UNITED STATES ATOMIC ENERGY COMMISSION

PRELIMINARY HAZARDS SUMMARY REPORT

Part B. License Application

April 1957

Yankee Atomic Electric Company
Boston, Massachusetts

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LICENSE APPLICATION

(Docket No. F-29)

Part B

TECHNICAL INFORMATION

HAZARDS SUMMARY REPORT

ACKNOWLEDGEMENT

This report was prepared as a joint effort by the personnel of Yankee Atomic Electric Company, Westinghouse Electric Corporation, and Stone & Webster Engineering Corporation. Special technical assistance was provided by Professor James M. Austin and Dr. Theos J. Thompson of Massachusetts Institute of Technology and Dr. Shields Warren of the New England Deaconess Hospital.

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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 525 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.

The reactor is controlled at operating temperature by 24 neutron absorbing control rods. In order to control excess reactivity in the cold reactor made available by its negative temperature coefficient, a neutron absorbing chemical compound is dissolved in the coolant moderator. Inherent stability and control is provided by the negative temperature coefficient of the reactor, and because the coolant moderator is operated near boiling temperatures. During large transient increases in reactivity, the moderator voids, caused by boiling, decrease reactivity. Simultaneously, the Doppler coefficient acts to limit more rapid reactivity transients.

A general view of the Yankee Plant and a simplified flow diagram of the primary and secondary systems are shown in Figures 1 and 2, respectively.

FIG. 1

GENERAL VIEW OF REACTOR PLANT

FIG. 2

SIMPLIFIED FLOW DIAGRAM PRIMARY AND SECONDARY SYSTEMS

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 CORE CROSS SECTION

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 $1/4$ 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 $1\frac{3}{4}$ in. and $1\frac{1}{4}$ 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 $1.03 \times 10^{-4} \Delta k/k$ 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:

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

$F_{\Delta T}$ 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 as 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 1MECHANICAL DESIGN DATA - INITIAL CORE

Total heat output, Btu/hr	$1,338 \times 10^6$ (392 mw)
<u>Coolant Flow</u>	
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
<u>Pressure Drop</u>	
Total across vessel, psi	28
Across core, psi	14
<u>Temperatures, F</u>	
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
<u>Heat Transfer</u>	
"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
<u>Hot Channel Factors</u>	
Heat flux	5.17
Film drop	7.36
Coolant rise	3.36
<u>Fuel Rod (Dimensions at room temperature)</u>	
Outside diameter, in.	0.336
Gap thickness, in.	0.003
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
<u>General</u>	
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
<u>Weights</u>	
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 $1,338 \times 10^6$ Btu per hr. The maximum reactor main coolant loop system capability is designed for 492 mw of heat or $1,680 \times 10^6$ Btu per hr. All of these values are based on four loop operation of the plant.

Nuclear Design of the Reactor CoreGeneral

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:

ρ_w = density of water
 V_w = volume of water

V_u = volume of uranium
 V_s = volume of structural material

Table 2

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NUCLEAR DESIGN DATA - INITIAL CORE

(Based on 525 F average water temperature)

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	525
Core average diameter, in.	74.4
Core active height, in.	90
Reactor buckling, B^2 , cm ⁻²	.000722*
<u>Active Volumes</u>	
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
<u>Weights</u>	
Fuel, lb	53,500**
Stainless steel, lb	9,000
Zircaloy-2, lb	2,980
Total, lb	65,500
Volume ratio, H ₂ O/UO ₂	1.33**
Volume ratio, H ₂ O/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
<u>Typical Performance Data</u>	
Initial conversion ratio	0.733
Cumulative conversion ratio	0.636
<u>k_{eff}</u>	
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
<u>Fraction of energy from:</u>	
U-235	0.704
U-238	0.068
Pu	0.228
<u>Nuclear Parameters</u>	
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 $J_0^2(B_{TR})$ where $J_0(B_{TR})$ 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,

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

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	525 F
Contribution from:		
Fast effect	$+0.5 \times 10^{-5}/F$	$+4 \times 10^{-5}/F$
Resonance escape	-2.5	-33
Thermal Utilization		
Without chemical neutron absorber	-0.3	+6
Leakage	<u>-0.4</u>	<u>-4</u>
Total, without chemical neutron absorber	$-2.7 \times 10^{-5}/F$	$-27 \times 10^{-5}/F$

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

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, 525 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 $+1 \times 10^{-3}/^{\circ}\text{C}$ (Nucleonics Vol. 10, No. 5, 64, 1952). Differentiating the expression for resonance escape with respect to temperature, the following expression is obtained:

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

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

Table 5

The Doppler Coefficients

$-.7 \times 10^{-5}$ per deg F, at 68 F

$-.8 \times 10^{-5}$ 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 sub-cooled 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. At operating temperatures, 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

FIG. 9

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

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 $3.0 \times 10^{-5} \Delta k$ per sec. This is approximately 1/3 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. The reactor is always subcritical at boron concentrations which could result in a positive void coefficient, even with all control rods withdrawn.

Effects of Plutonium Buildup

At the end of the core life, it is anticipated that approximately one-third of all fissions take place in the plutonium that has built up following capture of neutrons by U-238. Since plutonium has a wealth of resonance structure, it should make a contribution to the Doppler coefficient as should U-235 which has a number of fission resonances. 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. Also, the Doppler coefficient of a mixture of U-238 and U-235 does not go

UNIFORM VOID COEFFICIENT vs. BORON CONCENTRATION

to zero until the proportions are 1 to 1; thus the combined effect of U-235 and Pu-239 resonances should be small in the Yankee reactor where their combined concentrations will be less than 3 per cent of the U-238 present. Therefore, it is concluded that, throughout the core lifetime, there is no significant change in the overall Doppler coefficient due to the buildup of plutonium. The presence of higher plutonium isotopes should not change this conclusion because any positive contribution from fissionable Pu-241 should be nullified by the purely absorption resonance in Pu-240 at 1 electron volt. Similarly, investigations indicate that the contribution of resonance absorption in plutonium to the temperature coefficient is a negative effect. After 10,000 hr of full power operation, the magnitude of this effect is -4.6×10^{-5} $\frac{\Delta k}{k}$ per deg F.

The effect of plutonium in the reactor on the delayed neutrons and the consequent effect upon accidents is being investigated. It is anticipated that there will be negligible difference between an accident with plutonium, U-235, and U-238 in the reactor, compared with only U-235 and U-238, because no significant transient occurs until prompt criticality is passed and, after prompt criticality is passed, the delayed neutrons have no effect. It is anticipated that the magnitude of an accident is not significantly changed by plutonium in this core.

102 CRITICALITY CONSIDERATIONSCold 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 525 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 6Reactivity
(Control Rods Withdrawn)

	<u>k_{eff}</u>
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 525 F* would be 0.943 without the rods. If the rods are inserted, this decreases to 0.835. 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 rise in k_{eff} per unit time. This case occurs when the reactor is brought to full power during a maximum xenon override.

*All nuclear data are based on 525 F rather than 508 F used in thermal calculations. The effect on results is insignificant.

FIG. 10

EFFECTIVE MULTIPLICATION FACTOR vs TEMPERATURE

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 $\sum_a = .0071 \text{ cm}^{-1}$ at 8.1 hr after shutdown, compared to $.00415 \text{ cm}^{-1}$ at equilibrium. Upon starting up again, $\sum_a^{\text{Xe}}$ drops to about $.0032 \text{ cm}^{-1}$ after 8.9 hr and then rises towards equilibrium. The rate of change $\sum_a^{\text{Xe}}$ 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

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

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 VS NUMBER OF FUEL ASSEMBLIES
(NO CONTROL RODS)

103 CRITICAL EXPERIMENTSGeneral

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 scrambled 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 CONTROLGeneral

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 kilowatthour is dependent on the length of time a core can remain in the reactor.

The core is presently being designed for 10,000 hr 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 possibly be attained.

A core life of the order of 10,000 hr raises difficult problems of control, particularly in a pressurized water reactor with its large negative temperature coefficient. Approximately 19 per cent excess multiplication 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 hr life. This excess multiplication may be broken down as follows:

Cold to hot	7%
Fuel burnup	7%
Equilibrium	
Xenon and samarium	<u>5%</u>
Total	19%

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

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 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 1 per cent.

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 per cent. Accordingly, a very large number of control rods, possibly 75, would be necessary in order to cover the desired range of 24 per cent. 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 subdividing 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 per cent 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 by-passing 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 is proposed for this reactor, using a combination of mechanical control rods and a chemical neutron absorber dissolved in the coolant-moderator. 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 per cent subcritical with a clean core at operating temperature. To bring the reactor from this point to 5 per cent 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 per cent and 12 per cent. Experiments with such rods in critical assemblies, however, indicate

control rod worths higher by 30 per cent 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 per cent to 12 per cent on the basis of thermal neutron absorptions only, the control rods are not adequate to meet design objectives holding the hot clean core 3 per cent 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 per cent subcritical in the hot clean condition.

Leaving the chemical neutron absorbing compound in the main coolant during power operation is possibly 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 2.5 per cent 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 per cent 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 scrambled 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 sec

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 sec which, for a total rod worth of 10 per cent, provides a reactivity decrease of 16 per cent per second.

Chemical Control

The chemical control system is designed to shut down the cold clean reactor by a margin of at least 5 per cent 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 per cent to 3 per cent subcritical. In the present design, this would require about 1.6 gm 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 gm per liter of water, which means that approximately 8 gm 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 gm of natural boron per liter of water is sufficient to reduce k_{eff} to unity at any temperature above 250 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 rate of change in reactivity is $0.0005\% \Delta k/k$ per second, which is well within safe limits.

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:

The boric acid is stable in solutions at high temperature and pressure.

Ammonium borate solutions are likewise satisfactorily stable under these conditions.

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 of lithium metaborate at the interface between the water and vapor phases. This is known as the drying-up phenomenon.

FIG.13

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 phenomenon can occur as described above. The solubility of lithium borate decreases from 0.3 mole per liter at 500 F to approximately 0.2 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 and precipitation) 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.

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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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 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 (attenuation of approximately 10 through wall) of the pressure vessel integrated over 30 years of reactor operation is calculated to

SECTIONAL VIEW-REACTOR VESSEL AND CORE

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 at room temperature 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 min 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.E.V.C. = CHARGING & VOLUME CONTROL V.E.D. = VENTS & DRAINS

WESTINGHOUSE ELECTRIC CORPORATION
TITLE YANKEE ATOMIC ELECTRIC COMPANY
MAIN COOLANT SYSTEM
DPTN. SHAFFER MC-HAPP
DIV. C.A.P.A. PITTSBURGH, PA. 646J421

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. Pulse 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 pre-selected 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

FIG.16

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:

High Neutron Level. 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.

Short Reactor Period in the Start-up Range. 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.

Low Main Coolant Loop Flow. 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 two or more 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: Br₂, Rb, I₂, Xe, Cs, Kr, Ba, Fe, Ni, S, P, Si, Mn, U, Ta, Cu, Mo, Sn, B, NH₃, N₂, H₂, 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.

WESTINGHOUSE ELECTRIC CORPORATION
 TITLE YANKEE ATOMIC ELECTRIC COMPANY
 PURIFICATION SYSTEM

DESIGNED BY *W. J. ...* CHECKED BY *W. J. ...* DATE *1/4/67*
 DIV. C.A.P.A. PITTSBURGH, PA. **646J423**

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.

PRELIMINARY

WESTINGHOUSE ELECTRIC CORPORATION
 TITLE YANKEE ATOMIC ELECTRIC COMPANY
 CHARGING & VOLUME CONTROL SYSTEM

DESIGNED BY SHAFER 56 APP'D BY J. G. STANLEY 1/7/61
 DIV. C.A.P.A. PITTSBURGH, PA. 646J430

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

PRELIMINARY

WESTINGHOUSE ELECTRIC CORPORATION			
TITLE YANKEE ATOMIC ELECTRIC COMPANY			
PRESSURE CONTROL & RELIEF SYSTEM			
DATE	8/2/62	BY	646J422
DIV. CAPA	PITTSBURGH, PA.		

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

For present purposes of the Hazards Summary Report the waste disposal system is included with the assumption that the functions are correct, but the design detail requires further research and development work.

Function

The waste disposal system receives, contains, treats and safely disposes of all radioactive wastes in such a manner as to yield primarily concentrated solid wastes on resin material, fused solid material, or ashes. Gaseous wastes expected in small quantity, only, are dispersed to the atmosphere under favorable meteorological conditions.

The potential sources of radioactive wastes with necessary connecting drain lines to the waste disposal system are as follows:

- Main Coolant System
- Chemical Shutdown System
- Hotel and Laboratory Wastes
- Vent and Drain System
- Vapor Container Drain
- Steam Generator Blowdown
- Purification System
- Steam Plant Radioactive Drains
- Decontamination System
- Pressure Control and Relief System
- Shield Tank Cavity

Radioactive wastes from the reactor plant while operating at steady state are classified as low level activity wastes and appear as solids in suspension, solids in solution, gases in solution and gases out of solution. If no clad failures occur, only activated corrosion products and dissociated gases are directed to the waste disposal system for safe disposal. Plant operation, however, is anticipated to continue with some fuel assembly cladding failures depending upon the adequacy of shielding and the waste disposal system. With cladding failures, the important impurities for waste disposal handling are Br₂, Rb, I₂, Xe, Cs, Kr, Sr, Co, Ba, Fe, Ni, S, P, Si, Mn, C, Ta, U, Pu, Cb, Cu, Mo, Sn, B, NH₃, N₂, H₂, Na, Cr, and Ru.

Intermittently, other wastes carrying radioactive impurities are handled by the system. These include boron, used in cold shutdown of the plant and in the spent fuel pit; and decontamination solutions, used to reduce radioactivity

levels on equipment before maintenance. On the initial plant start-up, the boron containing water is not contaminated with radioactive isotopes; however, during subsequent start-ups the borated water is contaminated and boron separation from radioactive isotopes is accomplished by the waste disposal system with a minimum consumption of resin material.

Decontamination solutions for use throughout the plant are compatible with the waste disposal system concept. It is currently believed that certain acids and nonionic detergents are compatible with resin beds and an investigation program is under way to verify this concept.

Adequate monitoring of the waste disposal system is provided to assure safe operation, storage, and analysis. Adequate retention basin capacity is provided for radioactive decay and detailed isotopic analysis of liquid wastes before dilution and dispersion techniques are employed.

General Description

The waste disposal system consists of two waste hold-up tanks, a divided retention basin, ion exchange equipment, a resin storage tank, pumps, gas surge tank, hydrogen recombiner, gas storage tank, compressor, filter, and necessary piping, valves, and fittings. The system is shown on drawing 646-J-427.

Basis for Design

The system is designed to receive borated water, which may or may not contain corrosion and fission products.

The ion exchange resin temperature is limited to a maximum of 140 F. The normal operating temperature is any reasonable value below 140 F. To permit the temperature of the water to rise above 140 F results in partial decomposition of the resin and substantial loss of ion exchange capacity.

In waste disposal operation, radioactive liquids are cycled through the waste disposal demineralizers until the radioactivity is reduced sufficiently to be discharged to the retention basin.

Provisions are made for remotely removing and adding the ion exchange resin. Two ion exchangers are required, each emptied and refilled independently.

It is suggested that the waste disposal system tankage be located in an exclusion area so that shielding is accomplished by distance rather than by concrete or earthen structure. Isolation and operation of various elements of the system are performed by manually operated valves with reach rods, air operated diaphragm valves, or other dependable mechanisms.

PRELIMINARY

WESTINGHOUSE ELECTRIC CORPORATION			
TITLE: YANKEE ATOMIC ELECTRIC COMPANY			
WASTE DISPOSAL SYSTEM			
DRN. WEAVER	CHK. J.G. STUBBS	P/W	646-J-427
Div. CAPA	PITTSBURGH, PA		

The resin storage tank is partially underground and about 250 cu ft in size. Provision is made for the addition of further tanks. The final disposition of radioactive resin is left to further planning.

Two waste hold-up tanks are required, each sized to contain about 10,000 cu ft of water.

The gas surge tank, approximately 10 cu ft in volume, permits gas feed to the hydrogen recombiner at a nearly constant rate.

The hydrogen recombiner is of the platinum catalyst type.

The gas storage tank, approximately 50 cu ft in volume, receives gas at approximately 60 psi gage from the compressor. Preliminary calculations indicate that xenon and krypton will be the main sources of radioactivity in the exit gas stream. The half life of Xe-133 is 5.3 days; thus, in order for the Xe-133 activity to decay by a factor of 1,000, a decay period of 53 days is required in the gas decay tanks. The decayed gases are discharged from a hilltop stack with suitable dilution.

The filter is expected to be of glass fiber.

A hilltop stack is under consideration to discharge safely possible radioactive gases coming out of solution in the low pressure surge tank. The hilltop stack is located on the nearest hilltop sufficiently remote from the plant so that valley temperature inversions do not cause a hazard to the plant operations or to the public. A stack designed similar to a 50 ft high flagpole is adequate for this particular application. An exclusion area is established around this stack at a 1,000 ft radius.

A divided retention basin of 40,000 cu ft, total capacity, adequate to accommodate plant discharge for at least two months, is provided. This basin provides decay and identification holdup and is sampled before discharge to ultimate disposal.

References for ultimate disposal:

- a. Handbook 52, U.S. Bureau of Standards
- b. The Journal of the American Nuclear Society,
Volume 1, No. 6, Page 488, December 1956

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 de-generation 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.

WESTINGHOUSE ELECTRIC CORPORATION
 TITLE YANKEE ATOMIC ELECTRIC COMPANY
 SHUTDOWN COOLING SYSTEM
 DIV. C.A.P.A. PITTSBURGH, PA. 15201
 646J425

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

FIG. 18

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:

	<u>mr per hr</u>
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.

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PRELIMINARY

WESTINGHOUSE ELECTRIC CORPORATION	
TITLE YANKEE ATOMIC ELECTRIC COMPANY	
CHEMICAL SHUTDOWN SYSTEM	
DESIGNED BY <i>[Signature]</i>	APPROVED BY <i>[Signature]</i>
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.

