DOCKET NO. 50-29

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DRAFT

PROPOSED AMENDMENT NO. 4

To

Part B, License Application
AEC Docket No. 50-29

YANKEE ATOMIC ELECTRIC COMPANY

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All other figures and drawings previously included in sections that have been revised will still follow same page as before.

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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,

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

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

The effect of chemical neutron absorber in the coolant moderator during plant warm-up is such as to decrease the negative temperature coefficient. However, it is not anticipated that enough chemical poison will ever be present to cause the temperature coefficient to go positive. The concentration required for a zero coefficient is 2.6 grams of boron per liter of coolant in the hot reactor, or 2.3 grams per liter when the reactor is cold. On the other hand, 2.1 grams per liter provides 2 per cent shutdown of the cold clean core without any control rods.

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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. It should be noted that the reactor is always subcritical at boron concentrations which could result in positive temperature coefficients, even with all control rods withdrawn.

The result of these calculations are given in Table 3.

Table 3

Temperature Coefficient of Reactivity

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

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

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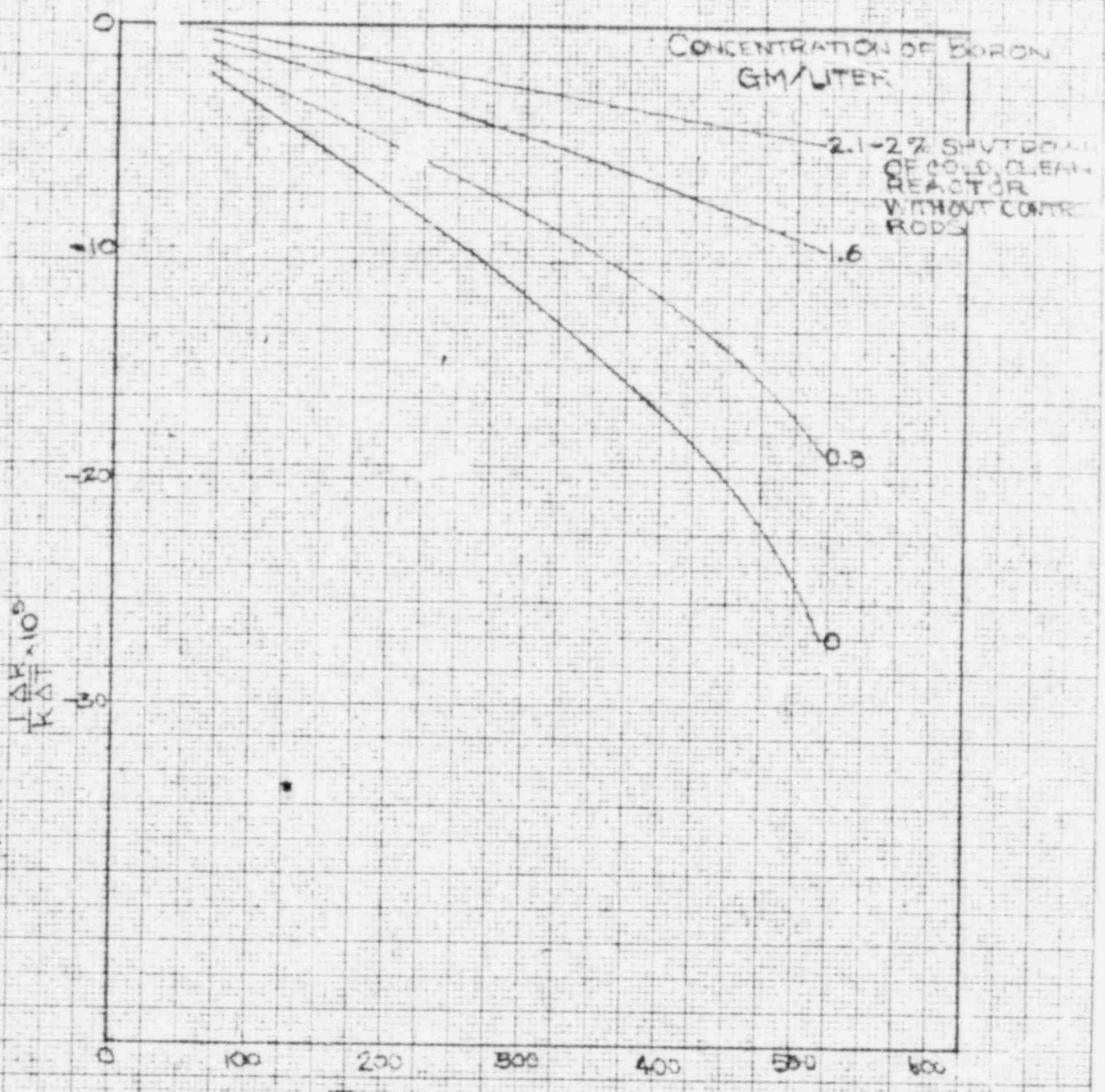
FIG. 8



