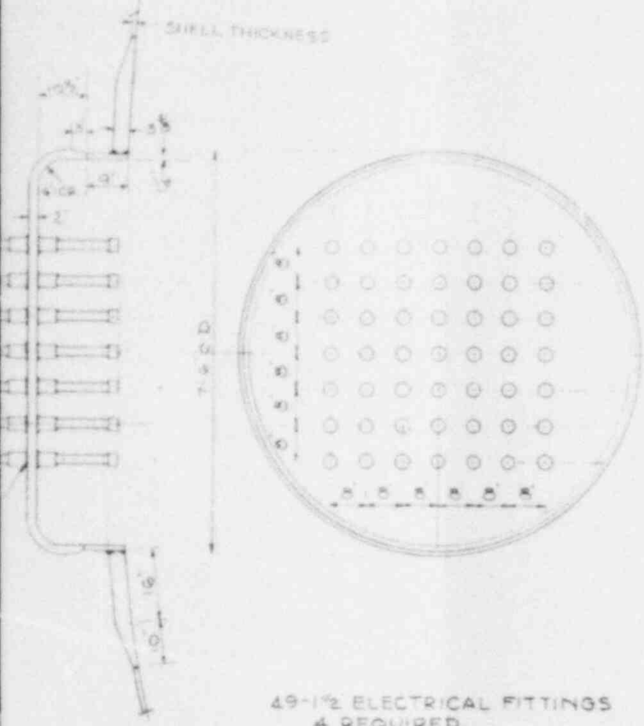
STONE & WEBSTER ENGINEERING CORPORATION

9699-FM-11 A

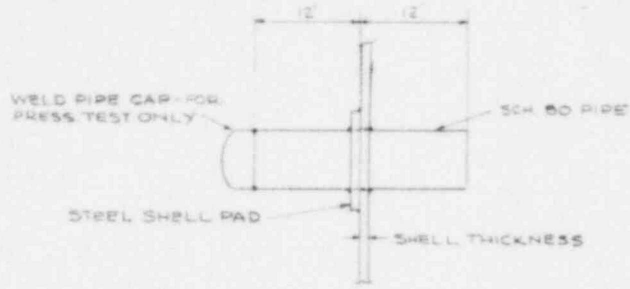


3000 LB SOCKET WELD ONE END, THREADED ONE END STL COUPLING
10½" LONG x .500 STEEL WELD SLEEVE
3000 LB SOCKET WELD STL COUPLING
STL PIPE PLUG THREADED FOR PRESS. TEST ONLY
SCH 80 C.STL PIPE 36" LONG
VARIES
DETAIL "A"
SCALE: NTS
SHELL THICKNESS
4" SCH 80 PIPE SEE DETAIL "A" FOR ARRGT
5.50 OD x 4.54 ID x 10½" LONG WELD SLEEVES
7'-6" OD
25-4" ELECTRICAL FITTINGS
SCALE - FEET
SHELL THICKNESS
2½" SCH 80 PIPE SEE DETAIL "A" FOR ARRGT
3.875 OD x 2.906 ID x 10½" LONG WELD SLEEVES
6'-6" OD
25-2½" ELECTRICAL FITTINGS
SCALE - FEET
1½" SCH 80 PIPE SEE DETAIL "A" FOR ARRGT
2.900 D x 1.915 I.D. x 10½" LONG WELD SLEEVE
SCH 80 C PIPE
INSULA
WELD PIPE C PRESS TEST
EXPA
PIPING SYST
MAIN STEA
FEED WAT
S.V. BLOWDOW TANK COOLI
BOILER SECOND BLOW DOW
SPENT FUEL CHU

164-165

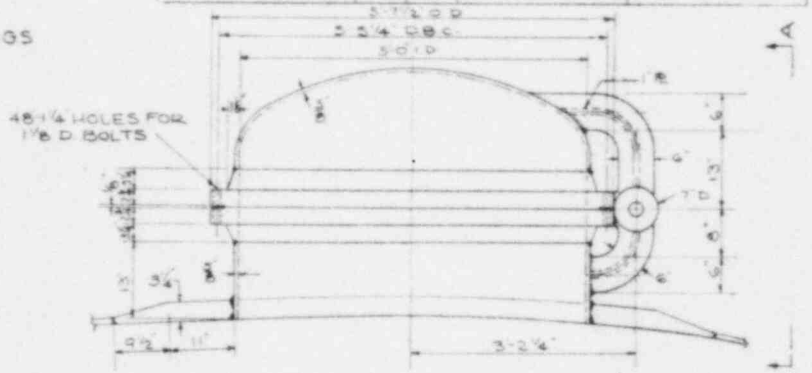


49-1½ ELECTRICAL FITTINGS
4 REQUIRED
SCALE - FEET

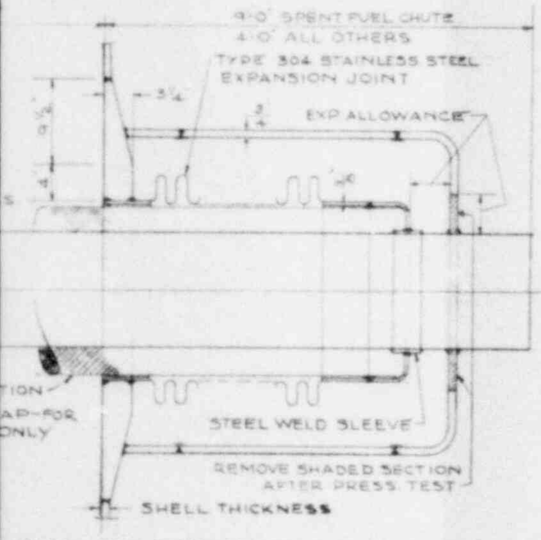


PIPE PENETRATIONS THRU SHELL
SCALE: N.T.S.

PIPING SYSTEM	SIZE	AMT	SCHED & TYPE	SHELL PAD DATA
P.C. MAKE UP	2½"	1	SCH 80 STNLS STL	7" O.D. × 2.94" I.D. × ⅜" THK
P.C. BLOWDOWN	2½"	1	SCH 80 STNLS STL	7" O.D. × 2.94" I.D. × ⅜" THK
P.C. COMP. COOLING INLET	4"	8	SCH 80 CARB STL	9" O.D. × 4.56" I.D. × ⅜" THK
P.C. COMP. COOLING DISCH	10"	1	SCH 80 CARB STL	16" O.D. × 10.88" I.D. × ⅜" THK
SHIELD TK. CAVITY	6"	1	SCH 80 CARB STL	11" O.D. × 6.72" I.D. × ⅜" THK
V.C. COOLER	4"	2	SCH 80 CARB STL	9" O.D. × 4.56" I.D. × ⅜" THK
V.C. COOLING SPRAYS	3"	1	SCH 80 CARB STL	7½" O.D. × 3.56" I.D. × ⅜" THK
V.C. DRAIN	6"	1	SCH 80 CARB STL	11" O.D. × 6.72" I.D. × ⅜" THK
V.C. VENTILATION	18"	2	.500 THK CARB STL	25" O.D. × 18.19" I.D. × ⅜" THK

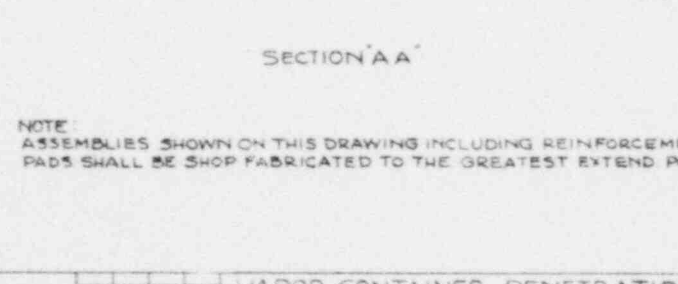


DETAIL OF MANHOLE - 1 REQ'D
SCALE-FEET



...NSION JOINT CONNECTIONS THRU SHELL
SCALE: N.T.S.

...EM	SIZE	AMT	SCHED & TYPE	EXP. JT SIZE	INSUL THK
...M	12"	4	SCH 80 C.S. PIPE	20"	3"
...ER	6"	4	SCH 80 C.S. PIPE	10"	1½"
...N ...NG	8"	2	SCH 80 C.S. PIPE	12"	1⅛"
...ARY ...N	2"	4	SCH 80 C.S. PIPE	10"	3"
...TE	14"	1	15" I.D. S.S. PIPE	90"	15" LEAD



SECTION "A-A"

NOTE:
ASSEMBLIES SHOWN ON THIS DRAWING INCLUDING REINFORCEMENT PADS SHALL BE SHOP FABRICATED TO THE GREATEST EXTEND POSSIBLE.

ISSUE	DESCRIPTION	CHKD.	INSP.	CORR.	APPR.
1	ORIGINAL ISSUE 12-6-56				

VAPOR CONTAINER - PENETRATIONS

YANKEE ATOMIC ELECTRIC COMPANY

STONE & WEBSTER ENGINEERING CORPORATION

9699-FM-12A

The inner concrete wall serves as the support for the reactor vessel, the water-filled neutron shield tank surrounding the reactor vessel and as a shield tank cavity above the vessel. The shield tank cavity, which is water-filled when handling fuel, is lined with a stainless steel membrane to assure complete watertightness.

When not otherwise metal covered, the surface of the concrete is protected with a smooth, hard finish plastic paint to prevent absorption of contaminated vapor and to assist in decontamination.

Vapor Container Tests

After the vapor container has been erected and all welding, radiographing and Magnafluxing have been completed, including manhole closures and shell penetrations, the vapor container is completely closed and subjected to field acceptance tests, including an air pressure test, leakage detection test, and a leakage rate test.

Air Pressure Test

The vapor container is pressurized with air to one and one-quarter times the design pressure, or 40 psi gage. This controlled pressure is held for a period of 6 hr. If leakage is detected by a bubble test, the vessel is depressurized, the leak repaired and the vessel retested. The air pressure test establishes the design integrity of the complete vapor container, including all penetrations and closures.

Leakage Detection Test

The purpose of the leakage detection test is to establish the leak tightness of all welded joints used in the erection of the vessel and gasketed closures required in the design, and to detect individual leaks from the vapor container in the order of .0001 cu ft per hr of air at a test pressure of 15 psi gage.

A leakage detection test by tracer gas is considered to be the most suitable, sensitive means of ensuring maximum vapor container integrity, and particularly for leakage around vapor container penetrations.

The leakage detection test is conducted with a halogen type leak detector equal to the General Electric Company Type H-1. This is a sensitive instrument capable of detecting leakage rates as low as .0001 cu ft per hr when the vapor container contains 1 per cent by volume of the tracer gas Freon-12. The vapor container is pressurized with air at 15 psi gage during testing, and Freon-12 is introduced into the container. All welded seams, penetration welded joints and closures of the vapor container are hand probed with the leak detector. Any leak detected is repaired and the area again retested. At the completion of this test, the vapor container is ready for the final leakage rate test.

Leakage Rate Test

Final evaluation of the vapor container is based on a leakage rate test. The vapor container is pressurized with air to 15 psi gage, and the temperature and pressure changes are recorded over a period of several days. The test pressure corresponds to the average pressure anticipated within the vapor container during a 24 hr period following a major rupture of the main coolant loop. When the average air temperature within the vapor container coincides, or nearly coincides, with the initial temperature conditions, the pressure change is recorded. If the leakage rate should be less than 0.1 wt per cent of the contained air during any 24 hr interval, corresponding to 70 cu ft per hr (STP), the vapor container is considered to be essentially leak-tight.

A leakage of this amount corresponds to a pressure decrease of 0.8 in. of water in 24 hr or a temperature decrease of 0.5 F during the same period. The magnitude of these measured quantities and the possible inability to measure the true average temperature of the contained air affect the accuracy of the leak rate demonstration.

Continuous Leakage Indication During Operation

In order to evaluate quantitatively the leakage rate from the vapor container during operation and to guard against the chance for gross leakage through improper closure after opening the container, the vapor container is continuously monitored. A proposed system provides that, before the reactor plant is made critical, the vapor container is closed and pressurized to about 1 psi gage by the station compressed air system. Thereafter, this pressure is controlled by a compressed air bottle system connected to the vapor container through a pressure reducing valve. The weight loss of air from the compressed air bottles is determined over an extended period and is a measure of the leakage from the vapor container. Effects of pressure and temperature fluctuations from changing atmospheric conditions balance out during long time intervals.

214 VAPOR CONTAINER VENTILATION SYSTEM

Heat Removal System

A system is provided for removing heat which is transferred from equipment and from the atmosphere to the air within the vapor container. The system consists of multiple units with air filters, cooling coils and motor driven fans, using Sherman Pond water in the cooling coils. Filters are designed for removal of approximately 50 per cent of particulate matter to minimize dust accumulations on cooling coils and surfaces exposed to container air. The design temperatures for the air within the vapor container are 100 F maximum in the summer and 50 F in the winter. During summer operation, Sherman Pond water at a maximum temperature of approximately 65 F is circulated through the air cooling coils and is discharged to Sherman Pond.

During normal operation, the vapor container is not accessible and no outdoor air is introduced into the vapor container for ventilation.

Monitoring devices are provided on the waste water and condensate lines to detect radioactivity in the event of pipe line leakage.

Heating System

Separate blower type unit heaters with steam coils are provided for heating the interior of the container during normal operation with one steam generator inoperative during cold weather and during maintenance periods. During plant shutdown, steam is taken from auxiliary steam boilers.

Purging Ventilation

A system is provided to exhaust air from the interior of the vapor container prior to, and during, maintenance operations. This system has a nominal capacity equivalent to one air change per hour. Following complete plant shutdown, air is introduced from outdoors into the interior of the vapor container and is exhausted through externally located air filters suitable for removal of particulate radioactive matter and through fans to the plant stack. Provisions are made for dilution of the exhaust air with outdoor air prior to discharge to the stack in order to control the concentration of radioactivity from nonfilterable fission products.

Facilities are provided to heat, by means of steam coils, the purging air which is introduced to the vapor container in cold weather.

The radioactive air filters consist of pre- and afterfilters, with afterfilters designed to remove more than 99 per cent of all particles larger than 3 microns.

Plant Stack

A plant ventilating stack is required on top of, or adjacent to, the main building. The stack is high enough to cause the emitted gases and vapors to clear the buildings. The stack discharges gases and vapors from the systems that are potential sources of radioactive contamination but are normally below measurable amounts. The discharge from this stack is monitored with an alarm. The operators can control the discharge to the stack by valves in the ventilation ducting. This operation is based upon meteorological conditions, and plant operations, and is monitored by an alarm system. In addition to the vapor container, steam jet air ejectors, exhaust hoods of the plant laboratory, and the incinerator stack discharge to the plant stack.

Duct connections to the purging system are provided with normally closed, manually operated, tight, rubber seated butterfly valves, open only during purging operation.

215 FUEL HANDLING SYSTEM

Function

In order that the reactor may be fueled and refueled, as required, without hazard to personnel, means are provided for underwater removal of fuel assemblies from the reactor, for transferring the assemblies from within the vapor container to a water filled storage pit located outside of the vapor container, for storing spent fuel assemblies under water for a sufficient period of time to allow them to decay to a tolerable level, for inserting the spent fuel assemblies into lead shielded shipping containers, and for removing the loaded shipping containers from the storage pit and loading them on freight cars. New fuel assemblies are unloaded from freight cars, removed from their shipping containers, deposited in the storage pit, transferred to the interior of the vapor container and installed in the reactor by reversing the mechanisms and procedures.

General Description

Drawing 9699-FM-19A shows the general arrangement of the fuel handling system currently under consideration.

The shield tank cavity located above, and joined to, the reactor vessel is a reinforced concrete, stainless steel lined container filled with 25 ft of borated water, and is provided primarily for the purpose of permitting the fuel assemblies to be handled under water as they are withdrawn from the core.

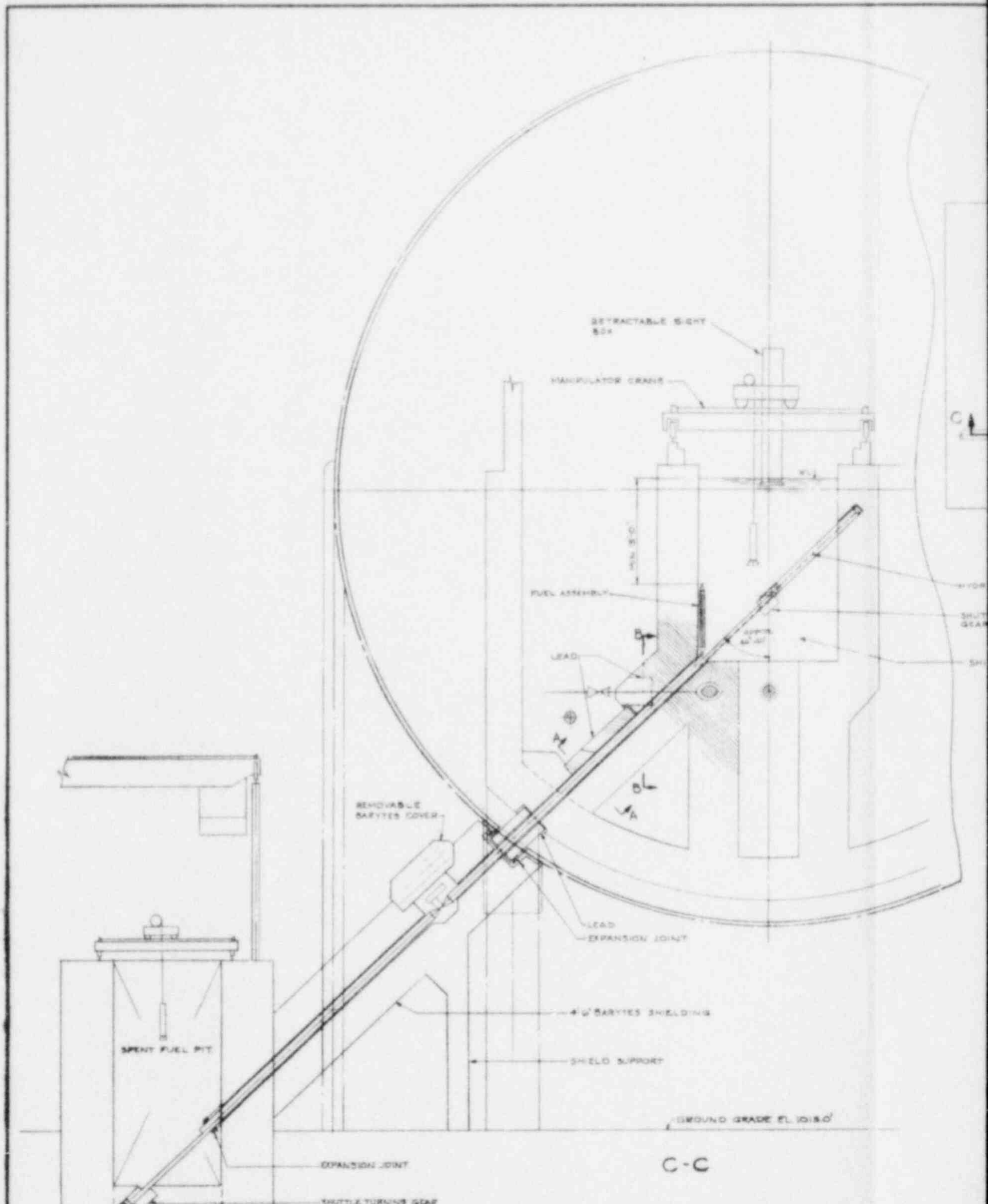
Located above the shield tank cavity is a manipulator crane for handling the fuel assemblies within this cavity and placing them in the spent fuel chute.

