Chap 3

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3 REACTOR

The reactor core is a three-region cycled core. The fuel rods are cold worked Zircaloy tubes containing slightly enriched uranium dioxide fuel.

The fuel assembly is a canless type with the basic assembly consisting of the RCC guide thimbles welded to the grids and the top and bottom nozzles. The fuel rods are held by the spring clip grips in this assembly which provide a very stiff support for the fuel rods.

Full length and part length rod cluster control assemblies and burnable poison rods are inserted into the guide thimbles of the fuel assemblies. The absorber sections of the control rods are fabricated of silver-indium-cadmium alloy sealed in stainless steel tubes. The absorber material in the fixed burnable poison rods is in the form of borosilicate glass sealed in stainless steel tubes.

The control rod drive mechanisms for the full length RCC assemblies are of the magnetic latch type. The latches are controlled by three magnetic coils. They are so designed that upon a loss of power to the coils, the rod cluster control assembly is released and falls by gravity to shut down the reactor.

The mechanisms for the part length control rods, which are normally in the core, are of a roller nut type mechanism which move at slow spe ' and stop motion on complete loss of power. Therefore complete loss of power to these mechanisms will not result in any significant reactivity change.

3.1 DESIGN BASES

3.1.1 PERFORMANCE OBJECTIVES

The initial reactor thermal power objective is 2758 MWt. The license application is for this power rating. Calculations indicate that hot channel factors will be considerably less than those used for design purposes in this application.

The turbine-generator and plant heat removal systems have been designed for a thermal rating of 3083 Mwt. The portions of the safety analysis dependent on heat removal capacity of plant and safeguards systems have assumed the maximum calculated power rating of 3216 Mwt as have the evaluations of activity release and radiation exposure.

The reactor core fuel loading and programing is designed to yield the first cycle average burnup of 14,200 MWD/MTU and equilibrium discharge burnup of 33,000 MWD/MTU. The fuel rod cladding is designed to maintain its integrity for the anticipated core life. The effects of gas release, fuel dimensional changes, and corrosion-induced or irradiation-induced changes in the mechanical properties of cladding are considered in the design of the fuel assemblies. 15

Rod Control Clusters are employed to provide sufficient reactivity control to terminate any credible power transient prior to reaching the design minimum departure from nucleate boiling (DNB) ratio of 1.30. This is accomplished by ensuring sufficient control cluster worth to shut the reactor down by at least 1% in the hot condition with the most reactive control cluster stuck in the fully withdrawn position.

Redundant equipment is provided to add soluble poison to the reactor coolant in the form of boric acid to maintain shutdown margin when the reactor is cooled to ambient temperatures.

In addition, the control rod worth in conjunction with the boric acid injection from the boric acid injection tank is sufficient to prevent return to critical as a result of the maximum credible steam break (one safety valve stuck fully open) even assuming that the most reactive control rod is in the fully withdrawn position.

Experimental measurements from critical experiments or operating reactors, or both, are used to validate the methods employed in the design. During design, nuclear parameters have been calculated for every phase of operation of the first core cycle and, where applicable, are compared with design limits to show that an adequate margin of safety exists.

In the thermal hydraulic design of the core, the maximum fuel and clad temperatures during normal reactor operation and at 112% overpower have been conservatively evaluated and found to be consistent with safe operating limitations.

3.1.2 PRINCIPAL DESIGN CRITERIA

Reactor Core Design

Criterion: The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated. (GDC 6)

The reactor core, with its related control and protection system, is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of the loss of reactor coolant flow (Section 14.1.6), trip of the turbine generator (Section 14.1.9), loss of normal feedwater and loss of all off-site power (Section 14.1.8).

The Reactor Control and Protection System is designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure a minimum Departure from Nucleate Boiling (DNB) ratio equal to or greater than 1.30.

The integrity of fuel cladding is ensured by preventing excessive fuel swelling, excessive clad heating, excessive cladding stress and strain. This is achieved by designing the fuel rods so that the following conservative limits are not exceeded during normal operation or any anticipated transient condition:

a) Minimum DNB ratio equal to or greater than 1.30
b) Fuel center temperature below melting point of UO_2
c) Internal gas pressure less than the nominal external pressure (2250 psia) even at the end of life
d) Clad stresses less than the Zircaloy yield strength
e) Clad strain less than 1%.

The ability of fuel designed and operated to these criteria to withstand postulated normal and abnormal service conditions is shown by analyses described in Chapter 14 to satisfy the demands of plant operation well within applicable regulatory limits.

The reactor coolant pumps provided for the plant are supplied with sufficient rotational inertia to maintain an adequate flow coastdown and prevent core damage in the event of a simultaneous loss of power to all pumps.

In the unlikely event of a turbine trip from full power without an immediate reactor trip, the subsequent reactor coolant temperature increase and volume insurge to the pressurizer results in a high pressurizer pressure trip and thereby prevents fuel damage for this transient. A loss of external electrical load of 50% of full power or less is normally controlled by rod cluster insertion together with a controlled steam dump to the condenser to prevent a large temperature and pressure increase in the Reactor Coolant System and thus prevent a reactor trip. In this case, overpower-overtemperature protection would guard against any combination of pressure, temperature and power which could result in a DNB ratio less than 1.3 during the transient.

In neither the turbine trip nor the loss-of-flow events do the changes in coolant conditions provoke a nuclear power excursion because of the large system thermal inertia and relatively small void fraction. Protection circuits actuated directly by the coolant conditions identified with core limits are therefore effective in preventing core damage.

Suppression of Power Oscillations

Criterion: The design of the reactor core with its related controls and protection systems shall ensure that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed. (GDC 7)

The potential for possible spatial oscillations of power distribution for this core has been reviewed. It is concluded that low frequency xenon oscillations may occur in the axial dimension, and part length control rods are provided to supress these oscillations. The core is expected to be stable to xenon oscillations in the X-Y dimension. Out-of-core instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon induced oscillations. (In-core instrumentation is used to periodically calibrate and verify the information provided by the out-of-core instrumentation.. The analysis, detection and control of these oscillations is discussed in reference 2) of Section 3.2.1.

Redundancy of Reactivity Control

Criterion: Two independent reactivity control systems, preferably of different principles, shall be provided. (GDC 27)

Two independent reactivity control systems are provided, one involving rod cluster control (RCC) assemblies and the other involving chemical shimming.

Reactivity Hot Shutdown Capability

Criterion: The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition. (GDC 28)

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes. The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning of life of the initial core.

The Rod Cluster Control (RCC) assemblies are divided into three categories comprising control banks, shutdown banks and a part length rod bank. The control banks used in combination with chemical shim control provide control of the reactivity changes of the core throughout the life of the core during power operation. These banks of RCC assemblies are used to compensate for short term reactivity changes at power that might be produced due to variations in reactor power level or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion and fission product buildup.

Reactivity Shutdown Capability

Criterion: One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn. (GDC 29)

The reactor core, together with the reactor control and protection system is designed that the minimum allowable DNBR is at least 1.30 and there is no fuel melting during normal operation including anticipated transients.

The shutdown groups are provided to supplement the control groups of RCC assemblies to make the reactor at least one per cent subcritical at the hot zero power condition (k_{eff} = 0.99) following trip from any credible operating condition assuming the most reactive RCC assembly is in the fully withdrawn position.

Sufficient shutdown capability is also provided to maintain the core subcritical assuming the most reactive rod to be in the fully withdrawn position for the most severe anticipated cooldown transient associated with a single active failure, e.g., accidental opening of a steam bypass, or relief valve, or safety valve stuck open. This is achieved by the combination of control rods and automatic boric acid addition via the emergency core cooling system. The minimum shutdown margin is calculated to be 1.95% assuming the maximum worth control rod in the fully withdrawn position allowing 10% uncertainty in the control rod cal lations.

Manually controlled boric acid addition is used to maintain the shutdown margin for the long term conditions of xenon decay and plant cooldown. Redundant equipment is provided to uarant ne capability of adding boric acid to the reactor coolant system.

Reactivity Holddown Capability

Criterion: The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public (GDC 30)

Normal reactivity shutdown capability is provided within 2 seconds following a trip signal by control rods with boric acid injection used to compensate for the long term xenon decay transient and for plant cooldown. As discussed in response to the previous criteria the shutdown capability prevents return to critical as a result of the cooldown associated with a safety valve stuck fully open.

Any time that the reactor is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds that quantity required for the normal cold shutdown. This quantity always exceeds the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay. Boric acid is pumped from the boric acid tanks by one of two boric acid transfer pumps to the suction of one of three charging pumps which inject boric acid into the reactor coolant. Any charging pump and either boric acid transfer pump can be operated from diesel generator power on loss of station power. Boric acid can be injected by one pump at a rate which takes the plant to 1% shutdown in the hot condition with no rods inserted in less than sixteen minutes In sixteen additional minutes, enough boric acid can be injected to compensate for xenon decay although xenon decay below the equilibrium operating level does not begin until approximately 15 hours after shutdown. If two boric acid pumps are available, these time periods are halved. Additional boric acid injection is employed if it is desired to bring the reactor to cold shutdown conditions.

On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability, independent of control rod clusters which normally serve this function in the short term situation. Shutdown for long term and reduced temperature conditions can be accomplished with boric acid injection using redundant components, thus achieving the measure of reliability implied by the criterion.

Alternately, boric acid solution at lower concentration can be supplied from the refueling water tank. This solution can be transferred directly by the charging pumps or alternately by the safety injection pumps. The reduced boric acid concentration lengthens the time required to achieve equivalent shutdown.

Reactivity Control Systems Malfunction

Criterion: The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits. (GDC 31)

The reactor protection systems are capable of protecting against any single credible malfunction of the reactivity control system, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

Reactor shutdown with rods is completely independent of the normal rod control functions since the trip breakers completely interrupt the power to the rod mechanisms regardless of existing control signals.

Details of the effects of continuous withdrawal of a control rod and continuous deboration are described in Section 14.1 and Section 9.2, respectively.

Maximum Reactivity Worth of Control Rods

Criterion: Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core. (GDC 32)

Limits, which include considerable margin, are placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals so as to lose capability to cool the core.

The reactor control system employs control rod clusters. A portion of these are designated shutdown rods and are fully withdrawn during power operation. The remaining rods comprise the control groups which are used to control load and reactor coolant temperature. The rod cluster drive mechanisms are wired into preselected groups, and are therefore prevented from being withdrawn in other than their respective groups. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming two of the highest worth groups to be accidentally withdrawn at maximum speed yielding reactivity insertion rates of the order of 4×10^{-4} Δk/sec which is well within the capability of the overpower-overtemperature protection circuits to prevent core damage.

No single credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than 80 steps per minute (~ 50 inches per minute).

3.1.3 SAFETY LIMITS

The reactor is capable of meeting the performance objectives throughout core life under both steady state and transient conditions without violating the integrity of the fuel elements. Thus the release of unacceptable amounts of fission products to the coolant is prevented.

The limiting conditions for operation established in the Technical Specification specify the functional capacity of performance levels permitted to assure safe operation of the facility.

Design parameters which are pertinent to safety limits are specified below for the nuclear, control, thermal and hydraulic, and mechanical aspects of the design.

<u>Nuclear Limits</u>

At full power (license application power) the nuclear heat flux hot channel factor, F_q^N = 3.12 as specified in Table 3.2.1-1, Line 17, is not exceeded.

The nuclear axial peaking factor F_z^N, and the nuclear enthalpy rise hot channel factor $F_{\Delta H}^N$ are limited in their combined relationship so as not to exceed the F_q or DNBR limits.

Potential axial xenon oscillations are controlled with part length rods to preclude adverse core conditions. The protection system ensures that the nuclear core limits are not exceeded.

Reactivity Control Limits

The control system and the operational procedures provide adequate control of the core reactivity and power distribution. The following control limits are met:

a. A minimum hot shutdown margin of 1.95 ΔK_{eff} is available assuming a 10% uncertainty in the control rod calculation.

b. This shutdown margin is maintained with the most reactive RCCA in the fully withdrawn position.

c. The shutdown margin is maintained at ambient temperature by the use of soluble poison.

Thermal and Hydraulic Limits

The reactor core is designed to meet the following limiting thermal and hydraulic criteria:

a. The minimum allowable DNBR during normal operation, including anticipated transients, is 1.30.

b. No fuel melting during any anticipated operating condition.