TEMPERATURE COEFFICIENT OF REACTIVITY VS TEMPERATURE

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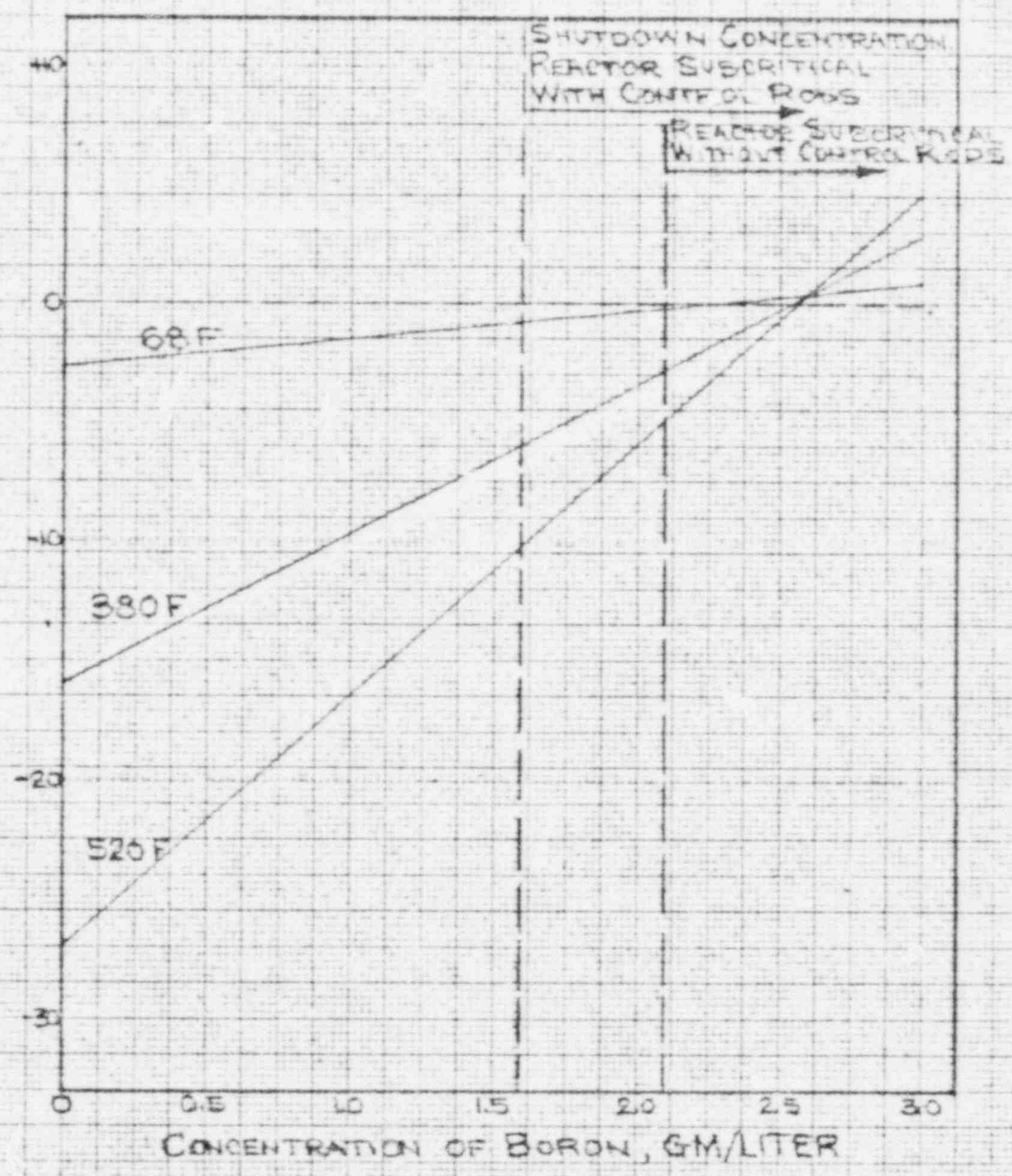
REV. 7/15/57