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.

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

8										ARRANGEMENT VAPOR CONTAINER SH-2
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1	ORIGINAL ISSUE									YANKEE ATOMIC ELECTRIC COMPANY
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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.

VAPOR CONTAINER SPHERICAL ARRANGEMENT				
YANKEE ATOMIC ELECTRIC COMPANY				
STONE & WEBSTER ENGINEERING CORPORATION				
9699-FM-11A				
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ISSUE	DESCRIPTION	CHKD.	INSP.	CORR. APPR.

PIPING SYSTEM	SIZE	AMT	SCHED. & TYPE	EXRJ.T SIZE	INSUL TH.
MAIN STEAM	12"	4	SCH. 80 C.S. PIPE	20"	3"
FEED WATER	6"	4	SCH. 80 C.S. PIPE	10"	1 1/2"
S.V. BLOWDOWN TANK COOLING	8"	2	SCH. 80 C.S. PIPE	12"	1 1/8"
BOILER SECONDARY BLOW DOWN	2"	4	SCH. 80 C.S. PIPE	10"	3"
SPENT FUEL CHUTE	14"	1	15" ID SS PIPE	60"	15' LEAD

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

PIPING SYSTEM	SIZE	AM'T	SCHED & TYPE	SHELL PAD DATA
P.C. MAKE UP	2 1/2"	1	SCH. 80. STNLS STL	7'0"D x 2.94" ID = 7/8" THK
P.C. BLOWDOWN	2 1/2"	1	SCH. 80. STNLS STL	7'0"D x 2.94" ID = 7/8" THK
P.C. COMP. COOLING INLET	4"	8	SCH. 80. CARB. STL	9'0"D x 4.56" ID = 7/8" THK
P.C. COMP. COOLING DISCH.	10"	1	SCH. 80. CARB. STL	16'0"D x 10.88" ID = 7/8" THK
SHIELD TK. CAVITY	6"	1	SCH. 80. CARB. STL	11'0"D x 6.72" ID = 7/8" THK
V.C. COOLER	4"	2	SCH. 80. CARB. STL	9'0"D x 4.56" ID = 7/8" THK
V.C. COOLING SPRAYS	3"	1	SCH. 80. CARB. STL	7'6" O.D x 3.56" ID = 7/8" THK
V.C. DRAIN	6"	1	SCH. 80. CARB. STL	11'0"D x 6.72" ID = 7/8" THK
V.C. VENTILATION	18"	2	500" THK. CARB. STL	25'0"D x 18.91" ID = 8" THK

SECTION "A A"

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

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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 after-filters, 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

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PROPOSED FUEL HANDLING ARRANGEMENT				
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SCALE - FEET				

PROPOSED FUEL HANDLING ARRANGEMENT

YANKEE ATOMIC ELECTRIC COMPANY
STONE & WEBSTER ENGINEERING CORPORATION

9699-FM-19A

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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 kw-hr. The average auxiliary power requirement at full load is 12,000 kw. The expected full load station heat rate is 12,210 Btu per kw-hr.

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.

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SERVICE BUILDING ROOF

OFFICE ROOF

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NOTE:
ALL GRADES AND PIPE SIZES TENTATIVE ONLY.
FOR CIRCULATING WATER SECTIONS SEE 9699-FM-22.B

5						<p>CIRCULATING WATER SYSTEM PLAN</p> <p>YANKEE ATOMIC ELECTRIC COMPANY</p> <p>STONE & WEBSTER ENGINEERING CORPORATION</p> <p>1-16-57</p> <p>SCALE - FEET</p> <p>9699-FM-22A</p>
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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 H₂ 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 stand-by 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 output of the Yankee Plant is transmitted at 115 kv over New England Power Company lines to the New England interconnection. Two 115 kv lines are provided; one running northerly to Harriman Station of New England Power Company, and the second running in an easterly direction toward Millbury connecting to the transmission network at Cabot Station of Western Massachusetts Electric Company. Normal operation is with both of these lines in service.

Three sources of station service power supply are provided. One source is taken from the generator leads through a station service transformer, the second source is supplied from the 115 kv line to Harriman Station, and the third is supplied from the 115 kv line running toward Millbury, Mass. In general, the main station auxiliaries are divided between the three sections of the station service bus which normally operates with

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bus section breakers open. Two of the main coolant pumps are supplied from the bus section which is connected to the generator leads, and the two main coolant pumps are supplied one from the Harriman line and one from the Millbury line. Station service bus sections are provided with automatic connection to the adjacent bus section in the event that power supply is lost to any section. This arrangement allows a minimum of two main coolant pumps to remain in operation on the failure of one source of station service power, even in the event that automatic transfer does not take place. The three essentially independent sources of station service power provide ample protection against complete loss of auxiliary power.

Main Transformer

The main 155,000 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

Three station service transformers rated 5,000/6,250 kva of the OA/FA type adjoin the main transformer outside the turbine room. These supply the 2,400 v auxiliary bus sections. In addition, three 1,000 kva dry type transformers supply the 480 v auxiliary bus which takes care of the smaller motors and station lighting.

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.

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 pent-house. 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.

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.

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 non-radioactive 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 INJECTIONFunction

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.

PRELIMINARY

WESTINGHOUSE ELECTRIC CORPORATION			
TITLE YANKEE ATOMIC ELECTRIC COMPANY			
CORROSION CONTROL SYSTEM-HYDROGEN INJECTION			
DESM	SHAFFER	12-5 APPD. SG I.G. STUART	1/452
DIV. C.A.P.A.		PITTSBURGH, PA.	
646J431			

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 SYSTEMFunction

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.

PRELIMINARY

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

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

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-30440V AC MOTOR DRIVEN
VALVES-125V DC MOTOR OPERATED

POWER SUPPLY-SAFETY INJECTION SYSTEM

YANKEE ATOMIC ELECTRIC COMPANY
 ROWE, MASS.

STONE & WEBSTER ENGINEERING CORPORATION

9699-QE-1

7-15-57

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 back-up 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

PRELIMINARY

WESTINGHOUSE ELECTRIC CORPORATION		
TITLE YANKEE ATOMIC ELECTRIC COMPANY		
VENT & DRAIN SYSTEM		
BY <i>W. H. White</i>	DATE <i>10/10/55</i>	646J428
BY <i>W. H. White</i>	DATE <i>10/10/55</i>	
PITTSBURGH, PA.		

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.

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.

3 SITE300 GENERALLocation

The site is located in the town of Rowe, Massachusetts, on the east bank of the Deerfield River at a point approximately three-quarters of a mile south of the Vermont-Massachusetts border. It is adjacent to the Sherman hydroelectric station of New England Power Company. The location is shown on Figure 19. United States Coast and Geodetic Survey Map, "Massachusetts-Vermont, Rowe Quadrangle", Scale 1:31680 shows the topographical features. This map is not attached.

Access

The site may be reached by a secondary road which runs from Massachusetts Route 2 at Charlemont to Vermont Route 9 at Wilmington. The distances by road to the site are 13 miles from Route 2 and 21 miles from Route 9.

The Hoosac Tunnel and Wilmington Railroad connects with the main line of the Boston and Maine Railroad at the eastern portal of the Hoosac Tunnel, about 7.5 miles east of North Adams, Massachusetts. From this point, the railroad follows the Deerfield River approximately 12 miles north to the town of Readsboro, Vermont, passing the site at the 6.5 mile point.

Population

The following tabulation, based on 1950 census data, shows population figures by zones. It shows clearly the influence of the city of North Adams. These zones have been indicated on Figure 20.

Distance from Site, Miles	Area, Square Miles	Population		Density - Persons/Sq Mile	
		Including North Adams	Excluding North Adams	Including North Adams	Excluding North Adams
0-1	3.14		174		55
1-5	75.6		1,862		25
0-5	78.7		2,036		26
5-10	23.5	26,946	5,379	115	23
0-10	314	28,982	7,415	92	24
10-20	946		75,311		80
0-20	1,260	104,293	86,726	83	66

FIG. 19

MAP OF PROPERTY OWNED BY
YANKEE ATOMIC ELECTRIC COMPANY
AND
NEW ENGLAND POWER COMPANY

GENERAL AREA MAP

FIG. 20

The towns within a 20 mile radius which have a population in excess of 2,500, together with their distances and directions, are as follows:

<u>Town</u>	<u>Population</u>	<u>Airline Distance, Miles</u>	<u>Direction from Site</u>
North Adams, Mass.	21,567	9	WSW
Greenfield, Mass.	17,349	19	SE
Bennington, Vt.	12,411	17	NW
Adams, Mass.	12,034	12	SW
Brattleboro, Vt.	11,522	20	NE
Williamstown, Mass.	6,194	13	WSW

Land Use

There are only three industrial developments within 10 miles of the site, excluding North Adams and small sawmills. These are a box company in Wilmington, Vermont, a hardwood products company in Readsboro, Vermont, and a paper company at Monroe Bridge, Massachusetts.

A knife manufacturing company and a steel products company are down river at Shelburne Falls. This leaves Greenfield and North Adams as the only important centers of manufacturing from the point of view of this report; North Adams, because of its proximity to the site, and Greenfield, because it is on the Deerfield River.

Closely populated areas are found only in the centers of each town, so that the total land area devoted to housing is small.

All of the remaining land is utilized as forest or cultivated crop land, except for railroads and highways.

Detailed land use figures by towns are not available, but the following data from the 1954 census of agriculture show the percentage of land devoted to crops in each of the four counties near the site:

<u>County</u>	<u>Total Land Area, Acres</u>	<u>Crop Land, Acres</u>	<u>Per Cent</u>
Berkshire, Mass.	602,880	71,000	11.8
Franklin, Mass.	452,480	56,500	12.5
Bennington, Vt.	430,080	36,800	8.6
Windham, Vt.	507,520	43,900	8.7

Public Water Supplies

The main stream of the Deerfield River travels a distance of 41.2 river miles between the Sherman Dam and its confluence with the Connecticut River.

There are no downstream towns which use water pumped directly from the main stream of the river for domestic purposes. In the Mill Village section of Deerfield, one gravel packed well, located within 1,000 ft of the river, feeds into a public water supply which probably serves a part of the town.

In other towns, only the village of Monroe Bridge and the towns of Shelburne Falls and Greenfield have public water supplies. These systems obtain water from springs, wells, or reservoirs on or near tributary streams.

Site Layout

Yankee and its parent company, New England Power Company, own approximately 2,000 acres located on both sides of the Deerfield River, as shown on Figure 19. All of this land with the exception of the roads indicated and a group of five houses in the Monroe Bridge area is in the form of forest and unused farm land. The location of the plant is planned presently at the easterly end of the Sherman Dam. This location was selected because of level nature of land, adequate foundation conditions, nearness to the Sherman Pond for cooling water supplies, convenient access by both highway and railroad.

In addition, proximity to the high tension switching substation at the Harriman hydroelectric station of New England Power Company, in Readsboro, Vermont, would facilitate the delivery of power to the interconnected transmission systems of the New England utilities which propose to purchase the output of the Yankee Plant.

The road which follows the westerly edge of the river is a state road of black-top construction. The road on the easterly side of the river from Monroe Bridge to the dam is a private right-of-way over land of New England Power Company and can be controlled from its intersection with the Monroe Bridge-Rowe road. Currently, there is no access to the plant site from the east and north by motor vehicles because of a range of steep hills which surround the plant site. The distance from the plant site to the nearest point on the state highway is approximately 1,000 ft. New England Power Company owns almost all land on both sides of this road thus making it unlikely that outside parties would construct homes or other permanent installations in this area. An existing house and barn adjacent to the Sherman hydroelectric station is occupied by the station attendant and is owned in fee by New England Power Company.

301 METEOROLOGY

Pollution Climatology of the Deerfield River Site

In October 1955, James M. Austin, Associate Professor of Meteorology, Massachusetts Institute of Technology, prepared for Yankee an analysis of the pollution climatology of the site, based on available data. As indicated by Professor Austin's report, complete meteorological data for a definite survey are not available now, and further observations are required.

Professor Austin's report is quoted verbatim as follows:

POLLUTION CLIMATOLOGY OF THE DEERFIELD RIVER SITE

By

James M. Austin
Associate Professor of Meteorology,
Massachusetts Institute of Technology

Topography

The most important factor to consider in this pollution survey is the unusual topography in the vicinity of the proposed site. The Deerfield River meanders through the hilly regions of western Massachusetts and southern Vermont. At the proposed site the elevation of the land is approximately 1150 ft. above sea level. Within a horizontal distance of one mile the hills on both sides of the valley rise to an elevation of 2000 ft., approximately. This steep-sloped character of the river valley exists to Charlemont, eight airline miles southeast of the site, and beyond Wilmington, Vt. to the branches of the river 12 miles north of the site. Between these two towns the valley takes a very erratic course with a general decrease in elevation to the south. The valley is densely wooded on both sides.

With such a deep river valley it is to be expected that the wind direction and speed will frequently differ markedly from that which prevails over the neighboring hills.

Availability of Meteorological Data

Pittsfield, which is located about 25 miles to the southwest of the site is the nearest regular weather bureau station. Prior to February 1947 Pittsfield took surface observations 24 hours a day but since that time the observations have been less frequent. The nearest station which has taken upper-air wind and temperature observations is Albany, New York, located about 40 miles

to the west. The upper-air temperature observations were discontinued in November 1951. Since 1951 upper-air temperature data must be obtained from Rome, N. Y., about 100 miles to the northwest of the site. For many years a cooperative observer station has been maintained at Hoosac Tunnel on the Deerfield River approximately three miles southwest of the proposed site. Since January 1955 this station has taken wind and temperature observations at 8 a.m., 4 p.m. and midnight. An inspection of the station showed that an anemometer was well exposed on the surge tank and was located about 240 feet above the river. The records of wind speed should give a good estimate of the wind in this deep valley. However wind directions have not always been observed in a systematic manner.

In view of the availability of data this climatological survey of the Sherman Dam site will be based primarily on 1945-47 data and will be analyzed in the following manner.

- (1) The recent wind observations from Hoosac Tunnel will be utilized to establish the conditions favorable for calm or very light winds in the valley. The expected importance of inversions for light winds will be established by taking the temperature difference between the valley bottom at Hoosac Tunnel and the 850-mb (5000 ft.) temperature at Rome, N. Y.
- (2) Since scant temperature data are available at Hoosac Tunnel prior to 1955, the Pittsfield surface temperatures will be used to determine the stability during the years 1945 to 1947. This procedure is justified by the comparison of Pittsfield with Hoosac Tunnel given in Table 1. The general stability of the air will be given by the temperature difference between the 5000 ft. temperature at Albany and the surface temperature at Pittsfield. Elevated low inversions will be noted by comparing the 3000 ft. temperatures with the 1500 ft. temperatures at Albany, N. Y.
- (3) The basic wind data will be the 2000 ft. pilot balloon observations at Albany for the years 1945 to 1947. It is the air motion at the level of the ridges which will carry contaminants to the populated areas hence the pilot balloon data give a more representative picture of the wind field than the surface winds at Pittsfield. Since these basic data are not surface observations the 40 mile distance from Albany to the

Sherman Dam site is not an important consideration. On the other hand when the data are used for pollution estimates it will be necessary to consider the topographical influences as well as the stability of the air.

Meteorological Conditions Favorable for Calm or Light Winds

Table 2 presents an analysis of the wind data taken at Hoosac Tunnel. This table shows the expected high frequency of calms and light winds during inversion conditions (lapse rate negative). It is apparent that under these conditions the valley is essentially isolated from the free atmosphere above and that contaminants will drift along the valley rather than be dispersed to the free atmosphere on either side of the valley.

With moderate to steep lapse rates (the dry adiabatic rate is 23°F) the low frequency of calms and the higher values of V_{sfc}/V_{2000} show that the air in the valley is mixing with the air aloft and that the air motion over the region is well represented by the 2000 ft. wind.

Wind Regime

The 2000 ft. pilot balloon observations from Albany, N.Y., provided the basic data for the wind analysis. In the event of a missing report the 1000 ft. observation was used when available and the speed was increased by 10 per cent to account for the normal increase of wind with height. The 1000 ft. wind was used in only 3 per cent of the cases. When inclement weather prevented a pilot balloon observation the surface wind at Pittsfield was used. In these cases the wind speed was increased by 60 per cent. Only 7 per cent of the time was it necessary to utilize the Pittsfield report.

The frequencies of occurrence of winds from various directions, grouped according to wind speeds and vertical stability, are shown in Tables 3 and 4. The daytime stability utilized the 1.30 p.m. temperatures while the night time stability used the 1.30 a.m. temperatures. The stability classes are defined as follows:

$T_{\text{sfc}} - T_{850}$	Class
$-\infty$ to 0° F	Inversion
$+1^{\circ}\text{ F}$ to 10° F	Stable
$+11^{\circ}\text{ F}$ to 20° F	Moderate lapse
$+20\text{ F}$ to ∞	Unstable

where T_{sfc} and T_{850} refer to the surface and 850 mb (or 5000 ft) temperatures, respectively.

From Tables 2, 3 and 4 it is apparent that calm conditions prevail in the valley on at least 30 per cent of the nights. This high estimate is supported by the recent June, July and August data from Hoosac Tunnel where calms were reported on 50 per cent of the nights. Undoubtedly the air is not motionless on these occasions. In all probability a down-valley wind of about 1-2 mph exists on these nights so that the most serious threat should occur at Monroe Bridge. With low wind speeds and stable air the turbulent diffusion is at a minimum. In view of the high frequency of these conditions it would be desirable to determine the precise nature of the flow in the valley by direct observation, such as through the use of a smoke generator. Meteorological theory would indicate that Readsboro is a much less hazardous region since it is located up the valley from the proposed site and would only be affected by winds from a direction of 150-180 degrees. Since daytime inversions are uncommon these calm conditions usually disappear after sunrise. In view of its distance from the site and the meandering of the valley it is improbable that Charlemont will be affected like Monroe Bridge during these calm nights.