To maintain fuel rod integrity and prevent fission product release, it is necessary to prevent clad overheating under all operating conditions. This is accomplished by preventing a departure from nucleate boiling (DNB) which causes a large decrease in the heat transfer coefficient between the fuel rods and the reactor coolant resulting in high clad temperatures.

The ratio of the heat flux causing DNB at a particular core location, as predicted by the W-3 correlation, to the existing heat flux at the same core location is the DNB ratio. A DNB ratio of 1.30 corresponds to a 95% probability at a 95% confidence level that DNB does not occur and is chosen to maintain an appropriate margin to DNB for all operating conditions.

Mechanical Limits

Reactor Internals

The reactor internal components are designed to withstand the stresses resulting from startup, steady state operation with any number of pumps running, and shut-down conditions. No damage to the reactor internals occurs as a result of loss of pumping power.

Lateral deflection and torsional rotation of the lower end of the core barrel is limited to prevent excessive movements resulting from seismic disturbances and thus prevent interference with rod control cluster assemblies. Core drop in the event of failure of the normal supports is limited so that the rod cluster control assemblies do not disengage from the fuel assembly guide thimbles.

The internals are further designed to maintain their functional integrity in the event of a major loss-of-coolant accident. The dynamic loading resulting from the pressure oscillations because of a loss-of-coolant accident does not cause sufficient deformation to prevent rod cluster control assembly insertion.

Fuel Assemblies

The fuel assemblies are designed to perform satisfactorily throughout their lifetime. The loads, stresses, and strains resulting from the combined effects of flow induced vibrations, earthquakes, reactor pressure, fission gas pressure, fuel growth, thermal strain, and differential expansion during both steady state and transient reactor operating conditions have been considered in the design of the fuel rods and fuel assembly. The assembly is also structurally designed to withstand handling and shipping loads prior to irradiation, and to maintain sufficient integrity at the completion of design burnup to permit safe removal from the core subsequent handling during cool-down, shipment and fuel reprocessing.

The fuel rods are supported at nine locations along their length within the fuel assemblies by brazed grid assemblies which are designed to maintain control of the lateral spacing between the rods throughout the design life of the assemblies. The magnitude of the support loads provided by the grids are established to minimize possible fretting without overstressing the cladding at the points of contact between the grids and fuel rods and without imposing restraints of sufficient magnitude to result in buckling of distortion of the rods.

The fuel rod cladding is designed to withstand operating pressure loads without rupture and to maintain encapsulation of the fuel throughout the design life.

Rod Cluster Control Assemblies

The criteria used for the design of the cladding on the individual absorber rods in the rod cluster control assemblies (RCCA) are similar to those used for the fuel rod cladding. The cladding is designed to be free standing under all operating conditions and will maintain encapsulation of the absorber material throughout the absorber rod design life. Allowance for wear during operation is included for the RCCA cladding thickness.

Adequate clearance is provided between the absorber rods and the guide thimbles which position the rods within the fuel assemblies so that coolant flow along the length of the absorber rods is sufficient to remove the heat generated without overheating of the absorber cladding. The clearance is also sufficient to compensate for any misalignment between the absorber rods and guide thimbles and to prevent mechanical interference between the rods and guide thimbles under any operating conditions.

Control Rod Drive Assembly

Each control rod drive assembly is designed as a hermetically sealed unit to prevent leakage of reactor coolant. All pressure-containing components are designed to meet the requirements of the ASME Code, Section III, Nuclear Vessels for Class A vessels.

The control rod drive assemblies for the full length rods provide rod cluster control assembly insertion and withdrawal rates consistent with the required reactivity changes for reactor operational load changes. This rate is based on the worths of the various rod groups, which are established to limit power-peaking flux patterns to design values. The maximum reactivity addition rate is specified to limit the magnitude of a possible nuclear excursion resulting from a control system or operator malfunction. Also, the control rod drive assemblies for the full length rods provide a fast insertion rate during a "trip" of the RCC assemblies which results in a rapid shutdown of the reactor for conditions that cannot be handled by the reactor control system.

The part length control rod drive mechanisms are used to position the part length rod control cluster assemblies within the reactor core. When required, the mechanisms drive the control clusters at a constant speed in either direction. Removal or loss of power causes the rods to stop all motion immediately.

3.2 REACTOR DESIGN

3.2.1 NUCLEAR DESIGN AND EVALUATION

This section presents the nuclear characteristics of the core and an evaluation of the characteristics and design parameters which are significant to design objectives. The capability of the reactor to achieve these objectives while performing safely under normal operational modes, including both transient and steady state, is demonstrated.

Nuclear Characteristics of the Design

A summary of the reactor nuclear design characteristics is presented in Table 3.2.1-1.

Reactivity Control Aspects

Reactivity control is provided by neutron absorbing control rods and by a soluble chemical neutron absorber (boric acid) in the reactor coolant. The concentration of boric acid is varied as necessary during the life of the core to compensate for: (1) changes in reactivity which occur with change in temperature of the reactor coolant from cold shutdown to the hot operating, zero power conditions; (2) changes in reactivity associated with changes in the fission product poisons xenon and samarium; (3) reactivity losses associated with the depletion of fissile inventory and buildup of long-lived fission product poisons (other than xenon and samarium); and (4) changes in reactivity due to burnable poison burnup.

The control rods provide reactivity control for: (1) fast shutdown; (2) reactivity changes associated with changes in the average coolant temperature above hot zero power (core average coolant temperature is increased with power level); (3) reactivity associated with any void formation; (4) reactivity changes associated with the power coefficient of reactivity.

Chemical Shim Control

Control to render the reactor subcritical at temperatures below the operating range is provided by a chemical neutron absorber (boron). The boron concentration during refueling has been established as shown in Table 3.2.1-1, line 29. This concentration together with the control rods provides approximately 10 per cent shutdown margin for these operations. The concentration is also sufficient to maintain the core shutdown without any RCC rods during refueling. For cold shutdown, at the beginning of core life, a concentration (shown in Table 3.2.1-1, line 37) is sufficient for one per cent shutdown with all but the highest worth rod inserted. The boron concentration (Table 3.2.1-1, line 29) for refueling is equivalent to less than two per cent by weight boric acid (H_3BO_3) and is well within solubility limits at ambient temperature. This concentration is also maintained in the spent fuel pit since it is directly connected with the refueling canal during refueling operations.

The initial full power boron concentration without equilibrium xenon and
15 samarium is 1160 ppm. As these fission product poisons are built up, the boron concentration is reduced to 780 ppm.

This initial boron concentration is that which permits the withdrawal of the control banks to their operational limits with the part length rods out. The xenon-free hot, zero power shutdown (k = 0.99) with all but the highest worth rod inserted, can be maintained with the boron concentration
15 of 677 ppm. This concentration is less than the full power operating value with equilibrium xenon.

Control Rod Requirements

Neutron-absorbing control rods provide reactivity control to compensate for more rapid variations in reactivity. The rods are divided into two categories according to their function. Some rods compensate for changes in reactivity due to variations in operating conditions of the reactor such as power or temperature. These rods comprise the control group of rods. The remaining rods, which provide shutdown reactivity, are termed shutdown rods. The total shutdown worth of all the rods is also specified to provide adequate shutdown with the most reactive rod stuck out of the core.

Control rod reactivity requirements at beginning and end of life are summarized in Table 3.2.1-2. The installed worth of the control rods is shown in Table 3.2.1-3

The difference is available for excess shutdown upon reactor trip. The control rod requirements are discussed below.

Total Power Reactivity Defect

Control rods must be available to compensate for the reactivity change incurred with a change in power level due to the Doppler effect. The magnitude of this change has been established by correlating the experimental results of numerous operating cores as mentioned above.

The average temperature of the reactor coolant is increased with power level in the reactor. Since this change is actually a part of the power dependent reactivity change, along with the Doppler effect and void formation, the associated reactivity change must be controlled by rods. The largest amount of reactivity that must be controlled is at the end of life when the moderator temperature coefficient has its most negative value. The moderator temperature coefficient range is given in Table 3.2.1-1, line 42, while the cumulative reactivity change is shown in the first line of Table 3.2.1-2. By the end of the fuel cycle, the nonuniform axial depletion causes a severe power peak at low power. The reactivity associated with this peak is part of the power defect.

Operational Maneuvering Band

The control group is operated at full power within a prescribed band of travel in the core to compensate for periodic changes in boron concentration, temperature, or xenon. The band has been defined as the operational maneuvering band. When the rods reach either limit of the band, a change in boron concentration must be made to compensate for any additional change in reactivity, thus keeping the control group within the maneuvering band.

Control Rod Bite

If sufficient boron is present in a chemically-shimmed core, the inherent operational control afforded by the negative moderator temperature coefficient is lessened to such a degree that the major control of transients resulting from load variations must be compensated for by control rods. The ability of the plant to accept major load variations is distinct from safety considerations, since the reactor would be tripped and the plant shutdown safely if the rods could not follow the imposed load variations. In order to meet required reactivity ramp rates resulting from load changes, the control rods must be inserted a given distance into the core. The reactivity worth of this insertion has been defined as control rod bite.

The reactivity insertion rate must be sufficient to compensate for reactivity variation due to changes in power and temperature caused either by a ramp load change of five per cent per minute, or by a step load change of ten per cent. An insertion rate of 3×10^{-5} $\Delta\rho$ per second is determined by the transient analysis of the core and plant to be adequate for the most adverse combinations of power and moderator coefficients. To obtain this minimum ramp rate one control bank of rods must remain inserted at least 10 per cent into the core at the beginning of life. The reactivity associated with this bite is 0.1 per cent.

Part Length Rods

The eight RCC assemblies with part length rods can be inserted into the core to control the axial power distribution. These assemblies do not drop when the reactor is tripped. The part length rods do not contribute to shutdown by themselves; they can, however, increase shutdown by flattening the power distribution at low power levels. No credit has been taken for this additional shutdown.

Xenon Stability Control

Part length control rods are provided to suppress xenon induced power oscillations in the axial direction, should they occur. Out-of-core instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon induced power oscillations. Extensive analyses, with confirmation of methods by spatial transient experiments at Haddam Neck, has shown that any induced radial or diametral xenon transients would die away naturally. A full discussion of xenon stability control can be found in Reference 2.

Excess Reactivity Insertion Upon Reactor Trip

The control requirements are nominally based on providing one per cent shutdown at hot, zero power conditions with the highest worth rod stuck in its fully withdrawn position or to prevent return to criticality following a credible steam-line break, whichever is the more limiting. The condition where excess reactivity insertion is most critical is at the end of a cycle when the steam break accident is considered. The excess control available at the end of cycle, hot zero power condition with the highest worth rod stuck out is 2.08% $\Delta\rho$ after allowing a 10% margin for uncertainty in control rod worth as shown in Table 3.2.1-3. 15

Calculated Rod Worths

The complement of 53 full length control rods arranged in the pattern shown in Figure 3.2.1-1 meets the shutdown requirements. Table 3.2.1-3 lists the calculated worths of this rod configuration for beginning and end of the first cycle. In order to be sure of maintaining a conservative margin between calculated and required rod worths, an additional amount has been added to account for uncertainties in the control rod worth calculations.

The calculated reactivity worths listed are decreased in the design by 10 per cent to account for any errors or uncertainties in the calculation. This worth is established for the condition that the highest worth rod is stuck in the fully withdrawn position in the core.

A comparison between calculated and measured rod worths in operating reactor shows the calculation to be well within the allowed uncertainty of 10%.

Reactor Core Power Distribution

In order to meet the performance objectives without violating safety limits, the peak to average power density must be within the limits set by the nuclear hot channel factors. For the peak power point in the core, the nuclear heat flux hot channel factor, F_q^N, is 3.1[illegible]. For the hottest channel the nuclear enthalpy rise hot channel factor, $F_{\Delta H}^N$ is 1.75.

Extensive power distribution analyses are performed for the core and support the assertion that the design objectives are achieved. Figures 3.2.1-2 through 3.2.1-5 show variation of hot channel factors for various rod positions. These calculations do not include the power flattening effect of equilibrium xenon and Doppler broadening. Analysis of the core power distribution with burnup is also performed. In-core instrumentation will be employed to check the power distributions throughout core lifetime.

Eight part length rods, which are similar to the standard control rods but which have absorber material in only the bottom three feet, are located in the core as shown in Figure 3.2.1-1. The function of these rods is to shape the axial power distribution and to control axial xenon oscillations.