K+E 10 X 10 TO THE CM. 359-14 KEUFFEL & ESSER CO.



TEMPERATURE COEFFICIENT OF
REACTIVITY vs. BORON CONCENTRATION

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as a pressure coefficient of reactivity. Since this factor is a function, among other things, of the total neutron absorption in the moderator, two values have been computed, one without chemical neutron absorber in the coolant and one with chemical neutron absorber. The data are shown in Table 4.

Table 4

Pressure Coefficient of Reactivity

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

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

The pressure changes in the primary system due to changes in the temperature within the pressurizer, which result from the on-off type of control, are smaller than those associated with plant transients and, in general, take place over a relatively long period of time. 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 boradening of the neutron absorption resonance peaks in U-238 with increasing temperature is a rapid effect and results in a "prompt" negative temperature coefficient of reactivity.

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

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

$$\frac{1}{p}\frac{\partial p}{\partial T} = -\left(\frac{N_o}{\rho\Sigma_s}\right)\frac{\partial \text{ Resonance Integral}}{\partial T}$$

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

TABLE 5

The Doppler Coefficients

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

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

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

Void Coefficient of Reactivity

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

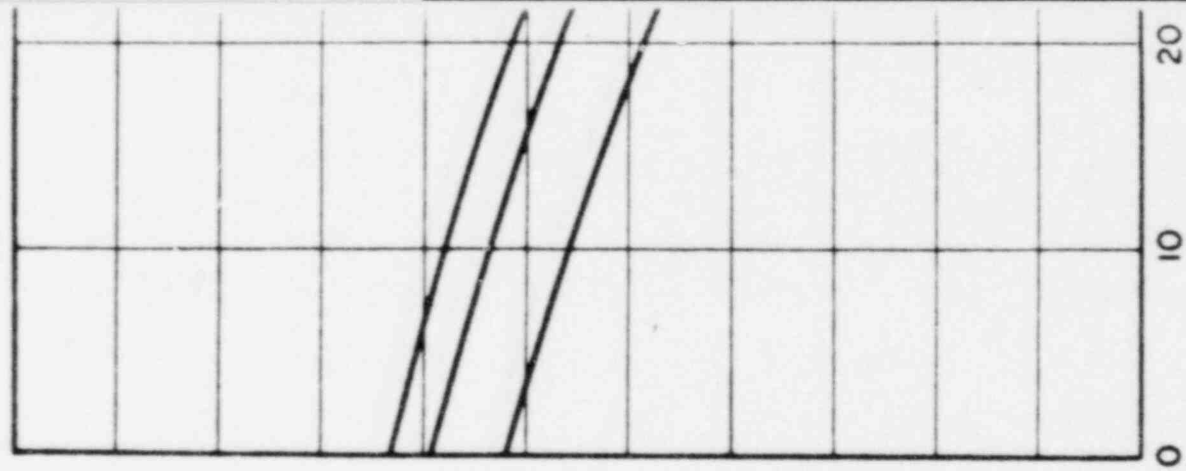
The reactor core is designed so that, in normal operation, bulk boiling does not occur even in the hottest channel. Under accident conditions, however, it is conceivable that bulk boiling may occur. The reactivity of the reactor then may be expected to be altered by the presence of steam voids. The effect of steam voids on reactivity is evaluated quantitatively and expressed as a coefficient of reactivity. In the operating range where the average temperature of the coolant is 508 F, the void coefficient of reactivity is negative with a value of -0.3% $\Delta k/k$ per % void. When there is no chemical poison in the coolant the effect of voids on the k_{eff} of the core without chemical poison is shown in Figure 9 in which curves are plotted for three mean core temperatures. As the temperature of the reactor is lowered, k_{eff} increases and more reactivity becomes available; therefore, a larger per cent void is required to shut down the reactor.