The spent fuel storage pit, located outside of the vapor container, is a reinforced concrete, stainless steel lined pit filled with 35 ft of water and is provided for receiving spent fuel assemblies from the chute and storing them under water for a specified decay period.

Located above the spent fuel storage pit is a manipulator crane for handling the fuel assemblies in the pit as received from, or transferred to, the chute.

The spent fuel storage pit is provided with an additional crane for removing or receiving fuel assemblies in their shipping containers.

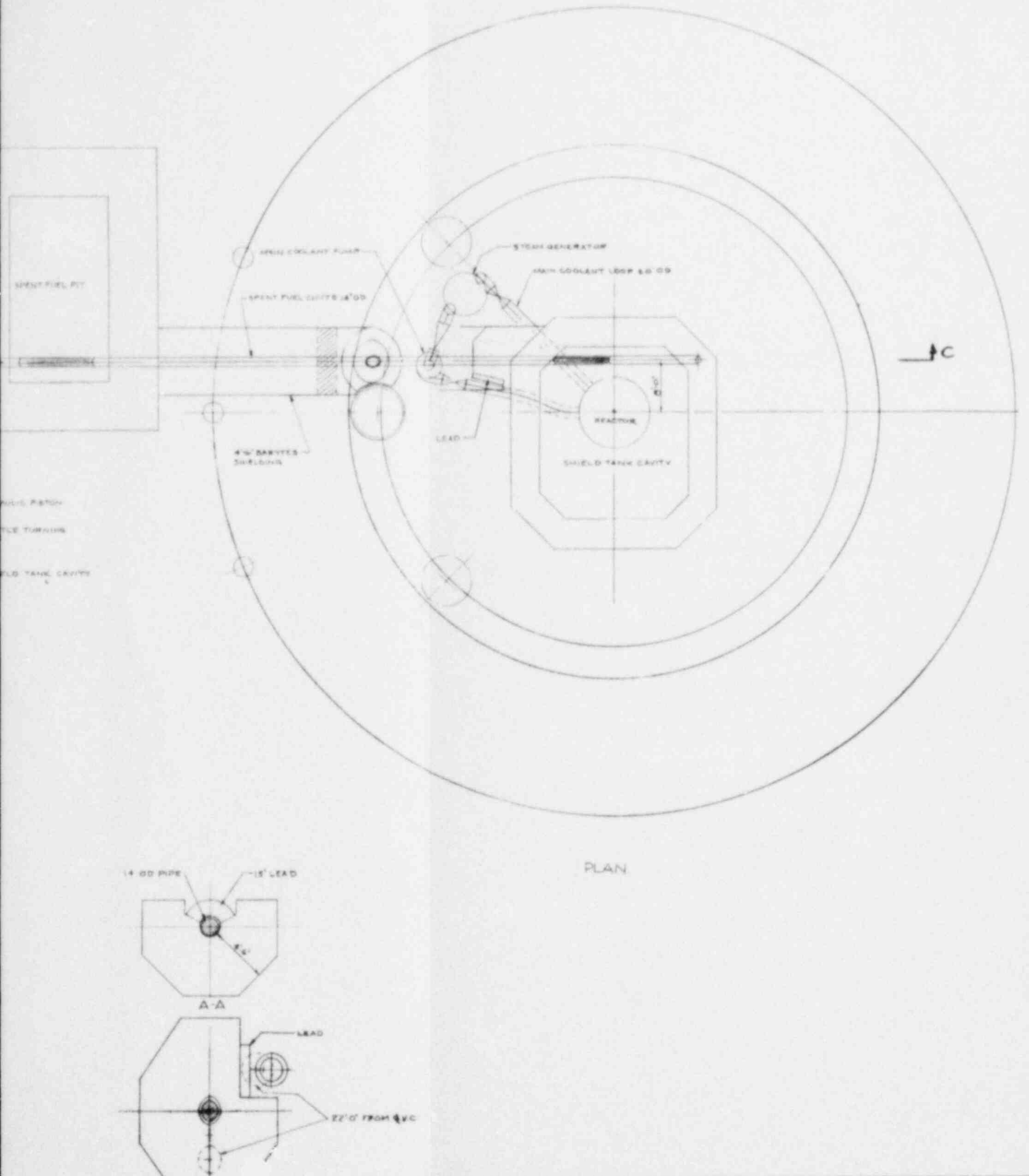
The fuel chute is an inclined 14 in. OD pipe interconnecting the spent fuel storage pit with the shield tank



RETRACTABLE SIGHT BOX
MANIPULATOR CRANE
FUEL ASSEMBLY
LEAD
REMOVABLE BARYTES COVER
LEAD
EXPANSION JOINT
4'-0" BARYTES SHIELDING
SHIELD SUPPORT
SPENT FUEL PIT
GROUND GRADE EL. 1018.0'
EXPANSION JOINT
SHUTTLE TURNING GEAR
HYDRAULIC PISTON

C-C

171-172



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PROPOSED FUEL HANDLING ARRANGEMENT

YANKEE ATOMIC ELECTRIC COMPANY

STONE & WEBSTER ENGINEERING CORPORATION

0 5 10 15
SCALE - FEET

9699-FM-19A

cavity within the vapor container. A hydraulically operated cone plug valve is provided with the fuel chute for the dual purpose of maintaining the integrity of the vapor container while the reactor is in operation and for preventing the free discharge of water from the shield tank cavity to the fuel storage pit while the transfer of a fuel assembly is in progress.

A double piston-ended shuttle within the fuel chute is provided for transporting a fuel assembly under positive control up or down the chute.

At the storage pit end of the fuel chute, a hydraulic cylinder is provided for injecting the loaded shuttle up into the fuel chute and for receiving and positioning the shuttle at the end of its travel down the chute. At the shield tank cavity end of the fuel chute, a hydraulic cylinder is provided for a similar purpose. Each hydraulic cylinder is provided with a turning mechanism to rotate the shuttle to a position which will facilitate handling by the manipulator cranes in the event that the shuttle misorients while traveling up or down the chute.

To remove fuel assemblies from the reactor vessel, the vessel head fastenings are first removed while the shield tank cavity is dry. The cavity is then filled with borated water and the head removed by the overhead polar crane and the head is stored under water within the shield tank cavity. The manipulator crane also removes the control rods and other internal components and stores them within the cavity. The manipulator crane then removes a fuel assembly and places it in the shuttle, which is held in position in the upper end of the fuel chute by means of hydraulic pressure applied to its lower end. A minimum of 15 ft of water is always maintained over the assembly for shielding purposes. The holding hydraulic pressure is then relieved and the upper hydraulic piston pushes the loaded shuttle down into the fuel chute until the lower end of the piston enters the mouth of the fuel chute, sealing it off.

At this time, the lower hydraulic piston is in its uppermost position sealing off the lower end of the fuel chute, and the fuel chute valve is open. Hydraulic pressure is then applied to the upper end of the shuttle forcing it to move down the chute.

The rate of descent is governed by a flow control valve which regulates the rate of water flow from the chute below the shuttle. When the lower end of the shuttle contacts the piston seal of the lower hydraulic cylinder, the fuel chute valve is closed and the hydraulic piston is retracted, guiding the shuttle to its full down position. The shuttle is turned as required and the storage pit manipulator crane removes the fuel assembly from the shuttle and inserts it into the spent fuel storage rack at the bottom of the storage pit. After the required decay period, the manipulator crane removes the fuel assembly from the storage rack and inserts it into its shipping container. A minimum of 15 ft of water is always maintained over the assembly.

When bringing a fuel assembly into the vapor container for insertion into the reactor vessel, the spent fuel pit manipulator crane lowers the assembly and inserts it into the shuttle. At this time, the fuel chute valve is closed and both the lower and the upper hydraulic cylinders are in their fully retracted positions. The lower hydraulic cylinder then advances, pushing the loaded shuttle up into the fuel chute until its piston seal enters the mouth of the fuel chute. The fuel chute valve is opened and hydraulic pressure is applied to the lower end of the shuttle, forcing it to move up the chute. When the upper end of the shuttle contacts the piston seal of the upper hydraulic cylinder, it is turned, as required, and the shield tank cavity manipulator crane removes the assembly. This manipulator crane is provided with a means for turning the assembly to assure its proper orientation prior to insertion into the reactor pressure vessel. The assembly is then placed in its proper location within the reactor vessel.

216 STEAM-ELECTRIC PLANT

Function

The secondary plant is designed to utilize, in a single turbine generator with condenser, the 492 mw expected heat output of the reactor plant to deliver to the New England Power Company transmission system 134 mw of electric power and to utilize any reduced heat output of the primary plant to produce correspondingly lower station outputs. The secondary plant is designed to receive and, through the cooling systems of the plant, dispose of the total heat existent or produced in the primary system following a sudden shutdown of the turbine generator from full load to no load.

The expected full load cycle heat rate at 1 1/4 in. Hg abs back pressure is 11,200 Btu per kwhr. The average auxiliary power requirement at full load is 12,000 kw. The expected full load station heat rate is 12,210 Btu per kwhr.

The component parts of the secondary plant are of types conventionally used in large central stations and are arranged to provide the best possible thermal economy of reactor plant output without sacrifice of safety or economy.

General Description

The secondary plant and equipment are shown on the following drawings:

9699-FM-18A - Machine Location Plan - Operating Floor

9699-FM-18B - Machine Location Plan - Basement Floor

9699-FM-18C - Machine Location - Elevation

9699-FM-18D - Machine Location Plan - Mezzanine Floor

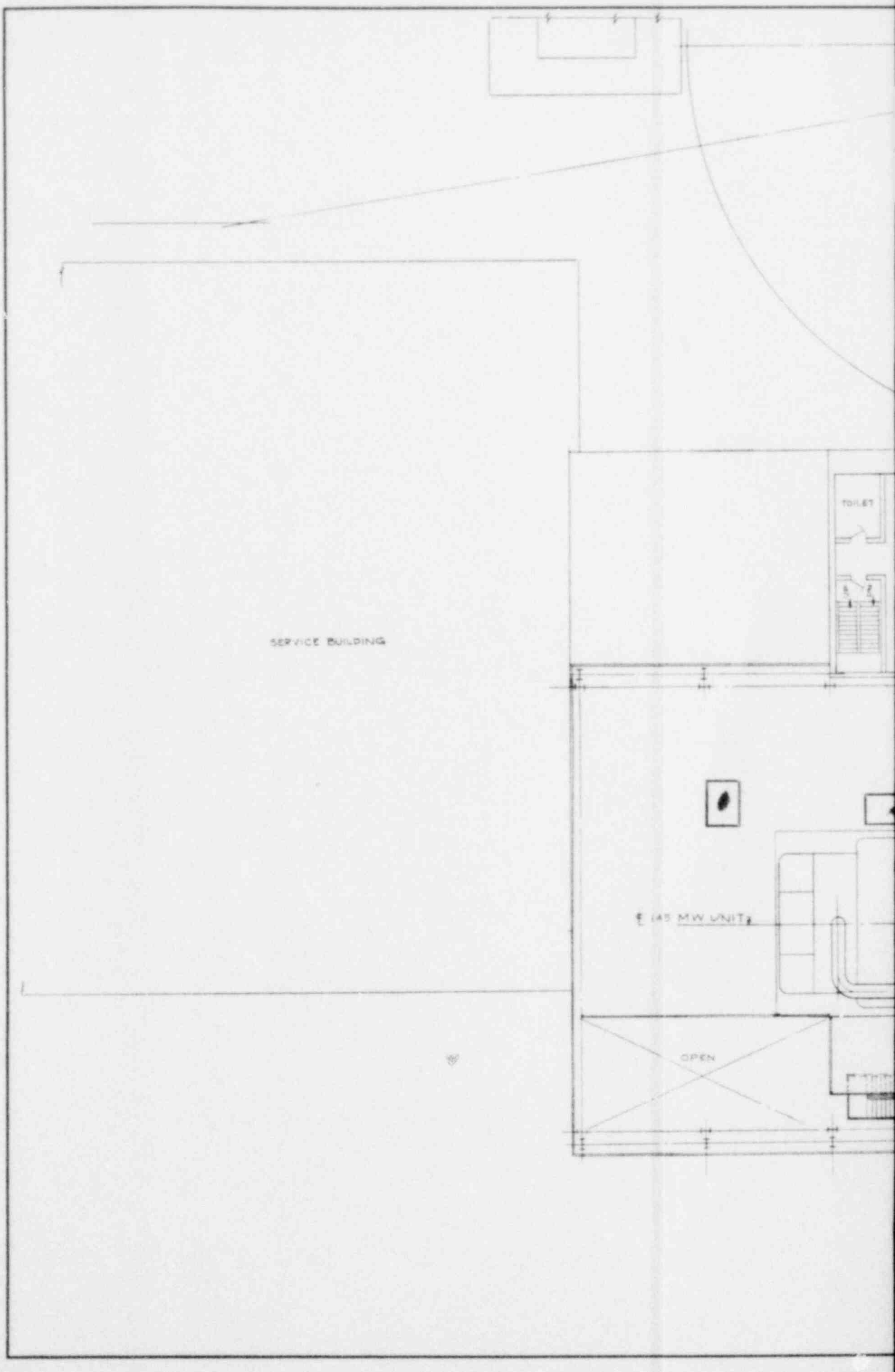
9699-FM-22A - Circulating Water System - Plan

9699-FM-22B - Circulating Water System - Sections

Grading and Fencing

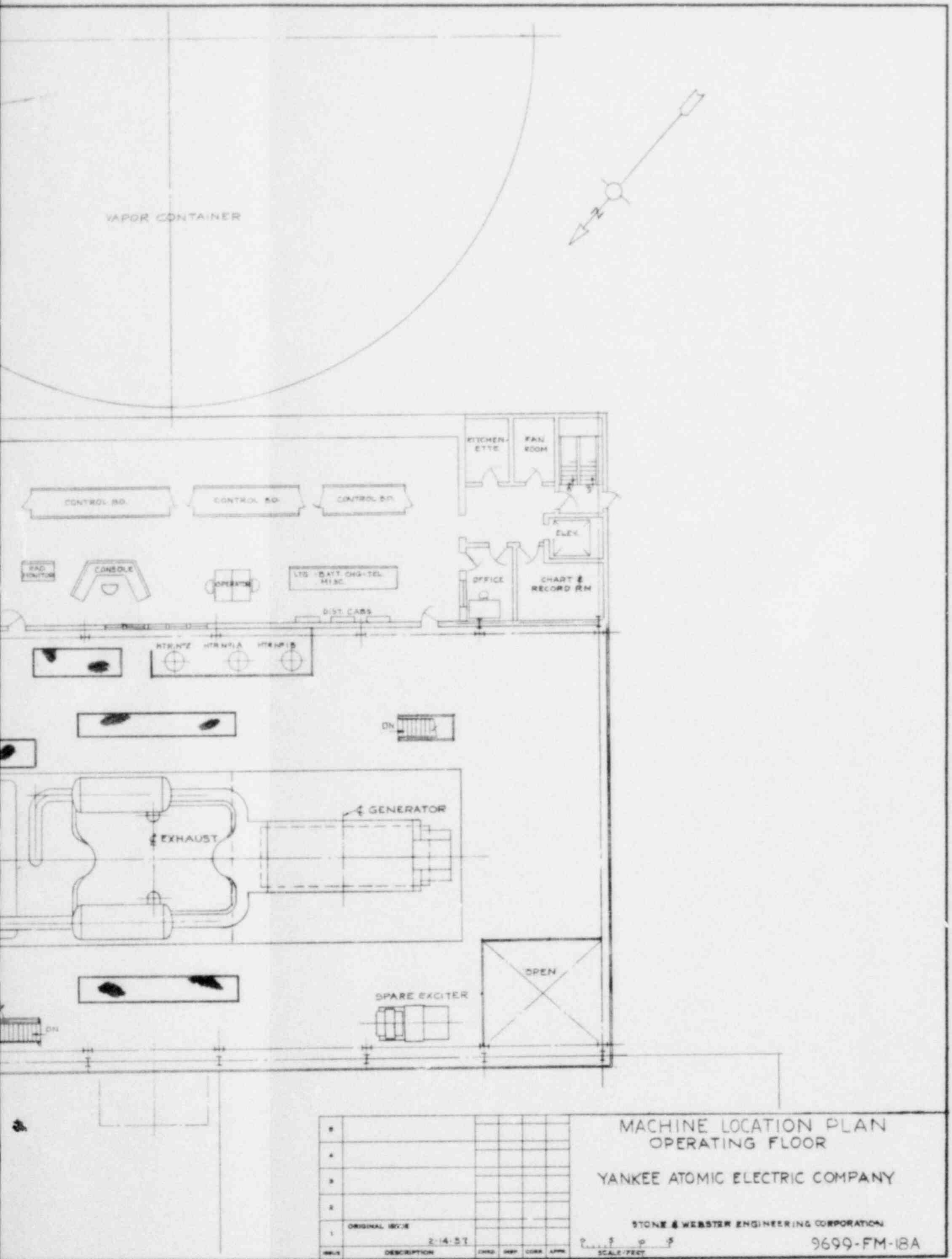
The general yard grade is 1,015 ft referred to New England Power Company Datum.

A barbed-wire perimeter fence at a 1,000 ft radius on the three wooded sides and a log boom and chain across the Deerfield River restrict approach to the plant. A chain link fence within this enclosure closely surrounds the structures and is provided with a guardhouse and necessary service gates.