The control system for axial power distribution control is based on manual operation of the part length rods. Administrative procedures, alarm functions, and automatic rod stops, guide and monitor the operator in performing these tasks. The out of core nuclear instrumentation system supplies the necessary information for the operator to control the core power distribution within the limits established for the protection system design. This information consists of a two pen recorder for each long ion chamber which displays the upper and lower ion chamber currents and an indicator which gives the difference in these two currents for each long ion chamber. These ion chamber currents to the recorders and indicators are calibrated against in-core power distribution obtained from the movable detector system so that the eight individual signals are directly related to the power generated in the adjacent section of the core. This essentially divides the core into eight sections, four in the upper half and four in the lower half, and the operator manually positions the part length rods to maintain a prescribed relationship between the power generated in the upper and lower sections of the core.

The relationship between core power distribution and out of core nuclear instrumentation readings will be established during the startup testing program. In core flux measurements will be made over the range of relative positions between part length rods and the full length rod control banks for reactor power in the range of 25% to 100%. These measurements, together with long ion chamber currents, will be processed to yield the relationships between core average axial power generation, the axial peaking factor and axial offset as indicated by the out of core nuclear instrumentation. These relationships can be checked during operation to assess the effect of core burnup on the sensitivity between in-core power distribution and out of core readings.

A more detailed discussion of the background, analytical and experimental, data which forms the basis for this approach, is given in reference 2.

Estimating Axial Power Peaks

To make use of part length rods for shaping axial power distributions, it is necessary to estimate the existing axial peak in the operating reactor.

Figure 3.2.1-6 shows that a conservative estimate of the axial peak can be made by comparing the readings of the long ion chambers which preferentially see the top and bottom halves of the core.

Figure 3.2.1-6 is a cross-plot of symmetric offset [(top reading - bottom reading) sum of the two] and the computed axial peaking factor. It shows that a high peak always gives a large value of symmetric offset. An enveloping line drawn above the plotted points then gives a conservative estimate of the axial peaking factor for any symmetric offset reading.

The data points on Figure 3.2.1-6 were gleaned from an extensive analysis of axial peaking due to duel depletion, xenon spatial transients, and non-base load operation. An extreme range of boron concentrations, and a complete range of part length rod positions are covered. All axial analyses reflect the non-uniform water density and fuel temperature distributions in the core.

Reactivity Coefficients

The response of the reactor core to plant conditions or operator adjustments during normal operation, as well as the response during abnormal or accidental transients, is evaluated by means of a detailed plant simulation. In these calculations, reactivity coefficients are required to couple the response of the core neutron multiplication to the variables which are set by conditions external to the core. Since the reactivity coefficients change during the life of the core, a range of coefficients is established to determine the response of the plant throughout life and to establish the design of the Reactor Control and Protection System.

Moderator Temperature Coefficient

The moderator temperature coefficient in a core controlled by chemical shim is less negative than the coefficient in an equivalent rodded core. One reason is that control rods contribute a negative increment to the coefficient and in a chemical shim core, the rods are only partially inserted. Also, the chemical poison density is decreased with the water density upon an increase in temperature. This gives rise to a positive component of the moderator temperature coefficient due to boron being removed from the core. This is directly proportional to the amount of reactivity controlled by the dissolved poison.

In order to reduce the dissolved poison requirement for control of excess reactivity, burnable poison rods have been incorporated in the core design. The result is that changes in the coolant density will have less effect on the density of poison and the moderator temperature coefficient will be reduced.

The burnable poison is in the form of borated pyrex glass rods clad in stainless steel. There are 1160 of these rods in the form of clusters distributed throughout the core in vacant rod cluster control guide tubes as illustrated in Figures 3.2.1-7 and 3.2.1-8. Information regarding research, development and nuclear evaluation of the burnable poison rods can be found in reference 1. These rods will initially control 7.2% $\Delta\rho$ of the installed excess reactivity and their addition will result in a 15
reduction of the initial hot zero power boron concentration in the coolant to 1306 ppm. The moderator temperature coefficient is negative at the operating coolant temperature with burnable poison rods installed.

The effect of burnup on the moderator temperature coefficient is calculated and the coefficient becomes more negative with increasing burnup. This is due to the buildup of fission product with burnup and dilution of the boric acid concentration with burnup. The reactivity loss due to equilibrium xenon is controlled by boron, and as xenon builds up, boron is taken out. The calculated net effect and the predicted unrodded moderator temperature coefficient equilibrium xenon at full power BOL is $-0.55 \times 10^{-4}/°F$. With core burnup, the coefficient will become more negative as a boron is removed because of a shift in the neutron energy spectrum due to the buildup of plutonium and fission products. At end of life with no boron or rods in the core, the moderator coefficient is $-3.0 \times 10^{-4}/°F$.

The control rods provide a negative contribution to the moderator coefficient as can be seen from Figure 3.2.1-9.

Moderator Pressure Coefficient

The moderator pressure coefficient has an opposite sign to the moderator temperature coefficient. Its effect on core reactivity and stabilty is small because of the small magnitude of the pressure coefficient, a change of 50 psi in pressure having no more effect on reactivity than half-degree change in moderator temperature. The calculated beginning and end of life pressure coefficients are specified in Table 3.2.1-1, Line 43.

Moderator Density Coefficient

A uniform moderator density coefficient is defined as a change in the neutron multiplication per unit change in moderator density. The range of the moderator density coefficient from BOL to EOL is specified in Table 3.2.1-1, Line 44.

Doppler and Power Coefficients

The Doppler coefficient is defined as the change in neutron multiplication* per degree change in fuel temperature. The coefficient is obtained by calculating neutron multiplication as a function of effective fuel temperature by the code LEOPARD.(3) The results are shown in Figure 3.2.1-10.

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* Neutron multiplication is defined as the ratio of the average number of neutrons produced by fission in each generation to the total number of corresponding neutrons absorbed.

In order to know the change in reactivity with power, it is necessary to know the change in the effective fuel temperature with power as well as the Doppler coefficient. It is very difficult to predict the effective temperature of the fuel using a conventional heat transfer model because of uncertainties in predicting the behavior of the fuel pellets. Therefore, an empirical approach is taken to calculate the power coefficient, based on operating experience of existing Westinghouse Atomic Power Division cores. Figure 3.2.1-11 shows the power coefficient as a function of power obtained by this method. The results presented do not include any moderator coefficient even though the moderator temperature changes with power level.

As the fuel pellet temperature increases with power, the resonance absorption in U-238 increases due to Doppler Broadening of the resonances. A large temperature drop occurs across the fuel pellet-clad gap. Under certain conditions, this gap may be closed, thus resulting in lower pellet temperature. The net effect is a lower effective fuel temperature, a higher (more negative Doppler coefficient, and a lower (less negative) power coefficient than that which exists with a pellet-clad gap. The power coefficient, which is determined using a closed gap model, is shown in Figure 3.2.1-12.

Calculations indicate the stability of the reactor to Xenon oscillations is relatively insensitive to the thermal model used to obtain the power coefficient. The damping factor associated with the fuel Doppler effect is

$$\alpha_f = \frac{\partial K_{eff}}{\partial T} \frac{\partial T}{\partial P}$$

where

T = fuel temperature

P = power

The quantity $\frac{\partial T}{\partial P}$ is larger for the gap model than for the no gap case but since the Doppler coefficient varies as $T^{-1/2}$ the term $\frac{\partial K_{eff}}{\partial T}$ is smaller. The net effect is that α_f is relatively insensitive to the thermal model in the range of power 0.5 to 1.5 of core average which is the range of interest for stability.

Nuclear Evaluation

The basis for confidance in the procedures and design methods comes from the comparison of these methods with many experimental results. These experiments include criticals performed at the Westinghouse Reactor Evaluation Center (WREC) and other facilities, and also measured data from operating power reactors. A summary of the results and discussion of the agreement between calculated and measured values is given in the following paragraphs.

Reactivity Analysis

Data from 55 oxide and 56 metal lattice critical and exponential experiments have been evaluated[4]. The results of these studies are summarized in Table 3.2.1-4. The values of neutron multiplication k are computed using experimentally measured material bucklings, and should equal unity. Table 3.2.1-4 demonstrates that much of the scatter can be attributed to variations in results from one experimental laboratory to another, whereas the evaluation demonstrates that errors do not develop with variations of certain significant parameters. As the calculational accuracy is independent of variations in hydrogen to uranium ratio, uranium enrichment, pellet diameter and buckling, extrapolation from experiments to operating cores or extrapolation from one operating core to another should not lead to any significant error.

It can be seen from Table 3.2.1-4 that if only WAPD experimental results are considered, the computational method predicts k to a standard deviation of 0.36 per cent which is a better estimate of the accuracy of the method because of the more detailed information available. Much of the additional scatter in the standard deviation for the other cases can be attributed to insufficient information on the dimensions and results of many of the cases published.

Depletion Analysis

Data from the Yankee spent core ar s have been compared with calculated data using the design techniques. The results are summarized in Figures 3.2.1-13 through 3.2.1-15. Uranium depletion and net plutonium production have a direct bearing on the core lifetime. The figures show the comparison between calculations (solid lines) and measured concentrations of the various isotopes. Although some small deviations can be observed between analysis and experiment, they are not considered serious.

Power Peaking Analysis

A series of critical experiments were carried out at the Westinghouse Reactor Evaluation Center (WREC) to determine the power peaking in fuel rods adjacent to water holes and to determine the effects of voids on power distribution.

The power peaking experiment was performed in a 30 x 30 array of 2.72 per cent enriched fuel with a water-to-uranium ratio of 3.5 with and without boron in the moderator. The pattern of 16 water holes was symmetrical about the center of the core. The core arrangement and pattern of fuel rods scanned are shown in Figure 3.2.1-16 for the unborated core and Figure 3.2.1-17 for the same core with 479 ppm boron in the water.

The analysis consists of PDQ calculations using two-group constants obtained from LEOPARD. Mixed Number Density thermal constants are used, and "soft spectrum" microscopic constants are used in the reflector and water holes. In the PDQ analysis, two mesh spacings per fuel rod are used. Also, in the unborated core a calculation is performed for one mesh space per fuel rod. The experimental data are normalized to the PDQ results using the average of the four central rods. The experimental and calculated results for the borated and unborated cores with two mesh spacers per fuel rod are shown in Figures 3.2.1-18 and 3.2.1-19, respectively, and in Figure 3.2.1-20 for the unborated core calculated with one mesh spacer per fuel rod. Each block in the figures represents a fuel rod. The experimental values correspond to the average values of counts taken at five positions on the fuel rod.

The agreement between analysis and experiment is within 2 to 3 per cent and is of the same order as the scatter in the experimental data. There is no consistent difference in over-estimating or underestimating peaking using the one mesh per fuel rod or two mesh per fuel rod representation.

The void experiments were performed for two different core configurations. The first series of experiments was carried out in a 47 x 47 square core of 2.7% enriched fuel with a W/U of 2.9, with no boron. The second series was performed using a 53 x 53 square core of 3.7% enriched fuel with a W/U of 2.9, and with 1046 ppm boron in the water. In both cores voids were simulated by empty 0.1875 inch O.D., 0.022 inch wall aluminum tubes inserted between fuel rods. The moderator in the voided region consisted of 11.52% aluminum, 16.29% void and 72.19% water. Data were taken for the following cases:

1. No void tubes
2. Four void tubes (2x2) located around the central fuel rod
3. Sixteen void tubes (4x4) at core center
4. One hundred ninety-six void tubes (14x14) at core center

The analysis again consisted of PDQ using two-group constants from LEOPARD, with MND thermal constants and "soft spectrum" water hole and reflector constants. The calculated power distribution is compared with the experimental power scans in Figure 3.2.1-21 and 3.2.1-22 for the unborated and borated cores for the four cases examined. The agreement between experiment and calculation is good except at the transition region between voided and non-voided regions. Here the calculated peaks are higher than those obtained by experimental measurements.

The reactivity effects of the void tubes were calculated assuming a constant axial reflector savings. Calculation and experiment for each case examined are compared in Table 3.2.1-5. Calculations overestimate the reactivity effect of the voids by approximately 10%, which is good agreement in view of the small magnitude of the effects being studied.