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

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



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

REV 7-15-57

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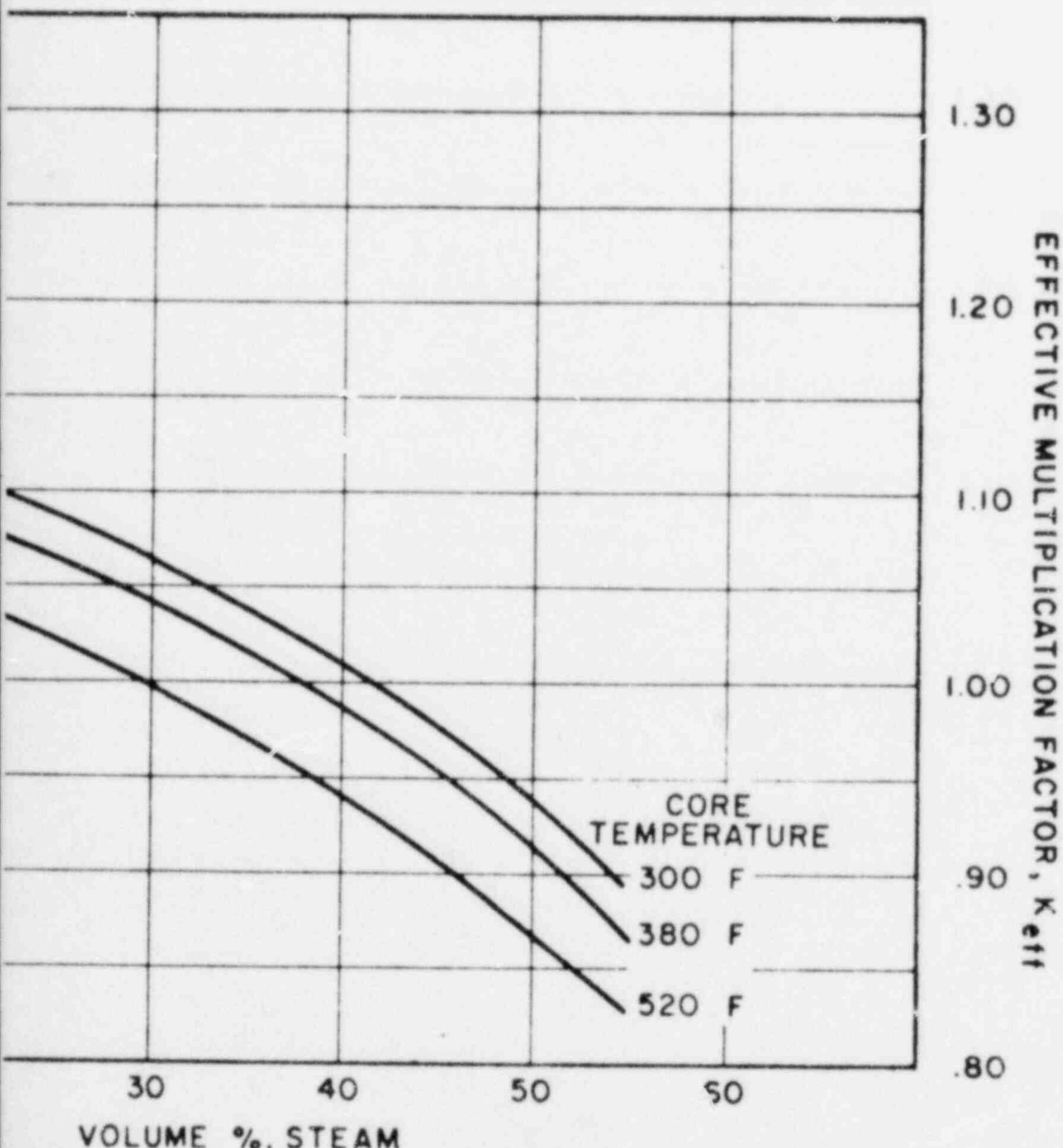