The analysis of the data in Table 2 shows that significant wind speeds often exist in the valley even when an inversion or a stable lapse rate prevails. It is most probable that the speed of the 2000 ft. wind is the important variable for distinguishing between calm conditions and light to moderate wind speeds in the valley. Below average 2000 ft. winds favor the former while above average 2000 ft. winds accompany the latter regime. Under these stable conditions the 2000 ft. direction may not always be representative of the wind direction in the vicinity of the site. The tendency of the air motion to take the path of least resistance through passes rather than over peaks is a well-established principle and makes it difficult to simulate atmospheric motion in a wind tunnel (1). The ridges east and west of the valley have few prominent passes so that there is no reason to anticipate that the wind directions, under stable conditions, differ markedly from the free-air wind direction at 2000 ft.

The tables show a maximum frequency of winds from a direction of 300 degrees so that the wind is most frequently blowing toward Shelburne Falls and Greenfield. The secondary summertime maximum of winds from 180° is directed up the river valley toward Readsboro and Wilmington, Vt. The remaining towns listed in Table 5 are infrequently downwind from the proposed site. With the exception of Monroe Bridge and Readsboro, all the populated areas are so located that the air must flow over a range of hills before it reaches the area. This motion over erratic terrain enhances the turbulent diffusion and thereby reduces the hazard.

The standard deviations of the wind speed have been computed for those groups which contain a large number of observations. In general the standard deviations are large compared with the mean wind speed and demonstrate that a wide range of speeds are observed in each category.

For the estimation of the travel of contaminants it is necessary to adjust the pilot balloon speeds. An extensive analysis of wind in the New England mountains (1) shows that this adjustment cannot be estimated with a high degree of accuracy. Except within the valley, it is apparent that the wind speed about the level of the ridges controls the dispersion of possible contaminants. In view of the higher elevations to the west meteorological theory would indicate that winds from a westerly direction are accompanied by speeds over the ridge to the east which are less than the free-air pilot-balloon speeds. The "site" factor is estimated as ranging between 0.6 and 1.0. The lower limit of 0.6 is based upon the observed winds in the valley and the observed winds at Pittsfield. With winds from an easterly direction the site factor will be higher. For up- or down-valley winds the ratios in Table 2 may be utilized to estimate the wind speed in the valley.

Temperature Inversions

The significance of inversions of temperature from the valley bottom to the free atmosphere above have already been discussed through the aid of Table 2. A second important inversion is the low but elevated subsidence or frontal inversion. An inversion about the mean elevation of the ridges will tend to suppress turbulent air motion and hence reduce the diffusion thereby leading to high concentrations downwind.

Radiosonde observations from Albany, N. Y., have been analyzed in order to determine the frequency of inversions between 1500 ft. and 3000 ft. In the winter half of the year inversions are present 20 per cent of the time while in the summer the frequency drops to 8 per cent. These elevated inversion cases are also included in the data which comprise Tables 3 and 4. As expected most periods with elevated inversions give a small temperature difference between the surface and 5000 ft. 80 per cent of the elevated inversions were accompanied by $T_{\text{surface}} - T_{850}$ values less than 10°F i.e., the "inversion" and "stable" categories of Tables 3 and 4. Hence the significance of the stability with inversions has already been recognized by the stability classification of Tables 3 and 4. The somewhat

unexpected "inversion" cases in Table 4 for summer days can be attributed to the presence of these elevated inversions.

Since the topography favors a high frequency of ground inversions and since the classification of Tables 3 and 4 serves to show the stable regime with elevated inversions, no further analysis of inversions will be undertaken. Topographical differences make it impossible to apply a detailed inversion analysis at Albany to the proposed site. It is considered that the method adopted here gives a reliable estimate of the frequency of stable and unstable air motion in the vicinity of the proposed site.

Precipitation

Since contaminants are washed out of the air by precipitation it is significant to determine the wind regime during periods of precipitation. The 6-hour precipitation amounts at Pittsfield and the 2000 ft. winds during the middle of the 6-hour period were utilized to determine the frequency of occurrence of precipitation with wind direction. A comparison of the precipitation amounts for the years 1945-1947 with the long-term average show that the three year period is representative of the long-term mean. The latter statistics are presented in Table 7.

Table 6 shows that 73 per cent of the wintertime precipitation and 58 per cent of the summertime precipitation is in the form of light rain showers or snow flurries. Much of the precipitation with west to north-west winds is of this type. The accompanying steep lapse rates of temperature and high wind speeds will favor rapid diffusion. Hence the fall-out concentrations will be minimized by unstable conditions and slight precipitation rates. The heavier and more prolonged periods of precipitation occur with northeast winds in winter and with northeast and southerly winds in summer. The northeast winds toward the cities of Adams and Pittsfield are blowing over erratic terrain so that they will give strong turbulent mixing thus reducing possible fall-out concentrations.

The principle fall-out hazard will occur with the southerly rain-bearing winds in summer toward Readsboro and Wilmington. However the relatively high wind speeds will act to minimize contaminant concentrations.

Hazards to Population

This analysis of potential hazards to the population will consider two possibilities, namely, a continuous emission of radioactive materials and an instantaneous release of radioactive material in the form of an explosion. Through the work of Sutton (2), Roberts (3) and others, equations have been prepared for the prediction of downwind concentrations. In the past decade these equations have been tested by field tests, such as conducted at the Brookhaven National Laboratory (4), (5). In general Sutton's theory appears to give satisfactory engineering estimates except under inversion or very stable conditions - a condition which prevails 35 per cent of the time in the Deerfield River valley. Of further significance is the fact that existing theory and empirical evidence are intended to apply to diffusion over reasonably homogeneous terrain. The terrain in the vicinity of the proposed site requires special consideration.

The erratic terrain favors two extreme conditions, namely, highly turbulent flow out of the valley and extremely stable flow within the valley. The expected concentrations will thus be estimated by using Sutton's equations and by introducing values of the parameters which recognize the unusual stability and instability. Continuity principles will be utilized to check the order of magnitude of the estimates.

Instantaneous Point Source

$$\chi_{(x,y,z,t)} = \frac{2Q}{\pi^{3/2} C_x C_y C_z (ut)^{3/2(2-n)}} \exp - \left[(ut)^{n-2} \left(\frac{x^2}{C_x^2} + \frac{y^2}{C_y^2} + \frac{z^2}{C_z^2} \right) \right] \quad (1)$$

(x,y,z,t) is the downwind concentration where x,y,z and t are measured from an origin moving with the cloud at constant speed u ; Q is the strength of the source; C_x , C_y and C_z are diffusion coefficients; and n is a parameter which varies with the turbulence. The concentration at the center of the puff ($x = y = z = 0$) is given by

$$\chi = \frac{2Q}{\pi^{3/2} C_x C_y C_z (ut)^{3/2(2-n)}} \quad (2)$$

Continuous Point Source

$$\chi_{(x,y,z)} = \frac{2Q}{\pi C_y C_z u x^{2-n}} \exp - \left[x^{n-2} \left(\frac{y^2}{C_y^2} + \frac{z^2}{C_z^2} \right) \right] \quad (3)$$

where x is distance downwind from the source. The maximum concentration at $y = z = 0$ is

$$\chi = \frac{2Q}{\pi C_y C_z u x^{2-n}} \quad (4)$$

These equations apply to the diffusion from a continuous point source placed at $x = y = z = 0$. The populated regions outside of the river valley are at elevations considerably lower than the height of the ridges in the immediate vicinity of the valley. The general air flow from above the sharp ridges to the lowlands will deviate from the general contour of the land so that the points of maximum concentration, defined by equations (2) and (4), will appear at a considerable elevation above ground level. The estimates of maximum concentrations at the ground can be made by setting $y = 0$ and $z = h$, where h is the estimated distance between the ground and the level of maximum concentration. Unquestionably h cannot be estimated with any degree of precision but trial computations show even a modest value of h greatly reduces the concentrations particularly near the ridge. As far as people at the ground are concerned the topographical effect is similar to that of emitting the pollutants from a very tall stack. This effect is an important one for reducing the hazard to people or animals living outside the river valley.

Estimates of the diffusion parameters are given in Tables 8 and 9. In view of the topographic influence on the turbulence itself two sets of parameters are presented. These estimates are based upon experimental evidence (2, 4, 5, 6) and the analysis of the proposed site. The values given for the "stable" flow outside the valley are based upon a consideration that the terrain will ensure strong mixing near the surface even though the lapse rate from the surface to 5000 ft. may belong in the stable category of Tables 3 and 4. Estimated concentrations can be computed for distances from the source by substituting for (ut) or x the distance from the source in meters. The units of χ/Q are (meters)⁻³.

Alternative Estimates

The diffusion project at M. I. T.'s Round Hill Field Station has recently found that satisfactory estimates of average downwind concentrations can be deduced from continuity principles. Effluent being emitted at a rate Q

spreads out as it is carried downwind. The edge of the cone which marks the boundary of the plume (defined as $\frac{1}{10}$ th of peak concentration) can be estimated from the standard deviation of the azimuth and elevation fluctuation of the wind.

For flow outside the valley the average concentration x meters from the origin is given by

$$Q = \frac{\pi}{2} u x^2 \tan 2\sigma_y \tan 2\sigma_z \chi_{av}$$

where σ_y and σ_z are the angular standard deviations of the wind in the horizontal and vertical, respectively. The reflection from the ground is considered here as in Sutton's equation (3). For unstable conditions outside the valley σ_y and σ_z should be of the order of 15° and 6° , respectively. These values give $(\chi/Q)_{av}$ equal to 2.7×10^{-9} at 21,000 m, the distance of Shelburne Falls from the site. The formula of Table 10 gives a maximum value of 1.2×10^{-8} .

Under inversion conditions within the valley the plume of contaminants will spread out in the vertical for a distance of 50 to 100 m and then will spread in the horizontal (see reference (7)). Hence, for distances beyond 100 m, the above formula becomes

$$Q = \pi u x \tan 2\sigma_y 100 \tan 2\sigma_z \chi_{av}$$

With inversion conditions in the valley σ_y and σ_z should be of the order of 4° and 2° respectively. These values give $(\chi/Q)_{av}$ equal to 2.5×10^{-4} at 1300 m, the distance of Monroe Bridge from the site. The formula of Table 10 gives a maximum value of 18×10^{-4} .

Maximum values may be expected to be about three times larger than average values. This continuity check thus gives dilution rates of the same order of magnitude as Sutton's formulae. From these checks and the previous consideration of topographical effects it can be concluded that the formulae in Table 10 give an upper limit to the concentrations at populated areas near the proposed site.

Conclusions

In all but one respect the proposed site is an excellent one from a pollution standpoint. The densely populated towns and also the isolated farm communities, out of the river valley, are shielded from possible contaminants by the ranges of hills on each side of the river. The necessity for contaminants to rise over these ridges before progressing toward populated regions ensures

a low dosage rate at ground level. The estimates of Table 10 are quite conservative to the extent that they represent maximum concentrations.

Within the valley, however, a possible serious hazard exists with the high frequency of inversion conditions. The employees of the New England Power Company, the residents of Monroe Bridge, and also of Readsboro could be subjected to high concentrations of contaminants. A detailed analysis of SO_2 concentrations in the Columbia River valley (7) demonstrates some of the peculiarities of air motion in valleys. For example, houses on a slope, such as at Monroe Bridge, may experience only slight concentrations at night but with sunrise, air with high concentrations of contaminants may be carried toward the slope. Inadequacies of theory and the lack of direct observational evidence make it impossible to assess, with any degree of reliability, the nature of the air flow during at least 30 per cent of the nights and early mornings. It is recommended, therefore, that direct experimental evidence be obtained of the air flow in the valley through the use of a device like a smoke generator. This recommendation is particularly worthy of consideration for any operation which involves a continuous emission of contaminants.

Table 1: Comparison of Hoosac Tunnel and Pittsfield Temperatures
(Fahrenheit degrees)

	Hoosac Tunnel	Pittsfield
Average Mean Temperature		
January	23.2	21.2
April	42.4	42.0
July	68.7	67.6
October	48.7	47.5
Av. Maximum Temperature		
3 Januarys	30.9	30.3
3 Julys	82.5	79.3
Av. Minimum Temperature		
3 Januarys	12.2	13.7
3 Julys	55.4	55.9

Table 2: Hoosac Tunnel Wind Data

January - June 1955

Lapse Rate of Temperature	Per cent of calms	<u>8 A. M.</u>		Per cent of calms	<u>4 P. M.</u>		Per cent of calms	<u>Midnight</u>	
		Av. Wind speed mph	V_{sfc}/V_{2000}		Av. wind speed mph	V_{sfc}/V_{2000}		Av. wind speed mph	V_{sfc}/V_{2000}
- ∞ to 0°F	40	2.7 (65)	0.21	38	5.1 (8)	0.21	36	2.7 (53)	0.11
+ 1°F to +10°F	28	4.2 (53)	0.32	28	5.2 (29)	0.33	45	3.0 (66)	0.14
+11°F to +20°F	12	8.6 (33)	0.48	3	9.0 (68)	0.50	6	10.1 (34)	0.37
> 20°F	0	15.2 (4)	0.67	5	10.4 (43)	0.70	no cases		

The numbers in parenthesis give the total number of observations in the class. V_{sfc} is the Hoosac Tunnel mean speed and V_{2000} is the mean 2000-ft wind speed at Albany, New York. The lapse rate of temperature is the temperature difference between Hoosac Tunnel and the 850-mb temperature at Rome, New York.

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Table 3A

Frequency of occurrence of winter night-time winds in various directions grouped according to stability. The lower left-hand number in each box is the average wind speed in m.p.h. (nautical) and the number in parenthesis is the standard deviation (when available) for the particular direction and stability class. Winter refers to the months of October to March, inclusive. All data are 2000 ft. winds at Albany, N.Y. for the years 1945, 1946 and 1947.

Stability				
Direction in Degrees	Inversion	Stable	Moderate Lapse	Totals
350,360,010	1.5 17.8	2.3 15.3	0.9 16.4	4.7
020,030,040	1.8 14.9	1.6 18.2	0.4 38.0	3.8
050,060,070	2.0 11.7	1.3 12.6	- -	3.3
080,090,100	0.9 10.0	0.4 14.0	- -	1.3
110,120,130	1.1 13.0	0.2 17.0	- -	1.3
140,150,160	2.3 17.1	1.8 20.6	0.5 26.0	4.6
170,180,190	5.0 21.0 (9.6)	3.2 27.5	0.9 32.4	9.1
200,210,220	3.7 20.4	3.6 22.6	0.5 19.7	7.8
230,240,250	3.5 14.3	4.5 21.3	1.5 27.3	9.5
260,270,280	2.8 16.5	6.6 22.3 (10.3)	5.9 28.7 (13.3)	15.3
290,300,310	3.5 17.1	11.6 27.8 (9.5)	12.3 27.3 (9.1)	27.4
320,330,340	2.3 16.7	7.0 22.0 (7.8)	2.8 24.3	12.1
Totals	30.4	44.1	25.7	100.0

Table 3B

Frequency of occurrence of summer night-time winds in various directions grouped according to stability. The lower left-hand number in each box is the average wind speed in m.p.h. (nautical) and the number in parenthesis is the standard deviation (when available) for the particular direction and stability class. Summer refers to the months of April to September, inclusive. All data are 2000 ft. winds at Albany, N.Y. for the years 1945, 1946 and 1947.

Stability				
Direction in Degrees	Inversion	Stable	Moderate Lapse	Totals
350,360,010	1.5 10.7	2.6 13.5	0.9 12.0	5.0
020,030,040	0.9 16.0	5.7 19.7 (9.3)	0.9 19.2	7.5
050,060,070	0.5 5.0	2.2 12.1	0.2 10.0	2.9
080,090,100	0.2 10.0	1.1 11.5	0.5 12.3	1.8
110,120,130	0.2 23.0	1.5 8.6	- -	1.7
140,150,160	1.3 21.4	3.7 15.8	- -	5.0
170,180,190	6.2 25.3 (8.8)	9.6 23.6 (10.6)	1.5 27.8	17.3
200,210,220	4.2 21.4	7.0 20.3 (8.5)	0.7 26.0	11.9
230,240,250	2.6 13.5	4.3 16.6	0.2 20.0	7.1
260,270,280	3.7 13.1	5.6 16.9 (8.5)	1.8 22.1	11.1
290,300,310	1.8 13.2	11.1 23.4 (8.6)	4.3 31.4	17.2
320,330,340	1.7 8.7	7.6 19.7 (7.2)	2.4 21.4	11.7
Totals	24.8	61.8	13.4	100.0

Table 4A

Frequency of occurrence of winter daytime winds in various directions grouped according to stability. See description to Table 3A.