Gross Power Distribution Analysis

The ability to evaluate power distributions in multiregion critical cores with no burnup has been evaluated in detail.(5) Agreement for all situations, including those with large enrichment variation and small regions, is found to be good as is illustrated in Figure 3.2.1-23. The ability to evaluate power distributions in depleted cores at power as been demonstrated by core evaluation programs using in-core instrumentation data from Yankee and Saxton. Other pertinent data will be obtained in the future from other Westinghouse reactors including large PWR cores controlled by chemical shim and will be factored into the final design.

As an example of such a comparison, a power distribution is shown in Figure 3.2.1-24 for the end of life in Yankee Core I, which was not controlled by chemical shim. A comparison of the burnup distribution is also presented in Figure 3.2.1-25.

In both cases two calculated values are given which show the effect of a rod program interchange during life.(6)

RCC Assembly Worth Analysis

In the control rod calculations performed by PDQ, the RCC rods are represented by internal boundary conditions (α's) in the fast and thermal groups. These boundary conditions applied to the unit cell in which the absorber rod, its clad and the associated water are homogenized. The values of these α's are determined to make the calculated rod worth of a single fuel assembly equal to that calculated by a more refined model. The better model represents each absorber rod explicitly and is used to analyze an extensive set of critical measurements. Approximately 30 different critical measurements were made for uniform and cluster arrays of absorber rods with different enrichments, rod diameters, water-to-uranium ratios and boron concentrations.

In the analysis of these measurements, the rods were represented by a theoretically determined thermal boundary condition and by a diffusion region in the single fast group. The fast absorption cross section was empirically determined from the measured rod worth to give agreement between analytical and experimental results. The development of this calculation scheme for RCC rod worth and a description of the measurements is given in Reference 72. Figures 3.2.1-26 and 3.2.1-27 are reproduced from this reference to show the fast absorption cross section as a function of the radius of the absorber which fits the experimental measurements for cluster and uniform cases, respectively. The solid lines were obtained by a least square fitting of the experimental data.

Moderator Coefficient Analysis

Inasmuch as the safe operation of any plant is closely associated with the ability to predict the behavior of that plant, correlation of analysis with experiment is presented to show that the moderator temperature coefficient is quite predictable. Measurements were made during the startup and operation of the SELNI core to get data for a core controlled by chemical shim. During the startup, the core was heated from room to operating temperature at a constant boron concentration of 1600 ppm. Figure 3.2.1-28 shows the results of the moderator coefficient measurements taken during this core heatup, and also the comparable calculated values. The calculations were performed with the one-dimensional AIM-5 code with LEOPARD input constants as described for neutron multiplication calculations. The agreement between calculation and experiment is good over the entire temperature range. In order to measure the moderator coefficient at different boron concentrations, control rods were traded for boron during the hot, no power startup tests. This procedure permitted moderator coefficient measurements to be made over a range of boron concentrations from 1300 to 1800 ppm.

The method of analysis for the case of trading rods for boron is, of necessity, different from the method discussed above. The AIM-5 code was again used, but an axial calculation was performed with an homogenized bank of absorber used to represent the moving control rods. The results of analysis and measurement are shown in Figure 3.2.1-29. The calculations were performed in the same manner as the measurement; i.e., the control group was inserted as boron was removed.

When the control group was fully inserted, further boron removal was compensated for by insertion of all rods banked. PDQ analyses were also performed for all the rods in and all rods out end points and the results are given in Figure 3.2.1-29. It can be seen that the one-dimensional calculations in which rods are represented by a homogenized absorber predicts the measured data very well.

The effect of burnup on the moderator coefficient has been measured in the core evaluation program performed on Yankee Core I.[8] Yankee Core I was controlled by cruciform blade rods, and so it was necessary to separate the effect of control rods from the effect of burnup on the moderator coefficient. Figure 3.2.1-30 illustrates these components and the agreement between analysis and measurement. The effect of rods was evaluated by treating the rods as an equivalent absorption area (approximation 1 in Figure 3.2.1-30 with a correlation for the effects of resonance absorption (approximation 2 in Figure 3.2.1-30). The results of the analysis lie within the experimental uncertainty and the burnup effect on the moderator coefficient results in a more negative coefficient with increasing burnup.

Doppler and Power Coefficient Analysis

As the fuel pellet temperature increases with power, the resonance absorption in U-238 increases due to Doppler broadening of the resonances. In order to predict the reduction in reactivity caused by this effect, it is necessary to know the temperature of the fuel as a function of power level, the position of burnup of fuel in the core, as well as the radial distribution of temperature within th individual fuel rods. However, uncertainties arise during operation at power which made it difficult to predict accurately the temperature of the fuel pellet. For example, pellets do not remain intact (i.e., uncracked) and in a concentric relationship with the clad, as has been observed from the Yankee spent fuel analysis.[9] In addition, the composition of gases in the gap changes with burnup because of diffusion of fission product gases to the gap. This generally results in an uncertainty in the temperature drop across the gap as a function of power level and burnup.

A semi-empirical model has been developed for calculating the effective fuel temperature (T_{eff}) based on fitting the measured power coefficients of the Yankee, Saxton, BR-3 and SELNI reactor cores. The measured power coefficient $1/k\ \delta k/\delta P$ can be written

$$\frac{1}{K}\ \frac{\delta k}{\delta p} = \frac{1}{k}\ \frac{\delta k}{\delta T_{eff}} \cdot \frac{\delta T_{eff}}{\delta P} \qquad (1)$$

The first term in the product on the right side of the Equation (1) is the Doppler coefficient which can be computed without knowing the heat transfer behavior of the fuel pellet or the relationship of T_{eff} and power. The second term on the right side of Equation (1) can then be related to the measured values of power coefficients. In this manner an empirical expression for the effective fuel temperature is obtained which makes it possible to relate T_{eff} to power, and thus calculate the power coefficient.

The method of analysis described in the preceding paragraph assumes accuracy of prediction of the Doppler coefficient as a function of the effective fuel temperature. This assumption indicates that the behavior of the U-238 resonance integral with a change in the fuel temperature is well known. Data is presented here to support this assumption. A correlation has been developed for the U-238 resonance integral which is known as the metal-oxide correlation.[4] This correlation has been found to agree with Hellstrand's uranium metal[10] and uranium dioxide[11] correlations for isolated rods. The correlation is also consistent with Hellstrand's temperature correlations.[12] Thus, a single correlation replaces the four Hellstrand correlations. The metal-oxide correlation is

$$R.I.^{28} = 2.16X + 1.48 + (0.0279X - 0.0537)\ T_{eff}^{1/2}$$

where T_{eff} is in degrees Kelvin and

$$X = \left[\frac{\Sigma so}{N_o^{28}} P_o + \frac{D}{l_o N_o^{28}}\right]^{1/2}$$

Σso = scattering cross section of the fuel (10.7 barns for uranium and 3.8 barns for oxygen)

N_o^{28} = U-238 number density in the fuel region

l_o = mean chord length in the fuel

D = shielding factor (calculated by Sauer's Method)(13)

P_o = $1 - P_c$ (P_c is tabulated in Reference 14)

This form of the resonance integral is not strictly rigorous, but its validity is demonstrated in Figure 3.2.1-31 where it is compared with Hellstrand's results for different temperatures.(4)

An extensive evaluation of power coefficient measurements has been made for the Yankee, Saxton, Br-3 and SELNI cores. The results of these measurements are given in Figure 3.2.1-32 which shows the change in the effective fuel temperature per kw/ft as a function of core average kw/ft. From these data an empirical equation for T_{eff} has been developed which will predict T_{eff} as a function of power level.(15) This equation for T_{eff} is given below.

$$T_{eff}(P/P_o) = 0.55\ \Delta T_{fuel} + \alpha(\bar{q}'')\ \delta\bar{q}'' + 1.571\ P/P_o\ \Delta T_o\ (\text{clad + film}) + T_{coolant}$$

where

P/P_o	=	fraction of full power
ΔT_{fuel}	=	difference between maximum and surface fuel pellet temperature (function of power)
$\alpha(\bar{q}'')$	=	empirical parameter dependent upon average heat flux
δ	=	ratio of the cold diametral gap to the inner diameter of the clad
$\bar{q}''$	=	average surface heat flux to the pellet
ΔT_o (clad + film)	=	temperature drop across clad and film (function of power)
$T_{coolant}$	=	average temperature of the coolant (function of power)

The empirically determined α is given in Figure 3.2.1-33 as a function of pellet surface heat flux. The difference in the effective temperature obtained from the experimental data of Figure 3.2.1-32 and from the correlation employing Figure 3.2.1-33 is shown in Figure 3.2.1-34 as a function of surface heat flux. It can be seen that even though there is some scatter in the experimental data (Figure 3.2.1-32), all the experimental points fall into a small band when the T_{eff} correlation is used. The most scattered experimental data points deviate from the predicted value (solid line) by no more than $\pm$ 80°F. It is concluded that the T_{eff} correlation can predict T_{eff} at any power level to within $\pm$ 80°F which constitutes less than $\pm$ 5% of the effective fuel temperature at full power.

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TABLE 3.2.1-1
NUCLEAR DESIGN DATA

STRUCTURAL CHARACTERISTICS

1.	Fuel Weight (UO_2), lbs.	216,600	
2.	Zircaloy Weight, lbs.	44,600	
3.	Core Diameter, inches	132.75	
4.	Core Height, inches	144	
	Reflector Thickness and Composition		
5.	Top - Water Plus Steel	∿ 10 in.	
6.	Bottom - Water Plus Steel	∿ 10 in.	
7.	Side - Water Plus Steel	∿ 15 in.	
8.	H_2O/U, (cold) Core	4.01	10
9.	Number of Fuel Assemblies	193	
10.	UO_2 Rods per Assembly	204	

PERFORMANCE CHARACTERISTICS

11.	Heat Output, MWt (initial rating)	2758	
12.	Heat Output, MWt (maximum calculated turbine rating)	3216	
13.	Fuel Burnup, MWD/MTU	14,200	10
	First Cycle		
	Enrichments, w/o		
14.	Region 1	2.2	
15.	Region 2	2.7	
16.	Region 3	3.2	
17	Equilibrium Enrichment	3.2	
18.	Nuclear Heat Flux Hot Channel Factor, F_q^N	3.12	
19.	Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^N$	1.75	

TABLE 3.2.1-1 (Cont'd)

CONTROL CHARACTERISTICS

	Effective Multiplication (Beginning of Life) With Rods in; No Boron		
20.	Cold, No Power, Clean	1.113	
21.	Hot, No Power, Clean	1.057	10
22.	Hot, Full Power, Clean	1.031	
23.	Hot, Full Power, Xe and Sm Equilibrium	1.001	
24.	Material	5% Cd; 15% In; 80% Ag	
25.	Full Length	53	
26.	Partial Length	8	
27.	Number of Absorber Rods per RCC Assembly	20	
28.	Total Rod Worth, BOL, %	(See Table 3.2.1-2)	
	Boron Concentration for First Core Cycle Loading With Burnable Poison Rods		
29.	Fuel Loading Shutdown; Rods in (k = .86)	2000 ppm	
	(k = .90)	1564 ppm	
30.	Shutdown (k = .99) with Rods Inserted, Clean, Cold	849 ppm	
31.	Shutdown (k = .99) with Rods Inserted, Clean, Hot	572 ppm	10
32.	Shutdown (k = .99) with No Rods Inserted, Clean, Hot	1405 ppm	
33.	Shutdown (k = .99) with No Rods Inserted, Clean, Cold	1370 ppm	
	To Maintain k = 1 at Hot Full Power, No Rods Inserted:		
34.	Clean	1160 ppm	10
35.	Xenon	860 ppm	
36.	Xenon and Samarium	780 ppm	
37.	Shutdown, All But One Rod Inserted, Clean Cold (k = .99)	915 ppm	
38.	Shutdown, All But One Rod Inserted, Clean Hot (k = .99)	677 ppm	10

TABLE 3.2.1-1 (Cont'd)

BURNABLE POISON RODS

39.	Number and Material	1160 Borated Pyrex Glass
40.	Worth Hot $\Delta\rho$	7.2%
41.	Worth Cold $\Delta\rho$	5.4%

KINETIC CHARACTERISTICS

42.	Moderator Temperature Coefficient at Full Power ($°F^{-1}$)	$-.25\times10^{-4}$ to -3.00×10^{-4}
43.	Moderator Pressure Coefficient (psi^{-1})	$+.2\times10^{-6}$ to $+3.00\times10^{-6}$
44.	Moderator Density Coefficient, $\Delta k/gm/cm^3$	$-.1$ to $.30$
45.	Doppler Coefficient ($°F^{-1}$)	-1.1×10^{-5} to 1.6×10^{-5}
46.	Delayed Neutron Fraction, %	.50 to .72
47.	Prompt Neutron Lifetime, sec.	1.5 to 10^{-5} to 2.0×10^{-5}

TABLE 3.2.1-2

REACTIVITY REQUIREMENTS FOR CONTROL RODS

Requirements	Per Cent Δρ Beginning of Life	End of Life
Control		
Power Defect	1.90	3.05
Operational Maneuvering Band	0.40	0.40
Control Rod Bite	0.10	0.10
X-Y Xenon Rods	0.20	0.20
Total Control	2.60	3.75

TABLE 3.2.1-3

CALCULATED ROD WORTHS, $\Delta\rho$

Core Condition	Rod Configuration	Worth	Less 10%*	Design Reactivity Require-ments	Shutdown Margin
BOL, HFP	53 rods in	8.46%			
	52 rods in; Highest Worth Rod Stuck Out	7.43%	6.69%	2.60%	4.09%
EOL, HFP (2nd Cycle)	53 rods in	7.98%			
	52 rods in; Highest Worth Rod Stuck Out	6.48%	5.83%	3.75%	2.08%**

BOL = Beginning of Life

EOL = End of Life

HFP = Hot Full Power

* Calculated rod worth is reduced by 10% to allow for uncertainties.