FIG. 9

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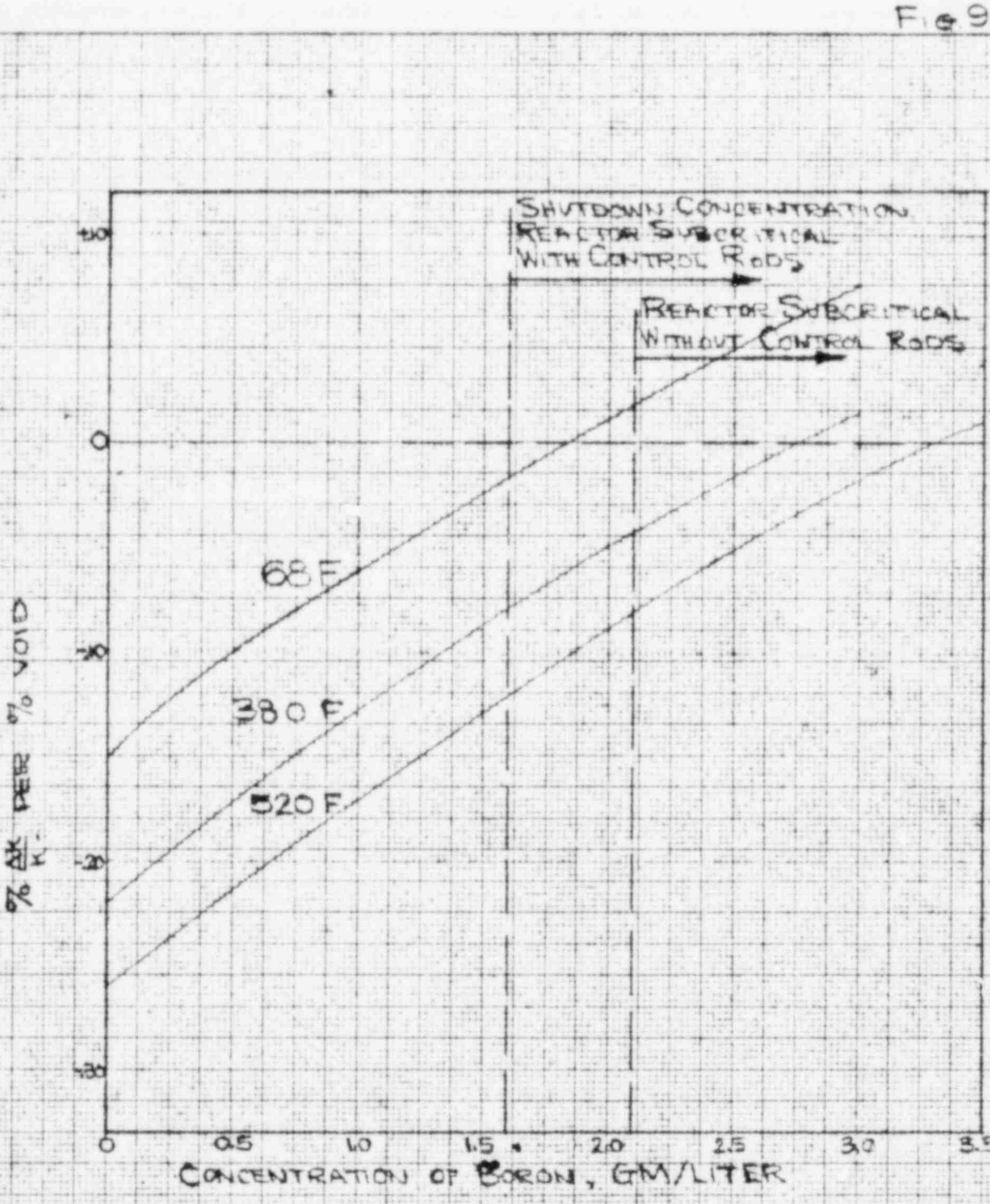
Fig. 9A