TOILET
SERVICE BUILDING
℄ 145 MW UNIT
OPEN

176-177



VAPOR CONTAINER
N
KITCHENETTE
FAN ROOM
CONTROL BD.
CONTROL BD.
CONTROL BD.
ELEV.
RAD MONITOR
CONSOLE
OPERATOR
LTG.-BATT. CHG.-TEL. MISC.
OFFICE
CHART & RECORD RM
DIST. CABS
DN
℄ GENERATOR
℄ EXHAUST
OPEN
SPARE EXCITER
DN
MACHINE LOCATION PLAN
OPERATING FLOOR
YANKEE ATOMIC ELECTRIC COMPANY
STONE & WEBSTER ENGINEERING CORPORATION
ORIGINAL ISSUE
2-14-57
DESCRIPTION
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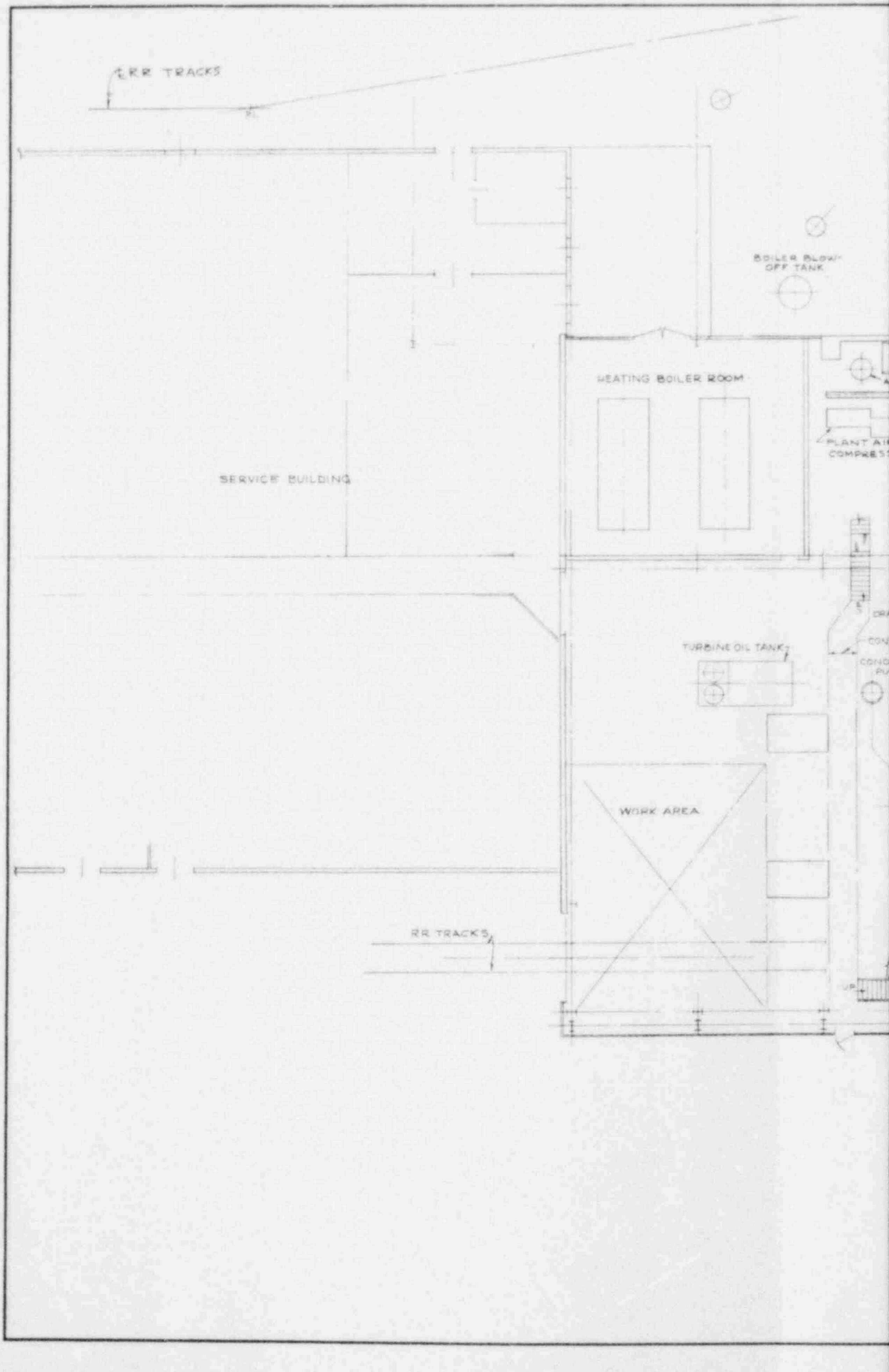
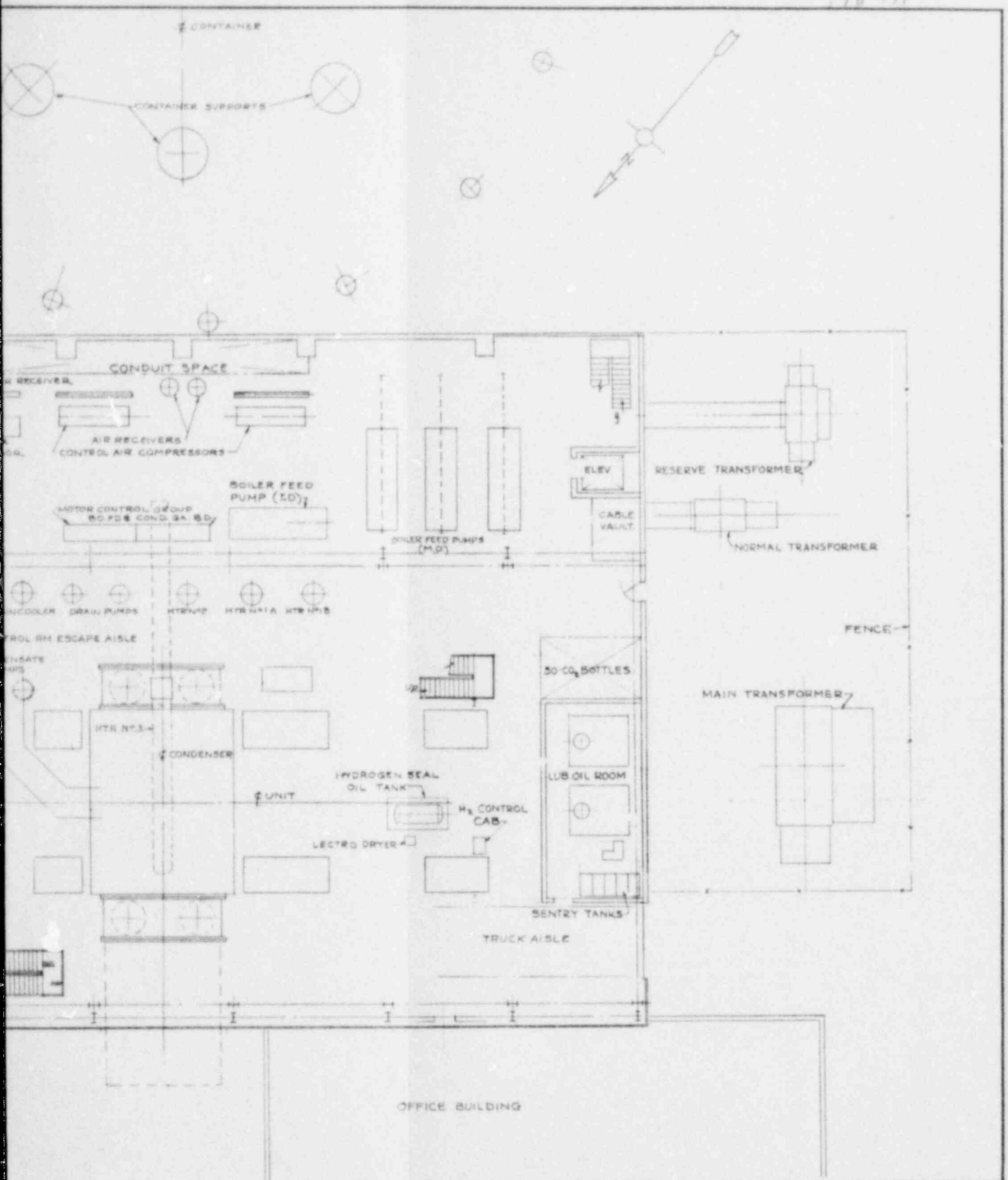
9699-FM-18A



℄ RR TRACKS
BOILER BLOW-OFF TANK
HEATING BOILER ROOM
SERVICE BUILDING
TURBINE OIL TANK
WORK AREA
RR TRACKS



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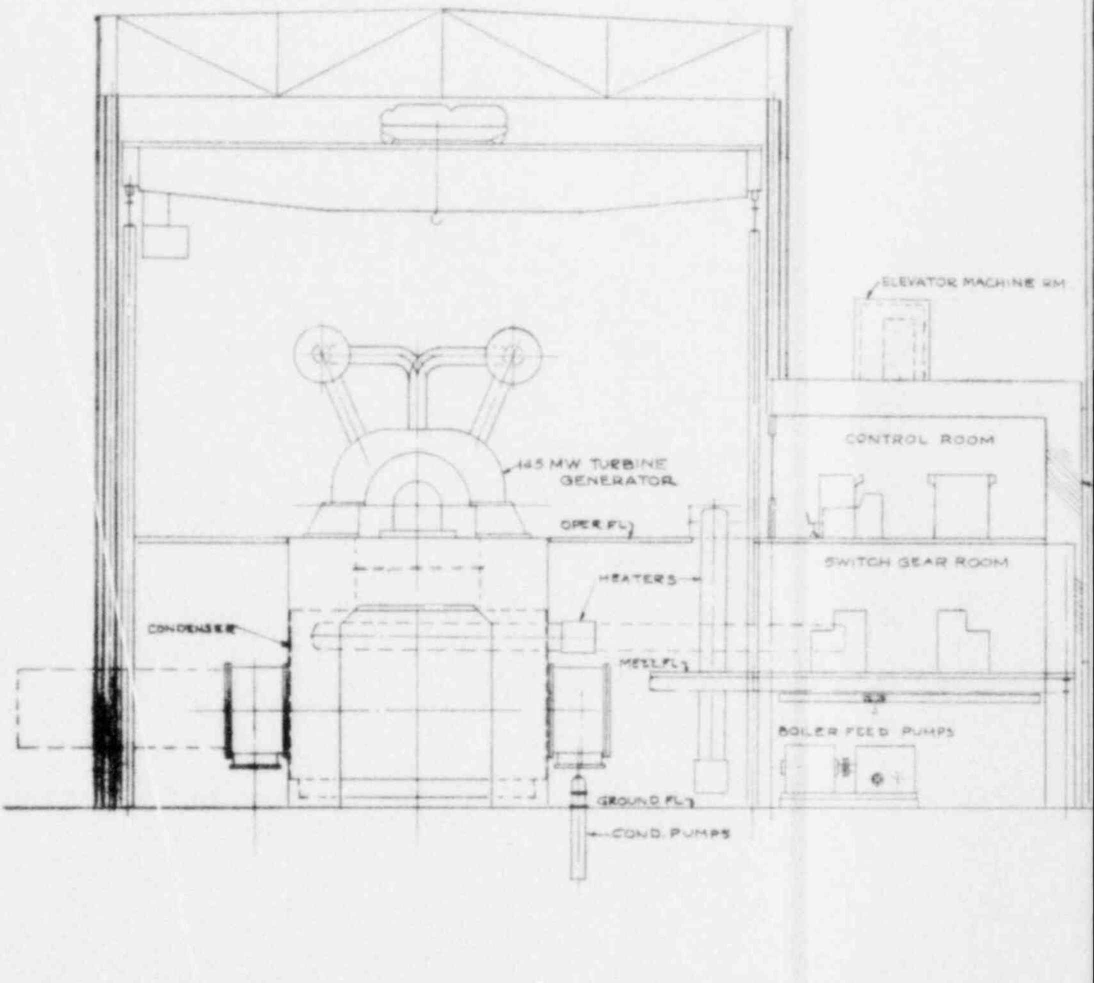
MACHINE LOCATION PLAN
GROUND FLOOR

YANKEE ATOMIC ELECTRIC COMPANY

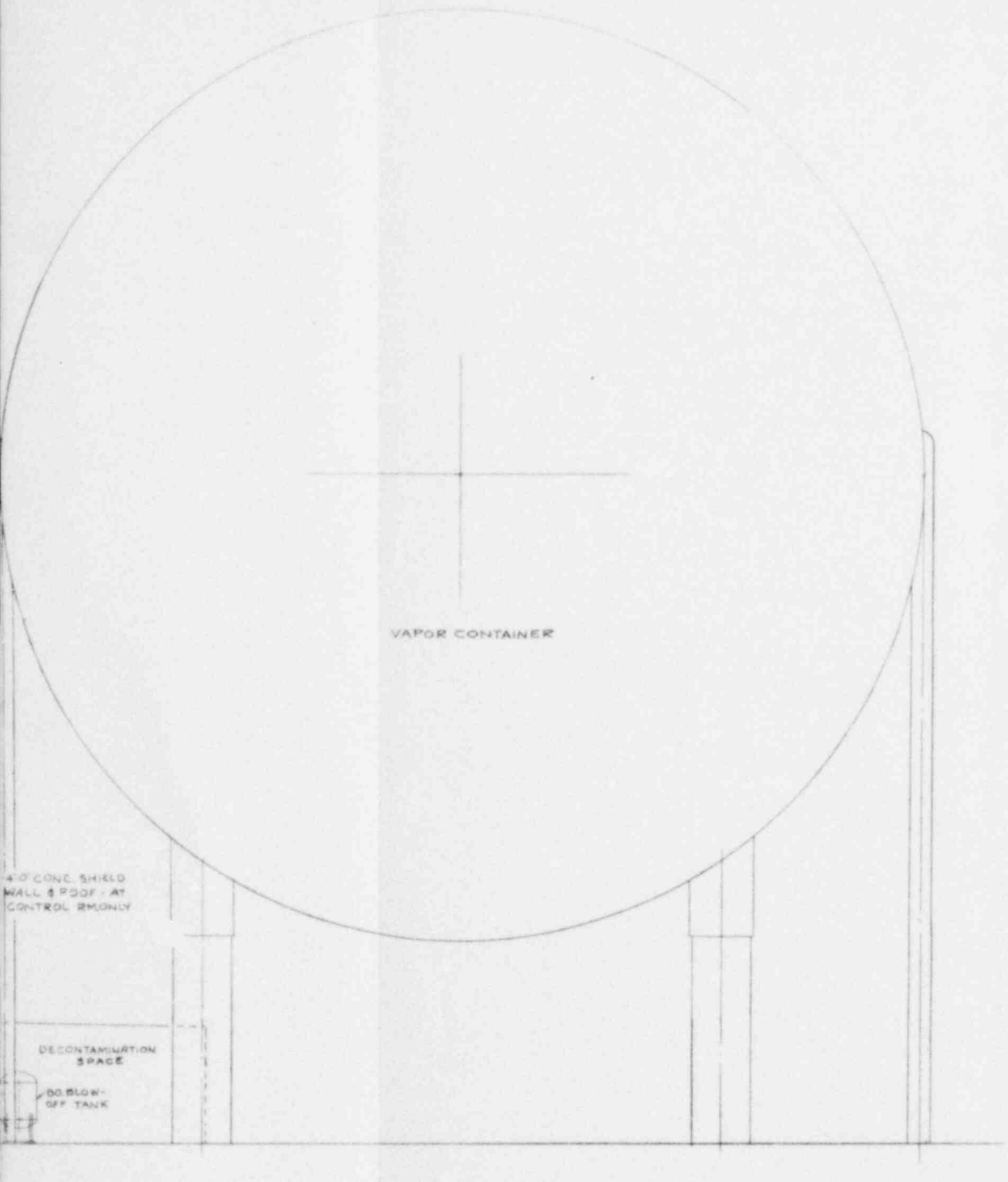
STONE & WEBSTER ENGINEERING CORPORATION

SCALE - FEET

9699-FM-18 B


ELEVATOR MACHINE RM
CONTROL ROOM
145 MW TURBINE
GENERATOR
OPER. FL.
SWITCH GEAR ROOM
HEATERS
CONDENSER
MEZZ. FL.
BOILER FEED PUMPS
GROUND FL.
COND. PUMPS

180-181



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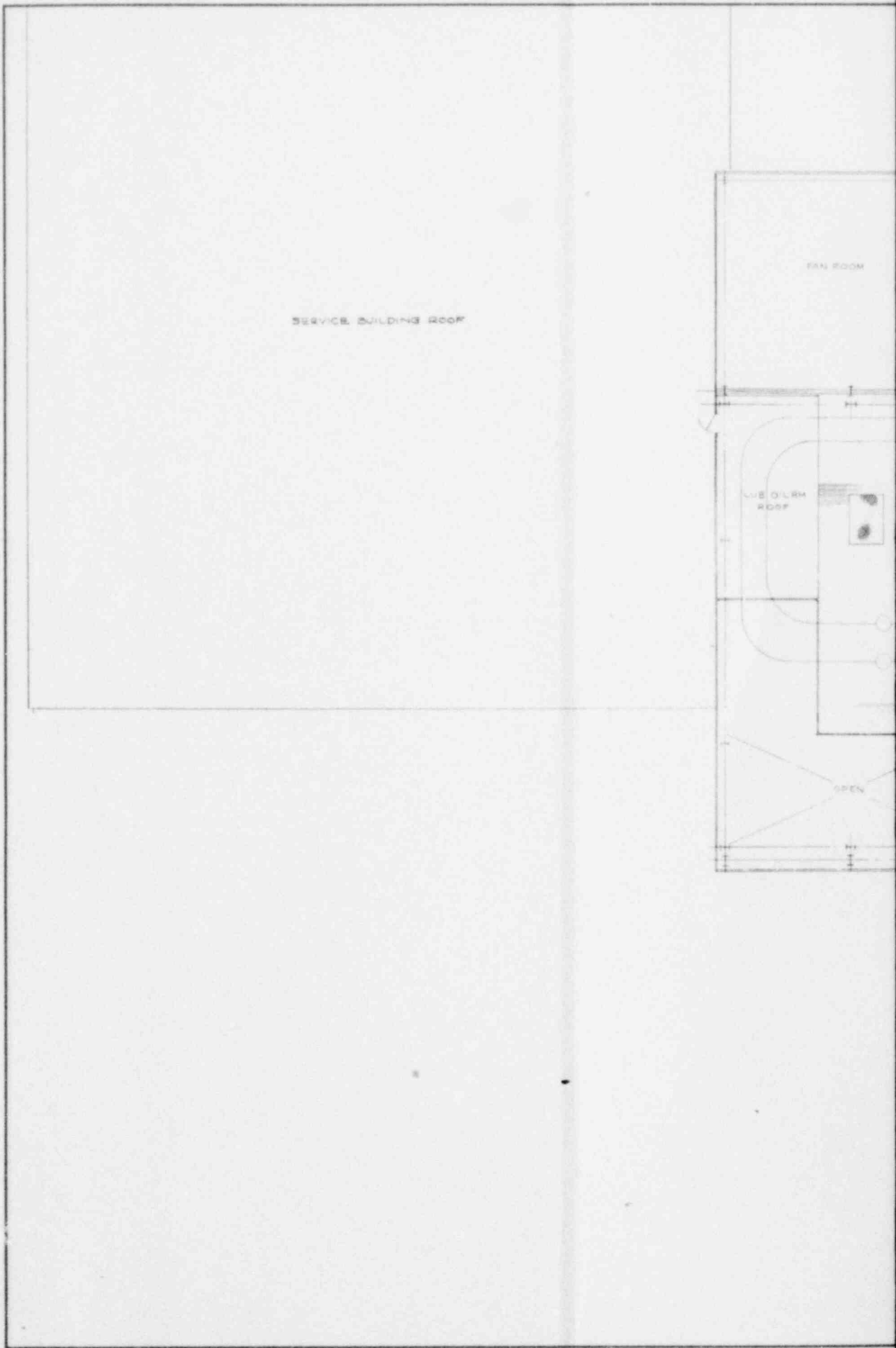
MACHINE LOCATION — ELEVATION