** The design basis minimum shutdown margin is 1.95%.

TABLE 3.2.1-4

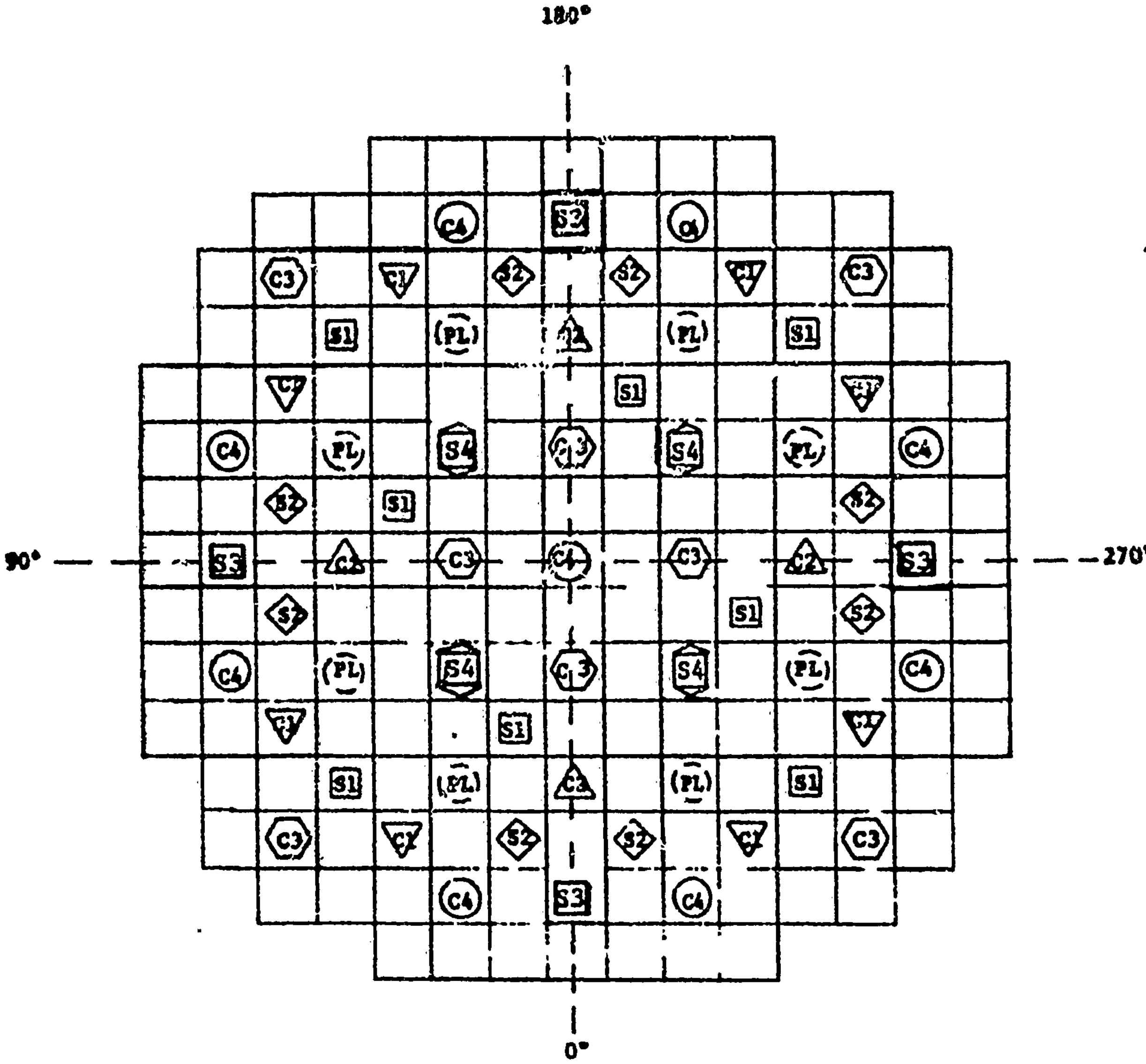
RESULTS OF CALCULATIONS AS A FUNCTION OF LABORATORY PROVIDING EXPERIMENTAL DATA[4]

Laboratory	Type of Experiment	No. of Experiments	Calculated $k \pm \sigma$
Westinghouse Atomic Power Division (WAPD)	Critical	16	0.9968 ± 0.0036
Bettis Atomic Power Laboratory	Critical	14	0.9940 ± 0.0022
Brookhaven National Laboratory	Exponential	35	0.9964 ± 0.0051
Hanford Atomic Products Operation	Exponential	20	0.9953 ± 0.0105
Babcock and Wilcox	Critical	26	0.9885 ± 0.0094
		111	

TABLE 3.2.1-5

CALCULATED AND MEASURED REACTIVITY EFFECTS OF VOID TUBES

Type of Core	No. of Tubes	Reactivity Change %Δk/k: Measured	Reactivity Change %Δk/k: Calculated
Unborated Core	0		
	4	-0.03	-0.034
	16	-0.11	-0.125
	196	-1.33	-1.416
Borated Core	0		
	4	-0.017	-0.020
	16	-0.076	-0.085
	196	-0.850	-0.943



GROUP	SYMBOL	NUMBER OF ROD CLUSTERS
S1	□	8
S2	◇	8
S3	▭	4
S4	⬣	4
C1	▽	8
C2	△	4
C3	⬡	8
C4	○	9
PL (Part Length)	◌	8
		61

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ROD CLUSTER GROUPS

Figure 3.2.1-1

CL

1.17							
1.09	1.22						
1.23	1.22	X 1.27					
1.11	1.24	1.20	1.19				
1.22	1.18	1.18	1.02	1.21			
1.12	1.14	1.06	1.02	.81	.98		
1.04	1.02	.94	.90	.80	.47		
.78	.82	.66	.52				

CL

CL - Center - Line
$F_{\Delta H}$ = 1.40 (X)

Assembly Average Power and Burnup, Cycle 1
(0 MWD/MT Burnup)

FIGURE 3.2.1-2

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CL

CL

1.03 16152							
1.07 16279	1.03 16438						
1.03 16468	1.08 17190	1.04 16705					
1.08 16402	1.04 16539	1.11 17114	1.09 16281				
1.06 16302	1.12 16864	1.08 16120	1.14 15741	X 1.19 17011			
1.12 16249	1.08 15632	1.13 15716	1.08 14680	1.08 13526	1.04 13811		
1.02 14068	1.12 14974	1.00 13271	1.04 13315	.88 11434	.58 7032		
.79 10620	.81 10984	.76 9472	.60 7607				

CL

CL - Center - Line
$F_{\Delta H}$ = 1.43 (X)

Assembly Average Power and Burnup, Cycle 1
(14,200 MWD/MT Burnup)

FIGURE 3.2.1-3
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0.671	1.090	1.439	1.343	1.362	1.113	0.882	0.603
1.090	1.340	1.447	1.487	1.309	1.088	0.777	0.579
1.439	1.447	1.544 ΔH → X N → X	1.437	1.308	0.944	0.[illegible]24	0.401
1.343	1.487	1.437	1.407	1.160	1.010	0.701	0.374
1.362	1.309	1.308	1.160	1.367	0.948	0.812	
1.113	1.088	0.944	1.010	0.948	1.109	0.535	
0.882	0.777	0.424	0.701	0.812	0.535		
0.603	0.579	0.401	0.374				

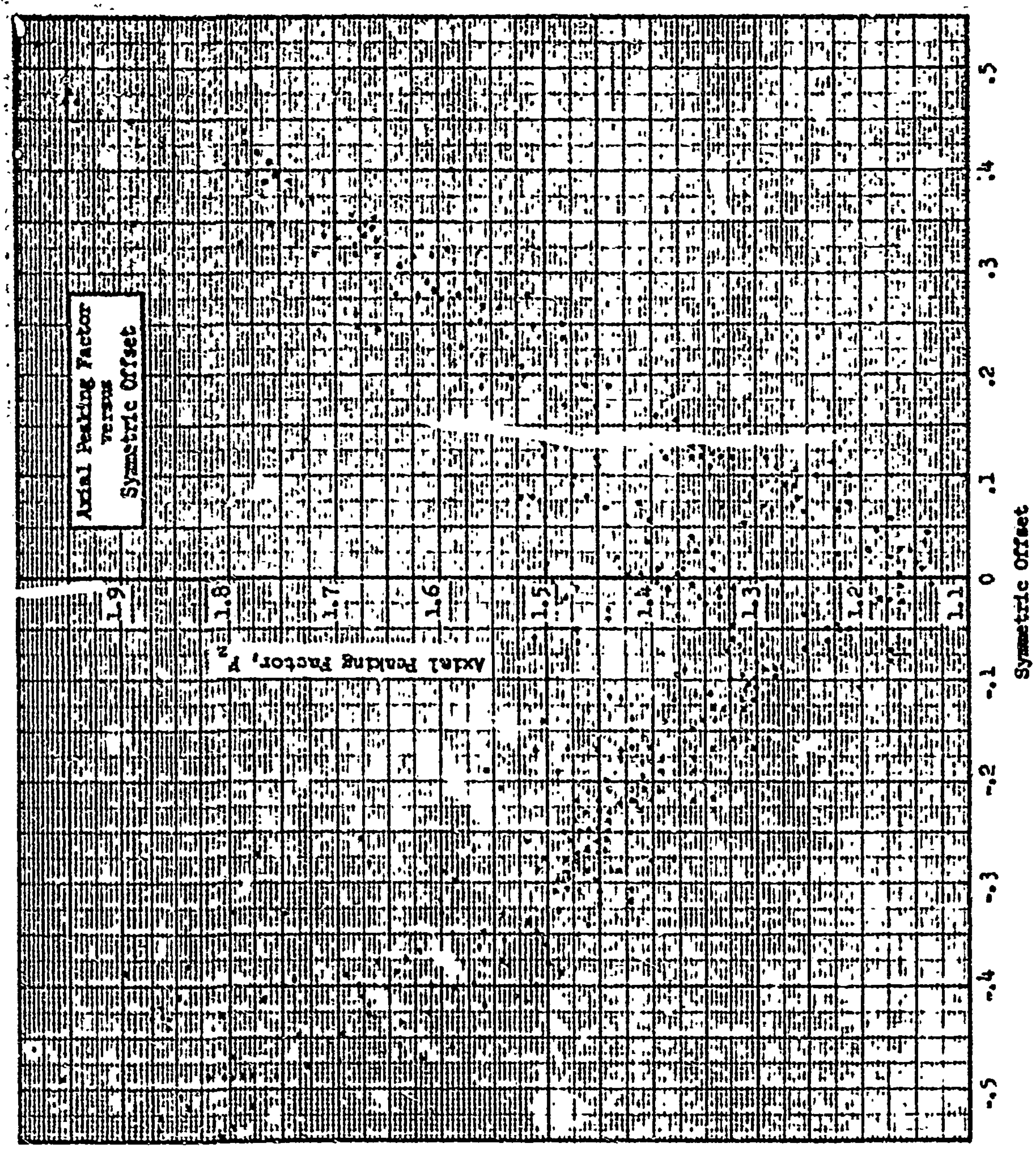
ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
BEGINNING OF LIFE, GROUP C4 INSERTED
2758 MWT $F^N_{\Delta H} = 1.70$