Stability					
Direction in Degrees	Inversion	Stable	Moderate Lapse	Unstable	Totals
350,360,010	0.7 9.0	2.0 14.8	1.6 12.2	1.6 9.1	5.9
020,030,040	0.5 9.0	0.9 15.0	1.1 16.7	0.5 9.7	3.0
050,060,070	1.8 13.6	1.3 15.9	0.5 3.0	0.2 5.0	3.8
080,090,100	0.4 16.0	0.4 9.0	0.2 5.0	- -	1.0
110,120,130	0.2 11.0	0.4 12.0	0.4 13.5	0.4 4.5	1.4
140,150,160	0.7 20.7	1.8 23.7	1.1 20.5	- -	3.6
170,180,190	1.1 40.2	4.6 23.5	4.0 17.6	0.4 6.5	10.1
200,210,220	0.4 27.0	3.1 23.1	4.3 16.7	0.9 8.8	8.7
230,240,250	0.5 18.0	3.1 18.1	4.4 15.4	2.2 14.5	10.2
260,270,280	- -	1.1 24.3	10.3 19.7 (8.7)	3.1 17.2	14.5
290,300,310	0.5 29.7	4.3 21.8	17.1 24.1 (11.1)	3.3 18.0	25.2
320,330,340	1.3 18.6	2.5 23.8	5.9 14.2 (5.9)	2.7 14.9	12.4
Totals	8.2	25.5	50.8	15.5	100.0

Table 4B

Frequency of occurrence of summer daytime winds in various directions grouped according to stability. See description to Table 3A

Stability					
Direction in Degrees	Inversion	Stable	Moderate Lapse	Unstable	Totals
350,360,010	- -	- -	2.2 12.7	4.2 8.2	6.4
020,030,040	- -	0.5 12.7	2.6 11.4	2.0 11.5	5.1
050,060,070	0.2 10.0	0.9 13.2	0.9 10.8	0.9 6.2	2.9
080,090,100	- -	0.9 6.0	0.9 7.2	0.4 5.5	2.2
110,120,130	0.4 12.5	0.2 13.0	0.9 9.8	0.7 5.8	2.2
140,150,160	0.5 23.0	3.0 15.3	3.3 12.6	1.5 8.5	8.3
170,180,190	0.4 10.0	2.0 23.5	10.1 16.1 (6.9)	4.2 10.6	16.7
200,210,220	- -	0.7 13.7	4.6 17.7	5.5 12.0 (7.5)	10.8
230,240,250	- -	0.4 18.0	3.5 16.2	5.2 11.0 (7.4)	9.1
260,270,280	- -	0.2 15.0	4.1 19.7	7.0 11.5 (7.3)	11.3
290,300,310	- -	0.9 16.2	6.6 17.6 (8.1)	9.6 13.4 (8.9)	17.1
320,330,340	- -	0.2 10.0	2.6 15.4	5.0 11.3 (6.9)	7.8
Totals	1.5	9.9	42.3	46.3	100.0

Table 5: Location of Populated Regions

	Distance in Meters	Direction from Proposed Site in Degrees	Elevation in Feet
Monroe Bridge	1300	240	1100
Readsboro	4900	345	1200
Charlemont	12000	150	600
North Adams	15000	260	700
Wilmington, Vt.	17000	20	1600
Adams	20000	230	800
Shelburne Falls	21000	130	500
Bennington	27000	305	700
Greenfield	30000	120	200
Brattleboro	32000	65	300
Pittsfield	40000	220	1000

Table 6A: Frequency of occurrence of winter precipitation within 4 class ranges of 6-hour amounts grouped according to wind direction. The table is based on Pittsfield data for the years 1945-1947. The lower number in each box is the average wind speed (nautical miles per hour) for the particular wind direction and precipitation class.

Winter					
Wind Direction	Precipitation Rate (in/6 hr)				Totals
	Trace	0.01-0.05	0.06-0.25	> 0.25	
350,360,010	0.9 14.6	1.4 16.9	0.8 10.3	0.2 11.5	3.3
020,030,040	0.7 17.5	1.3 18.9	1.5 13.6	0.4 15.0	3.9
050,060,070	0.8 8.0	1.7 14.6	3.5 15.0	1.3 14.7	7.3
080,090,100	0.2 11.0	1.0 11.7	0.8 12.6	0.1 14.0	2.1
110,120,130	0.4 12.3	0.4 13.0	1.0 14.1	- -	1.8
140,150,160	1.8 21.2	1.7 20.9	1.6 24.9	0.6 22.2	5.7
170,180,190	4.5 24.6	2.6 20.6	2.6 29.0	0.8 30.1	10.5
200,210,220	2.4 31.3	2.6 29.1	1.7 22.3	0.7 30.7	7.4
230,240,250	3.0 21.6	3.2 19.9	1.6 23.6	0.6 14.6	8.4
260,270,280	6.5 24.5	5.5 21.8	2.0 24.2	0.1 41.0	14.1
290,300,310	15.0 26.7	9.0 30.4	3.4 25.4	0.5 25.8	27.9
320,330,340	4.0 20.7	2.7 24.0	1.1 27.7	- -	7.8
Totals	40.0	33.1	21.6	5.3	100.0

Precipitation occurs within a 6-hour period 30 per cent of the time in winter.

Table 6B: Frequency of occurrence of summer precipitation within 4 class ranges of 6-hour amounts grouped according to wind direction. The table is based on Pittsfield data for the years 1945-1947. The lower number in each box is the average wind speed (nautical miles per hour) for the particular wind direction and precipitation class.

Summer					
Wind Direction	Trace	Precipitation Rate (in/6 hr)			Totals
		0.01-0.05	0.06-0.25	>0.25	
350,360,010	1.2 8.5	1.7 13.5	1.8 14.4	0.6 11.0	5.3
020,030,040	0.9 14.2	1.7 12.2	2.6 14.2	1.1 16.0	6.3
050,060,070	0.8 8.2	2.6 11.6	1.7 11.5	2.1 14.9	7.2
080,090,100	0.5 5.0	0.8 10.4	1.5 9.2	0.5 15.3	3.3
110,120,130	0.9 9.8	1.1 13.1	0.9 12.3	0.3 11.0	3.2
140,150,160	2.3 21.5	4.8 16.2	2.6 15.6	1.7 15.5	11.4
170,180,190	4.7 21.0	5.1 21.8	4.7 25.2	2.9 23.3	17.4
200,210,220	2.6 20.7	2.7 21.5	2.6 21.5	1.5 22.0	9.4
230,240,250	2.1 13.1	2.1 17.3	2.7 18.3	2.7 20.4	9.6
260,270,280	3.2 17.7	2.6 18.5	2.0 22.0	0.5 12.0	8.3
290,300,310	6.1 22.6	4.2 24.0	1.8 20.3	0.9 21.5	13.0
320,330,340	1.8 16.1	1.8 19.4	2.0 16.6	0.3 13.5	5.9
Totals	26.9	31.2	26.9	15.1	100.0

Precipitation occurs within a 6-hour period 20 per cent of the time in summer.

Table 7: Precipitation Statistics from Pittsfield, Mass.

Month	Average Precipitation (inches)	Average Number of Days with Precipitation (> 0.01 inch)
January	3.1	17
February	2.5	15
March	3.2	15
April	3.6	15
May	3.8	15
June	4.6	12
July	4.9	13
August	4.3	10
September	4.2	10
October	2.9	9
November	4.0	13
December	3.0	14

Table 8: Values of Diffusion Parameters for Flow within the Valley

<u>Temperature Lapse Rate</u>	u (m/sec)	n	C _x	C _y	C _z
Inversion	1	0.6	0.15	0.15	0.1
Moderate Lapse	6	0.25	0.25	0.25	0.2
Unstable	5	0.15	0.3	0.3	0.3

Table 9: Values of Diffusion Parameters for Flow out of the Valley

Temperature Lapse rate	u (m/sec)	n	C _x	C _y	C _z
Stable	8	0.3	0.25	0.25	0.2
Unstable	6	0.15	0.3	0.3	0.3

Table 10: Estimates of the Dilution Factor χ/Q

A. Within the Valley (Monroe Bridge and Readsboro)

Maximum concentrations from equations 2 and 4

	Instantaneous Source	Continuous Source
Inversion	$160/(ut)^{2.1}$	$42/X^{1.4}$
Moderate Lapse	$29/(ut)^{2.63}$	$2.1/X^{1.75}$
Unstable	$13/(ut)^{2.78}$	$1.4/X^{1.85}$

B. Outside the Valley

Maximum concentrations from equations 2 and 4

	Instantaneous Source	Continuous Source
Stable	$29/(ut)^{2.55}$	$1.6/X^{1.7}$
Unstable	$13/(ut)^{2.78}$	$1.2/X^{1.85}$

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- (4) P. H. Lowry: Microclimate Factors in Smoke Pollution from Tall Stacks, Meteorological Monographs Vol.1, No. 4, 1951
- (5) I. A. Singer: A Comparison of Computed and Measured Ground-level Dose-rates from Radio-argon Emitted by the Brookhaven Reactor Stack, Brookhaven National Laboratory, 1954
- (6) D. G. Friedman: The Height Variation of Lateral Gustiness and Its Effect on Lateral Diffusion, Journal of Meteorology, Vol. 10, No. 5, 1953
- (7) E. W. Hewson: Atmospheric Pollution by Heavy Industry, Quart. Journal Roy. Meteor. Soc., Vol. 71, 1945

(End of Professor Austin's report)

Meteorological Measurements Program

A weather observation station is located at No. 5 Station, New England Power Company, four miles downstream from the site. The data available at this location consist of temperature and precipitation at two-hour intervals, and wind velocity and direction at four-hour intervals. These have been helpful in preliminary studies, but the tortuous nature of the valley and the position of the anemometer and wind vane result in considerable uncertainty in interpretation of results.

An automatic weather recording system has been installed at the Yankee site. This equipment records wind velocity, direction and air temperature at one-hour intervals. Three observation points are used as follows:

Point 1 - At southeasterly end of Sherman Dam, on top of a 30 ft utility pole; this location is approximately 470 ft northwest of the reactor. This station will be the reference point for in-valley winds and will be the base for the temperature profile determination.

Point 2 - On a hillside 1,400 ft south-southwest of the reactor and 276 ft above the crest of the dam; El. 1,290 local datum or 1,395 mean sea level. This station is situated to monitor down-valley winds.

Point 3 - On top of highest hill in immediate area. The nearest hill of equal or greater height is located 1 3/4 miles to the northeast, away from prevailing winds. This location is 4,200 ft southeast of the reactor and 956 ft above the crest of the dam; El. 1,970 local datum or 2,075 mean sea level. This station is intended to determine prevailing winds above the valley and degree of correlation with winds in the valley. It will also be the upper level point for the temperature profile determination.

Information from these three locations is telemetered by land line to the Sherman Hydroelectric Station where it is recorded on standard five channel chadless punched tape. A printed symbol is also provided for visual checking of results.

From the punched tape, data can be reproduced on pages in columnar form or can be set up on a punched card system.

Detailed analysis of wind patterns at the site will begin with data taken in February 1957 and will be available for incorporation in subsequent amendments to the hazards evaluation.

302 HYDROLOGYPlant Site

The plant site is entirely located on the watershed of the Deerfield River. Surface and subsurface drainage is from the high lands east and south of the plant site toward the river. Observations of ground water level in the borings show a drop of 5 ft from Boring 2 to Boring 1 and a drop of 10 ft from Boring 1 to Boring 7. Thus, at the time these borings were made, the gradient of the ground water table was downward toward the northwest which indicated flow of the ground water toward the Deerfield River. The glacial tills of the site contain considerable fine sand, and the lower members contain some silt and clay size particles. While free draining, permeabilities of these soils are lower than most fluvial deposits of sand and gravel. The boring and seismic survey plan is shown in Figure 21.

Deerfield RiverDrainage Area

The Deerfield River rises near Sunderland, Vermont, follows a winding course in a southerly direction 30 miles to the Massachusetts-Vermont state line, and then continues south about 7 miles into Massachusetts where it turns and follows a meandering but general easterly course for about 36 miles through Shelburne, Deerfield and Greenfield to its confluence with the Connecticut River. It has a total drainage area of 664 square miles, 347 square miles in Massachusetts and 317 square miles in Vermont. The drainage area above Sherman Pond is 236 square miles.

Improvements

There are eight hydroelectric generating plants and two large storage reservoirs along the Deerfield River. Both the reservoirs and two of the hydroelectric plants are above the site, while the remainder are downstream. Pertinent data pertaining to these hydroelectric developments are as follows:

Hydroelectric Generating Stations

<u>Station</u>	<u>Ownership</u>	<u>Nominal Plant Capability, kw</u>	<u>Elevations*</u>		<u>Dam Location Miles from Mouth of River</u>
			<u>Full Pond</u>	<u>Normal Tailwater</u>	
Searsburg	N.E. Power	4,800	1,650	1,416	60.0
Harriman	N.E. Power	45,000	1,392	1,000	47.2
Sherman	N.E. Power	6,500	1,002	921	41.2
No. 5	N.E. Power	15,000	922	676	40.6
No. 4	N.E. Power	6,000	368	299	18.8
No. 3	N.E. Power	6,000	297	229	16.0
Gardner Falls	West. Mass.	3,700	229	189	14.9
No. 2	N.E. Power	7,000	189	123.5	12.9

*All elevations on local datum: 0=105.66 ft above MSL

Storage Reservoirs

	<u>Drainage Area - sq miles</u>		<u>Reservoir Contents</u>	
	<u>Gross</u>	<u>Net</u>	<u>acre-ft</u>	<u>Billion cu ft</u>
Somerset	30.0	30.0	57,345	2.498
Harriman	184.0	154.0	116,075	5.056

River Flow

The United States Government operates a river gage station one mile downstream from Charlemont, Massachusetts. Detailed public records at this point are available from 1913 to date and show the effect of storage reservoir operation.

BORING AND SEISMIC SURVEY PLAN

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303 GEOLOGYPlant Site

The site lies in a small valley entering the Deerfield River Valley from the southeast approximately opposite the east end of Sherman Dam. This dam was constructed as a hydroelectric project by the New England Power Company in 1926 and has a maximum height of approximately 90 ft. Except for the Deerfield River Valley the site is surrounded by the Berkshire Mountains, which rise to heights of about 1,000 ft above the site to either side and immediately behind it. This area was overridden by continental ice during Wisconsin glaciation, at which time continental ice reached the central portion of Long Island. It is probable at the location of the site that the surface of the ice sheet was at least 3,000 ft above sea level. The ice sheet almost totally removed all residual soils and the present soil mantle found in this vicinity is predominantly glacial till and drift.

The surface of the bedrock at the site is extremely irregular, solid ledge outcropping in a small hill along the northeast side and again in a large hill to the southeast. Consequently, one of the concerns of this investigation was to establish bedrock elevations within the area as a guide to design in order to keep rock excavation to a minimum. Three borings, Nos. 1, 2 and 7, were made at the locations shown on Figure 21. A small gravel pit, which had been opened by a local highway department, afforded an examination of the upper soil. A seismic survey was made to check depths to bedrock. The seismic survey was run using the refraction technique, the depth of bedrock being determined at the end of each seismic survey line. The elevation at each point where it was determined is shown on Figure 21. These elevations indicate the surface of the rock generally slopes toward the Deerfield River.

The soils disclosed by the borings as shown on the logs, Figure 24, are primarily medium to fine sands with gravel, cobbles and boulders. These soils are glacial tills, most probably laid down as bed moraine by the ice sheet. They comprise a heterogeneous mass of soil dumped into place by the glacier and compacted by its weight. Some individual boulders are 10 to 12 ft in size. Figure 22 shows typical soils exposed in the gravel pit. Figure 23 shows large glacial boulders exposed along the shore of Sherman Pond.

The borings indicate that the deeper lying soils are somewhat more compact and contain a slightly greater percentage of clay and silt size particles than the upper soils. The seismic survey also indicates the deeper lying soils to be somewhat more compact as velocities were higher than in the surface materials.

FIG 22
TYPICAL SOIL CONDITION

FIG 23
LARGE GLACIAL BOULDER

LEGEND

SAMPLE: NUMBER OF BLOWS OF 140 LB HAMMER FALLING 30" REQUIRED TO DRIVE SAMPLE 12" OR LESS IF INCHES DRIVEN ARE INDICATED.

WATER LEVEL IN DRILL HOLE AFTER COMPLETION OF BORING.

EL. ALL ELEVATIONS REFER TO NEW ENGLAND POWER CO. DATUM WHICH IS 105.66 FT ABOVE MEAN SEA LEVEL, USGS DATUM.

LOG OF BORINGS

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The borings were carried to a depth of approximately 50 ft. Each boring was started using 4 in. casing and NX core-bits, the largest size available, stepping down as boulders which required coring were penetrated. Even with these techniques, it was necessary to try two different locations at both Borings 1 and 2 in order to reach this depth.

The bedrock is composed of Archean Metamorphics predominantly schists and gneiss. The rock is fresh and free of weathering. Although jointed, this is a strong, stable rock.

Foundation Design

The soils underlying the site are strong, stable materials and when the surface mantle of humus and top soil has been removed there would be little, if any, advantage in foundation stability and strength to be gained from deep foundations. Accordingly, it is recommended that all structures and equipment be founded upon spread footings at such depths below ground surface as are necessary to ensure protection against frost.

The shearing strength of granular soils is proportional to the stress normal to the plane of shear. The rupture plane of footing failure follows a logarithmic spiral shape starting at one edge of the footing and reaching a depth below surface which is roughly proportional to the width of the footing. Consequently, bearing pressures for footings on granular soils should be smaller for narrow or shallow footings than deep wide footings.

Recommended bearing values for footings on these soils are shown on Figure 25. A maximum bearing pressure of 6 tons per sq ft for large, deep footings is recommended with suitable reduction for shallow or small footings indicated graphically.

Removal of boulders from footing excavations might be difficult and might result in disturbance of the soil below nominal footing grade. Soil which is loosened or disturbed in these operations would compress more under load than undisturbed soil; for example, a zone of loosened soil under one edge of a footing might cause undesirable tipping. Care in excavation would be required to minimize disturbance. All disturbed and loosened soils below footing grade should be excavated with side slopes not steeper than 45 deg and the hole backfilled to footing grade with lean concrete.

SOIL BEARING VALUES FOR FOOTINGS

304 SEISMOLOGY

The general area is seismically stable. Only two earthquakes (1), (2), of sufficient intensity to be felt by any considerable number of people have epicentered within 50 miles of the site; one on July 28, 1875 near Canon Mountain, Connecticut, and the second November 23, 1884 in Southern New Hampshire. In addition, the site has undoubtedly experienced some motion attendant upon shocks epicentered in other regions but which were felt over widespread areas, such as the earthquake of February 28, 1925 which epicentered in the St. Lawrence Valley and which was felt as far south as Virginia. While this earthquake was felt over a considerable region, damage was entirely limited to a narrow belt on either side of the St. Lawrence River, predominantly in areas of soft, rather unstable soils.

Linehan (2) indicates the site is in one of the areas of least seismicity in the Northeastern United States and that the risk of shock is very slight, but consideration should be given to the possibility of a weak or moderate earthquake.

Experience has indicated that damage from earthquakes is greater in areas which are overlain by soft, unstable soils, especially if these are of considerable depth. Soils at this site are compact glacial tills and bedrock is at shallow to moderate depths. Experience has shown that, with these favorable conditions, earthquakes of moderate intensity will not cause structural damage to modern framed structures designed to withstand reasonable wind loads. Accordingly, it is recommended that no special provisions be made for seismic design in this plant.