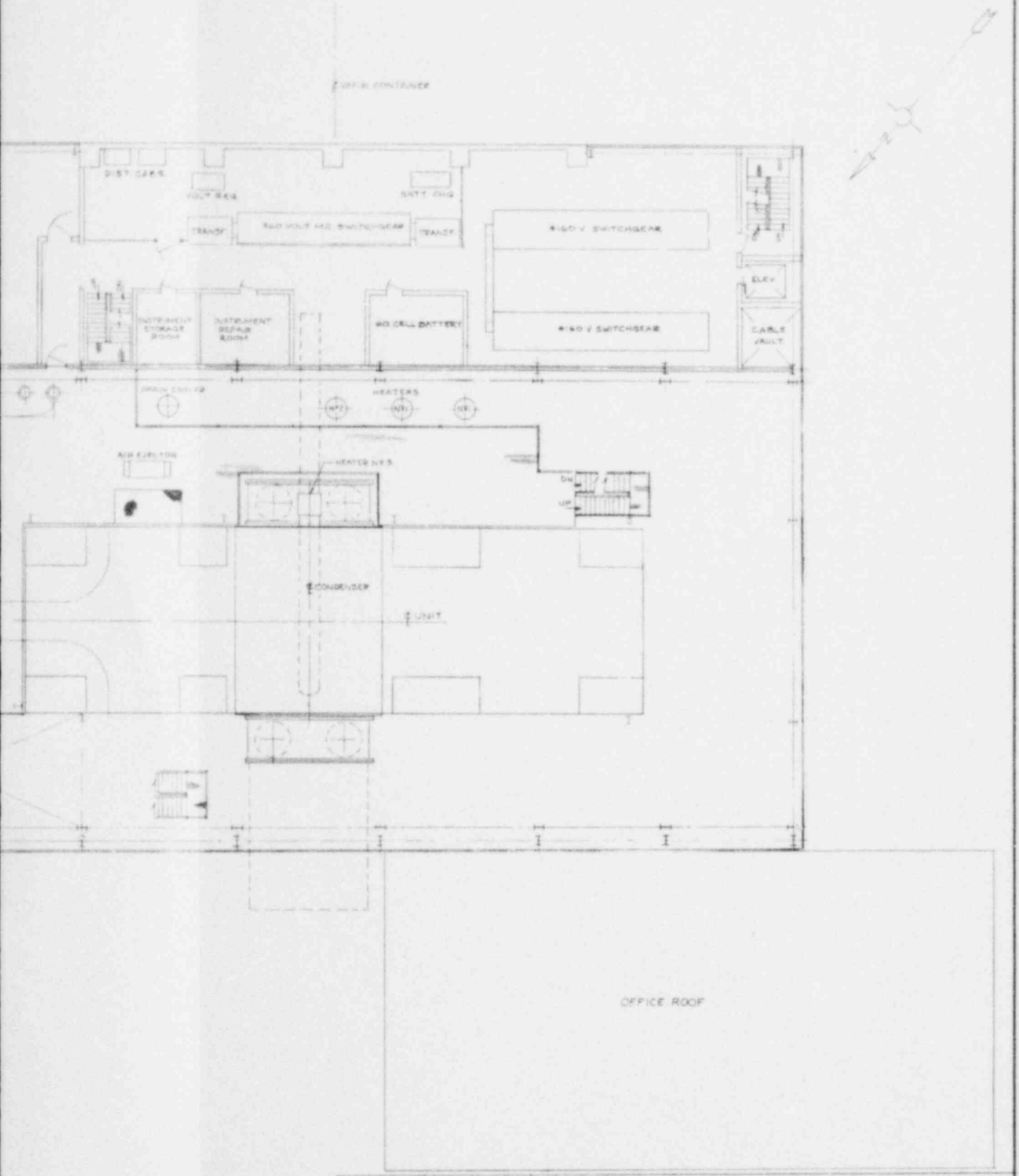
YANKEE ATOMIC ELECTRIC COMPANY

STONE & WEBSTER ENGINEERING CORPORATION

0 5 10 15
SCALE FEET

9699-FM-18C


SERVICE BUILDING ROOF
FAN ROOM
LUB OIL RM
ROOF
OPEN



MACHINE LOCATION PLAN
MEZZANINE FLOOR

YANKEE ATOMIC ELECTRIC COMPANY
STONE & WEBSTER ENGINEERING CORPORATION

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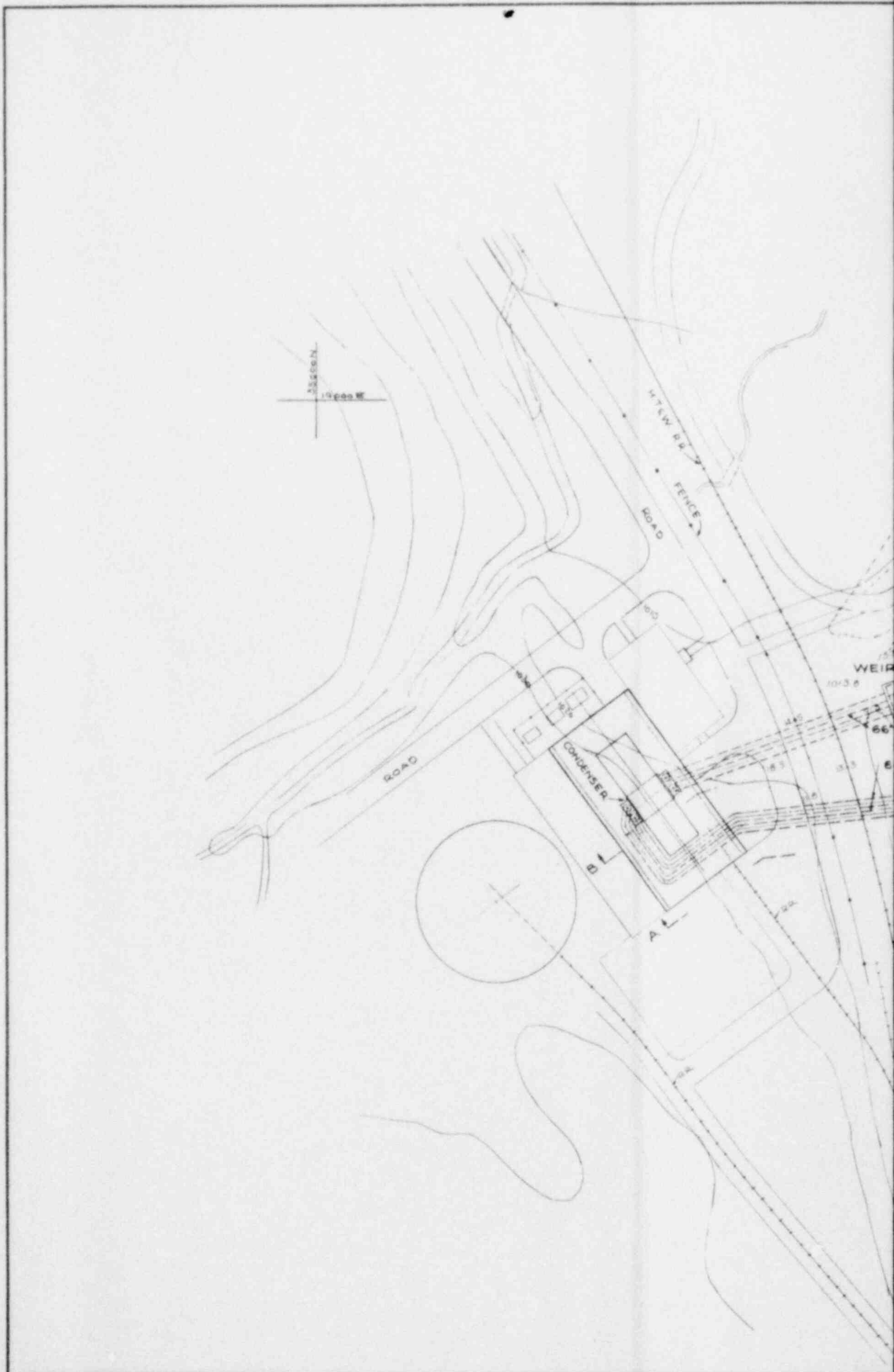
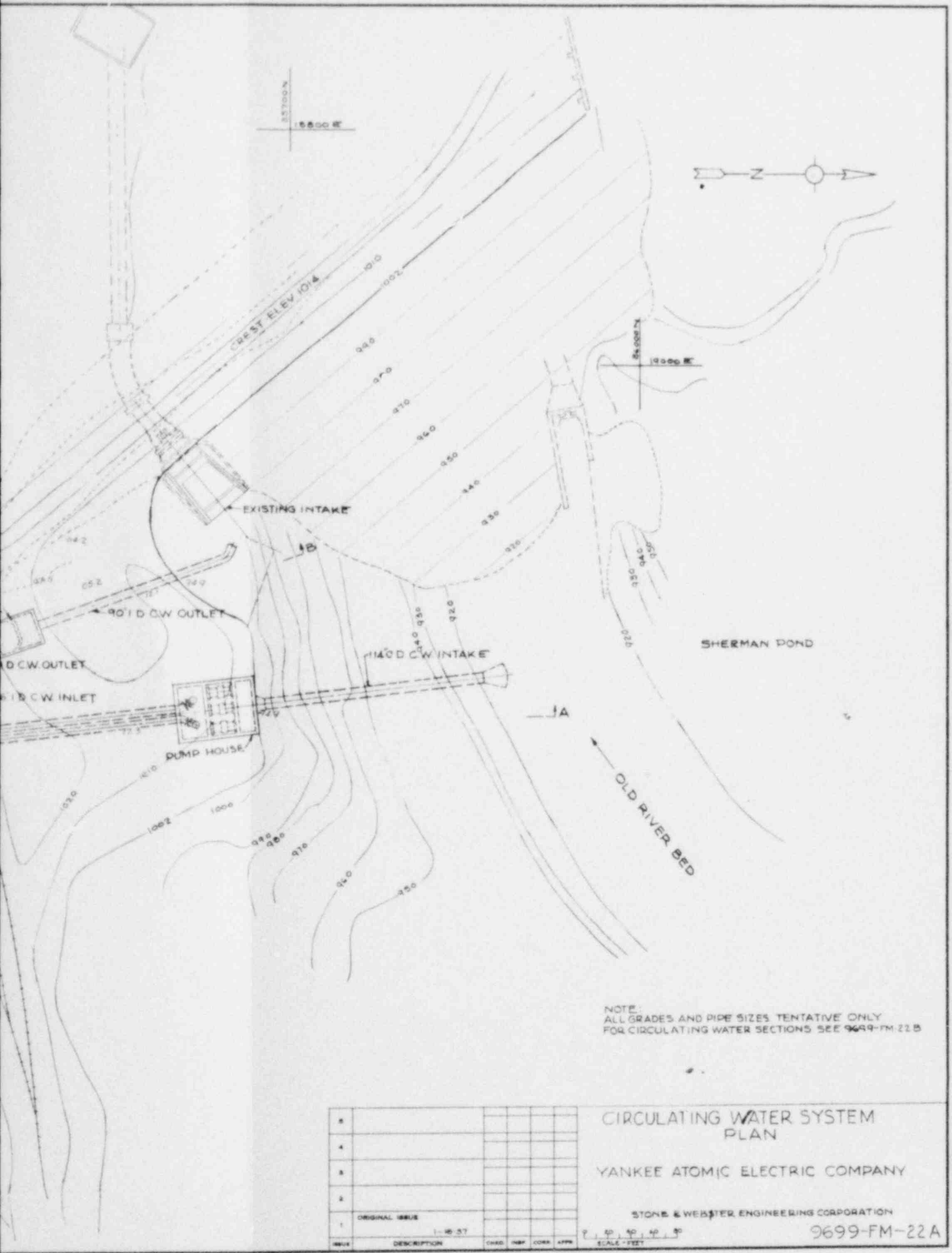
0 5 10 15 SCALE - FEET

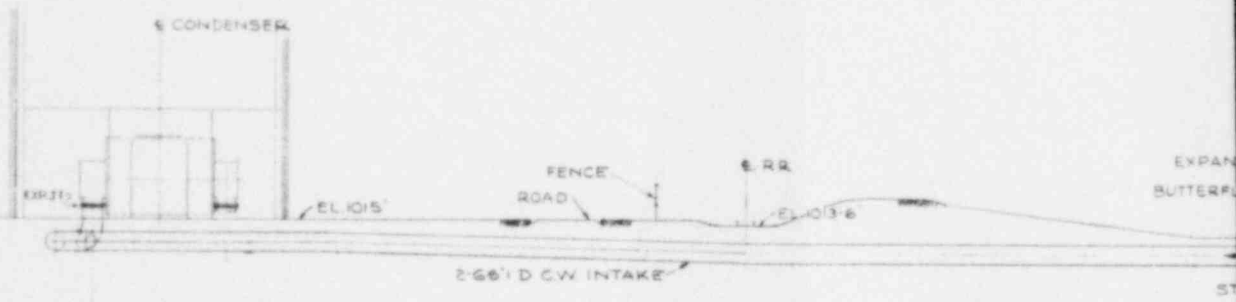
9699-FM-18D



FENCE
ROAD
ROAD
CONDENSER
B
A

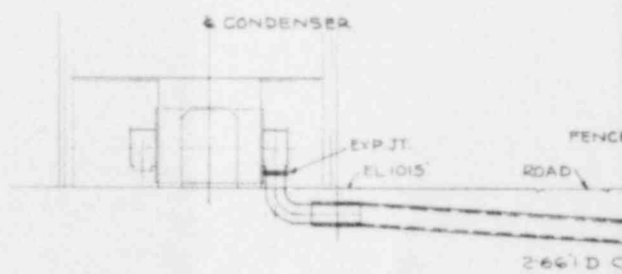


184-185
CREST ELEV 1014
EXISTING INTAKE
90' I D C.W. OUTLET
D C.W. OUTLET
I D C.W. INLET
114' O D C.W. INTAKE
PUMP HOUSE
SHERMAN POND
OLD RIVER BED
NOTE:
ALL GRADES AND PIPE SIZES TENTATIVE ONLY
FOR CIRCULATING WATER SECTIONS SEE 9699-FM-22B
ORIGINAL ISSUE
1-16-57
ISSUE
DESCRIPTION
CIRCULATING WATER SYSTEM
PLAN
YANKEE ATOMIC ELECTRIC COMPANY
STONE & WEBSTER ENGINEERING CORPORATION
SCALE - FEET
9699-FM-22A

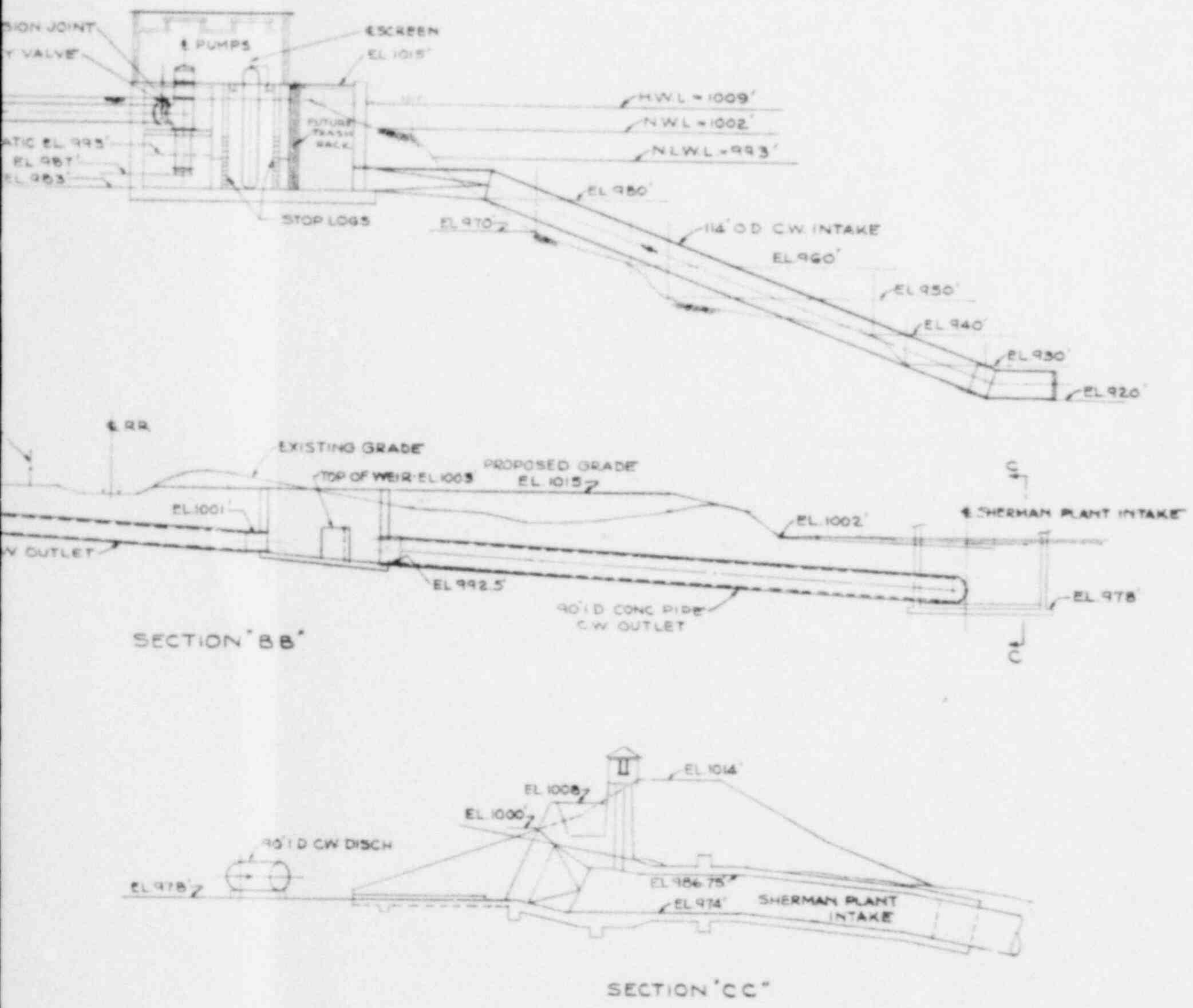


℄ CONDENSER
EXP.JT.
EL.1015'
FENCE
ROAD
℄ R.R.
EL.1013'-6"
2-66"I.D. C.W. INTAKE

SECTION "A A"



℄ CONDENSER
EXP.JT.
EL.1015'
ROAD
FENCE



NOTE:
ALL GRADES AND PIPE SIZES TENTATIVE ONLY

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CIRCULATING WATER SYSTEM
SECTIONS

YANKEE ATOMIC ELECTRIC COMPANY

STONE & WEBSTER ENGINEERING CORPORATION

SCALE FEET 0 10 20 30 40

9699-FM-22B

Embankment is stabilized by using slopes flatter than 33 deg.

Footings and Foundations

The vapor container footing rests on undisturbed soil approximately 30 ft below original ground grade. In general, concrete is designed to produce a compressive strength of at least 2,500 psi after 28 days, while the turbine support is designed for a compressive strength of at least 3,000 psi. All reinforcing steel conforms to Tentative Specifications for Billet-Steel Bars for Concrete Reinforcement ASTM-A15 and Specifications for Minimum Requirements for the Deformations of Deformed Steel Bars for Concrete Reinforcement ASTM-A305. Concrete design is based on the American Concrete Institute Building Code Requirements for Reinforced Concrete, "ACI" 318-56.

Structural Steel

Structural steel conforms to the Specifications for Structural Steel for Bridges and Buildings, ASTM A7.

Siding and Roof Deck

The walls and roof of the demineralizing vault, and the walls of the spent fuel pit, screen and pump well and seal pit are concrete.

The control room is adequately shielded with concrete walls and roof deck to provide protection for the operators and other personnel who are present in the control room at the time of an accident.