UNIFORM VOID COEFFICIENT
VS.
BORON CONCENTRATION

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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 U-238 - U-235 mixture does not go 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 negated 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 hrs of full power operation the magnitude of this effect is -4.6×10^{-5} $\frac{\Delta k}{k}$ /°F.

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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. The reasons for this are that 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.

After shutdown from full power, the Xe-135 concentration rises to a peak at 8.1 hours. 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 $\Sigma_a = .0071 \text{ cm}^{-1}$ at 8.1 hours after shutdown, compared to $.00415 \text{ cm}^{-1}$ at equilibrium. Upon starting up again, Σ_a^{Xe} drops to about $.0032 \text{ cm}^{-1}$ after 8.9 hours and then rises towards equilibrium. The rate of change Σ_a^{Xe} is plotted and converted into dk/dt. The rate of change of k_{eff} with time is maximum at start-up with a value of $+3.5 \times 10^{-6}\%$ $\frac{\Delta k}{k}$ per sec. This can be handled easily by the control rod system and thus presents no possibility of a runaway.

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

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

The oscillation has a 30 hour 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 k 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.

104 CONTROL

General

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

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

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

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

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

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

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

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

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

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

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

Control Rods

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

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

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

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

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

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

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

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

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

Excess neutron level

Short period during reactor start-up

Low main coolant loop flow

High or low main coolant loop pressure

Manual scram

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

Alarms are provided for:

High reactor outlet temperature

Loss of turbine generator load

Reactor period less than 20 seconds

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

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

Chemical Control

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

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

The present design makes use of a bleed-feed system to change the concentration of the chemical neutron absorber. The present maximum rate of water injection for this system is about 100 gpm. Since the mechanical control rods can handle xenon and samarium transients, there is no need for faster action by the chemical control system. At a bleed-feed rate of 100 gpm in a 3,000 cu ft system, the maximum 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.

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

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

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

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

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

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 I, "Rules for Construction of Power Boilers". 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 d wn 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. Leaktightness is secured from gaskets with provision for a backup seal weld. Operating experience will show whether seal welding of the reactor vessel head is required.

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

The fast neutron flux at the inside wall (attentuation of approximately 10 through wall) of the pressure vessel integrated over 30 years of reactor operation is calculated to be 10^{20} neutrons per square centimeter. Experimental data exist at Oak Ridge which state that changes in the properties of steel which has been exposed to 2×10^{18} neutrons per square centimeter are measurable. These effects have to do with increase in hardness and decrease in ductility of the material. However, it has also been found that these effects can be annealed out in approximately 30 minutes at 600 F. Since the steel enclosing the main coolant loop for the Yankee reactor will be approximately 500 F, a diffusion calculation has been made using the experimental point at 600 F as a check.