- (1) N.H. Heck "Earthquake History of the United States"
Special Publication No. 149, U. S. Department of
Commerce, Coast & Geodetic Survey
- (2) Rev. Daniel Linehan S.J., Director, Weston
Observatory, Memorandum dated April 29, 1955

305 PREOPERATIONAL RADIATION MONITORING

A program of continuous monitoring of airborne activity began in October 1956. A scintillation type detector with energy discriminator is directly connected to an automatic time print out. By this means, the counting rate integrated over the energy spectrum is obtained as a permanent record. These data are correlated with weather conditions, particularly precipitation and snow cover, by which means many of the variations which occur can be explained.

Dust samples are being obtained several times per month, using a high volume air sampler. These are counted to obtain information about the amount and nature of radioactivity carried on dust particles.

This work will be extended to include periodic sampling of river water and soil in order to develop a complete record of natural radioactive background before nuclear fuels are delivered to the site.

4 PLANT OPERATION

400 GENERAL

Since the Yankee project has progressed only to the preliminary engineering and design stage, plant operation is covered in broad and general terms. Rules and procedures are indicated, and the concept of safe, effective and regulated operation is outlined. The Yankee Plant Operators' Manual will replace this section when the Final Hazards Summary is filed as an amendment to this application.

Location of the Yankee atomic electric plant in the Commonwealth of Massachusetts subjects the plant and its operating personnel to license requirements of Massachusetts, as well as of the Atomic Energy Commission. Currently, the areas of licensing activity of these agencies are only slightly overlapping. The newly formed Massachusetts Commission on Atomic Energy has not yet defined its position in regard to licensing. Plant operations, in any event, are conducted in such a way as to meet the requirements of federal and local licensing authorities and inspection programs. In addition, the entire plant is operated in a manner consistent with the best industrial practices for large electric generating facilities.

401 PLANT ORGANIZATION

Sixty-two to sixty-six full time employees are necessary for plant operation. The larger number may be required depending upon the number of guard stations, manning of those stations, and the national security situation. The plant organization chart is shown on Figure 26. All employees within the fenced periphery of the plant have appropriate security clearances, since this is a restricted access area under control of the plant security officer.

Based on current regulations, the minimum background training and license requirements which are adhered to in staffing the plant are as follows:

<u>Job Category</u>	<u>Training and License</u>
Plant Superintendent	Nuclear plant experience, technical degree and license as required in Part 55, AEC Regulations.
Chief Engineer	Nuclear plant experience, technical degree and license as required in Part 55, AEC Regulations; 1st class operating engineer's license, Commonwealth of Massachusetts.
Reactor Supervisor	Nuclear plant operation experience; license as required in Part 55, AEC Regulations; technical degree desired.
Maintenance Supervisor	Nuclear plant background and steam plant maintenance experience; technical degree preferred.
Technical Services Supervisor	Technical degree; industrial hygiene and radiation background; nuclear and steam plant instrumentation background desired.
Watch Engineers	2nd class operating engineer's license, Commonwealth of Massachusetts; license as required in Part 55, AEC Regulations.

YANKEE PLANT ORGANIZATION CHART

<u>Job Category</u>	<u>Training and License</u>
Radio-Chemist	Technical training as required.
Control Engineer	Technical training as required.
Turbine Operators	Operating engineer's license, Commonwealth of Massachusetts, as required.
Control Room Operators	Two per shift; at least one with license as required in Part 55, AEC Regulations.
Nurse (Male)	Medical and radiation dosimetry background.
Other Employees	Specific technical or administrative training not a prerequisite. These employees are trained under a Company training program to perform satisfactorily the duties of their stations.

The Chief Engineer is in a position of overall operating authority in the plant. Pending clarification of operating authority between Massachusetts steam plant operator licensing and AEC nuclear plant operating licensing, it is considered possible that both licenses can be acquired by him.

The plant organization is adequate for plant operation, routine maintenance and refueling, and for performing most maintenance within the vapor container, except steam generator replacement. The plant is not staffed on a permanent basis to do major repair or overhaul of the primary plant, since this is anticipated to be infrequent and unpredictable.

Normally, the plant operating crews do the refueling of the reactor while the regular maintenance crew performs primary and steam plant maintenance during the shutdown. If either primary plant or steam-electric plant maintenance is expected to be extensive during a particular shutdown period, sufficient outside maintenance workers are brought in to take care of the steam-electric work and thus release the plant maintenance force for work on the primary plant, only.

Currently, the Yankee technical staff includes six experienced nuclear engineers. In addition, consulting arrangements are in effect with the following individuals and organizations:

Dr. Theos J. Thompson, Nuclear Engineering, M.I.T.
Professor James M. Austin, Meteorology, M.I.T.

General K. D. Nichols, Washington, D.C.
Nuclear Development Corporation of America, White Plains, N.Y.
Dr. Shields Warren, Deaconess Hospital, Boston, Massachusetts

It is planned that, after plant operation begins, Yankee will continue to maintain a central office in Boston. The officers and staff of this office will include competent engineering and management personnel. In addition, the Company will retain consultants. Specialized technical services will be acquired through contract arrangements, as needed.

402 PERSONNEL TRAININGPlant Supervisory and Technical Training

Nuclear plant training of supervisory personnel and technical graduates of utility companies, now owners of Yankee, was initiated during June, 1951, when technical graduates were lent to AEC contractors for training. Yankee now has technical personnel, directly employed or on call with member companies, with theoretical, design, materials, construction, test and operation experience on the STR, EBR I and II, BORAX I and III, EBWR, APDA-PRDC, LMFR, and with the Nuclear Metals Corporation. Supervisory-technical training of Yankee employees is continuing at APDA-PRDC and Argonne National Laboratory. As the plant operation phase approaches, Yankee plans to arrange with the AEC for additional personnel training required under Part 55, AEC Regulations.

Plant Operator Training

Plant operator training in nuclear theory, plant devices and operation will be initiated about one year prior to plant start-up. Plant operators will be drawn from owner companies' steam plant operating staffs. At the time of reassignment to the Yankee plant, these operators will be thoroughly experienced in large steam-electric plant operation, and will hold appropriate operators' licenses issued by the Massachusetts Department of Public Safety. Four nuclear plant training programs for operators will be instituted as follows:

Lecture courses conducted by Yankee and Westinghouse Commercial Atomic Power Activities engineers and scientists.

On-the-job training by construction liaison and preoperational component and systems test programs. Operators will perform pre-nuclear start-up tests on the entire plant under the direction of Westinghouse and Yankee engineers.

Additional training on pressurized water reactor plants similar to Yankee, as required by the AEC leading to Part 55 licenses.

Continuing educational programs for all employees on plant hazards, radioactivity and security, conducted by plant supervisory personnel. Continuing training programs will be conducted for plant personnel and member company personnel leading to the licensing of a pool of technical personnel and operators in anticipation of a steady growth of the atomic electric business.

403 INITIAL PLANT INSPECTION AND TESTING

Plant inspection and testing leading to full scale operation is divided into three phases.

Manufacturer's Component Tests and Inspections

Test, construction code and cleanliness requirements accompany each specification or purchase order for equipment. Hydrostatic, leak, metallurgical, electrical and other tests performed by the supplying manufacturer are described in detail on the specification sheet, together with the requirements for test witnessing by an inspector, when and if required. Fabrication and cleanliness standards, including final cleaning and sealing, are described, together with shipping procedures.

The specifications and tests are in accordance with State Regulations, National Society Codes, and the best industrial practices.

Plant Systems Inspection and Tests

As each component is received, it is carefully checked to determine that it was shipped according to specifications, had the proper inspection approval and that no damage occurred in transit. Construction, field fabrication, and erection are handled only by competent personnel under construction permits issued by the appropriate authorities, using materials and techniques that conform with applicable State and National Society Codes.

Following erection, a plant component and systems test program will be carried out prior to loading nuclear fuel. Checkout test specifications will be written for all systems, and tests will be performed by the plant operating staff under the direction of the technical staff of Yankee and its contractors. Systems test approval is by appropriate authority, and is coordinated with the AEC inspection operation.

Initial testing of the vapor container is performed by the fabrication and erection contractor under the direction of the Stone & Webster Engineering Corporation.

Nuclear Start-up and Tests

Following completion of the systems test program, including main coolant loop flushing and hydrostatic testing, the nuclear start-up program is initiated. A summary of the initial tests following loading of the nuclear fuel is as follows:

Cold Critical
Control Rod Worth
Boron Coefficient of Reactivity

Pressure Coefficient of Reactivity
Temperature Coefficient of Reactivity
Effects of Coolant Flow
Control Rod Drive
Scram Circuitry
Heat Balance vs Nuclear Instrument Calibration
Low Power
Part Power
Shutdown Cooling by Natural Circulation
Water Conditioning
Shielding
Monitoring and Alarm
Manual Control
Automatic Temperature Control
Overall Plant and Core Evaluation

404 NORMAL OPERATING PROCEDURES

During the final stages of construction and preliminary systems testing, the manufacturer's manuals on all plant equipment will be accumulated. These manuals will be used as a basis for preparing the Yankee Plant Operators' Manual and for the on-the-job training program of systems testing and operator education. Normal operating procedures affecting each plant station will be detailed in the manual, together with start-up and emergency procedures.

The minimum number of plant operators for normal operation is six, which includes: 1 guard; 2 control room operators; 1 watch engineer; 1 turbine operator; 1 monitoring and auxiliary system man. Other supervisory personnel are "on call" at all times at residences in close proximity to the plant. The plant is manned on a 24 hr a day basis, 365 days per year, to produce electric power as a base load station.

405 EMERGENCY PROCEDURES

Emergency procedures are put into effect whenever the monitoring and alarm system indicates the presence of excessive radioactivity or that a primary plant rupture has occurred. When such alarms are given, the control room operator reduces load and shuts down the entire plant. Even in the absence of any alarm, the control room operator may shut down the plant if, in his judgment, an unsafe or abnormal condition exists. The sequence of events following a major accident, such as a rupture of the primary plant, is as follows:

The steam and nuclear plant is secured while putting the chemical shutdown or safety injection system into operation.

All plant personnel, not essential for securing the plant, proceed immediately by predetermined evacuation routes to the guardhouse at the periphery fence 1,000 ft from the plant proper.

Plant shift personnel considered essential for safely securing the steam and reactor plant, including the watch engineer, two control room operators, the turbine operator and the monitoring and auxiliary system man, assemble at the shielded control room.

All plant supervisors not on duty at the time, including the superintendent, chief engineer and reactor supervisor, are called to the guardhouse where they receive protective equipment and take charge in order to prevent further damage.

The monitoring man provides himself and the emergency essential operating crew with oxygen tank type gas masks and then makes a radioactivity survey of the plant area using portable and plant monitoring equipment.

The reactor operators follow established emergency procedures, as outlined in the operators manual, to minimize plant damage, determine conditions within the vapor container, determine whether core melting has occurred and determine radioactivity levels within the vapor container. The operators establish telephone contact with the guardhouse and further direct the continued evacuation of nonessential personnel from the guardhouse, based on area monitoring readings and the probable condition of the plant.

The site weather stations and area monitoring stations give indications in the control room of weather conditions and radioactivity levels at the plant site and neighboring areas. Calculations of the direct radiation hazard following a major loss of coolant accident indicate that a 2 1/2 hr exposure period is allowable at the periphery fence, 1,000 ft from the vapor container. At the nearest private dwelling exposed to direct radiation, 4,000 ft away, an exposure period of several weeks is allowable. The air-borne radiation hazard must be evaluated at the time of an accident based on the leak rate and prevailing meteorological conditions. Portable radiation monitoring equipment is available in the guardhouse for use by technical services personnel for local monitoring operations.

Essential personnel evacuate the plant proper upon relief or when the plant is deemed secured. Relief shifts of essential personnel are made up at the guardhouse where the necessary protective equipment is issued. Relief personnel are rotated as required to restrict radiation exposure to less than maximum permissible emergency doses until the plant is completely secured.

Return to the plant proper following a major accident is governed by radioactivity levels and the extent of damage.

406 REFUELING PROCEDURES

The detailed description of refueling procedures will appear in the Yankee Plant Operators' Manual. Normally, the refueling operation is performed by the reactor operating crews under direction of the reactor supervisor. Reactor shutdown for refueling is determined by the remaining excess reactivity at power, statistical radiation damage information on fuel damage vs. burn up obtained from in-pile tests on similar fuel assemblies, and the radioactive contamination of the primary systems due to fuel assembly cladding failures. The general sequence of events following determination of scheduled shutdown is as follows:

Reduce the hydrogen level in the main coolant system to equilibrium concentrations at one atmosphere and room temperature by operation of the purification system after shutting down the corrosion control-hydrogen injection system.

Secure plant and start controlled cool down to 100 F while injecting boric acid into the main coolant system to the desired concentration.

Open the vapor container stack valve and purge the container of any accumulated air-borne and gaseous radioactivity at allowable release rates which are based on AEC criteria and meteorological conditions. Open the access hatch and enter the vapor container.

Loosen the reactor vessel head as described in the operators' manual.

Flood the reactor shield tank cavity to a depth of 25 ft with borated water containing 1.6 g of boron per liter as boric acid.

Release the reactor control rod drive shafts from their mechanisms, leaving the control rods in a down position in the core.

Remove the reactor vessel head and internals, such as control rod drive shafts, core hold-down ring, and upper core support barrel, and store in the shield tank cavity under water.

Remove the spent fuel by the fuel handling system under water, using proper health and safety monitoring equipment and maintaining adequate ventilation. Remove single fuel assemblies beginning at the center of the core and working outward. Changing the location of fuel assemblies from the outer region of the core to the center may or may not be

done based on further considerations of two-region or single-region core loadings.

Transfer removed fuel assemblies to the spent fuel pit and store under water in the racks provided. Normally, a dissolved chemical neutron absorber is not required in the spent fuel pit water in order to maintain the stored fuel assemblies subcritical.

Introduce new fuel from the storage vault, through the spent fuel pit, to the shield tank cavity, and lower into the reactor by reverse operation of the fuel handling system.

Replace any reactor control rod required. Control rod withdrawal is monitored by period meter instrumentation. The control rods and extensions are withdrawn from the core into the shield tank cavity where the control section is separated from the extension at the break joint. The control rod is handled by the fuel handling system in the same way as a spent fuel assembly.

Start-up of the plant after refueling is outlined in detail in the Yankee Plant Operators' Manual.

Spent fuel is stored at the plant for a period of time, after which the fuel assemblies are shipped in air-cooled coffins, each coffin carrying several fuel assemblies. Shipping cask loading is performed by the spent fuel pit manipulator and is accomplished under water. Shipment of spent fuel to the re-processing plant is by rail or truck. Appropriate fuel transfer and accountability procedures are used throughout.

407 RADIOLOGICAL HEALTH AND SAFETY

The plant equipment for radioactivity monitoring is used by the plant technical services personnel to maintain safe working conditions at the plant proper. The basic criterion for shielding design is 1/10 the allowable dosage of 300 mrem per 40 hr week at all stations manned full time. A shielding evaluation program after nuclear start-up ascertains that the basic design criterion is not exceeded. Maintenance operations may subject plant personnel to higher levels of radioactivity for short periods of time; however, complete records on each employee are maintained to insure that no individual receives radiation doses in excess of tolerance.

Four area monitoring stations for air-borne activity are located external to the plant fenced periphery. One station is located near the plant hilltop stack. The other three stations are at selected locations with reference to the plant and hilltop stack stations. These stations are simple beta-gamma counter installations which are provided principally, for determining an unsafe condition following an accident. The signals from these stations and the weather stations are sent to the radiation monitoring panel in the control room.

Control and monitoring of liquid discharges are described in Sections 208 and 210.

A site radioactivity monitoring program for background determinations is currently in operation which will monitor liquid, gaseous and solid discharges during normal plant operation to insure that radioactivity levels do not exceed the maximum permissible concentrations prescribed in Handbook 52, U.S. Bureau of Standards.

The radiological health and safety activities are under the direction of the technical services supervisor, assisted by a radio-chemist, a male nurse, and five monitoring and auxiliary system men, on a one per shift basis. Medical services of a consulting nature are available on a contract basis.

Film badges are worn by all plant personnel during normal plant operation. During maintenance periods, both dosimeters and film badges are worn. The film badge service is available on a contract basis with no developing facility provided at the plant site. Continuous personnel exposure records are maintained by the male nurse.

408 PLANT MAINTENANCE AND ACCESS

During normal operation, the plant vapor container is closed. Temperature and radioactivity buildup within the container are monitored. When access into the vapor container is desired, general safeguards apply, details of which will be incorporated in the Plant Operators' Manual. The reactor is shut down. The shielding design of the reinforced concrete structure supporting the reactor does not provide for access to the vapor container with the reactor at high power levels. Likewise, the closed vapor container ventilation system limits access.

If only minor adjustment or maintenance is required, which can be accomplished without hazard from the high pressure primary plant system, the following sequence applies:

Following reactor shutdown, the ventilation system is operated with monitored and controlled discharge to the plant stack.

The main coolant system is not depressurized, but slowly drops from 2,000 psia because of heat losses. Boric acid solution is injected into the main coolant in sufficient quantity to keep the reactor shut down.

The main coolant pumps remain in intermittent service, as required to limit the rate of temperature drop and equalize temperatures in the system.

The pressurizer is operated manually to maintain the system pressure at levels suitable for pump operation.

The closed television circuit is operated to determine conditions within the vapor container.

The maintenance crew entering the vapor container is outfitted with proper clothing and tank type gas masks. Its entry is preceded by a monitoring and decontamination man from technical services.

Entrance is made by way of the double door personnel access hatch, and minor repairs are accomplished.

Major maintenance within the vapor container, such as steam generator tube plugging, is performed on a deferred maintenance schedule at the time of complete plant shutdown and depressurization. The plant power level is reduced by closing the inlet and outlet main stop valve in any defective main coolant loop.

All maintenance operations on contaminated equipment or in contaminated areas are supervised by technical services to see that proper decontamination procedures and radioactivity safeguards are observed.