Floors

Supported floors are reinforced concrete on structural steel framing designed for the following live loads:

Office Area - 2,000 lb concentrated, or	100 psf
Stairways	100 psf
Control Room - Weight of equipment plus 50 psf overall clear floor space	-
Turbine Room - Equipment weight, or	150 psf

Ground floors are reinforced concrete on undisturbed soil or compacted fill and are designed for the following live loads:

Laboratory	100 psf
Toilet and Locker Room	100 psf
Work Area (where required for heavy live load)	As required
Mechanical Equipment - Equipment Weight, or	250 psf
Electrical Equipment - Equipment Weight, or	250 psf
Demineralizer Vault - Equipment Weight, or	250 psf

Turbine Generator

The turbine is tandem compound, double flow design 1,800 rpm, rated 145,000 kw at 3 1/2 in. Hg abs exhaust pressure when dry and saturated steam is supplied to the throttle at 465 psia.

The generator rating is 170,000 kva, or 161,500 kw at 30 psi gage H_2 pressure and .95 pf, 18,000 v, 60 cycles.

Standard turbine generator auxiliary equipment, including controls, exciter equipment, lubrication facilities and operational and supervisory instruments, are provided.

The turbine exhaust scroll is fitted with rupture diaphragms to protect against overpressure in the scroll and condenser in the event of loss of vacuum resulting from stoppage of the circulating water flow and if the low vacuum trip mechanism should fail to close the turbine throttle trip valves.

Condenser

The turbine exhausts to an 80,000 sq ft single pass surface condenser. Tubes are arsenical admiralty, 7/8 in. OD, No. 18 BWG and 30 ft overall length. The hot well is of the deaerating type, with oxygen in condensate not to exceed .01 cc per liter.

The condenser is provided with a twin element, 2-stage steam jet air ejector for normal air removal from the condenser. A hogging jet is provided for quickly developing vacuum in the condenser when starting the plant. The air ejector after-condenser is vented and the hogging jet discharged to the plant stack.

A steam line to the condenser with control valve is provided. The condenser is equipped with nozzles in the condensing zone to receive wet steam discharged from the steam generator during start-up, shutdown and any transient plant operating conditions. These nozzles are equipped with baffles to protect the condenser tubes from erosion by the high velocity steam.

Lubricating Oil System

A lubricating oil conditioner is provided on the mezzanine grade to filter and clarify continuously the turbine lubricating oil. A small by-pass stream of lubricating oil is continually discharged from the turbine oil reservoir, by means of a gear type motor driven pump, to the conditioner from which it returns by gravity to the oil reservoir.

A fireproof walled central lubricating oil room contains a centrifugal oil separator and two steel oil tanks in which alternately new, used or purified turbine oil is stored.

Circulating Water System

Three traveling screens in the circulating water intake from Sherman Pond are arranged for continuous operation and continuous flushing under manual control during the fall pond turnover when large quantities of debris and leaves may enter the intake.

Two motor operated screen washing pumps are provided in the pump well.

Two screens have capacity to pass the circulating water requirements of the station. Stop log slots are provided before and after each traveling screen, and two sets of stop logs are provided to permit unwatering any well for maintenance of a screen.

Two vertical motor driven circulating water pumps manually controlled from the control room are operated continuously, except during maintenance periods. A motor operated butterfly valve is installed in the discharge of each pump and arranged to close automatically on pump motor failure. Each pump discharges through an independent concrete pipe line to the condenser. The two halves of the condenser inlet water box are tied together through a normally closed, manually controlled, motor operated butterfly valve which is opened to permit operating both sides of the condenser when only one circulating pump is available. For brief periods during condenser tube cleaning operations, one pump and one side of the condenser only are operated.

The circulating water discharges from the condenser to a reinforced concrete seal pit with weir to maintain a siphon in each half of the condenser and to limit the vacuum at the top of the condenser outlet water box. The seal pit discharges through a buried concrete line to Sherman Pond, with the outlet located at the Sherman hydroelectric plant intake.

Condensate and Feed Water System

Three stages of steam extraction are provided from the turbine for heating condensate and feed water in closed feed water heaters. A drain cooler between No. 3 and No. 2 feed water heaters receives separated moisture from the crossover between the high and low pressure turbines.

Secondary system make-up is added to the condenser hot well under low level float control from a 100,000 gal steel demineralized water storage tank at ground grade, provided with a floating seal to minimize oxygen absorption. Excess water in the secondary system is returned under hot well high level control by the main condensate pump to the demineralized water storage tank.

Condensate is normally discharged from the condenser hot well by one of two vertical motor driven condensate pumps through the air ejector condenser and through the No. 3 and No. 2 closed feed water heaters to the boiler feed pump suction. A turbine gland steam condenser is provided after the air ejector.

Three half-size motor driven boiler feed pumps, two of which are required to carry full load, discharge feed water through the No. 1 heaters to the steam generators.

Condensate from No. 1 high pressure heaters discharges under level control to the No. 2 heater from which the combined drains are pumped by one of two motor driven heater drain pumps to the boiler feed pump suction. The turbine crossover drain cooler condensate discharges to the No. 3 closed feed water heater which, in turn, drains to the main condenser, under level control.

Feed water heater vents cascade downstream similar to the heater condensate drains.

Provision is made for by-passing condensate drains and vents around a heater not in service.

Heater tube sides are provided with a relief valve to protect against a closed-off water side with shell side heat leakage. Heater shell sides are provided with relief valves having capacity to relieve the flow from one ruptured tube, two tube ends, except in cases where the shell can withstand the maximum tube side pressure.

Service Water Supply

Service water for make-up and cooling is obtained normally from two of three motor driven pumps installed in the circulating water pump well. Because of the importance of cooling water supply to nuclear plant components, essential services can be maintained by one of the three pumps. A standby pump is started automatically by a discharge pressure switch on failure of a running pump. This arrangement ensures a supply of water for station service when the circulating water pumps are not running.

Coolers supplied with water direct from the service water pumps, through strainers, include:

Primary component heat exchangers
Turbine oil coolers
Generator gas coolers
Hydrogen seal oil cooler
Spent fuel pit cooler

The service water pumps also supply the water treatment plant filters. Filtered water pumps supply the following services:

Demineralizers for power plant make-up and main coolant make-up

Filter backwash

Vapor container coolers

Station utility service

Spent fuel pit

Miscellaneous small cooling services

Supplementary chemical feed equipment for the secondary plant feed water system

Chemical decontamination equipment for the main coolant system

Chromate solution feed to recirculated cooling water systems for corrosion control

Cooling Systems

Service water is pumped through the turbine oil cooler, the generator gas coolers and the seal oil cooler. These coolers operate in parallel. The turbine is provided with two oil coolers, only one of which is normally in service. The generator is provided with four gas cooler sections, all of which are normally in service, but the generator can be operated at load with one gas cooler section out of service. The seal oil cooler is by-passed, if necessary. The service cooling water is discharged to the circulating water outlet line.

Each cooler is provided with inlet and outlet isolation valves and with necessary drain and vent valves.

Secondary Plant Drains

Ultimate disposal of all secondary plant drains is made through two one-day hold-up tanks to permit sampling and monitoring the drains for radioactivity prior to discharge to the circulating water outlet line. Steam generator blowdown and steam line drains are discharged to the blow-off tank which drains to the hold-up tanks. Pump base plate drains and other cold water drains also discharge directly to the hold-up tanks.

Compressed Air Systems

One motor driven 500 cfm service air compressor is provided complete with inter and after coolers and air receiver to supply air for general utility service.

Two motor driven control air compressors are provided, of the nonlubricated, carbon ring type, each complete with an aftercooler and air receiver.

Because of the isolated nature of this single unit plant, two full-sized control air compressors are provided, with arrangement for automatic starting of the stand-by compressor and a secondary automatic backup through a normally closed tie line.

Dry type air filters and silica gel air dryers are provided ahead of the control air piping system.

Piping

Power plant piping is generally carbon steel. Cast iron is used for buried cooling water and drain lines and where otherwise required for corrosion protection.

Buried steel piping is coated and wrapped to provide external corrosion protection.

Brass pipe is used for water and control air lines, 2 in. and smaller.

Controls and Instruments

The principal equipment is started and controlled from the control room, with the exception of the turbine generator which is started and brought up to operating speed by local control. Major equipment is also provided with local manual starting.

In most cases, a stand-by motor-driven pump is automatically started by a discharge pressure switch on failure of the running pump.

A pressure gage is installed at the discharge of each pump, near the inlet to each safety valve and near the inlet to each pressure control valve. Vital pressures in the secondary system are recorded and, when needed for control, are indicated in the control room.

An indicating thermometer or a thermometer test well is installed adjacent to each temperature recorder well. Vital temperatures in the secondary system are recorded and, when needed for control are indicated in the control room.

Each steam generator is provided with a 3-element level control in the feed water supply.

All tanks, including waste storage tanks, are provided with level indication. Vital tanks, where the level must be under continual observation by the operator, have levels remotely indicated in the control room. High and low level alarms are provided on all vital tanks.

Feed water heater shells are provided with high and low level alarms.

The secondary main steam flow from the feed water flow to each steam generator is measured for main coolant heat output determination and as part of the 3-element steam generator level control.

Standard controls for synchronizing, load maintenance, voltage and frequency regulation and controls for station service auxiliaries are provided.

General Description of Electrical Equipment

The electrical system is shown in the one line diagram 9699-FE-1A.

The net output of the Yankee plant is normally transmitted at 115 kv over New England Power Company lines to Millbury, Mass. The output of the plant may also be transmitted at 115 kv and, at reduced capacity, to the Harriman Station of the New England Power Company located about three miles distant. The 115 kv oil circuit breaker to Millbury operates normally closed and the 115 kv oil circuit breaker for the Harriman line, normally open. In the event of a 115 kv Millbury line trip out, service is maintained with the fast closing 115 kv oil circuit breaker to Harriman.

Heating Systems

During normal operation, no heat is required for the interior of the vapor container since the heat losses from interior equipment are expected to offset transmission losses to the outdoors through the container shell. During shutdown periods in cold weather, air heated by steam coils is supplied to the interior of the vapor container.

Heat for laboratories, offices and similar spaces containing windows in exposed walls is provided by finned pipe radiators using hot water as a heating medium. A circulator, steam-to-water heat exchanger and expansion tank are located in the equipment penthouse.

Heat for the control room is provided by a steam heating coil contained in the heating and ventilation system supply unit.

Heating for the turbine and condenser rooms is provided by steam unit heaters.

Heat for toilet, locker and shower rooms is provided by the ventilation equipment through steam heating coils located within the ventilation units.

Heat for miscellaneous areas is provided either by finned pipe radiators, using steam or hot water as the heating medium, or by steam or hot water unit heaters.

Sources of Steam for Heating

During normal operation of the plant, turbine extraction steam reduced to a minimum pressure of 10 psi gage provides heat for all heated spaces. During shutdown periods, steam for heating is taken from a reducing station served by a light-oil-fired auxiliary steam generator operating at a pressure of 100 psi gage.

Steam at 100 psi gage is furnished to the laundry and to steam hoses in decontamination areas.

Drainage Systems

A drainage system for the vapor container is provided to collect water used during cleaning or decontamination operations and to permit removal of water from the container in the event of spillage or leakage. Vapor container drainage is monitored and if radioactive above tolerance, it is pumped to the radioactive waste disposal facility prior to disposal. If radioactivity of the container drains is below tolerance, disposal is directly to the storm sewer system.

Two sources of station service power supply are provided. One source is taken from the generator leads through the normal station service transformer and the second source is supplied from the 115 kv line to the Harriman Station, through the reserve station service transformer. The main station auxiliaries are equally divided between two sections of the station service bus which normally operates with the bus section breaker open. Station service bus sections are provided with automatic throwover in the event that voltage is lost on either bus. The arrangement allows a minimum of two main coolant pumps to operate at all times on failure of one alternative power supply without loss of time required for breaker closing. Time delay on large motor transfer is provided to avoid momentary drop in station service voltage. With two reliable and separate sources of station service supply, there should be no hazard to the plant due to temporary loss of electric power.

Main Transformer

The main 165,000 kva, 3 phase, 18/115 kv, 60 cycle transformer is forced oil, water-cooled type and located outside the turbine room wall.

Station Service Transformers

One normal and one reserve 10,000/12,500 OA/FA kva station service transformer adjoin the main transformer outside the turbine room. Two 480 v, 1,000 kva dry type transformers are supplied for station service.

Emergency lighting is provided for all control rooms, laboratories, stairways, and service areas from a 125 v d-c station battery source.

Fire Alarm

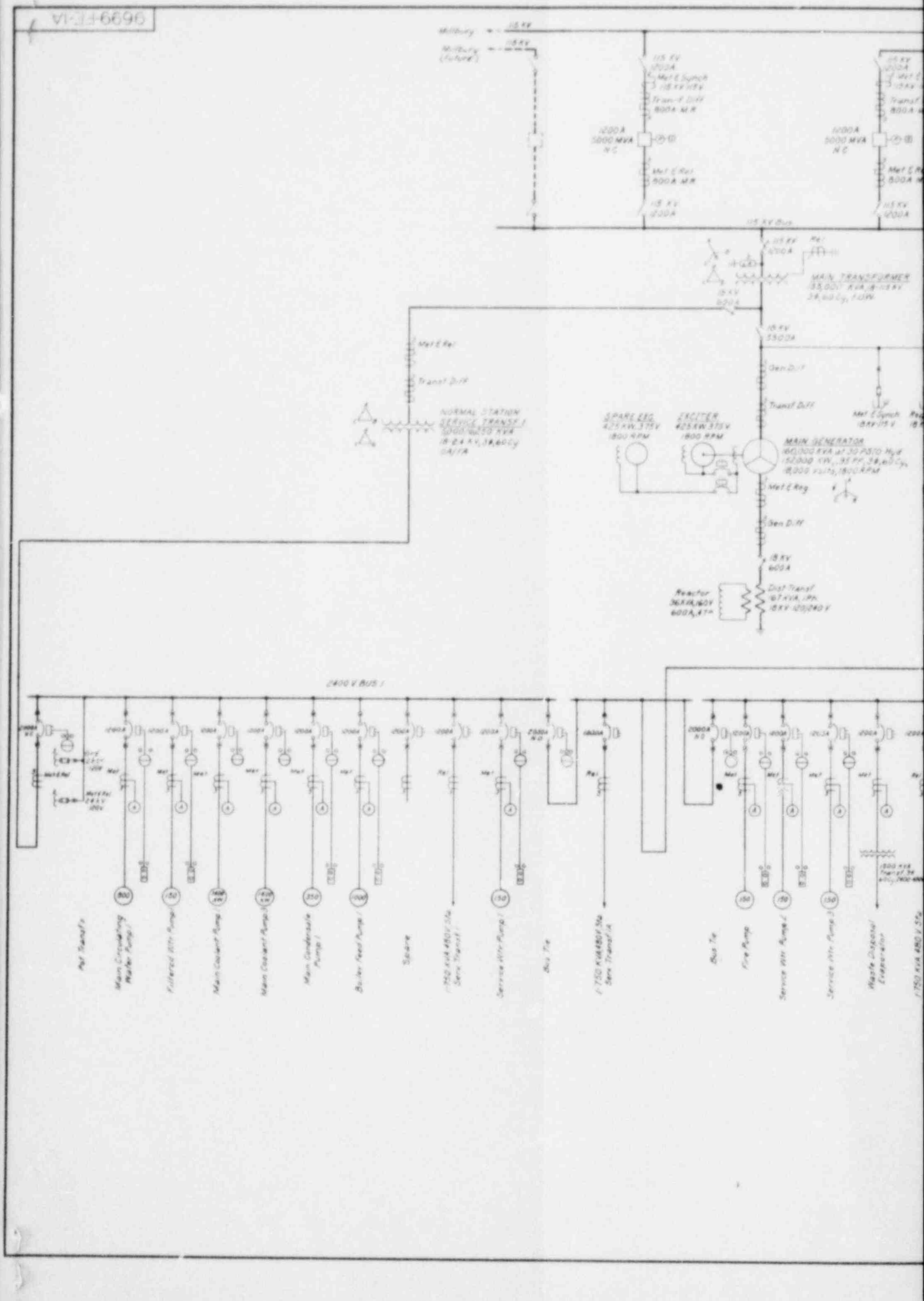
A fire alarm system consisting of 12 noncode alarm boxes is included to sound alarm and transmit signal to nearest fire department.

Miscellaneous Electrical Equipment

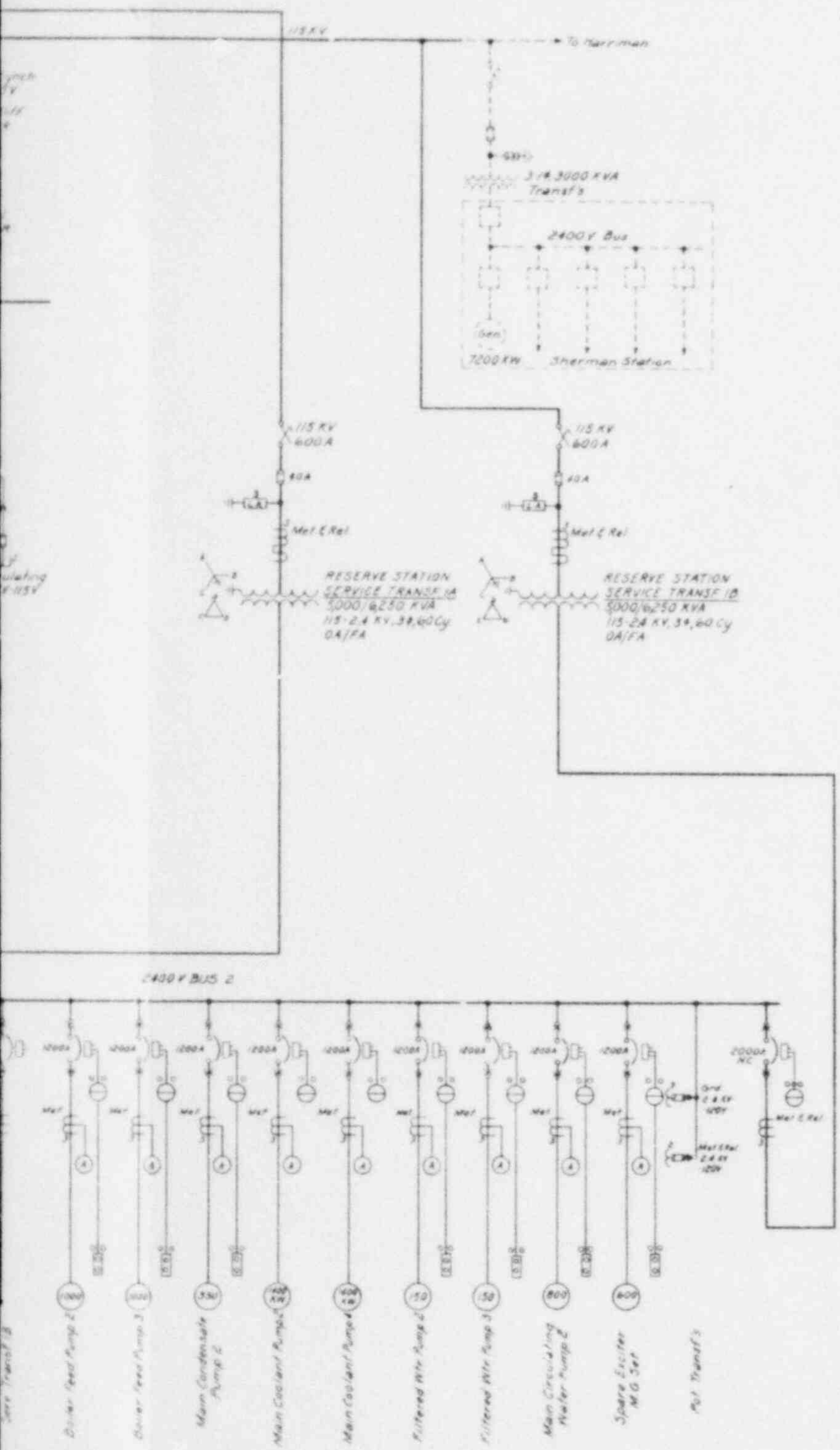
Annunciator alarms, PBX telephone system, closed circuit TV system and public address system are provided for convenience and safety.