FIGURE 3.2.1-4

X X 1.402 X X	$\leftarrow F^N_{\Delta H}$ 1.290	1.349	1.130	1.169	1.157	1.171	0.917
1.290	1.395	1.286	1.161	0.989	1.110	1.149	0.967
1.349	1.286	1.227	0.962	0.568	0.929	1.045	0.799
1.130	1.161	0.962	0.940	0.830	0.999	1.000	0.633
1.169	0.989	0.568	0.830	1.189	0.954	0.938	
1.157	1.110	0.929	0.999	0.954	1.159	0.584	
1.171	1.149	1.045	1.000	0.938	0.584		
0.917	0.867	0.799	0.633				

ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
BEGINNING OF LIFE, PART LENGTH RODS IN
2758 MWT $F^N_{\Delta H}$ = 1.54

FIGURE 3.2.1-5

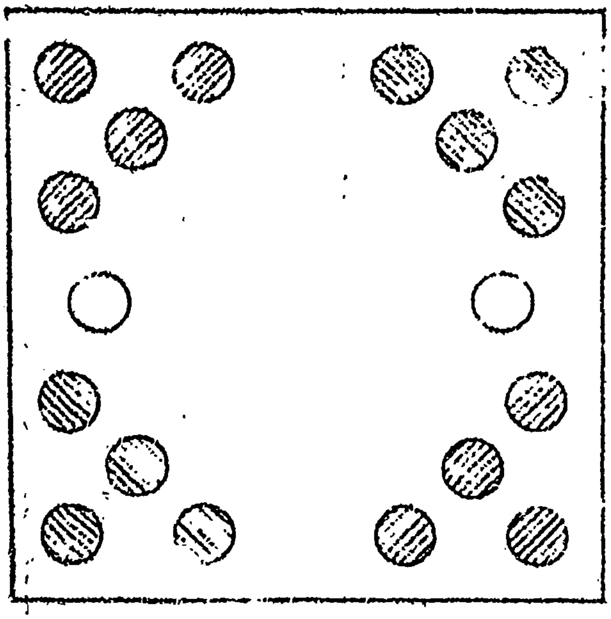


AXIAL PEAKING FACTOR Vs. SYMMETRIC OFFSET
FIG. 3.2.1-6

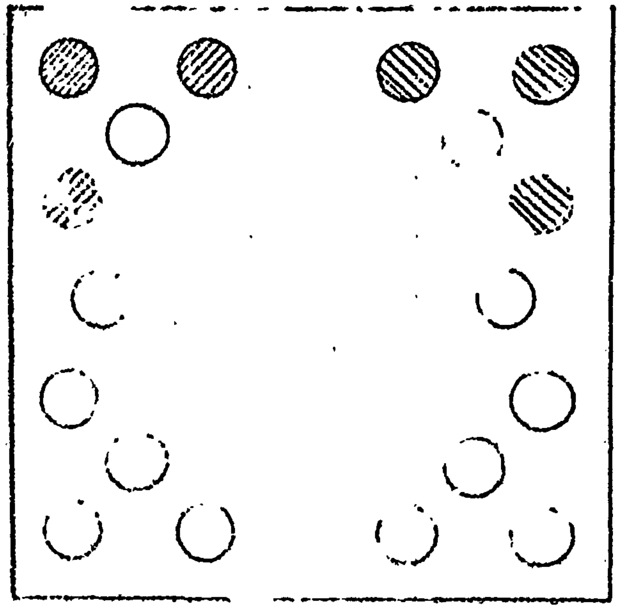
					6		6		6					
		6		12		16		16		12		6		
	6		16		12		12		12		16		6	
		16		16		12		12		16		16		
	12		16		12		16		12		16		12	
6		12		12		12		12		12		12		6
	16		12		12		16		12		12		16	
6		12		16		16		16		16		12		6
	16		12		12		16		12		12		16	
6		12		12		12		12		12		12		6
	12		16		12		16		12		16		12	
		16		16		12		12		16		16		
	6		16		12		12		12		16		6	
		6		12		16		16		12		6		
					6		6		6					

Distribution of Burnable Poison Rods -
Number of B. P. Rods per Assembly

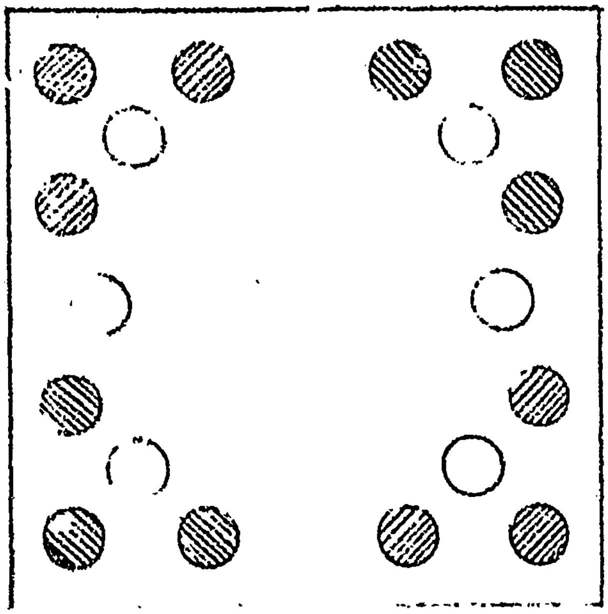
FIGURE 3.2.1-7



16 RODS



6 RODS



12 RODS

ARRANGEMENT OF BURNABLE POISON
RODS WITHIN AN ASSEMBLY

FIGURE 3.2.1-8

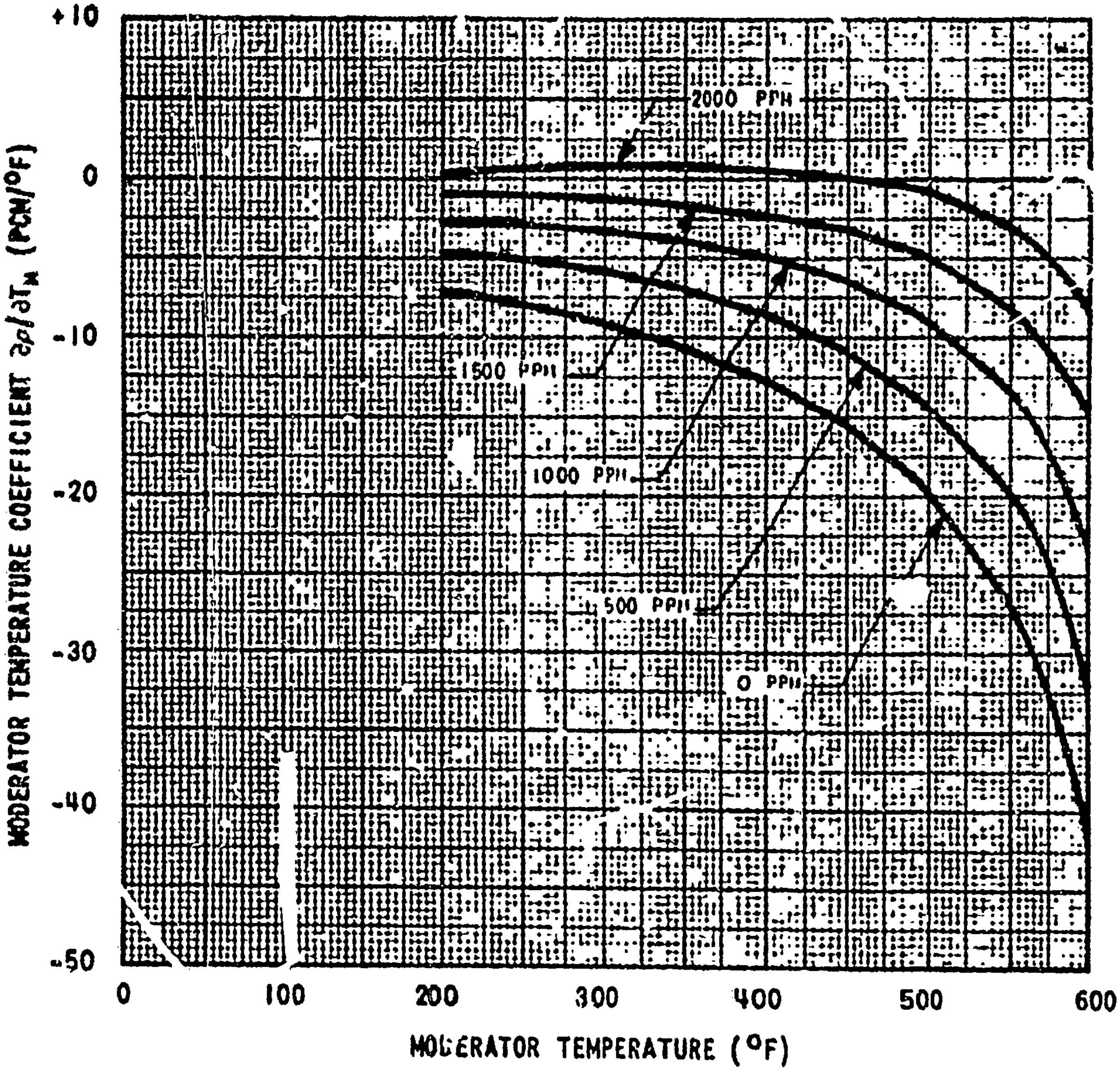


Figure 3.2.1-9a

Moderator Temperature Coefficient versus Moderator Temperature (EOL, Cycle 1)