This calculation indicates that the irradiation effects will be annealed out as they occur and that there will be no serious effect to any portion of the primary coolant-moderator container.

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. Thus 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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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

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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.

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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,

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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.

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219 SAFETY INJECTION-SHIELD TANK CAVITY SYSTEM

Function

The functions of the safety injection-shield tank cavity system are to supply borated water for flooding the shield tank cavity during refueling operations, and to the reactor vessel for cooling 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.

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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 two minutes 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

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

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

408 PLANT MAINTENANCE AND ACCESS

During normal operation, the plant vapor container is closed. Temperature and radioactivity build up 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 radiation 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 a equalize temperatures in the system.

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

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. Their 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.

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 hideout" 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 an 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. Since a deposit of this size is highly improbable and since it corresponds approximately to prompt criticality, it is concluded that no possibility of hazard results from a boron hideout accident.

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 1.03×10^{-4} $\Delta k/k$ per sec and minimum temperature coefficient of reactivity, with a chemical neutron absorber in the system, of -1.6×10^{4} $\Delta k/k$ per deg F, the temperature rises at the rate of 0.38 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 100 sec. The condition of smooth boiling is expected to persist for 100 sec or up to approximately 200 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.

Since only the four pump failure without scram exceeds temperature limitations, a four pump failure with scram has been analyzed by means of an analog computer, with the following additional assumptions:

All control rods 60 per cent withdrawn at the beginning of the transient

The control rods fall 5 ft in 0.58 sec

The reactivity decrease due to scram is .05 per cent $\Delta k/k$

Equilibrium decay heat present

The temperature-time relationships are similar to those given for the previous cases and are shown in Fig. 33. There is no serious effect in this accident within the first few seconds. Assuming no heat transport to the steam generators as the pumps coast down, boiling occurs at the outlet of the hottest channel in approximately 180 sec. Thereafter, temperatures rise slowly as decay heat continues to be generated, until heat transport conditions are established. Heat transport by thermo-siphon circulation through the main coolant loops to the shell side of the steam generators is the subject of a study now in progress. It can be shown, however, on an overall conservation of energy basis, that it takes approximately 4.3 hr for decay heat to evaporate all of the water on the shell side of the steam generators and approximately 7.1 hr to evaporate all of the water in the steam generators plus all the water in the main coolant system down to the level where the core would be partially uncovered. During the first 4.3 hr after the loss of all the main coolant pumps, the evaporated water is discharged as steam to the atmosphere through the steam generator safety relief valves and the plant stack. During the period from 4.3 hr to 7.1 hr, steam escapes from the safety relief valves in the pressure control and relief system and is discharged to the low pressure surge tank. Initially, this steam is quenched by the cool water in the low pressure surge tank. Eventually, this water is heated to saturation pressure and temperature and steam is discharged through the 150 psi gage safety relief valves on the low pressure surge tank into the vapor container. Since there are three essentially independent sources of station service power and two of them are not affected by reactor scram, a total interruption of power to all four pumps is highly improbable. If such an interruption did occur, however, partial service could be restored in a matter of minutes, and this is sufficient to get at least one pump back in service.

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 rupturing of a 20 in. OD main coolant pipe.

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

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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.

Fig. 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.

Fig. 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 = Net heat release, Btu
A = Reciprocal of mechanical equivalent
W = Mechanical work performed, ft-lb
ΔE = 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 $\Delta E = 0$.

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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	
Reactor	1,600
Pressurizer	150
Steam generators	800
Piping	548
Pumps	20
Miscellaneous	52
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	20.0
Total	49.2
Total, psi gage	34.5
Final temperature, F	249

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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:

Activity	Curies
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:

Meteorological Condition	Wind Velocity, Fps	Radio-Iodine Concentration, Microcuries/ml
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:

Meteorological Condition	Thyroid Dose, rem in Year Following Exposure
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.

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.