Air Conditioning

An air conditioning system is provided for the radioactivity laboratory and the counting room to permit accurate analytical work and to make possible the control of air flows within hoods in order to minimize the spread of radioactive contamination.


9699-FE-1A
Millbury
115 KV
Millbury (future)
115 KV
1200A
Met & Synch
Transf Diff
800A M.R.
1200A
5000 MVA
N.C.
Met & Rel
800A M.R.
115 KV
1200A
115 KV Bus
MAIN TRANSFORMER
Met & Rel
Transf Diff
NORMAL STATION SERVICE TRANSF. 1
Gen Diff
SPARE EXC.
425 KW 375V
1800 RPM
EXCITER
425 KW 375V
1800 RPM
MAIN GENERATOR
Met & Reg
Gen Diff
18 KV
600A
Reactor
Dist Transf
2400 V BUS 1
Pot Transf's
Main Circulating Water Pump 1
Filtered Wtr Pump 1
Main Coolant Pump 1
Main Coolant Pump 3
Main Condensate Pump 1
Boiler Feed Pump 1
Spare
Service Wtr Pump 1
Bus Tie
Bus Tie
Fire Pump
Service Wtr Pump 2
Service Wtr Pump 3
Waste Disposal Evaporator

195-196



LEGEND

- Air Circuit Breaker Manually Operated
- Air Circuit Breaker Electrically Operated
- Oil Circuit Breaker Pneumatically Operated
- Synchronizing Switch
- Interrupter Switch Manually Operated
- Disconnecting Switch
- Pole Top Switch Manually Operated
- Current Transformer
- Bushing Type Current Transformer
- Potential Transformer
- Bushing Type Potential Transformer
- Fuse
- Disconnect Device
- Ammeter
- Induction Motor Numerals indicate Horsepower
- Control Switch with Red & Green Indicating Lights
- 2-Button PB Station, Maintaining Contacts "Start Stop" with Red & Green Indicating Lights
- LA Lightning Arrester
- NC Normally Closed
- NO Normally Open

ISSUE	DESCRIPTION	CHKD	DSP	CORD	APPD
5					
4					
3					
2	REVISED TO AGREE WITH 9699-FE-1				
1	ORIGINAL ISSUE	ESK			

MAIN ONE LINE DIAGRAM

POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.

STONE & WEBSTER ENGINEERING CORPORATION

7-15-57

DRAWING NUMBER 9699-FE-1A

This system utilizes 100 per cent outside air at all times and is controlled in conjunction with a radioactive hood exhaust system to prevent outward leakage from areas containing radioactive material.

Ventilation Systems

A radioactive exhaust system is provided for the hoods and general areas of the radioactivity laboratory. This system consists of radioactive air filters at hoods and glove boxes in the radioactivity laboratory, as well as a system of ducts connected to a single fan located in an equipment penthouse. The fan discharges to atmosphere through the plant stack.

A general ventilation supply system is provided to furnish filtered air, heated in the winter, for interior toilet, locker and shower rooms, as well as for the laundry.

A nonradioactive hood exhaust system, without radioactive air filters, is provided for the fume hood in the plant laboratory.

An exhaust system without radioactive air filters is provided for the laundry.

A system of supply and exhaust ventilation is provided for the control room to maintain air conditions within tolerable limits. The supply system contains a steam heating coil for winter heating.

Spent fuel assemblies from the reactor are handled in the shield tank cavity, only when the vapor container is open. Hooded and ducted vents discharging to the outer atmosphere are provided over this cavity to remove contaminated vapor released incidental to spent fuel handling.

Spent fuel assemblies are temporarily stored after removal from the reactor in a water filled spent fuel pit outside the vapor container. This pit is provided with a cover, and the space over the water is permanently vented to the plant stack.

Ventilation of the turbine and condenser area is provided by natural means supplemented in local areas, if necessary, by motor-driven equipment. In general, air is exhausted from the roof of the turbine area through gravity ventilators and flows into the area through open windows and doors during the summer.

Ventilation for other general service areas is provided by natural means wherever possible.

Storm drainage systems for roofs, roads and yard areas of conventional design are provided, utilizing natural run-off ditches where feasible and underground piping elsewhere.

Urinals and water closets in potentially contaminated areas drain to either of two tanks, each of one day holdup capacity to permit daily sampling and analysis of sewage for radioactivity prior to discharge to the septic tank and leaching field. Sanitary drainage from uncontaminated areas discharges to the septic tank and leaching field without intermediate holdup.

Drainage from normally nonradioactive fixtures, such as lavatories in the contaminated washroom, showers and nonradioactive sinks is collected in a basin and pumped to the waste disposal area for monitoring and, if necessary, for treatment prior to final disposal.

Drainage from fixtures and areas handling radioactive wastes is collected by a system of accessible and replaceable piping to a tank and pumped to the waste disposal system for monitoring and treatment prior to disposal.

Potable Water

Cold water for use in showers, toilet rooms and for drinking purposes is taken from a well.

Well water, softened if necessary, is used as the source of hot water for use in sinks, showers and for the laundry.

During normal plant operation, hot water is heated by extraction steam and, during plant shutdowns, by low pressure steam from the auxiliary steam generator.

Water for sinks and showers is heated to 140 F in a hot water storage heater, and water for the laundry is heated to 180 F in a separate hot water storage heater.

217 FIRE PROTECTION SYSTEM

Yard Hydrants

Yard hydrants are served by an underground fire protection line connected to the discharge of fire pumps.

Valved branches from the yard main serve the lubricating oil room, turbine generator and interior hose station systems.

Transformer Fire Protection

The plant transformers are not considered to be a source of trouble based on past experience. The ground area under and around the transformers is covered with sufficient crushed rock to contain and cool insulating oil in the event of a leak. Fire hydrants with hose and fog nozzles and portable fire fighting equipment are located within reach of the transformer area.

Lubricating Oil Room

An automatic system of water spray fire protection with deluge valve is provided for the central lubricating oil room.

Area Beneath Turbine Generator

A system of water spray fire protection is provided beneath the turbine generator in the vicinity of the seal oil tank and lubricating oil tank. This system is manually operated, with valves located at a distance from the protected equipment and in areas shielded from potential fires.

Interior Hose Station Protection

Fire hose stations are provided for interior areas of the plant, except within the vapor container. Hose stations generally consist of hose reels and 75 ft of noncollapsible rubber hose with fixed water spray nozzles, suitable for extinguishing oil fires.

In areas such as offices where oil fires are unlikely and where space is limited, the hose stations consist of hose racks with unlined linen hose and solid stream nozzles mounted in cabinets.

Water for the interior hose stations is taken from valved connections to the yard fire protection main.

Hose station protection for the radioactivity laboratory is omitted in order to minimize the possible spread of radioactive contamination which might prevail if a hose stream were used for extinguishing a fire in this area.

No hose stations for fire protection duty are provided for the interior of the vapor container since this area is inaccessible during operation. During maintenance periods, portable extinguishers are used inside the vapor container.

Source of Water

Two vertical turbine type fire pumps, located in the screen well structure, take water from Sherman Pond. Each pump has a nominal capacity of 1,000 gpm at 100 psi gage and is driven by an electric motor.

Pressure is maintained on the yard fire protection piping constantly by a make-up pump of approximately 75 gpm capacity, arranged to start automatically when the pressure in the yard piping drops below a predetermined value. A hydropneumatic tank with compressed air supply is connected to the yard piping and is provided with level and pressure controls to actuate the make-up pump, main fire pumps and facilities for admission of compressed air.

Electric power for the main fire pumps is supplied through two buses, one of which is always energized. One pump is connected to one bus and the other pump is connected to the second bus.

Portable Extinguishers

Portable carbon dioxide and pressurized water extinguishers are distributed within the plant for extinguishing small fires normally.

Codes and Standards

All fire protection systems and equipment conform to local codes and the standards of the National Board of Fire Underwriters.

218 CORROSION CONTROL SYSTEM - HYDROGEN INJECTION

Function

One function of the corrosion control system is to introduce hydrogen gas into the main coolant for the purpose of preventing excessive corrosion of components of the primary plant. Other methods of corrosion control may become necessary pending the outcome of the Research and Development Program.

The hydrogen injection system operates intermittently throughout the life of the plant.

The system under normal conditions operates with automatic pressure differential control but it is designed so that it may be operated both automatically and manually.

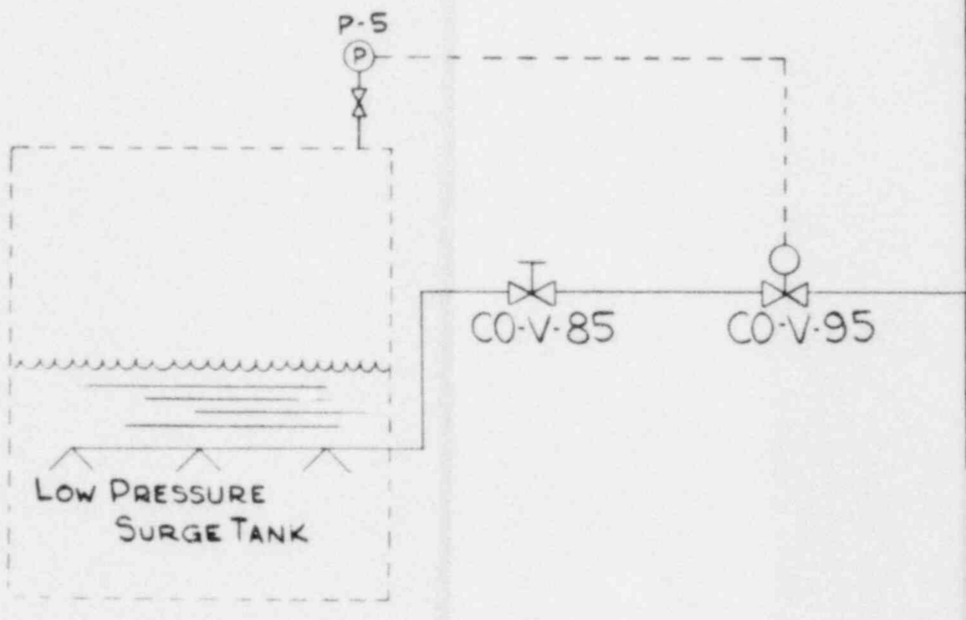
General Description

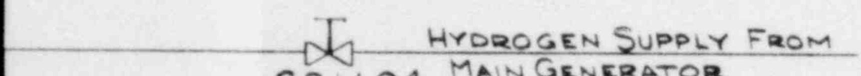
The system, shown on drawing 646-J-431, consists of one hydrogen gas feeder pipe drawing main generator quality hydrogen from the turbine electric plant. This single line contains the system isolation valves, an automatic pressure control valve and in the surge tank, a gas dispersing nozzle with necessary taps.

Basis for Design

The system injects hydrogen gas into the surge tank vapor space up to a partial pressure of 30 psia. At this pressure, and at the normal operating temperature of the water in the tank of 120 to 130 F, 25 to 30 cc (STP) of hydrogen are dissolved in 1 kg of water. This concentration of hydrogen reduces the corrosion rate of the metal surfaces of the primary plant to or below the design rate of 10 mg per sq dm per month.

The principal mechanism that operates this system is a pressure control valve, which is set to maintain any set pressure in the tank from atmospheric pressure to 30 psia. The surge tank pressure may, in the course of plant operation, gradually and periodically develop an internal pressure exceeding 30 psia due to release of fission product and fission product decay gases that collect in the main coolant. When the internal pressure reaches 40 psia, a relief valve operates to blowdown the tank to a pressure near 20 psia. This action, in turn, trips the pressure control valve, admitting hydrogen to the tank until the 30 psia minimum is obtained. The pressure control valve is located near the surge tank.


P-5
P
CO-V-85
CO-V-95
LOW PRESSURE
SURGE TANK



PRELIMINARY

WESTINGHOUSE ELECTRIC CORPORATION				
TITLE YANKEE ATOMIC ELECTRIC COMPANY				
CORROSION CONTROL SYSTEM-HYDROGEN INJECTION				
DFTM. SHAFFER	12-5-56	APPD. I.G. STUDDART	1/4/57	646J431
DIV. C.A.P.A.		PITTSBURGH, PA.		

204-205

A manually operated globe valve is provided in the system line, connected in series with the pressure control valve. This valve isolates the hydrogen supply in the event that the pressure control valve fails and is readily accessible for this purpose.

The spray nozzles is located near the normal water surface in the tank to facilitate attaining equilibrium concentrations of dissolved hydrogen.

The system is designed to ASA B31.1-1955, Code for Pressure Piping, Sections 2 and 6.

219 SAFETY INJECTION-SHIELD TANK CAVITY SYSTEM

Function

The functions of the safety injection-shield tank cavity system are to supply borated water for flooding the shield tank cavity during refueling operations, and to supply borated water to the reactor vessel for cooling of the core in the unlikely event of a major loss of water accident.

General Description

The system consists of a safety injection-shield tank cavity water storage tank, two dual purpose pumps and miscellaneous piping, valves and fittings, as shown on drawing 9699-QM-1. Remotely operated pumps and valves permit control of this system from the control room.

Basis for Design

The system is sized for handling 110,000 gal of demineralized water containing 1.6 g of boron per liter as boric acid. This volume of water is sufficient for flooding the shield tank cavity to a depth of 25 ft, providing 15 ft of shield water over fuel assemblies while they are transferred to the spent fuel pit during refueling operations. One of the 1,200 gpm injection-fill pumps provides for mixing the stored boric acid solution, filling the shield tank cavity in approximately 1 1/2 hr, and pumping shield tank cavity water to the waste disposal system for cleanup if it should become slightly contaminated when it is mixed with the main coolant in the shield tank cavity during the refueling operation.

The safety injection function of the system is accomplished by using shield tank cavity water storage and fill equipment. Safety injection is provided to each of the four main coolant lines outboard of the main stop valves in order to cool the core following a main coolant system rupture of any size which can not be compensated for by the charging system pumps. Cooling is provided to prevent core meltdown due to decay heat.

The safety injection system is started manually, but with partially automatic follow-through thereafter. To minimize the chances of erroneous start-up, a single covered starting switch is provided. System functioning will occur only when the reactor pressure falls below the shutoff head of the safety injection pumps, approximately — psi gage.

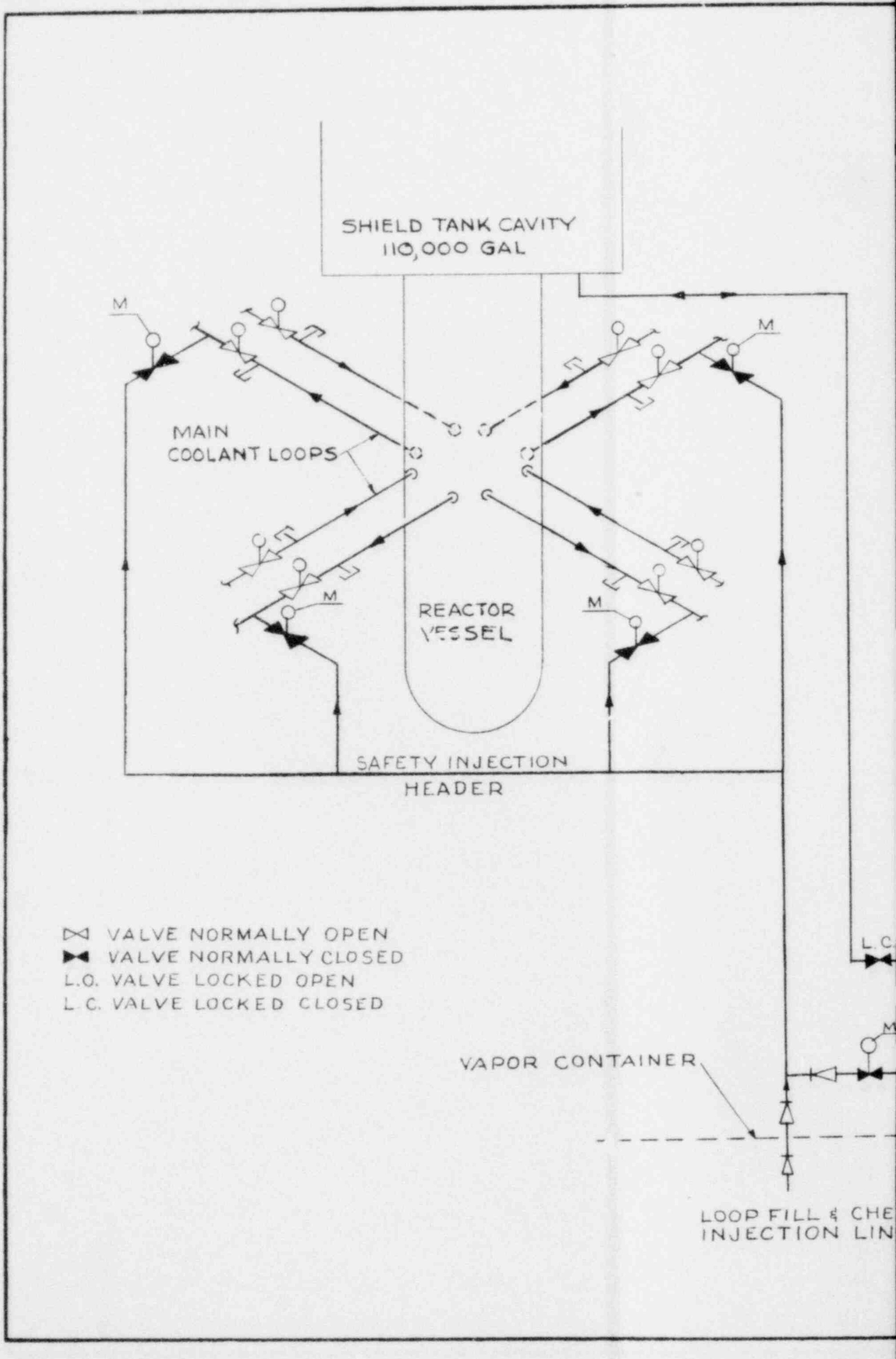
Injection with two pumps at a rate of 2,400 gpm fills the reactor vessel to the top of the core in approximately 3 1/2 min. Assuming that 2 min are required for the initiation of the system, this action prevents core meltdown even after an assumed instantaneous loss of all main coolant.