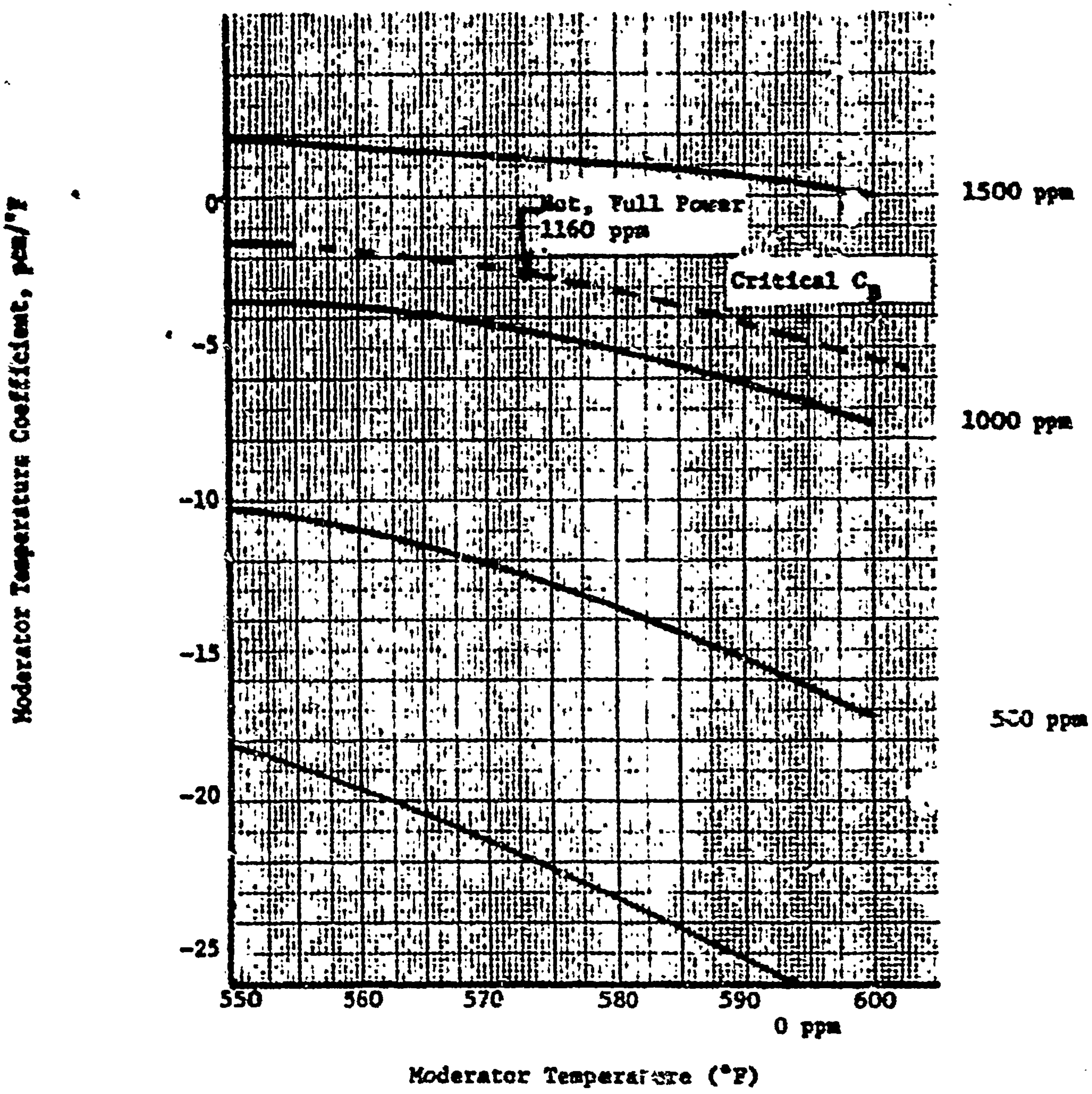


Figure 3.2.1-9b

Moderator Temperature Coefficient versus Moderator Temperature

(BOL, Cycle 1) Hot-to-Power Swing Calculated at Full Power

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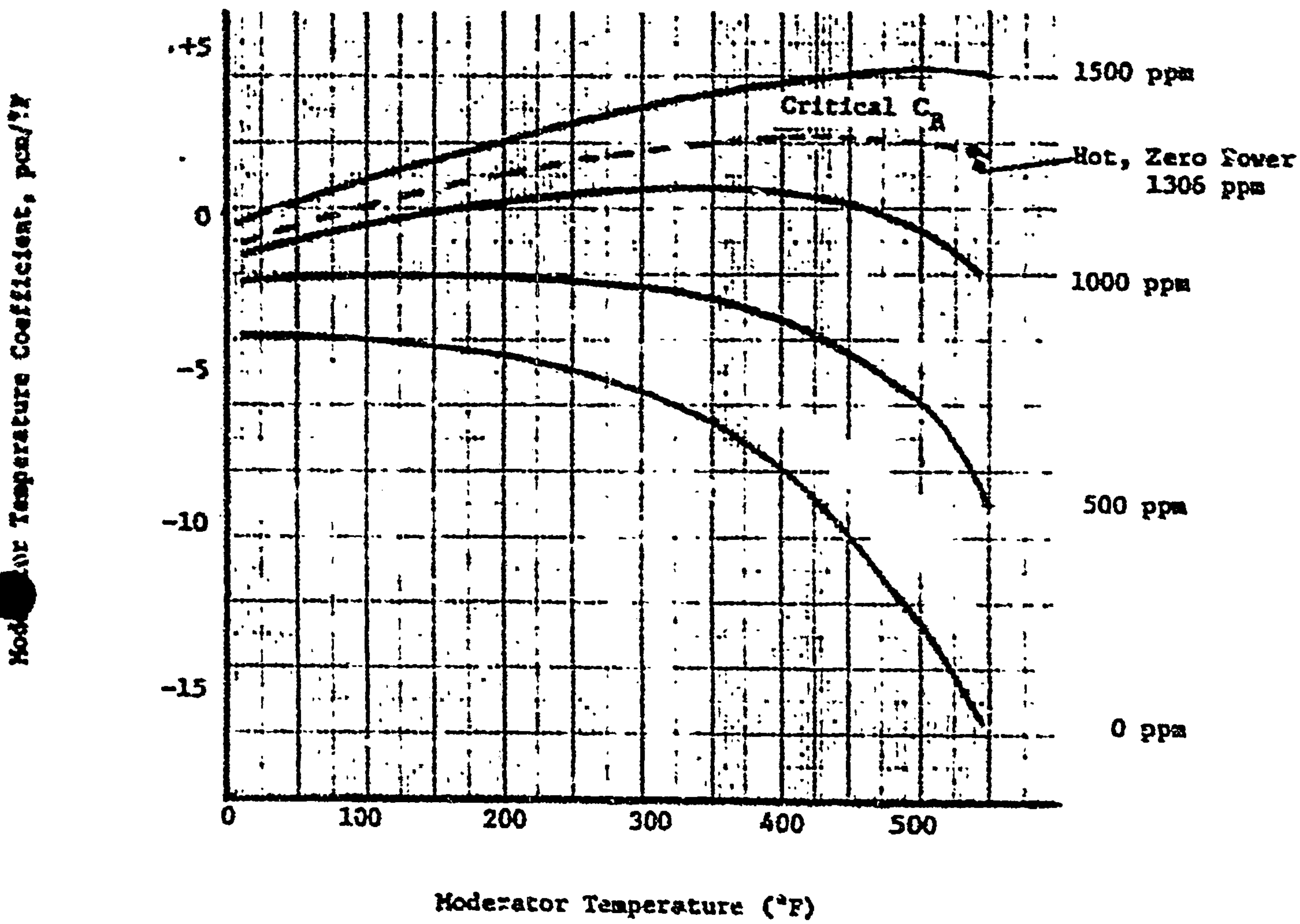


Figure 3.2.1-9c

Moderator Temperature Coefficient versus Moderator Temperature

(BOL, Cycle 1) Cold-to-Hot Swing Calculated at Zero Power

DOPPLER COEFFICIENT

vs

EFFECTIVE FUEL TEMPERATURE

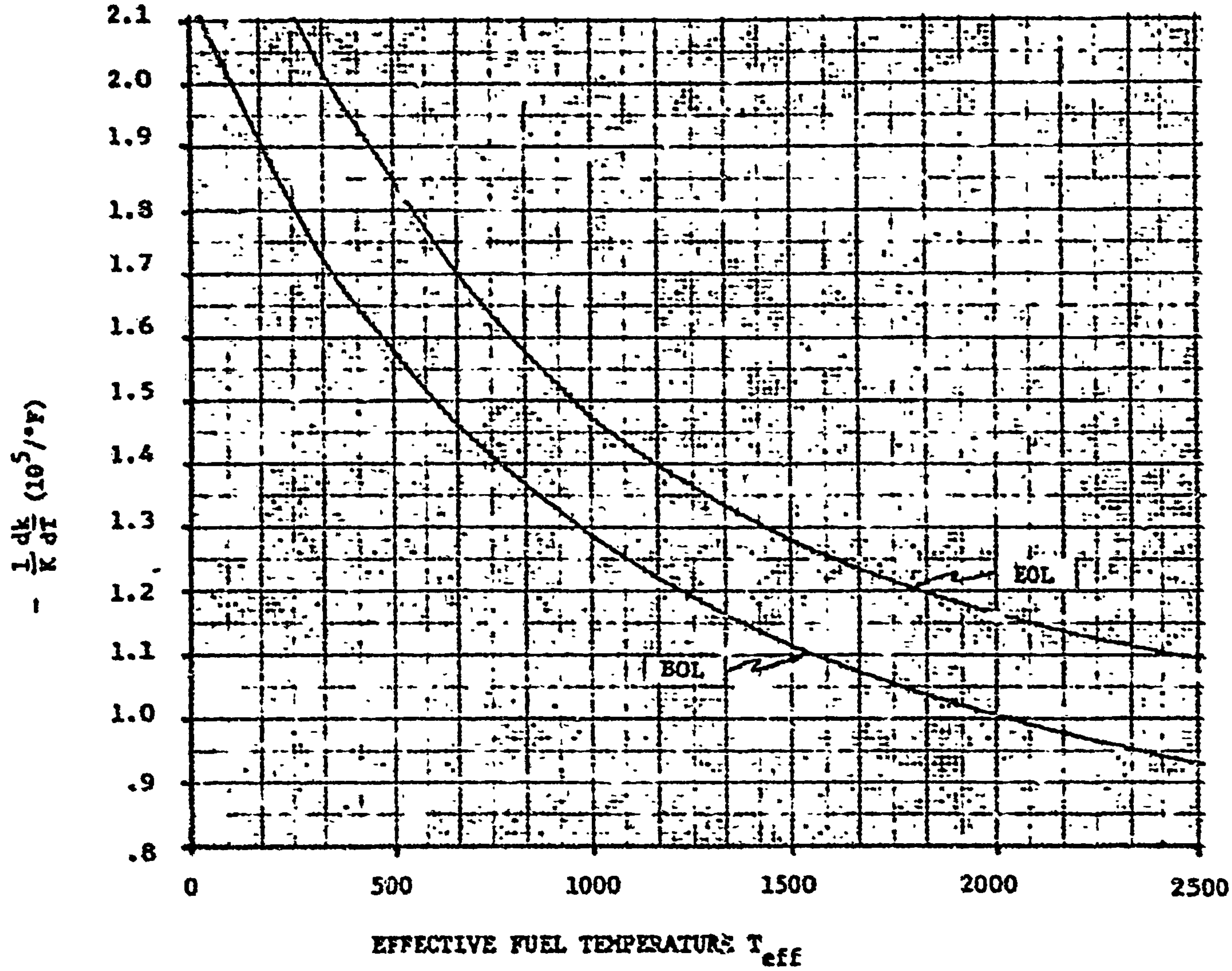
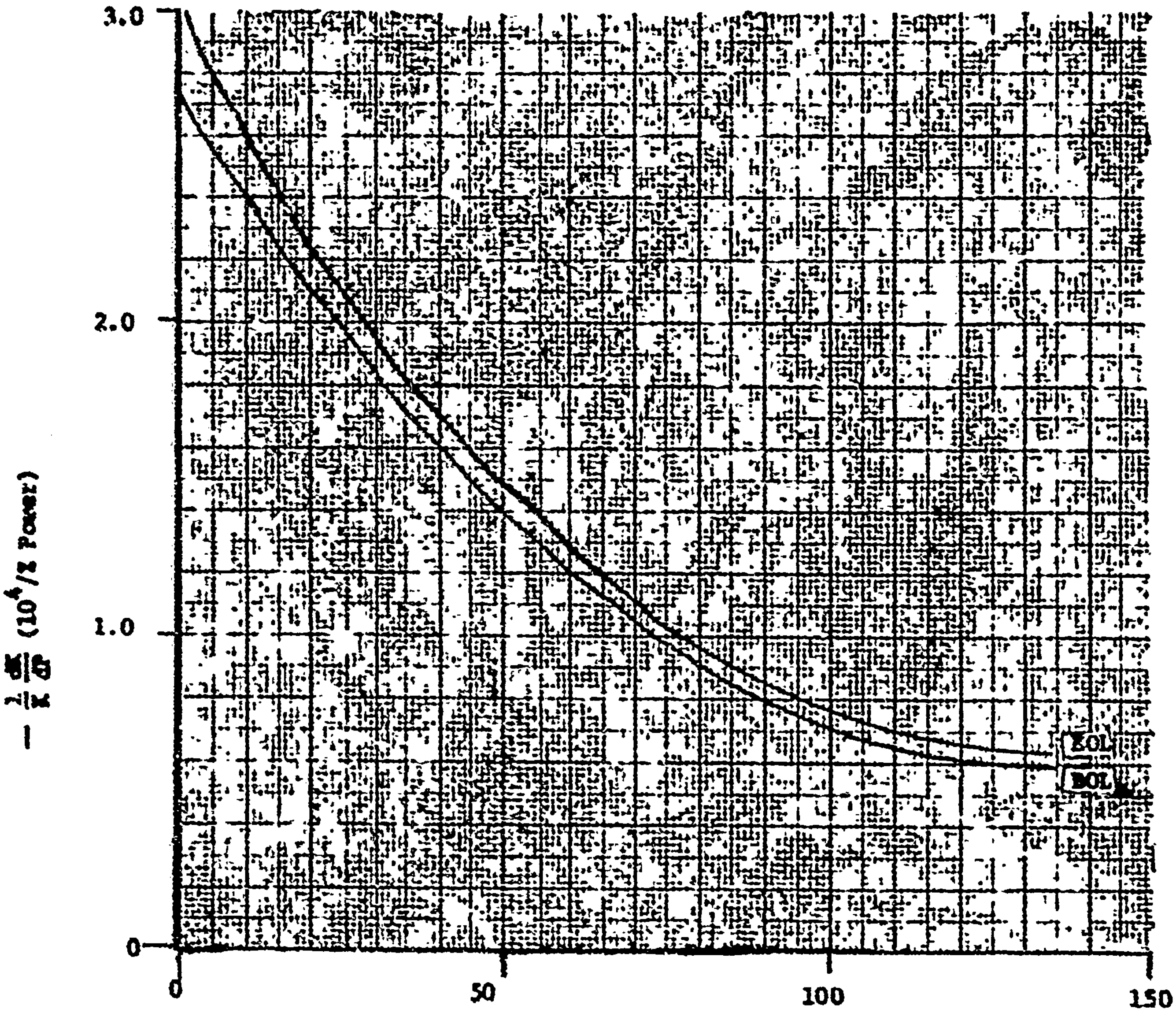