409 ROUTINE TESTING PROGRAMS

In addition to initial plant start-up testing, routine testing is scheduled continuously.

Main Coolant System Hydrostatic Testing

Any alterations to the high pressure coolant system subsequent to the start-up hydrostatic test are followed by a hydrostatic test to prove the integrity of the new weld or component. Whenever the reactor vessel head is removed for refueling, the primary plant is hydrostatically tested.

Steam Generator Testing and Inspection

The steam generators are deemed by the Massachusetts Department of Public Safety to come under the ASME Boiler and Pressure Vessel Code, Section I, and must receive annual physical inspection in addition to the main coolant system hydrostatic tests. Hand holes are incorporated in the steam generator design, and the decontamination system is operated to allow physical inspection.

Radioactivity

A continuous program of site radioactivity testing began in October 1956. Soil, water, air and plant and animal matter testing for radioactivity are carried on throughout the life of the plant, and thereafter, as required. Following plant start-up, additional radioactivity testing is carried on by the plant Technical Services Department and by subcontract. Plant discharges are monitored on a continuous basis by the monitoring system and, in addition, local monitoring and testing is carried on to ascertain the nature and concentration of the liquid, gaseous and solid discharges. Test records are maintained.

Instrumentation and Control

Nuclear and steam plant instrumentation testing is on a continuous basis. Automatic and manual controls of the plant are checked through a continuous test program and records are maintained.

Water Chemistry

All water in use in the nuclear and steam plant undergoes daily testing for the determination of gaseous, dissolved, and suspended impurities. Log sheet records are maintained.

Fuel Inspection and Testing

Both new and irradiated fuel assemblies are inspected for defects. A research program for detecting failed fuel assemblies after irradiation is being undertaken. A research program for determining isotopic concentration of fissionable materials is being studied, particularly in connection with safety.

Performance

Core capability evaluation is carried on by the preparation of a continuous plant performance record. Performance testing and data on fuel assembly cladding failures are used as a basis for subsequent core design changes.

New Core Nuclear Testing

Each replacement of the core or fraction thereof, or of reactor control rods where changes in material or design parameters have been made, is followed by a program of start-up testing to determine the temperature coefficient, excess reactivity and other nuclear characteristics.

Vapor Container Leakage Rate Testing

Leakage rate testing is described in Section 213.

410 PLANT SECURITY AND SPECIAL NUCLEAR MATERIALS
TRANSFER AND ACCOUNTABILITY

Plant security is under the direction of the security officer who has charge of security education and handling of classified material. All security procedures at the plant are consistent with AEC regulations.

Since there is only one normal point of access for traffic through the chain link fence only one guardhouse is provided at the facility. The guardhouse is manned at all times by at least one properly authorized guard. Normal turnstile access for individuals and gate access for motor vehicles are provided. Exclusion area details are shown on drawing 9699-FY-5A.

Normally, a second guard or continuous chain link periphery patrol are not required. The exclusion area plan provides for an interior chain link fence and a three strand, barbed wire perimeter fence, which is considered to be a barrier to prevent trespassing on the plant property.

Access for a slow speed freight railroad, the Hoosac Tunnel & Wilmington Railroad, is provided through the perimeter fence by automatic gate control from the guardhouse. Likewise, during freight deliveries to the plant proper, the security officer supervises railroad operations through the normally locked gates in the chain link fence.

The Yankee Plant, as a Transfer and Accountability Station, will be approved by appropriate authority, and proper personnel for handling special nuclear materials will be provided. A suitable reinforced concrete storage vault for new fuel is provided in the plant design.

Automobile parking for employees and visitors to the plant is provided outside the chain link fence periphery.

EXCLUSION AREA PLAN				
YANKEE ATOMIC ELECTRIC CO				
STONE & WEBSTER ENGINEERING CORPORATION				
5				
4				
3				
2				
1	ORIGINAL ISSUE			
ISSUE	DESCRIPTION	CHKD.	INSP.	CORR. APPR.
	1-3-57			
<div> <div>0 50 100 150 200 250</div> <div>SCALE - FEET</div> </div> <div>9699-FY-5A</div>				

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5 ACCIDENTS AND HAZARDS

500 GENERAL

It is generally recognized that pressurized water reactors exhibit a high degree of inherent stability due primarily to the large negative temperature coefficient of reactivity associated with the change in density of the coolant moderator with temperature, superimposed on the somewhat smaller negative coefficient of the fuel itself. In this reactor, the instrumentation and control system is designed to take full advantage of the inherent stability and, furthermore, the full power, 392 mw, operating condition is chosen in such a manner as to maximize this effect. At full power, the mixed mean temperature of the coolant leaving the reactor is 525 F and that leaving the hottest channel is 599 F. Since saturation temperature at 2,000 psia is 636 F, even the water leaving the hottest channel is about 37 deg below saturation temperature, with the result that there is no bulk boiling and only a negligible amount of nucleate boiling in a small portion of the core at the hottest channels. Temperature changes, coupled with the negative temperature coefficient of the reactor, act to limit smaller transients, while bulk boiling in the hot channels operates to reduce and limit reactivity in larger transients where the power increases slowly.

The Doppler coefficient, approximately $10^{-5} \Delta k/k$ per deg F, serves to minimize short, fast transients. Even with this small Doppler coefficient, about 5 per cent in negative reactivity is theoretically available starting from the cold condition before the fuel temperature has risen to its melting point of 5,000 F. Because the temperature of some of the centrally located fuel rises to this value before such temperatures are reached in the balance of the core, the effect of the Doppler coefficient is less than 5 per cent. Approximately one fifth of this amount, a 1 per cent change in reactivity, is available at full power to limit a fast rising transient. Under these conditions, the average temperature of the fuel rises from the normal value of 1,000 F to about 2,000 F.

Twenty-four mechanical control rods are provided for regulating the power level of the reactor, compensating for fission product buildup, counteracting the effects of fuel burn-up, and for scramming the reactor either manually or automatically. The control rods are capable of shutting down the hot clean reactor to about 3 per cent subcritical and of shutting down the hot reactor to about 8 per cent subcritical after operation at power has continued long enough to reach equilibrium xenon and samarium concentration. Additional control to bring the reactor to cold shutdown is provided by the chemical shutdown system through which negative reactivity can be introduced at a rate of 0.6 per cent per minute. Very long-time transients, such as xenon oscillations, are easily taken care of by the control rods.

A scram by control rods alone and with equilibrium poison present brings the reactor to approximately 8 per cent subcritical at operating temperatures. The process of bringing the scrammed reactor from operating temperature to the cold condition by introducing chemical neutron absorber into the primary system is aided by the buildup of transient xenon which provides increments of negative reactivity for a period of about 8 hr. Some of the control rods can be left in the out position during the cooling-down period and are thus available for safety at all times and at all temperatures.

Reactor accidents are of three general types; those associated with reactivity insertion, those associated with release of chemical energy in the core, and those associated with mechanical failures of the main coolant system.

Two types of reactivity accident are considered; a low power level accident which might occur at start-up, and a cold water accident which could result either from the addition of cold water to the hot reactor at power or from the addition of pure water to a reactor which is operating with highly borated water. Accidents of this type are minimized by slow operating valves, by having high neutron flux and short period automatic scrams available, and by interlocks arranged so that the difference in water temperatures across a closed stop valve must be 50 F or less before that valve can be opened.

The possibility of a chemical reaction between water and the various constituents of the reactor core has been considered. Experimental results to date indicate that there is no problem of this nature.

One type of mechanical accident is the failure of coolant pumps which will reduce the flow through the core and may cause harmful or dangerous heating of the core. For the present, a procedure is proposed of scrambling the reactor, operating at or near full power, after the failure of three or more pumps.

Another mechanical accident is one which involves a rupture of the main coolant system and the loss of large quantities of water. Without proper functioning of the safety injection system, partial or complete meltdown of the core may follow with resultant release of gaseous and volatile fission products from the fuel. Other accidents, such as clad failures, may release fission products, but only in the case of a rupture of the main coolant system can these fission products escape into the vapor container. This accident, therefore, has been used as a basis for vapor container design.

The results of a partial or complete core meltdown, no matter how improbable, are analyzed to determine possible criticality of the fuel at the bottom of the reactor vessel. Finally, the subsequent buildup of pressure and radioactivity within the vapor container is calculated to establish conditions therein, following an accident of this type.

501 REACTIVITY ACCIDENTS

Start-up Accident

In one type of start-up accident, it is assumed that the control rods are withdrawn at the maximum design rate up to, and beyond, criticality. For such an accident to occur, there must be a series of multiple failures in the nuclear instrumentation and scram systems and, at the same time, there must be errors or negligence on the part of the reactor operators. Ordinarily, the reactor would be scrammed automatically on either a short reactor period or on high neutron flux level. In addition, both flux level and reactor period are displayed on instruments located on the operating console and the operator can take corrective measures.

Because of the importance and relative frequency of start-up operations, start-up accidents have been studied in considerable detail for all research and power reactors. In the case of the pressurized water type reactor, extensive analytical work has been done for this type of accident using analogue computers. The pattern of the accident is, therefore, well understood.

The large negative temperature coefficient in pressurized water reactors results in a large reactivity change from cold shutdown to operating temperature. In spite of this, the negative temperature coefficient of reactivity of a pressurized water-reactor is not effective in the start-up operation until significant power levels are reached.

Fig. 27 shows results of a typical start-up accident involving control rod withdrawal at maximum design rate. Neutron flux level, ϕ , relative to flux level at the beginning of the rod withdrawal, ϕ_0 , is plotted on a logarithmic scale up to $\phi/\phi_0 = 2 \times 10^7$. Above this value, power is plotted on a linear scale in per cent of the designed thermal output, 392 mw. The abscissa is time in seconds after initiation of control rod withdrawal. The reactor design limits the rate of reactivity addition by control rod withdrawal to $1.03 \times 10^{-4} \Delta k/k$ per sec. This rate of reactivity addition is assumed to continue through criticality until the negative temperature coefficient limits the initial power rise of the reactor. In the case analyzed, the maximum value reached in the initial transient is 135 per cent of full power, or 530 mw, and the time available for the operator to take corrective action is several minutes.

The reactivity increase rate assumed for the curve of Fig. 27 is the present design value. This value, $1.03 \times 10^{-4} \Delta k/k$ per sec, is well within safe limits since it corresponds to 74 sec to go from delayed critical to prompt critical. The maximum time to return to criticality from shutdown following a scram from full

START-UP ACCIDENT
WITH
CONTINUOUS ROD WITHDRAWAL

FIG. 27

power would, under the most unfavorable circumstances, be less than 20 min. Both of these times are acceptable from the standpoint of plant operation.

Cold Water and Boron Concentration Accident

Since the reactor has a relatively large negative temperature coefficient of reactivity, a lowering of temperature represents an addition of reactivity. Such a downward temperature change might come about through opening valves which previously had isolated a coolant loop containing water at a temperature below that of the water in the reactor core. A single isolated main coolant loop section contains approximately 400 cu ft of water, while the remainder of the primary system contains 2,600 cu ft.

The reactor may at some time be operated hot with small quantities of boron in the water which are nevertheless significant in terms of reactivity. If a cold water accident occurred under such conditions, it might be aggravated if the water in the blocked-off loop, in addition to being at a lower temperature, contained a concentration of boron lower than that of the water in the reactor. Also, an accident similar to a cold water accident could be initiated by opening stop valves when a difference in boron concentration exists between the water in a previously isolated loop and that in the reactor, even though both are at the same temperature.

A cold water accident is prevented by incorporating in the design a system of interlocks for the main loop stop valves. These interlocks prevent opening of the valves if there is a difference of more than 50 F in water temperature between the loop isolated by the valves and the reactor itself.

Analyses of cold water accidents based on present design concepts have been made. They indicate that, even with a failure of the protective interlocks, the rate of opening of the stop valves will be slow enough so that on opening, with the maximum possible temperature difference between the reactor and the blocked-off loop, an accident of any consequence can not occur. The reactivity effect is limited by the mixing of the small volume of water in an isolated loop with the larger volume of water remaining in the interconnected primary system. In the present design, the time required for the water to complete its travel through a loop is approximately 13 sec, whereas the time required to open stop valves to substantially full flow is greater by a factor of 4. Thus, the cooler water enters the core over a finite time interval during which adequate mixing takes place. Analysis shows that the possible rate of increase of reactivity is less than in the case of the continuous rod withdrawal accident during start-up which has been described. The course of the transient is similar, and the maximum power level reached following a continuous rod withdrawal will not be exceeded.

The possibility of an accident due to a difference in boron concentration is minimized by an operating procedure which calls for sampling and chemical analysis before cutting in a previously isolated loop. Conductivity cells and neutron meters are also possibilities for this purpose. Concentration in the isolated loop is matched to that in the rest of the system on the basis of this analysis before opening the stop valves.

If, through operating error, the concentration of boron in the loop to be cut in is lower than in the rest of the system, the excursion is limited by the slow opening stop valves. In the worst case, if the loop to be cut in contains no boron, there is sufficient reactivity in withdrawn control rods to shut down the reactor.

Loss of Chemical Neutron Absorber

Another type of reactivity accident is an unscheduled decrease in the concentration of the chemical neutron absorber. This could result from the introduction of pure water into the primary system to compensate for loss through a small leak, or by the removal of the dissolved neutron absorber by an ion exchanger. A change in reactivity of less than $.00755 \Delta k/k$, equivalent to going from delayed to prompt critical in 30 sec, presents no serious operational problems. To increase reactivity at this rate would require replacement of 10 per cent of the system volume per minute, about 2,500 gpm, with unborated water. This dilution rate applies to the cold reactor with an initial boron neutron absorber concentration of about 1.6 g per liter. At higher temperatures, the required boron concentration is lower and the dilution rate for the same reactivity change is greater. Since both the maximum pure water make-up rate and the

flow through the ion exchanger are limited to approximately 100 gpm, an unscheduled cleanup of the neutron absorber in the primary system can not cause a significant increase in reactivity.

Another accident is the loss of chemical neutron absorber caused by leakage from the main coolant system slightly greater than charging pump capacity at a reactor temperature and with a core condition that requires dissolved chemical neutron absorber for reactivity control. A leak greater than 100 gpm results in depressurization of the plant and will necessitate shutdown. This is accomplished by injecting water containing chemical neutron absorber from the safety injection system.

Another possible accident might be called a "boron hide-out" accident in which a deposit of chemical absorber which has precipitated within the core is suddenly dislodged and swept out. This would cause an increase in reactivity equal to the amount tied up in the absorber. This is somewhat similar to the reactivity tied up in the voids of a boiling reactor. A deposit of absorber would be essentially black to thermal neutrons and would thus have the same reactivity effect as an equal surface area of control rod. A comparison calculation with control rod worths shows that only 0.5% $\Delta k/k$ effect would result from losing a deposit corresponding to a 4 sq ft neutron absorbing surface from the center of the reactor. A deposit of this magnitude that could be instantaneously released is believed to be impossible, and, since it requires at least this much deposit to effect prompt criticality, it is concluded that no hazard exists.

Continuous Rod Withdrawal at Power

Another type of reactivity accident is continuous rod withdrawal at power. In this case, the reactor is initially operating at or near full power, and a continuous withdrawal of control rods at design speed occurs. It is conceivable, though highly improbable, that such an accident could occur through a combination of equipment and personnel failures.

If a continuous withdrawal of rods occurs, power level increases and reactor temperatures rise as a result of the reactivity addition. With the design reactivity addition rate of $1.03 \times 10^{-4} \Delta k/k$ per sec and minimum temperature coefficient of reactivity, with a chemical neutron absorber in the system, of $-1.6 \times 10^{-4} \Delta k/k$ per deg F, the temperature rises at the rate of 0.64 F per sec. At these slow rates, even if overtemperature control rod insertion devices and high neutron flux level scrams fail to function, the operator still has ample time to shut down the reactor before any damage results. The scram circuitry, including that of the manual scram, is independent of the circuitry which normally programs the rods and, hence, is not affected by failures of the rod programming system.

If the automatic controls fail and if, in addition, the reactor operator does not promptly initiate a manual scram, bulk boiling occurs in the reactor core, thereby compensating for further reactivity additions after the temperature of the water has exceeded saturation. With forced circulation, the boiling is expected to be steady up to 1 per cent reactivity in the voids. The void volume corresponding to 1 per cent reactivity is approximately 3 per cent, and boiling occurs only in a relatively small portion of the core. With the reactor initially at full power, bulk boiling begins in approximately 58 sec. The condition of smooth boiling is expected to persist for 100 sec or up to approximately 160 sec after the continuous rod withdrawal is initiated. This allows sufficient time for the operator to halt rod withdrawal or take other corrective action. Even without such corrective action, it is believed that the bulk boiling effect will limit the transient and terminate the accident at safe reactor temperatures.

502 CHEMICAL ACCIDENT

Even in the event of core meltdown, no release of chemical energy by the reaction of water with the stainless steel cladding of the fuel is expected. While a reaction between components of stainless steel and water is thermodynamically possible, an energetic reaction is not to be expected on the basis of experience in industrial plants where molten stainless steel is mixed with water to produce a fine mesh powdered material.

In the production of fine mesh stainless steel powder by the water granulation method, molten stainless steel is poured in a stream about 1 1/2 in. in diameter. The stream of metal is hit with a high velocity water stream. A whole spectrum of particle sizes is produced in this way, varying from 400 mesh powder to globules 1 1/2 in. or so in diameter. The particles are so slightly oxidized that most of the powder is sold without further processing. No energetic chemical reaction between the steel and the water has been observed.

503 MECHANICAL ACCIDENTS

Loss of Coolant Flow Accident

It is important that thermal damage to the reactor core be prevented even under the most unlikely accident conditions, wherever this is practicable. One type of accident which might result in thermal damage to the core is that involving loss of coolant flow in the primary loop. In order to be certain that the plant design assures the integrity and continued serviceability of the core, various types of accidents involving a rapid decrease in coolant flow have been analyzed. The loss of coolant flow accidents described here assume complete loss of power to all pumps in the primary system. It is highly improbable that such an accident would occur since the four pump motors are divided, 1, 1 and 2, between three sources of electrical power supply, a transformer fed from the main generator leads and two transformers connected to separate, incoming 115 kv lines. These may be considered to be essentially independent sources of power; however, for the purpose of these studies, it has been assumed that both sources would be lost simultaneously.