The injection flow rate is sized to provide for the loss of 25 per cent of the total pump discharge through a single ruptured injection line or main coolant pipe. Adequate missile protection is provided for the safety injection header, and the individual injection lines are divided compartmentally by reinforced concrete partitions. After the reactor vessel is filled to capacity following the rupture, the 1,200 gpm injection flow from one pump is adjusted remotely by control valve arrangement to replace just the water in the reactor vessel that is boiled off into the vapor container by the release of decay heat.

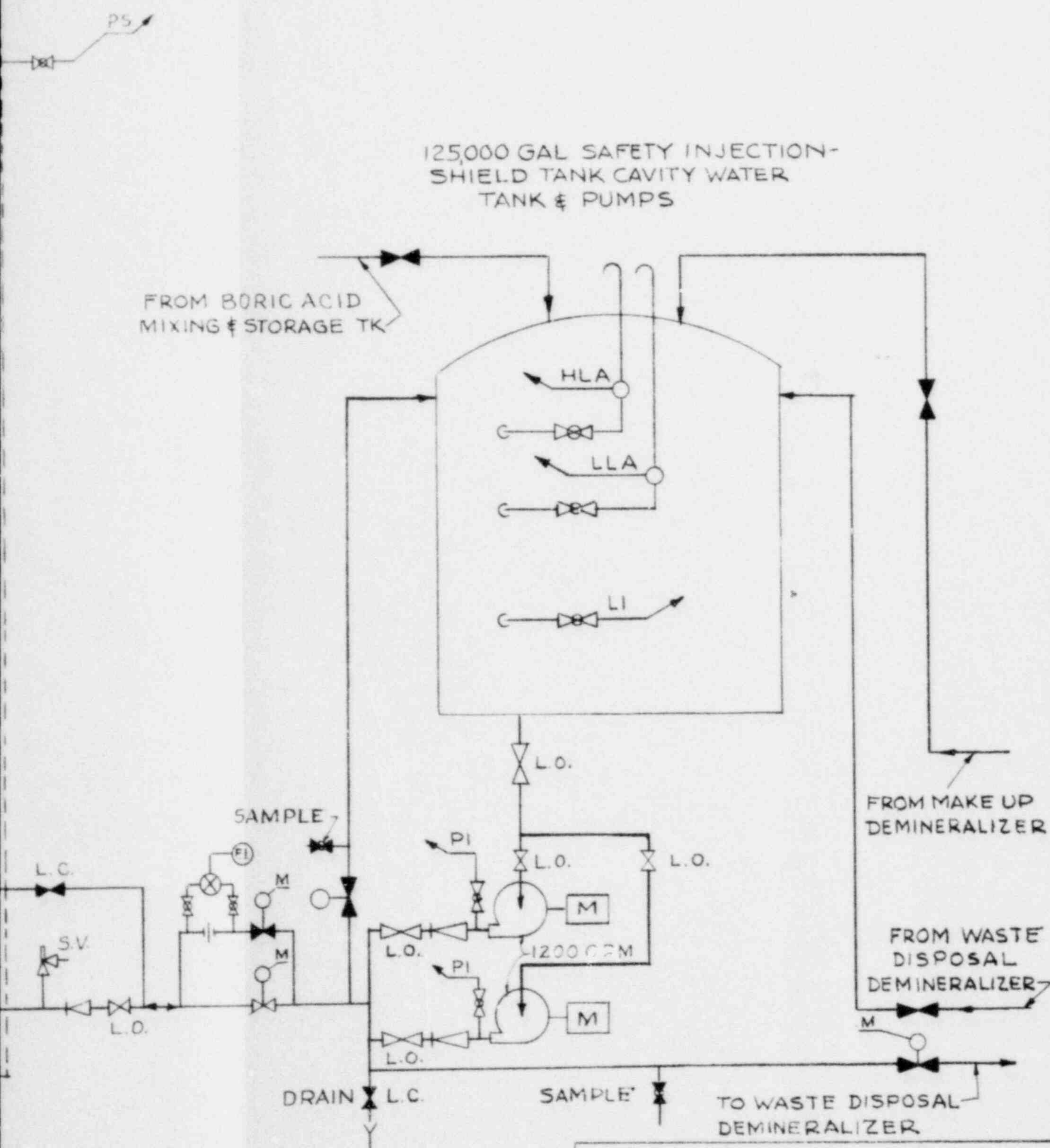
The 125,000 gal safety injection-shield tank cavity storage tank provides sufficient water to replace decay heat losses for approximately 300 hr after reactor shutdown. The tank is refilled, if it should be necessary, to continue borated water injection at rates less than 5 gpm for more than 300 hr. The vapor container is designed to hold 4,500,000 lb, approximately 580,000 gal, of safety injection water.

Maximum system reliability is provided by independent power supplies to each safety injection pump as shown on drawing 9699-QE-1. One pump is supplied by bus section and transformer connected to the Harriman 115 kv transmission line and the other from a similar bus section and transformer connected to the Millbury 115 kv line. These power supplies are not only essentially independent of each other but are entirely separate from a third source of station power, a transformer connected to the turbine generator leads. Automatic switching is provided to pick up any section of the station service bus in the event of a power failure in approximately one-third of a second. While details of the electrical diagram are not finally settled at this time, these concepts will be adhered to and the final scheme will be as reliable as that shown on drawing 9699-QE-1.

The motor operated valves of the safety injection system operate on 125 v d-c station battery supply. Operating controls of all valves and motors for the safety injection system



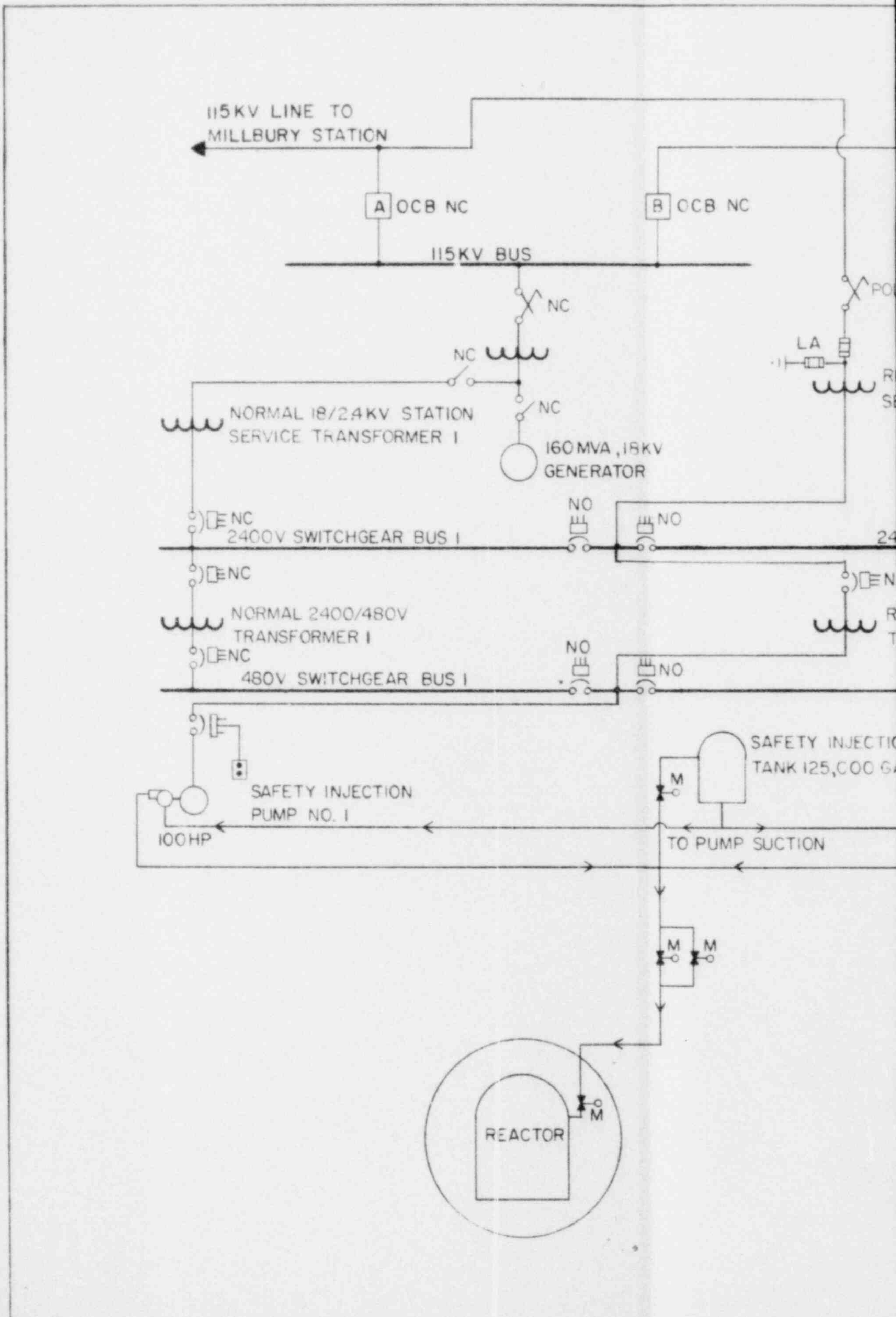
SHIELD TANK CAVITY
110,000 GAL
M
M
M
M
MAIN
COOLANT LOOPS
REACTOR
VESSEL
SAFETY INJECTION
HEADER
VALVE NORMALLY OPEN
VALVE NORMALLY CLOSED
L.O. VALVE LOCKED OPEN
L.C. VALVE LOCKED CLOSED
VAPOR CONTAINER
LOOP FILL & CHE
INJECTION LIN

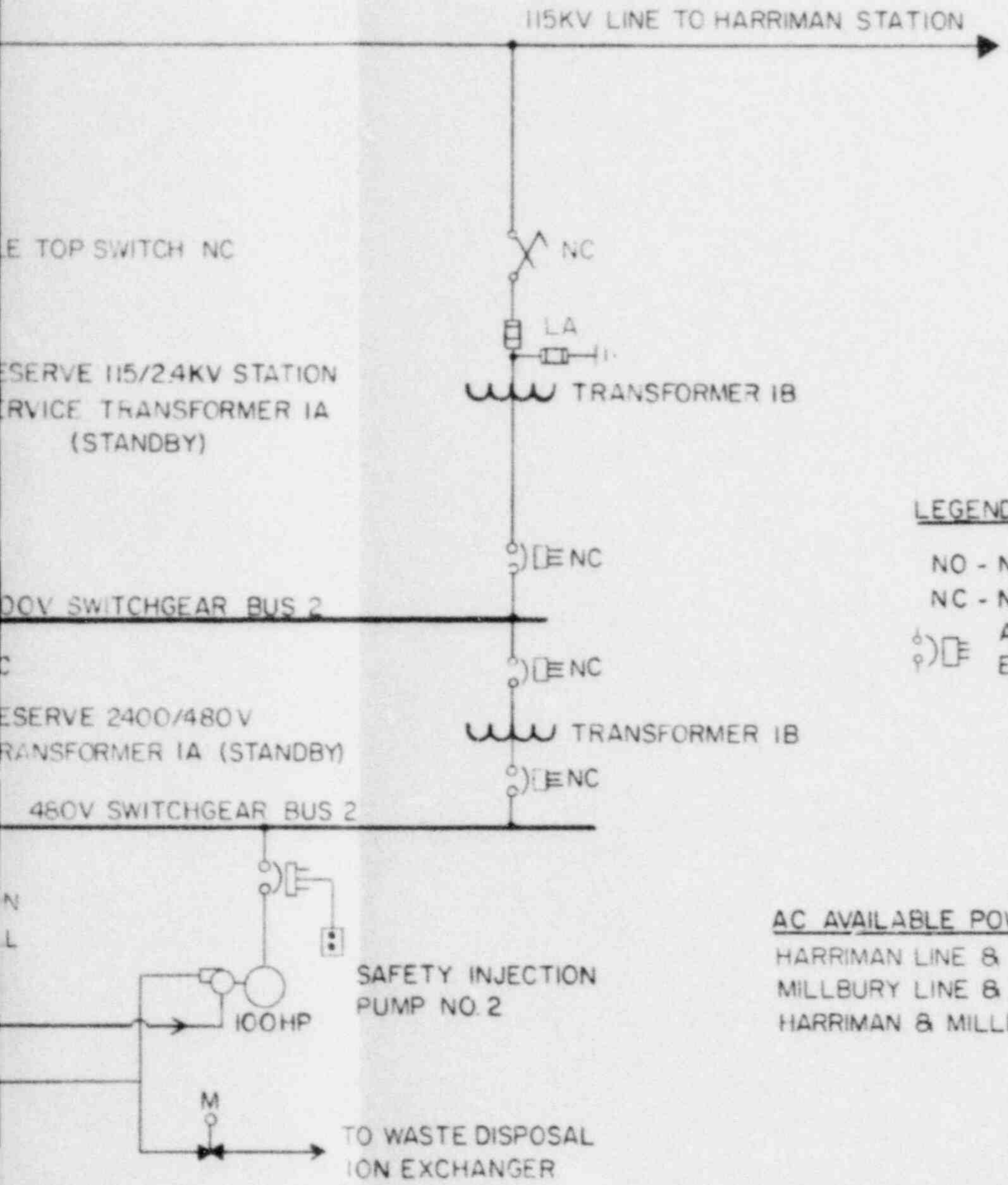


PRELIMINARY

SAFETY INJECTION-SHIELD TANK
CAVITY SYSTEM
YANKEE ATOMIC ELECTRIC COMPANY
STONE & WEBSTER ENGINEERING CORPORATION
7/15/57
9699-QM-1

208-209


115KV LINE TO
MILLBURY STATION
A OCB NC
B OCB NC
115KV BUS
NC
NC
NC
LA
NORMAL 18/2.4KV STATION
SERVICE TRANSFORMER I
160 MVA, 18KV
GENERATOR
NC
NO
NO
2400V SWITCHGEAR BUS I
NC
NORMAL 2400/480V
TRANSFORMER I
NC
NO
NO
480V SWITCHGEAR BUS I
SAFETY INJECTION
PUMP NO. 1
100HP
M
TO PUMP SUCTION
M
M
M
REACTOR



LEGEND

NO - NORMALLY OPEN
NC - NORMALLY CLOSED
AIR CIRCUIT BREAKER
ELECTRICALLY OPERATED

AC AVAILABLE POWER SUPPLIES-FULL LOAD

HARRIMAN LINE & GENERATOR
MILLBURY LINE & GENERATOR
HARRIMAN & MILLBURY LINES NO GENERATOR

NOTES

PUMPS-3Φ 440V AC MOTOR DRIVEN
VALVES-125V DC MOTOR OPERATED

POWER SUPPLY-SAFETY INJECTION SYSTEM

YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.

STONE & WEBSTER ENGINEERING CORPORATION

7-15-57

9699-QE-1

2.12-213

are grouped on one starting switch so that a single operation energizes all components of the system.

At periodic intervals, the system pumps and motor operated valves are individually operated and checked, and the safety-injection water sampled and analyzed for boron concentration.

Throughout the period of system operation, an operator is available in the control room to run the system manually in conformance with pre-established drills and procedures and as assisted by suitable plant instrumentation, if in his judgment it is necessary.

220 SAMPLING SYSTEM

Function

The function of the sampling system is to take samples periodically of the main coolant for evaluation of pH, conductivity, boric acid concentration, and dissolved hydrogen gas concentration. Sampling is a manual operation, except for remotely controlled isolation valves in inaccessible areas.

The system operates intermittently throughout the life of the plant.

General Description

The sampling system consists of three sampling lines: the first, from the inlet header of the purification system demineralizers; the second, from the outlet header of the demineralizers; and the third, from inside the isolation valves of the purification system and from the drain header for each of the main coolant loops. Each line contains an isolation valve and a combination isolation-needle valve, and terminates in the sampling cubicle which is located in the plant laboratory. The third line contains a cooler to lower the liquid sample temperature from about 500 F to about 70 F. All isolation valves in this line are required to withstand 2,500 psi gage. The manually operated needle-isolation valve is provided as back-up for the isolation valves.

A ventilation hood is provided for venting radioactive gases which might be released from the sample to the waste disposal system. A sink is provided to retain and conduct all water purged from the system to the waste disposal system prior to taking samples. The hood and sink are located in the sampling cubicle.

Basis for Design

The system is capable of removing 1 gpm of water from either the inlet or outlet demineralizer headers or from the shutdown cooling and main coolant systems.

The system is designed so that the condition of each demineralizer resin bed can be determined independently by manual analysis. The take-off lines connect to the terminal ends of the headers and not between the demineralizers.

The design pressure for the shutdown cooling and main coolant sample line is 2,500 psia, and in the other two lines, it is 150 psi gage. All sample lines discharge to atmospheric pressure downstream of the needle valves.

The design temperature is 140 F in all lines, except the line for the shutdown cooling and main coolant systems, when the design temperature is 500 F.

All material used in the system is fabricated of Type 304 stainless steel or equivalent.

221 VENT AND DRAIN SYSTEM

Function

The vent and drain system is designed to provide suitable facilities for discharging all radioactive fluids to the waste disposal system during filling draining, and flushing of the main coolant system, isolated loops or reactor plant auxiliary systems; to provide a suitable means for discharging radioactive water and gases from relief and safety valves to the low pressure surge tank; to provide totally enclosed facilities for venting radioactive gases from the primary plant and its auxiliaries to the waste disposal system; and to provide a suitable means for venting air from the primary plant and its auxiliary systems.

General Description

The vent and drain system, as shown on drawing 646-J-428, employs the following equipment:

High pressure drain header to drain the isolated main coolant loop, the shell side of the steam generator, and the tube and shell side of the feed and bleed heat exchanger.

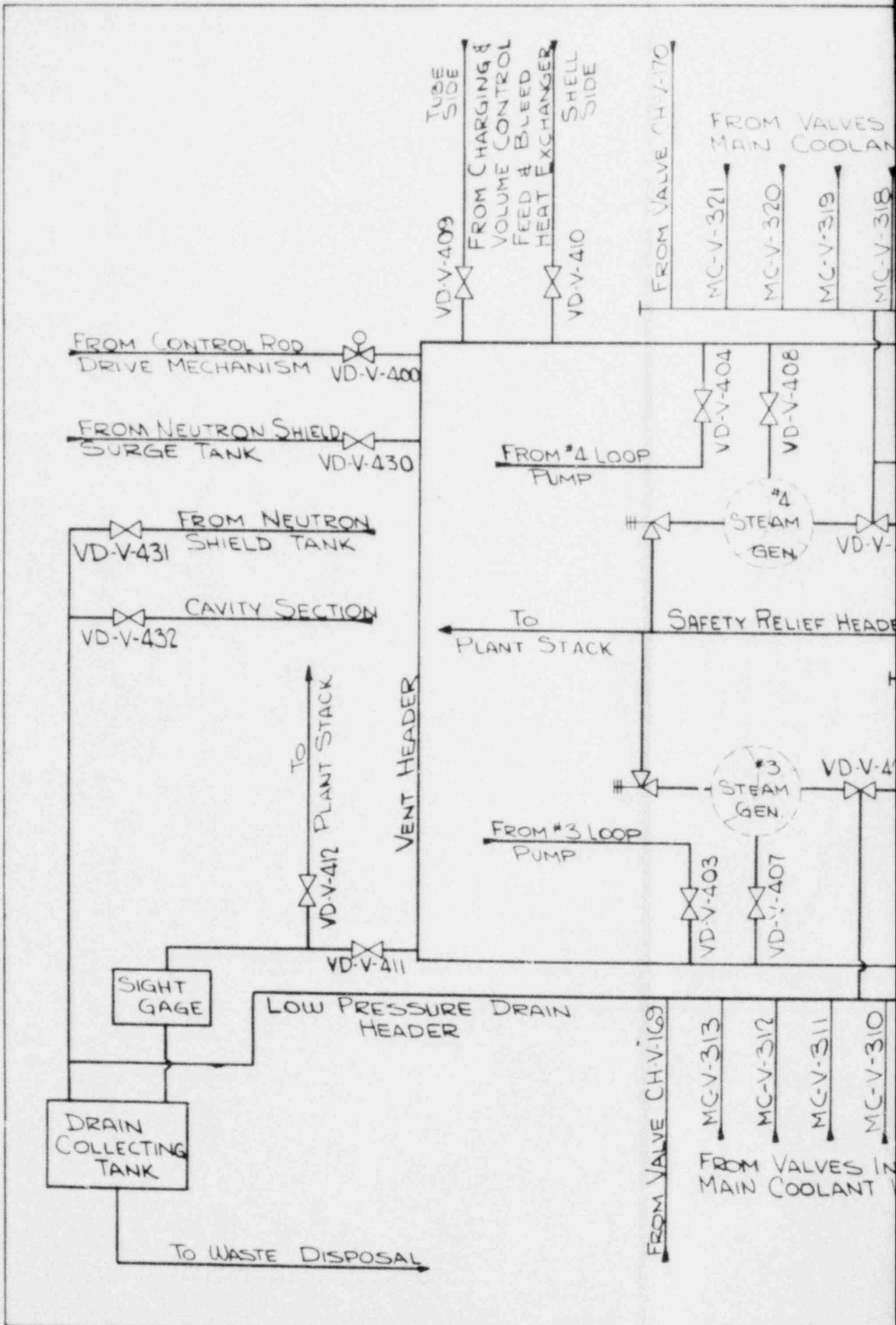
Low pressure header to drain the lantern ring valve stem glands of valves in the main coolant system, charging and volume control system, high pressure drain header, and pressure control and relief system.

Vent header to collect the vents from the main coolant pumps, feed and bleed heat exchanger, shell side of steam generator, neutron shield tank, and reactor vessel.

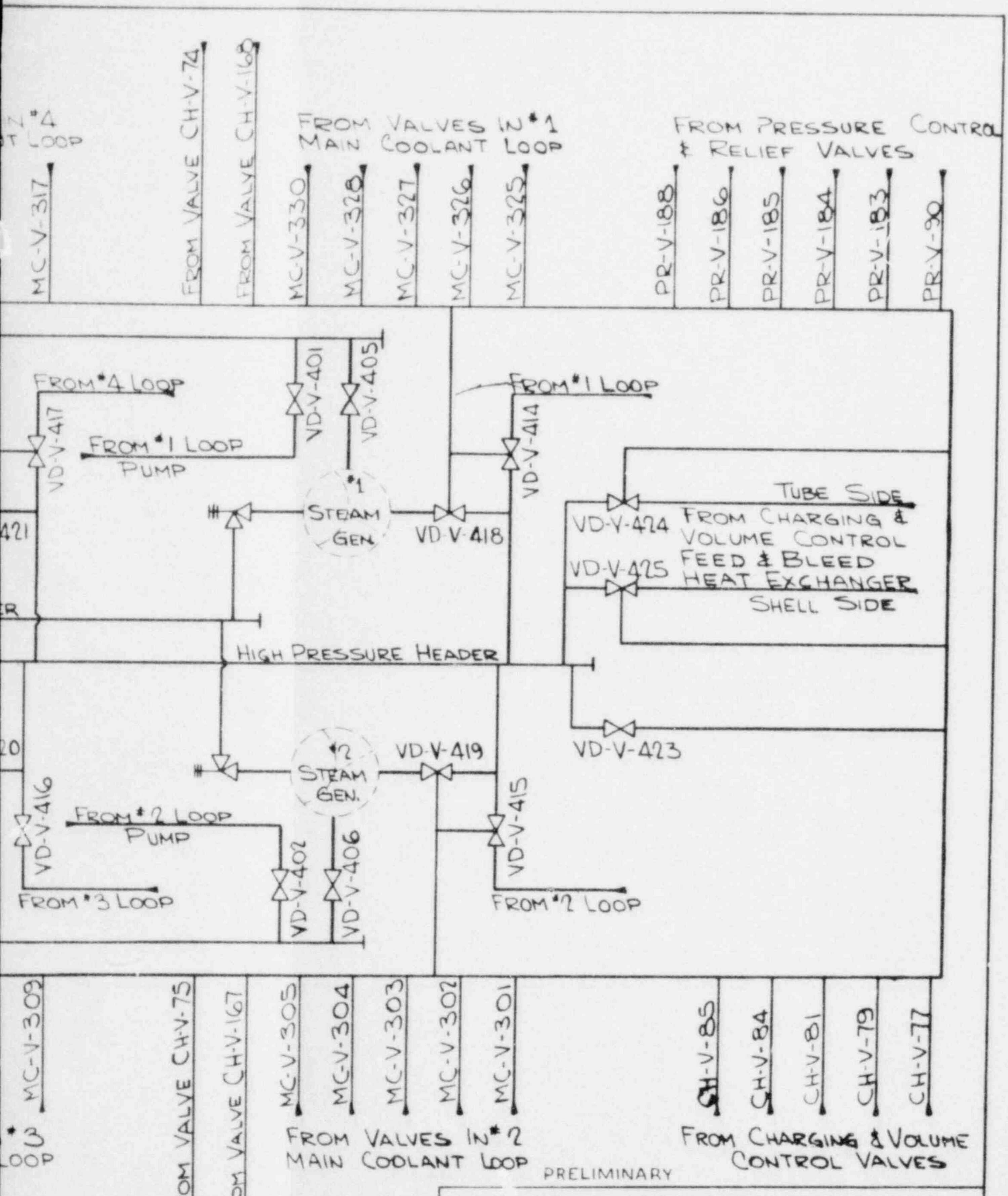
Safety relief valve header to collect the blowdown of the safety relief valves on the steam side of the steam generator and discharge it to the plant stack.

The high pressure header discharges to the low pressure header through a manual stop valve, which acts as a backup valve for each valve connected with the high pressure drains. This is consistent with the practice of backing up all stop valves between high and low pressure systems.

The main loop relief valves, the high pressure valve lantern rings, and the cavity section between the reactor vessel and the neutron shield tank drains to the low pressure header. The lantern ring valve stem gland drainage is from those valves which have high pressure on both sides of the valve disc. Those



TUBE SIDE
FROM CHARGING & VOLUME CONTROL FEED & BLEED HEAT EXCHANGER
SHELL SIDE
VD-V-409
VD-V-410
FROM VALVE CH-V-170
FROM VALVES IN MAIN COOLANT
MC-V-321
MC-V-320
MC-V-319
MC-V-318
FROM CONTROL ROD DRIVE MECHANISM
VD-V-400
FROM NEUTRON SHIELD SURGE TANK
VD-V-430
VD-V-404
VD-V-408
FROM #4 LOOP PUMP
#4 STEAM GEN
FROM NEUTRON SHIELD TANK
VD-V-431
CAVITY SECTION
VD-V-432
TO PLANT STACK
SAFETY RELIEF HEADER
TO PLANT STACK
VENT HEADER
#3 STEAM GEN.
FROM #3 LOOP PUMP
VD-V-412
VD-V-403
VD-V-407
VD-V-411
SIGHT GAGE
LOW PRESSURE DRAIN HEADER
FROM VALVE CH-V-169
MC-V-313
MC-V-312
MC-V-311
MC-V-310
DRAIN COLLECTING TANK
FROM VALVES IN MAIN COOLANT
TO WASTE DISPOSAL



PRELIMINARY

WESTINGHOUSE ELECTRIC CORPORATION

TITLE YANKEE ATOMIC ELECTRIC COMPANY

VENT & DRAIN SYSTEM

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DIV. C.A.P.A. | PITTSBURGH, PA.

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valves which have high pressure on the inlet and low pressure on the outlet side do not require lantern ring gland drainage as the leakage past the valve disc drains to the low pressure system. Manual stop valves to control the drainage are installed in the drain line from the cavity section between the reactor vessel and the neutron shield tank and from the neutron shield tank. The low pressure drain header discharges to the drain collecting tank.

The vent connections are located at the highest possible points on the primary plant equipment. Vent header connections provide for venting to the plant stack and discharging to the drain collecting tank.

The reactor vessel is vented through the vent connections of the control rod motor mechanisms. Twenty-three mechanisms discharge to the vapor container. A line with a remotely controlled stop valve is installed on the remaining motor mechanism, which permits controlled venting when personnel can not enter the vapor container. This vent line discharges to the vent header. The maximum rate of vent gases is discharged when the main coolant system is being filled.

The safety valves on the steam side of the steam generator are connected to a line which discharges to the plant stack.

The vent and drain section piping is designed in accordance with ASA B31.1-1955 Sections 1 and 6.

222 COMPONENT COOLING SYSTEM

Function

The functions of the component cooling system are:

To remove heat from the various reactor plant components in order to maintain them at their required operating temperature and to transfer the absorbed heat from the component cooling water to a raw water supply.

General Description

The component cooling system, shown on drawing 646-J-424, consists of two coolers, two circulating pumps, a surge tank, piping, valving, and fittings. This equipment is connected to two main piping headers from which branch lines are connected to the equipment being cooled. River water is used to cool the component cooling water. The coolers, pumps, and surge tank are located outside the vapor container.

Both pumps and coolers are full size and cross connected to protect against reactor plant shutdown in the event of failure of one of the pumps or coolers.

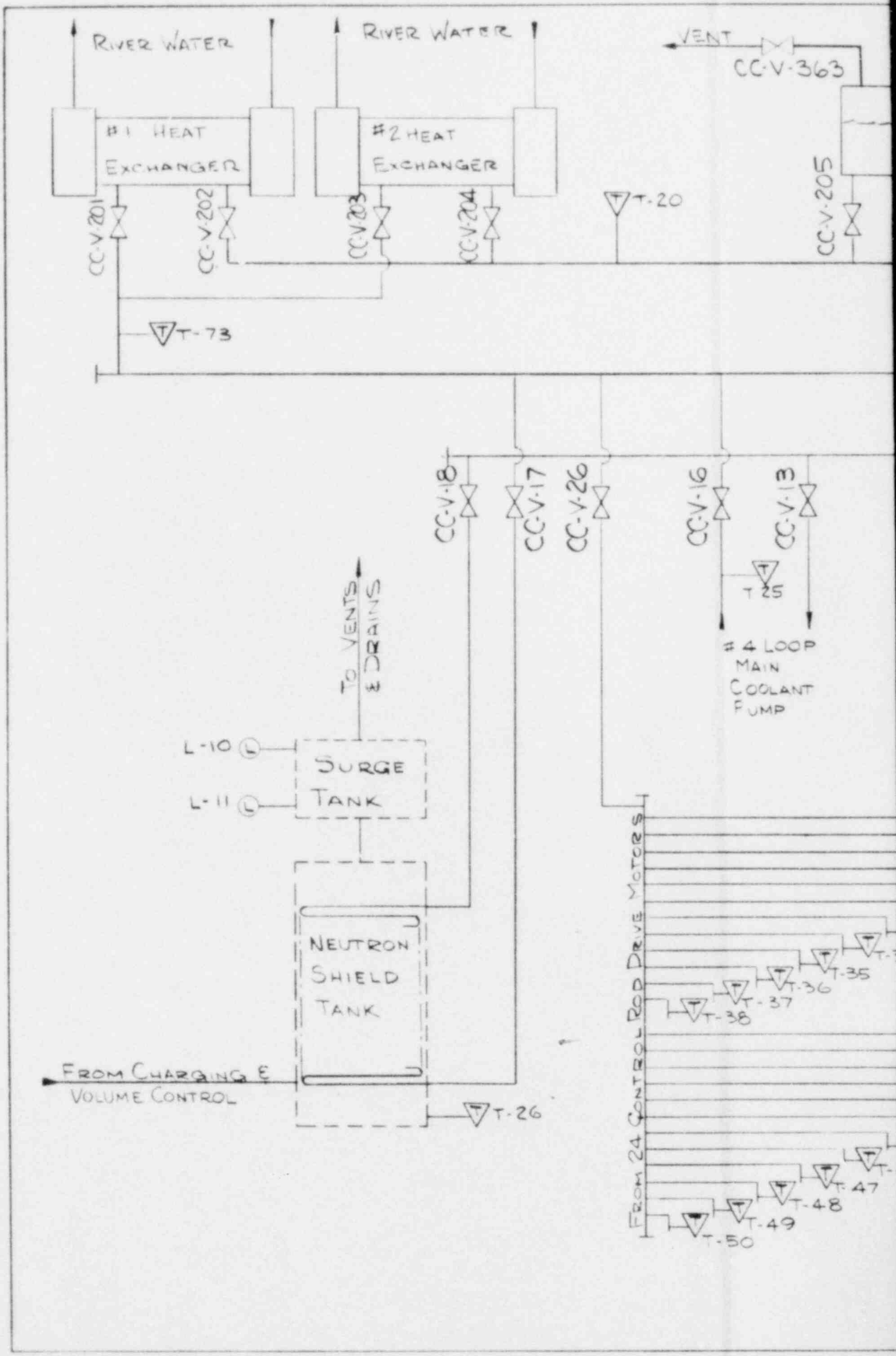
Basis for Design

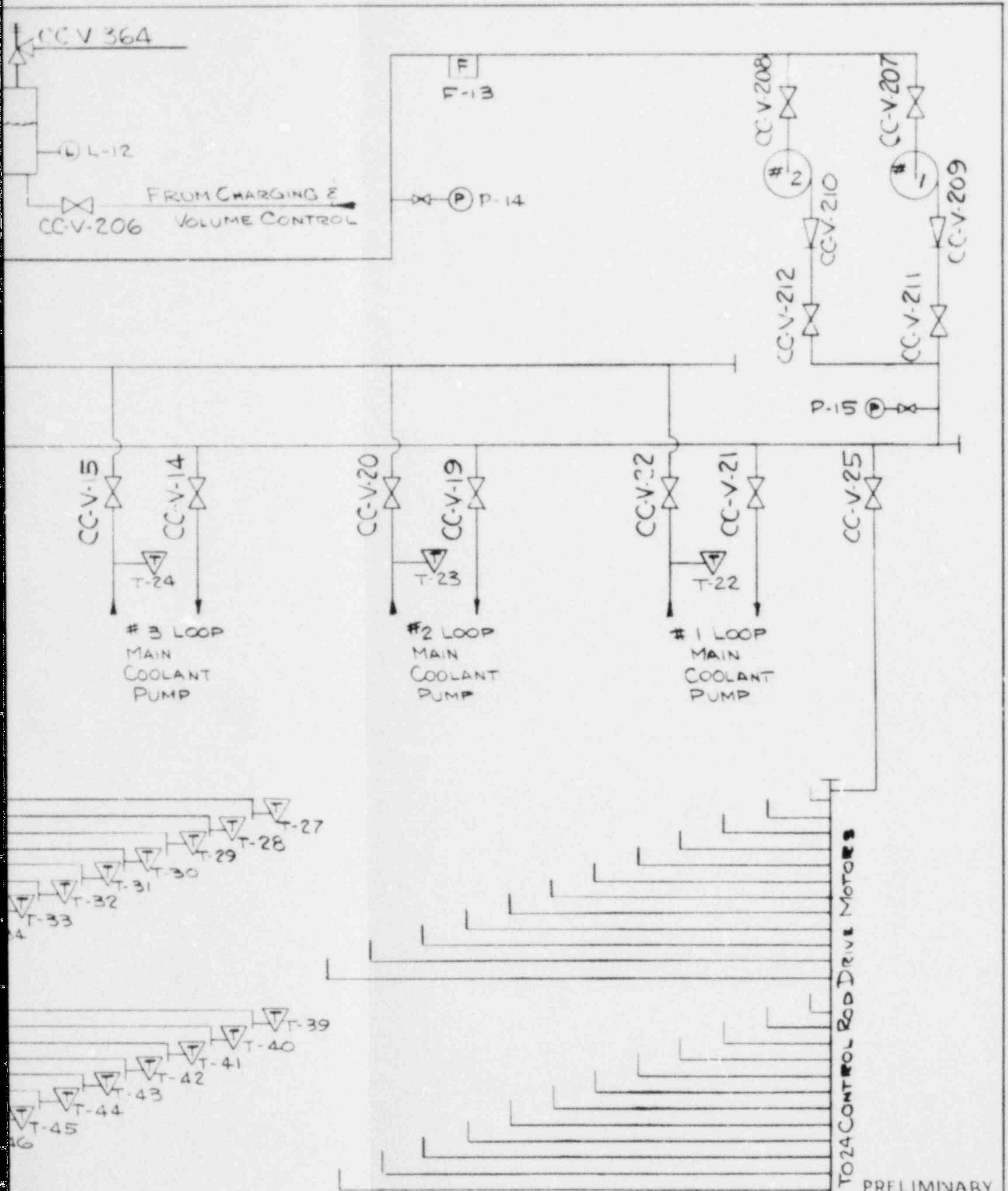
Two motor driven circulating pumps are employed in the component cooling system, one of which is a spare. Each pump can be isolated from the line for repairs. A check valve is used in the discharge line of the pumps to prevent backflow. The pump motors have independent power supplies and, in the event of failure of the operating pump, the stand-by pump starts automatically by a pressure switch on the pump discharge header.

Two coolers are provided to transfer heat from the component cooling water to the raw water, one of which is a spare. Each heat exchanger is designed for the full cooling capacity at normal plant operation.

River water enters the tube side of the cooler at 65 F and discharges from the cooler at approximately 80 F. Cooling water enters the shell side at 125 F and discharges at approximately 65 F.

A surge tank is used in the component cooling system to provide make-up water and to accommodate for the expansion and the contraction of the water in the system. The water level in the tank is maintained by level control with high and low water level alarms.


RIVER WATER
RIVER WATER
VENT
CC-V-363
#1 HEAT EXCHANGER
#2 HEAT EXCHANGER
CC-V-201
CC-V-202
CC-V-203
CC-V-204
T-20
CC-V-205
T-73
CC-V-18
CC-V-17
CC-V-26
CC-V-16
CC-V-13
T-25
#4 LOOP MAIN COOLANT PUMP
TO VENTS & DRAINS
L-10
L-11
SURGE TANK
NEUTRON SHIELD TANK
FROM CHARGING & VOLUME CONTROL
T-26
FROM 24 CONTROL ROD DRIVE MOTORS
T-35
T-36
T-37
T-38
T-47
T-48
T-49
T-50



PRELIMINARY

WESTINGHOUSE ELECTRIC CORPORATION

TITLE YANKEE ATOMIC ELECTRIC COMPANY

COMPONENT COOLING SYSTEM

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The system is initially filled from the charging and volume control system with demineralized water. A corrosion inhibitor is used in the component cooling water to minimize corrosion.

The maximum allowable working pressure of the system is 125 psi gage and the design temperature of the system is 250 F. Material in contact with the component cooling water is carbon steel, or equivalent, and all material in contact with the main coolant conforms to ASTM A-312, Grade TP 304 or equivalent.

Suitable valving is provided to isolate the equipment being cooled from the component cooling system.