In these loss of coolant flow accident studies, it has been assumed that there would be no thermal damage to the reactor core if no bulk boiling occurs at the outlet of the hottest channel in the core at any time during the accident. This is a conservative assumption adopted because a detailed knowledge is not now available concerning redistribution of flow within the core when bulk boiling occurs. Without such knowledge, it must be assumed that a redistribution of flow would occur such that burnout would take place at some point within the core.

When pumping power is lost, decrease of flow with time is determined by the initial conditions, the number of pumps failing, the kinetic energy of the system, and the rate of dissipation of energy due to frictional forces within the loop. Figure 29 gives coolant coastdown curves for loss of electric power to all four pumps for a case of no added inertia and the case of added inertia having a kinetic energy of approximately 8 mw sec. Inertia may be added, as an example, by connecting motor-generator sets electrically to the pump motors.

The physical mechanism of the accident and the related thermal and neutron kinetics of the reactor core is as follows:

As the coolant flow decreases, the core outlet coolant temperature increases and the film heat transfer coefficient decreases. The increase in average coolant temperature causes a decrease in reactivity which, in turn, puts the reactor on a negative period and results in a reduction in power level.

MAIN COOLANT FLOW vs TIME AFTER
LOSS OF POWER TO FOUR PUMPS

The uranium dioxide fuel has, relatively speaking, a low thermal conductivity and high heat capacity. The specific power of the reactor and the geometry of the fuel, that is, fuel rod diameter, are such that a high temperature gradient is established in the fuel during steady state operation at or near full power. Consequently, the average fuel temperature is well above the average coolant temperature. During a flow coastdown transient, the large amount of energy stored in the core tends to hold up the heat flux while the film heat transfer coefficient is decreasing. The result of these two effects is to approach a burnout condition at the point of highest specific power and highest heat flux in the core.

Figures 30 and 31 present data obtained from an analogue computer study. The system of equations developed for the study is based on the following assumptions:

Reactor power constant, at 100 per cent, until
scram, at which time power is reduced
instantaneously to zero

Flow coastdown can be represented as an exponential
function approximating the curves of Figure 29

The fuel cladding is in thermal equilibrium with
the fuel and the coolant

All physical properties of the core materials,
except the fuel thermal conductivity, remain
constant and are evaluated at the average core
temperature. The value for thermal conductivity
of the fuel is that corresponding to the highest
fuel temperature

Coolant pressure is constant at the initial value

Figure 30 represents the case of no added inertia. Bulk boiling in the hottest channel occurs in approximately 1 sec for both no scram and scram at time zero. Figure 31 represents the case of added inertia having a stored energy of approximately 8 mw sec. With the added inertia, bulk boiling in the hot channel occurs 3 sec after loss of power to the pumps. Bulk boiling will, however, be prevented if a scram is initiated within 2 to 3 sec after the loss of electric power to the pumps. One second appears to be an unreasonably short time to effect a scram in view of the assumption that the power is reduced instantaneously to zero following scram. On the other hand, it appears practicable to consider a control system which will initiate a scram within 2 sec after failure of electric power to the pumps.

REACTOR CORE OUTLET TEMPERATURES vs
TIME AFTER LOSS OF POWER TO FOUR
PUMPS WITH NO ADDED INERTIA

**REACTOR CORE OUTLET TEMPERATURES vs
 TIME AFTER LOSS OF POWER TO FOUR PUMPS
 WITH ADDED INERTA OF 8 MW SEC**

The present system design does not provide for the coupling of additional inertia to the fluid system. Figure 31 indicates, however, that, for the system constants assumed, added inertia will be required to prevent bulk boiling at the outlet of the core with scram delay times which are reasonable. There is, however, considerable uncertainty with respect to the system constants used in the study. Flow coastdown characteristics of the main coolant pumps are not presently available, neither are accurate values of the thermal conductivity of the fuel; and, as has been pointed out, the relationship between bulk boiling, burnout, and damage to the core is not known. The loss of coolant flow accident, its effects on reactor performance, and design alterations which may be required to eliminate core damage as the result of such an accident will therefore be the subject of detailed analysis under the Yankee Research and Development Program.

Loss of Water Accidents

General

The effects of loss of water accidents without any insertion of borated water from the safety injection system, but including release of contaminated vapor from the flashing of fluid in the primary coolant system, are considered from the following points of view:

- Core again becoming critical
- Core melting down when uncovered
- Resultant pressure rise in the vapor container

In any case involving loss of primary system pressure, automatic scram is effected by the control system. To investigate the possibility of a return to criticality, a series of breaks of increasing size is assumed, a small break equivalent to a 1/4 in. diam opening, a medium break equivalent to a 1/4 in. to a 4 in. diam opening, and a large break equivalent to the complete severance of a 20 in. OD main coolant pipe.

With the condition of the core melting down when uncovered, the largest break results in the most rapid uncovering of the core and the possibility of meltdown.

The maximum pressure in the vapor container results from a large break which releases substantially all the main coolant fluid and allows it to flash almost instantaneously into the vapor container. Pressure builds up within the vapor container before conduction through the sphere, absorption of the heat by interior concrete and other mechanisms for dissipating heat become effective. In order to select for design purposes a size of break that has physical reality, it is assumed that the large break is a complete rupture of a main coolant pipe, plus the simultaneous rupture of one secondary steam line. For this purpose, it is immaterial whether a pipe line or a vessel of the main coolant system ruptures, as long as the break is large and the loss of coolant is rapid and complete in a few seconds.

Small Break

The reactor is equipped with a high pressure charging system having a capacity of 100 gpm. The system has three pumps, one of which is always in service. If there is a leak so small that the loss of fluid is less than, or equal to, the capacity of the charging system, there is no loss of system pressure. For example, it has been calculated that a 1/4 in. diam opening will discharge approximately 25 gpm at 2,000 psia. Accordingly, a leak of this size, or smaller, anywhere in the primary system does not affect reactor operation.

Medium Size Break

If a break larger than 1/4 in. diam opening occurs, a single charging pump can not maintain system pressure, and the primary system pressure will drop.

A medium size break is defined as one equivalent in size to an opening between 1/4 in. and 4 in. in diameter. The 4 in. diam corresponds to an opening area of approximately 1/10 sq ft. Entrainment of water in the steam escaping from such an opening is not an important factor in removing water from the system. A break of these proportions will expel water in three more or less distinct stages:

Solid discharge of subcooled water caused by the pressure

Flash-flow of steam entraining some water

Flow of steam only, once the level of water in the reactor is below the outlet and inlet nozzles

Flow calculations for a medium size break show that the time required to complete solid water discharge is 30 sec with 67 per cent of the weight of water remaining at the end of this time. Two-phase flow is completed in the next 52 sec, with 33 per cent of the original weight of water remaining. Steam production from flashing of water and from decay heat causes the top of the core to uncover in approximately 30 sec more. The core is completely uncovered in an additional period of 110 to 600 sec.

Because so little water is entrained during a blowdown through a 1/10 sq ft or smaller opening, the temperature of the water drops in an orderly fashion, thereby increasing reactivity because of the negative temperature coefficient.

Large Break

A large break is defined as one ranging in size from an opening 4 in. in diameter to a 20 in. pipe severance, 16 in. ID, with two open ends. The corresponding areas are 1/10 sq ft and 3 sq ft.

For a 1 sq ft break, in the middle range of large breaks, pressure blowdown requires 3 sec and will eject one-third of the water from the vessel. The water level drops to the outlet nozzles of the reactor in 5 more seconds. The flashing mixture entrains such large quantities of water that the core is uncovered in 17 more seconds, or 25 sec after rupture.

With a full 20 in. pipe severance, sufficient water is ejected to uncover the core in 12 sec and essentially no water remains after 18 sec.

Calculations show that complete scram of the reactor does occur despite the flow of water and steam upward through the reactor core. A pressure differential greater than 77 psi through the core is required to exceed the gravitational force on the control rods. This pressure drop can not be achieved even under this extreme condition.

Criticality of the Core During Blowdown

A return to criticality may occur during blowdown if there is insufficient control in the rods to hold a new core subcritical at temperatures reached during the transient without a chemical neutron absorber present. However, although this can occur theoretically, an event of this type is highly improbable.

Reactivity increases during a blowdown transient because of the decrease in temperature. Two factors present in this reactor, void production and uncovering of the core, tend to counterbalance the reactivity increase.

The multiplication factor k_{eff} as a function of temperature is shown in Figure 10. k_{eff} as a function of void volume is shown in Figure 9. k_{eff} as a function of height of water in the core is shown in Figure 34.

The change in these variables with respect to time has been calculated as a function of size of opening. Breaks smaller than 1/4 in. constitute no problem. In a 1/10 sq ft, medium-size break, the k_{eff} goes below unity owing to void production and reactor scram. The k_{eff} with a clean core drops to a value of less than 0.96 approximately 1 min after the rupture, as the temperature falls. After this, the reactor is held subcritical by voids and control rods as core uncovering proceeds. With large openings, 3 sq ft, the rapidity with which the water is blown out of the reactor causes the water level to drop abruptly. The void coefficient, approximately 0.03% $\Delta k/k$ per vol % steam, is the controlling factor, and there is no return to criticality.

Decay Heat

Following reactor scram, which should be completed within 2 sec after a drop of system pressure, decay heat will be given off at the rate indicated in Figure 35. As long as the core remains covered, the decay heat will be extracted by boiling water. The rate of heat loss to the boiling water is such that the core will be cooled and the temperature of the fuel will drop.

Boiling will take place in the core and fuel temperatures will decrease as the water approaches saturation temperature during the pressure discharge. As the water level falls in the core, the temperature of the fuel tends to rise but can be held within safe limits by use of the safety injection system.

Vapor Containment

The vapor container is designed to retain all vapors, gases, liquids and solid materials released as a result of a loss of coolant accident. The maximum loss of coolant accident employed in the vapor container design consists of:

Complete severance of one 20 in. main coolant line,
with two open pipe ends

Simultaneous rupture of one secondary main steam line inside the vapor container. The placement of each main coolant loop in a separate concrete shielded compartment and the installation of a nonreturn valve in the main steam line from each steam generator limit this part of the accident to the rupturing of a single secondary main steam line

REACTIVITY VS HEIGHT OF WATER

FIG. 34

FIG. 35

Detachment of an object or metal fragment from the pressurized system in such a way that it acquires kinetic energy, which, unless restrained or stopped by a barrier, might perforate the steel shell of the containment vessel, thus releasing contaminated vapor following the loss of water accident.

Figure 36 shows the initial pressure transient following the release of 186,000 lb of fluid from the main coolant system and one secondary coolant circuit into the net volume of the vapor container of 840,000 cu ft. The maximum differential pressure between the concrete compartment and the vapor container is 6 psi, and this pressure is reached in 0.2 sec. A port area of 400 sq ft in any one loop shield compartment is provided to limit the pressure differential across the concrete walls to this value. The concrete walls are designed for a maximum differential pressure of 8 psi. All coolant is released from the main coolant system within approximately 18 sec and equilibrium is attained inside the vapor container at a maximum pressure of 34.5 psi gage, or 49.2 psia. The corresponding vapor temperature is 249 F and the energy released is 94×10^6 Btu.

Figure 37 shows the long-time effect after the release of vapor and initial pressure rise to 34.5 psi gage. During the first 2 hr, there is a marked decrease in pressure due to thermal radiation and convection from the uninsulated vapor container shell and due to the diffusion of heat into the inner concrete structure. Subsequently, there is a gradual decrease in pressure with a small secondary rise, peaking in 4 hr at 15 psi gage, due to the continued release of decay heat from the reactor core.

The air-vapor mixture pressure within the vapor container after the maximum loss of coolant accident is based on the assumption that the total internal energy of the fluid remains the same before and after the rupture. This is based on the conservation of energy relation:

$$Q = AW + \Delta E$$

Where

$$Q = \text{Net heat release, Btu}$$

$$A = \text{Reciprocal of mechanical equivalent}$$

$$W = \text{Mechanical work performed, ft-lb}$$

$$\Delta E = \text{Change of internal energy, Btu}$$

During the brief interval after the initial burst, it is assumed that there is no heat loss, or $Q = 0$. There is no work done, since the fluid begins and ends in a state of rest, or $AW = 0$. Therefore, the internal energy before the accident is the same as that after the accident, or $\Delta E = 0$.

PRESSURE RISE IN VAPOR CONTAINER
vs.
ELAPSED TIME AFTER RUPTURE

PRESSURE IN VAPOR CONTAINER
FOLLOWING A NUCLEAR ACCIDENT
NO INSULATION ON VAPOR CONTAINER SHELL

The summary of the principal data for the major loss of water accident is as follows:

Main coolant pressure, upper operating limit, psia	2,150
Average temperature main coolant, upper operating limit, F	518
Total volume of water in main coolant system, cu ft	1,600
Reactor	150
Pressurizer	800
Steam generators	548
Piping	20
Pumps	52
Miscellaneous	<u>52</u>
Total	3,170
Total volume of steam in main coolant system, cu ft	110
Total volume of water in one secondary loop, cu ft	570
Total volume of steam in one secondary loop, cu ft	590
Gross volume of vapor container, cu ft	1,020,000
Net effective volume of vapor container, cu ft	840,000
Weight of fluids in main coolant system and one secondary circuit, lb	186,000
Internal energy of released fluids, Btu	94,000,000
Vapor flashed from main coolant, per cent	32
Final pressure, psia	
Vapor	29.2
Air	<u>20.0</u>
Total	49.2
Total, psi gage	34.5
Final temperature, F	249

Subsequent to the initial pressure release, the following heat transfer effects proceed simultaneously, with the net integrated effect of these on the vapor container pressure shown in Figure 37. These effects are as follows:

Decay heat is released from the reactor core in accordance with the following relation:

$$\frac{P}{P_0} = .076 \quad \Theta^{-.2}.$$

Where P = rate of heat release after time Θ , mw
 P_0 = initial rate of heat release, 482 mw
 Θ = time after reactor shutdown, sec

The rate of release of decay heat is dependent upon the number of hours which the core has operated; the longer the operating period, the greater the rate of release of decay heat.

This relationship is based on an infinite time of operation, which corresponds substantially to the rate of heat release after many hours of operation, and is thus conservative.

Heat is lost by radiation and convection through the spherical shell. This rate of heat release depends on the ambient temperature within the vapor container, the outside ambient temperature and a radiation and convection coefficient which available data indicate for large spheres is 2.2 Btu per sq ft, per hr, per degree F. The outside ambient temperature is taken as 70 F, the average of a summer day. The initial ambient temperature within the vapor container prior to the accident is 120 F.

Heat is absorbed by the vapor container metal. This rate of absorption is proportional to the ambient temperature within the vapor container. The weight of the containment vessel is approximately 2,500,000 lb and the specific heat is 0.12 Btu per lb, per degree F.

Heat is slowly released from metal parts which have been operating at normal temperature. This rate of heat release is proportional to the ambient temperature within the vapor container. The insulated metal parts weigh approximately 1,500,000 lb, and have a specific heat of 0.12 Btu per lb per degree F. Insulation is provided by 4 in. of Foamglas with an average thermal conductivity of 0.55 Btu per sq ft, per degree F, per hr, per in. of thickness.

Heat is absorbed by the internal concrete structures. The rate of diffusion of heat into the concrete with time is dependent on the ambient temperature within the vapor container. The temperature-time relationship for the concrete was determined by use of the Schmidt method, using a specific heat of 0.22 Btu per lb, per degree F, a thermal conductivity of 0.5 Btu per sq ft, per degree F, per hr, per ft of thickness, and a density of 150 lb per cu ft.

The ambient temperature is an independent variable in all of these factors, except decay heat, contributing to the redistribution of heat. This permits the determination of the new vapor temperature and total pressure of the air-vapor mixture after any elapsed time. Only the first two and last items in this list contribute importantly to the redistribution of heat. The results of the calculation are shown in Figure 37. The conservative assumption has been made that there is no condensation during the first few minutes after the initial rupture to reduce the calculated initial pressure.

Missile Protection

Although it is believed that no plausible missile could be released by the main coolant system, protection is provided by the inner concrete structure. It is considered that the ductile austenitic stainless materials of construction of the main coolant system piping will not fail in a brittle manner when in contact with the hot compressed coolant. It is likewise considered that the stainless clad, carbon steel reactor vessel, fabricated and tested according to the best techniques and under the proper codes and in contact with the hot compressed fluid, will not fail, and is not a feasible source of missiles.

The internal reinforced concrete structure serves as a secondary biological shield, as structural support for equipment and, in addition, provides a missile barrier. For biological radiation dose considerations, the walls and bottom of this structure consist of 4.5 to 6 ft of reinforced concrete, and the top or upper floor level consists of a minimum of 3 ft.

Loss of Load Accident

If flow of steam from the steam generators is accidentally stopped by malfunction and closing of the steam throttle valves or by turbine trip-out, it is important that thermal damage to the primary system be prevented. A study of this problem is currently in progress, however, preliminary information has been developed. Present design is predicated upon no control interlocking of turbine trip-out to effect automatic control rod run-in (or reactor scram), and only audible and visible alarm signals are energized. However, if this design becomes unreasonable, interlocking will be provided. Preliminary figures indicate a steam dump bypass flow of 7 per cent of full steam flow as adequate for safe maximum temperatures in the primary and secondary systems utilizing only the negative temperature coefficient for shutdown.

504 CORE MELTDOWN

General

Partial or complete melting of the reactor core following rupture and the loss of all water from the main coolant system would result in release of fission products into the vapor container. To prevent such an occurrence a safety injection system with ample storage of borated water, a highly reliable means of introducing this borated water into the reactor vessel, and a dependable power supply is provided.

Whether melting of the core can be prevented may depend on the time available for the operator to introduce the borated water for cooling. The time interval between the initiation of the accident and the start of core melting is available to start injection of borated water into the reactor vessel. An analysis has been made to ascertain the length of this time interval. The core meltdown event has been carried to its conclusion, assuming no injection of borated water, to determine the rate of melting, fission product release and possible criticality of the melted fuel.

Mechanism of Core Meltdown

Under steady state full power conditions, 392 mw thermal, the average center fuel temperature is 1,362 F; the corresponding fuel cladding surface temperature is 570 F. The maximum fuel temperature at the center of the hottest pellet in the reactor is 4,500 F, while the corresponding cladding temperature is 642 F. The steep gradient in temperature between the center of the pellet and the fuel cladding surface is the result of the high heat flux which prevails and the low thermal conductivity of sintered UO_2 . A significant decrease in the rate of heat removal from the surface of the fuel cladding will cause the temperature gradient to decrease and the cladding temperature to approach the fuel center temperature. The melting point of stainless steel is 2,800 F, while that of uranium dioxide is approximately 5,000 F. Therefore, stainless steel cladding will melt before the fuel melts if heat generation continues within the fuel while the rate of heat removal from the cladding surface decreases, as is the case if the fuel cladding tube is surrounded by steam. As soon as any point on a fuel rod reaches the melting point of 2,800 F, the cladding will rupture, allowing some of the gaseous and volatile fission products to escape. When a sufficient portion of the structural material in the fuel assembly has reached the melting point of stainless steel, the assembly fails and fuel pellets fall to the bottom of the reactor vessel.

Rate of Core Melting

Calculations have been made to determine the rate at which the core melts and the data are presented in Figs. 38 and 39.

CORE MELTDOWN vs TIME AFTER RUPTURE
1/10 FT² RUPTURE

CORE MELTDOWN vs TIME AFTER RUPTURE
3 FT² RUPTURE

The fractions of the core melted for 1/10 sq ft and 3 sq ft breaks are shown as a function of time. The calculations are based on the steady state power distribution within the core from which the variation in decay heat generation rate follows.

Assuming no injection of borated water following a 1/10 sq ft rupture which uncovers the core, the temperature of the dry core rises in accordance with the decay heat, as shown in Fig. 35, and heat capacity of the system. The first tube melts in 12.5 min and many other tubes melt shortly thereafter. As shown in Fig. 38, half the fuel cladding tubes melt in 24.5 min. It is important to note that a time interval of 12.5 min is available before any melting occurs. This period of time is considered sufficient for the operator to take corrective action and inject borated water into the reactor vessel to cover a substantial portion of the core.

A 3 sq ft break without use of borated water injection follows the same pattern on a shortened time scale. Cooling occurs by convective boiling for 12 sec, after which there is no further heat removal. At the end of the cooling period, the average temperature of the pellets is approximately 510 F. The first tube starts to melt 5.4 min after the accident, as indicated in Fig. 39, and this time is available for the operator to take corrective action. The first pellets are released and fall to the bottom 7 min after the accident. Twelve minutes later, 40 per cent of the pellets are in the bottom of the vessel.

Criticality Consideration Following Melting of the Core

Calculations have been made to determine whether a criticality problem exists after the fuel pellets have fallen to the bottom of the reactor vessel. If all the fuel pellets are stacked in the bottom of the vessel, the maximum k_{eff} is 1.06 at 300 F, and the maximum k_{eff} , 1.04 at 380 F. This calculation assumes that the fallen fuel contains no steel or fission products present in the pellet water mixture, although, if melting occurs by decay heat, some fission products must be present. Optimum geometry and a water-to-equivalent uranium volume ratio of 0.82 (random packing) is also assumed.

Since the maximum value of k_{eff} is greater than 1.0, criticality is attainable with water present if some quantity less than all the fuel drops to the bottom of the pressure vessel. At 300 F, however, at least 65 per cent of the fuel must fall to be critical; at 380 F, 75 per cent is required.

Therefore, the time required for sufficient fuel pellets to fall to the bottom of the vessel to achieve criticality is approximately 20 min. (See Figure 39.) In view of the relatively long time period and the conservative assumptions made in the criticality analysis, it is believed that adequate time is available for remedial action to be taken. However, if later refined calculations indicate that criticality of pellets in the bottom of the vessel is a problem, a neutron poison grid can be installed to eliminate the problem.

Fission Product Activity at the End of Core Cycle

At the end of the core cycle, a variety of fission products is present in the fuel material. An analysis of the gross gamma activities of the gaseous and volatile fission products has been made, since fission products of this type can be released if core melting is not prevented. It is assumed that the core has been operated at full power of 392 mw for an infinite length of time. This is a conservative assumption which yields somewhat higher activities than those which might actually be present.

The noble gases and the halogens are considered to constitute the gaseous and volatile fission products. Altogether, 12 isotopes of bromine, krypton, iodine and xenon are adjudged to possess significant gamma activity to be considered. Five elements are volatile at, or below, 450 C in combination with oxygen and/or one of the halogens. These elements are arsenic, molybdenum, antimony, tin, and tellurium. Of these elements, the isotopes of arsenic and tin have a low yield in the fission process. Fifteen isotopes of the other three elements have been examined and significant activities have been included in the totals. Gross fission product gamma activities are given in Table 7 at three different times after reactor shutdown from full power, 392 mw.

Table 7

Fission Product Gamma Activity Following Shutdown

(In units of 10^{18} mev/sec)

Time after shutdown	<u>0</u>	<u>5 min</u>	<u>1 hr</u>
Gases	7.64	4.31	2.66
Elements volatile as compounds	<u>.81</u>	<u>.73</u>	<u>.44</u>
Total	8.45	5.04	3.10

Fission product gamma activity is plotted graphically as a function of time up to 1 hr after shutdown in Fig. 40. The lower curve scale of 10^{19} gives the gamma activity of all the fission products following shutdown; the upper curve scale of 10^{18} includes only the gaseous and volatile fission products and may be compared with the data presented in Table 7.

Fission Product Release to Vapor Container

The fission product gamma activity which is present in the core after infinite time of operation at 392 mw is shown in Fig. 40. Following an accident in which no use is made of the safety injection system and complete core melting results, the activity attributable to gaseous and volatile fission

products could be released into the vapor container. If the safety injection system is used but is not fully effective, partial melting occurs and the activities released to the vapor container are the product of the values shown in Fig. 40 and the fraction of the core melted. For a 1/10 sq ft break in the main coolant system, the fraction of the core melted as a function of time is given in Fig. 38. Similar fractions for a large break of 3 sq ft are given in Fig. 39.

FIG. 40

FISSION PRODUCT GAMMA ACTIVITY AFTER INFINITE
TIME OF OPERATION AT 392 mw POWER

505 HAZARDS FROM REACTOR ACCIDENTSMaximum Credible Accident

In foregoing sections analyses have been made of a number of accidents the most serious of which is the large loss-of-water accident. Such an accident might occur through rupture or severance of a 20 inch main coolant line resulting in depressurization and virtually complete loss of water from the primary system. The danger here, of course, is the possibility of core meltdown and release of fission products to the vapor container, thereby causing a radiation hazard to the public. To prevent such an occurrence a safety injection system which is described in Section 219 is provided. It is believed that this system is designed and can be administratively controlled to assure instant availability in case of need and cannot in any credible way be disabled or rendered inoperative by the primary effects of the accident. In the opinion of Yankee Atomic Electric Company and its technical advisors, the maximum credible accident can be defined as this large loss-of-water accident, with core melting prevented by the safety injection system, no release of fission products from the core, and therefore no hazard to the public.

Hypothetical Accident

The unique danger from a nuclear reactor installation is the accidental release of fission products from the plant and the creation thereby of external radiation hazards. The present state of reactor technology demands that all reasonable measures be taken to guard against even the most unlikely event by incorporating effective safety features in the plant design. Accordingly, even though the maximum credible accident does not result in the release of fission products and cause external radiation hazards, a hypothetical accident in which such a release does occur has been postulated and analyzed in order to evaluate the effectiveness of containment and other safety features which are incorporated in the plant design.

The hypothetical accident is based on the following assumed conditions and sequence of events:

A 20 in. pipe severance occurs in the primary system resulting in depressurization and virtually complete loss of water from the primary system.

Blowdown of the primary system results in an initial pressure rise in the vapor container to 34.5 psi gage, decreasing to approximately 15 psi gage after 2 hr.

For unexplained reasons, partial core meltdown occurs.

No criticality of the fuel pellets released by clad melting occurs.

20 per cent of the gaseous and volatile fission products present in the core after 10,000 hr of reactor operation at 392 mw are released to the vapor container and dispersed homogeneously therein.

Gaseous and volatile fission products are released instantaneously from the fuel into the vapor container, although the release actually would occur after a finite time interval which would begin several minutes after the start of the accident.

The vapor container has a leak rate of 70 cu ft per hr with the vapor container internal pressure at 15 psi above atmosphere, and this leak rate and driving head are assumed to continue indefinitely.

Direct Radiation On-site

In the hypothetical accident, fission products are released into the vapor container, and a radiation source exists at the site. Calculated radiation levels at a point external to the vapor container are based only on that activity which is not attenuated by the internal secondary shield. To obtain radiation levels, no credit is taken for any delay time due to slow melting of the core before release of the fission products begins. Figures 41 and 42 show gamma dose rates and integrated doses as a function of distance

GAMMA DOSE RATE FOR OUTSIDE EXPOSURE

INTEGRATED GAMMA DOSE
FOR
OUTSIDE EXPOSURE

from the sphere. The control room, which has additional shielding, provides a place where essential plant operating personnel can gather for protection from radiation. The total direct dose received under these conditions is less than 200 mr in the first hour and less than 2 r in the first 24 hours following the release. All plant personnel other than essential personnel will proceed to the guardhouse on the sounding of a clearly audible signal. Since a period of several minutes is available before a significant portion of the core melts, plant personnel can take stations or evacuate the plant without receiving harmful direct radiation doses.

Direct Radiation Off-site

Dose rate due to direct radiation emanating from the vapor container in the case of a large primary system rupture followed by partial core meltdown is shown in Fig. 41 as a function of distance from the source. Integrated doses for 5 minutes and for 1 hour after the instantaneous release as a function of distance are shown in Fig. 42.

The dose at the public road across from Sherman Pond, 1,300 ft from the vapor container, is 5 r during the first hour after release. Hence, several hours are available to remove persons and vehicles which might be on the road at the time to a safe distance from the site. This is based on the once-in-a-lifetime direct radiation dose limit of 25 r indicated in National Bureau of Standards Handbook 59.

Because the power plant is located at the bottom of a deep narrow valley, direct radiation from the vapor container does not reach inhabited buildings, except for one dwelling approximately 4,000 ft distant in the down-river direction. Since the integrated dose for the first hour is 20 mr, the occupants of this house could remain there for several weeks without serious exposure. In the up-river direction, there are no buildings within 9,000 ft,

and beyond that point all buildings are shielded by hills.

All buildings east of the river and within one mile of the site are owned in fee by Yankee Atomic Electric Company or New England Power Company and are considered to be under administrative control of these two companies.

Vapor Container Leakage and Air-borne Radiation

In the hypothetical accident, 20 per cent of the gaseous and volatile fission products are assumed to be homogeneously dispersed in the vapor container. Leakage from the vapor container at the assumed leak rate will release these fission products to the atmosphere and, under certain meteorological conditions, they can be carried to populated areas where they may be inhaled or ingested.

Of the volatile and non-volatile fission products in the core, radio-iodine and radio-strontium provide the controlling activities with respect to the inhalation dose, with iodine being selectively absorbed by the thyroid and strontium by the bone. For the purpose of this report, it has been conservatively assumed that 20 per cent of all the iodine and strontium are released from the core even though the release of strontium has been reported more nearly 1 to 5 per cent. The total activity of iodine and strontium assumed to be in the core is:

<u>Activity</u>	<u>Curies</u>
Iodine - 131	1.0×10^7
Iodine - 132	1.6×10^7
Iodine - 133	2.3×10^7
Iodine - 134	2.7×10^7
Iodine - 135	2.1×10^7
Strontium - 89	1.3×10^7
Strontium - 90	2.4×10^5

Based on KAPL - 1178, it has been shown that the integrated 60-day dose to the thyroid from the inhalation of iodine - 131 is approximately a factor of 10 greater than the dose to the bone from the inhalation of a curie-equivalent of strontium - 89. Since the radio-iodine activity as iodine - 131 equivalent is approximately 1.8×10^7 curies as compared to the strontium - 89 activity of

1.3×10^7 curies, and since the dose to the bone due to strontium - 89 is comparable to the dose due to strontium - 90, the iodine - 131 dose to thyroid was selected as the controlling dose.

The total radio-iodine activity emanating from the vapor container is assumed to have a concentration of 2.8×10^7 microcuries per cu ft. Based on the Sutton Continuous Point Source Equation and using in-valley meteorological conditions presented in Professor Austin's report the concentrations of radio-iodine, as iodine-131 equivalent, 100 ft below the center of a radioactive cloud, which is over the nearest inhabited area 4,000 ft away, are as follows:

<u>Meteorological Condition</u>	<u>Wind Velocity, Fps</u>	<u>Radio-Iodine Concentration, Microcuries/ml</u>
Inversion	3.3	1.6×10^{-6}
Moderate Lapse	19.7	6.9×10^{-9}
Unstable	16.4	2.3×10^{-9}

This tabulation and additional meteorological information indicate that the highest concentration of activity would occur under an inversion condition with a low velocity down-valley air movement. If the accident occurs under these conditions, the leading edge of the radioactive cloud reaches the nearest inhabited area approximately 20 minutes after release of fission products from the vapor container begins.

The once-in-a-lifetime off-site dose for ingestion and inhalation of air-borne radioactivity has not yet been established by the AEC, and there exists some difference of opinion on the subject. Lacking a definitive allowable dose, values suggested by K. Z. Morgan, W. S. Snyder, and Mary R. Ford in their paper, Maximum Permissible Concentration of Radioisotopes in Air and Water for Short Period Exposure, presented in 1955 at the Geneva Conference on the Peaceful Uses of Atomic Energy, have been adopted. These are:

Dose Criterion
Following Exposure

Maximum Permissible Radio-Iodine
Concentration for 8 Hr Exposure,
Microcuries/ml

0.3 rem in week	7.0×10^{-8}
15.7 rem in year	1.7×10^{-6}
150 rem in 70 yr	1.7×10^{-5}

Dr. Shields Warren has stated that he believes a dose of 50 rem to the thyroid may show clinically detectable effects, while a dose of 15.7 rem would probably provide no clinical indication. On this basis, 15.7 rem in the year following exposure has been taken as the off-site, once-in-a-lifetime internal dose.

Based on the assumption that a person is 4,000 ft from the plant and 100 ft below the center of the radioactive cloud, and taking no credit for radioactive decay, the doses received under various meteorological conditions are as follows:

<u>Meteorological</u> <u>Condition</u>	<u>Thyroid Dose,</u> <u>rem in Year</u> <u>Following Exposure</u>
Inversion	15.
Moderate Lapse	0.064
Unstable	0.021

Comparison of this tabulation with the 15.7 rem dose limit adopted shows that in all cases the dose received in 8 hours is below the limit. Thus, from this analysis it is clear that, even under the worst meteorological conditions, the hypothetical accident does not result in excessive concentrations of radioactivity in the nearest inhabited area.

506 CONCLUSIONS

This pressurized water reactor possesses inherent stability because of its negative temperature and Doppler coefficients. Since it is normally operated with the coolant moderator near its saturation temperature, further stability results from the formation of steam voids in any extensive rising power excursion. In addition to this inherent stability, mechanical control rods, capable of making the hot reactor subcritical, are provided to regulate power level and to control reactivity throughout the power production lifetime of the core. A supplementary chemical control system is provided to bring the reactor to cold shutdown.

The plant design and the selection of materials provide four sequential barriers to the escape of fission products. These barriers, in order, are:

Oxide Fuel - The noncorrosive UO_2 fuel acts as a first barrier to contain large percentages of the fission products within its matrix.

Stainless Steel Cladding - The fuel rod cladding with only two end welds per full length tube acts as a second barrier to escape.

Main Coolant System - The third barrier to escape is the high integrity main coolant system.

Vapor Container - This barrier acts as a fourth line of defense in the event of fission product release from the main coolant system.

An additional geographical barrier is inherent in the site selected for the plant. The nearest privately owned and occupied dwelling is approximately 4,000 ft away, and the population density within the first five-mile radius is 25 people per square mile.

Several reactor plant accidents have been investigated and analyzed. Accidents involving reactivity additions during start-up and at full power result in transients but give every indication of leveling off at power levels that are neither harmful nor dangerous. Accidents involving release of energy through chemical reaction between the water and the metallic constituents of the core are considered impossible because of the materials employed. Mechanical accidents in the form of pump failures with ensuing decrease or loss of coolant flow can be handled without dangerous power or reactivity excursions. In failures of one or two pumps, stability is regained with practically no increase in temperature level, even without scram. In failures of three or four pumps, however, a low flow scram occurs and the reactor is kept under control by this means.

Among the mechanical accidents that have been analyzed is one caused by a break in a 20 inch main coolant line at the worst possible location and involving loss of all water from the main coolant system. This is considered to be the maximum credible accident.

In none of these accidents is there any melting of the core, any release of gaseous and volatile fission products to the vapor container, nor any hazard to the public.

However, an analysis has been made of a hypothetical accident in which core melting and fission product release are assumed. An accident has been examined in which it is assumed that a large break occurs in the main coolant system; virtually all water is lost from the system; partial core meltdown occurs; and 20 per cent of the gaseous and volatile fission products are released to the vapor container. The analysis shows that there would be no hazard to the general public because of direct radiation from the vapor container. Since the vapor container has a finite leak rate, some of the fission products may escape to the atmosphere and, under certain meteorological conditions, the escaping fission products may be carried to nearby inhabited areas. At the nearest community, however, an 8 hour exposure to the indicated concentration of radioactivity, under the most unfavorable meteorological conditions, would result in less than tolerable once-in-a-lifetime inhalation and ingestion doses.

Yankee Atomic Electric Company, therefore, concludes that this reactor can be operated without undue hazard to the public health and safety.