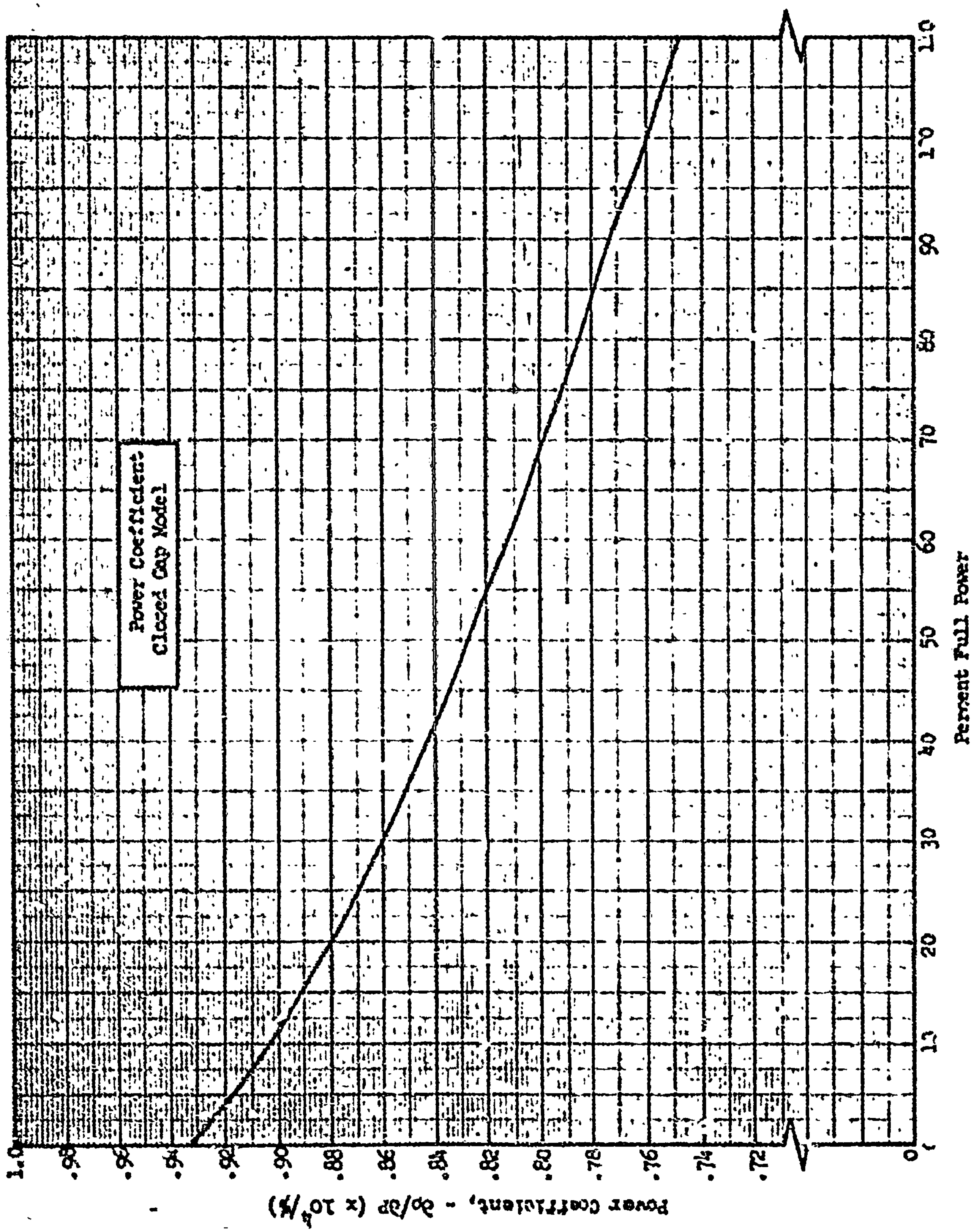
FIGURE 3.2.1-10

POWER COEFFICIENT VS PERCENT POWER
WITH T_{MOD} = 572. °F
E = 2.7 W/0
BOL = 2100 ppm

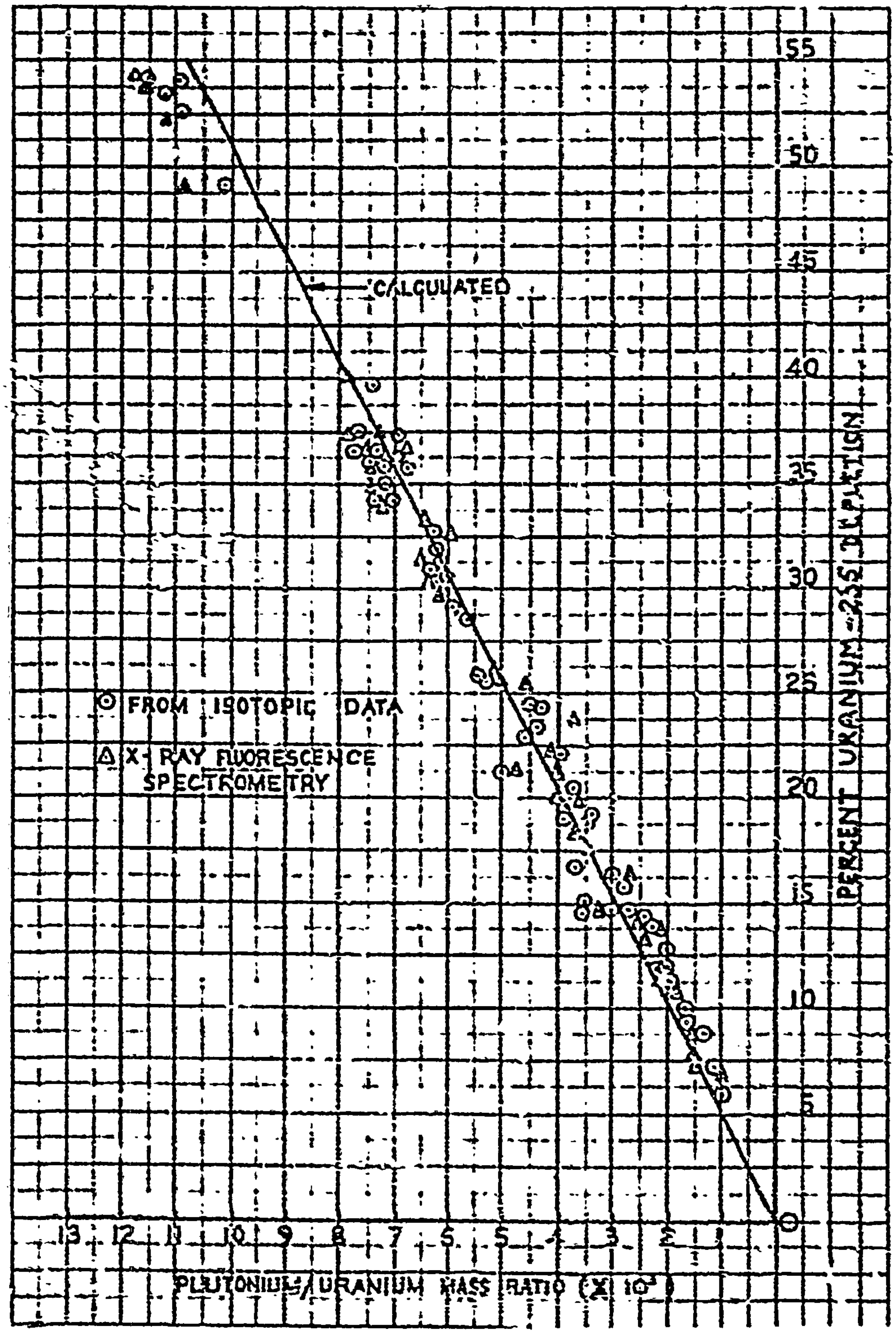


PERCENT OF FULL POWER, 2758 MWT

FIGURE 3.2.1-11

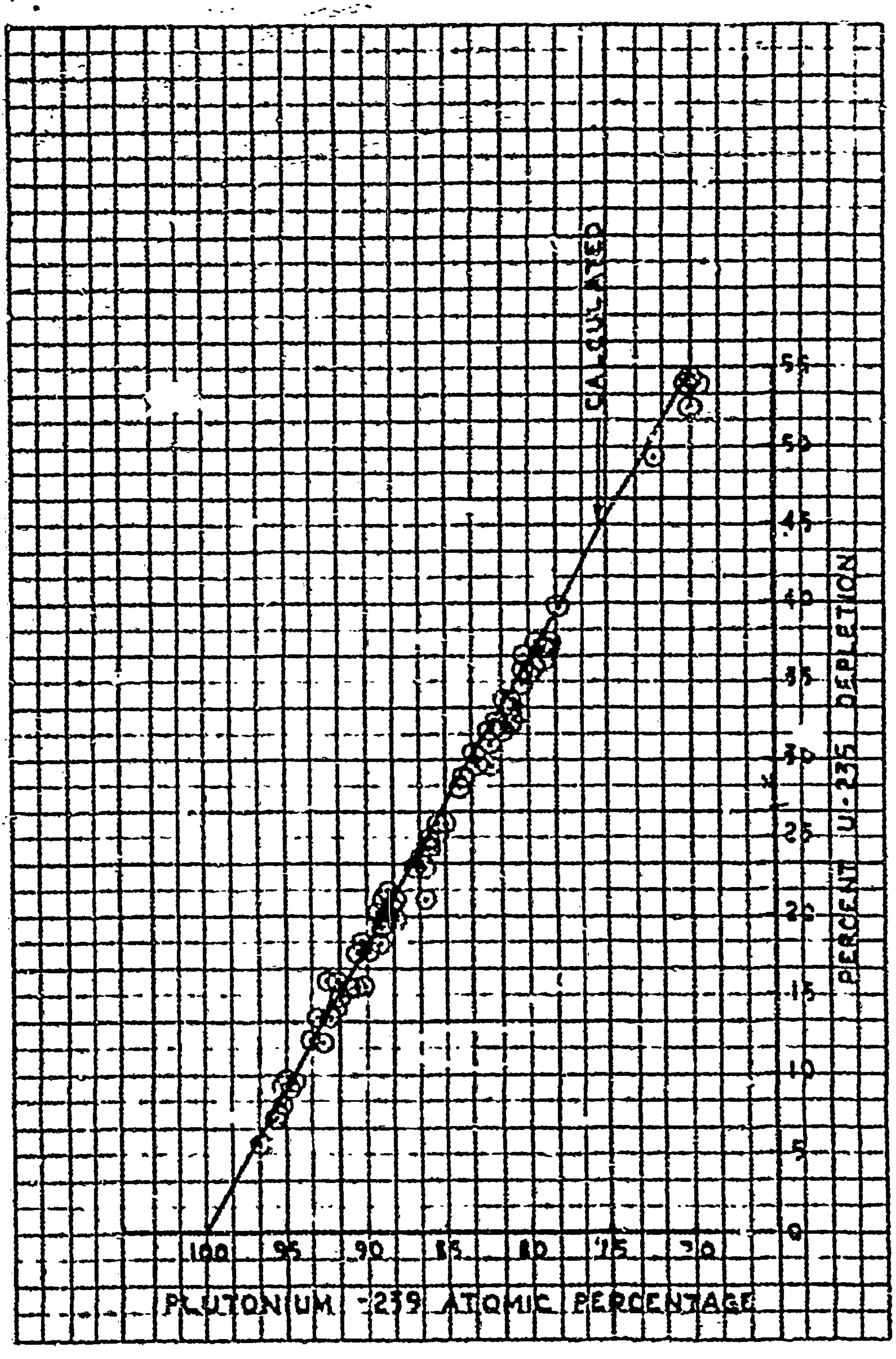


POWER COEFFICIENT CLOSED GAP MODEL
FIG. 3.2.1-12

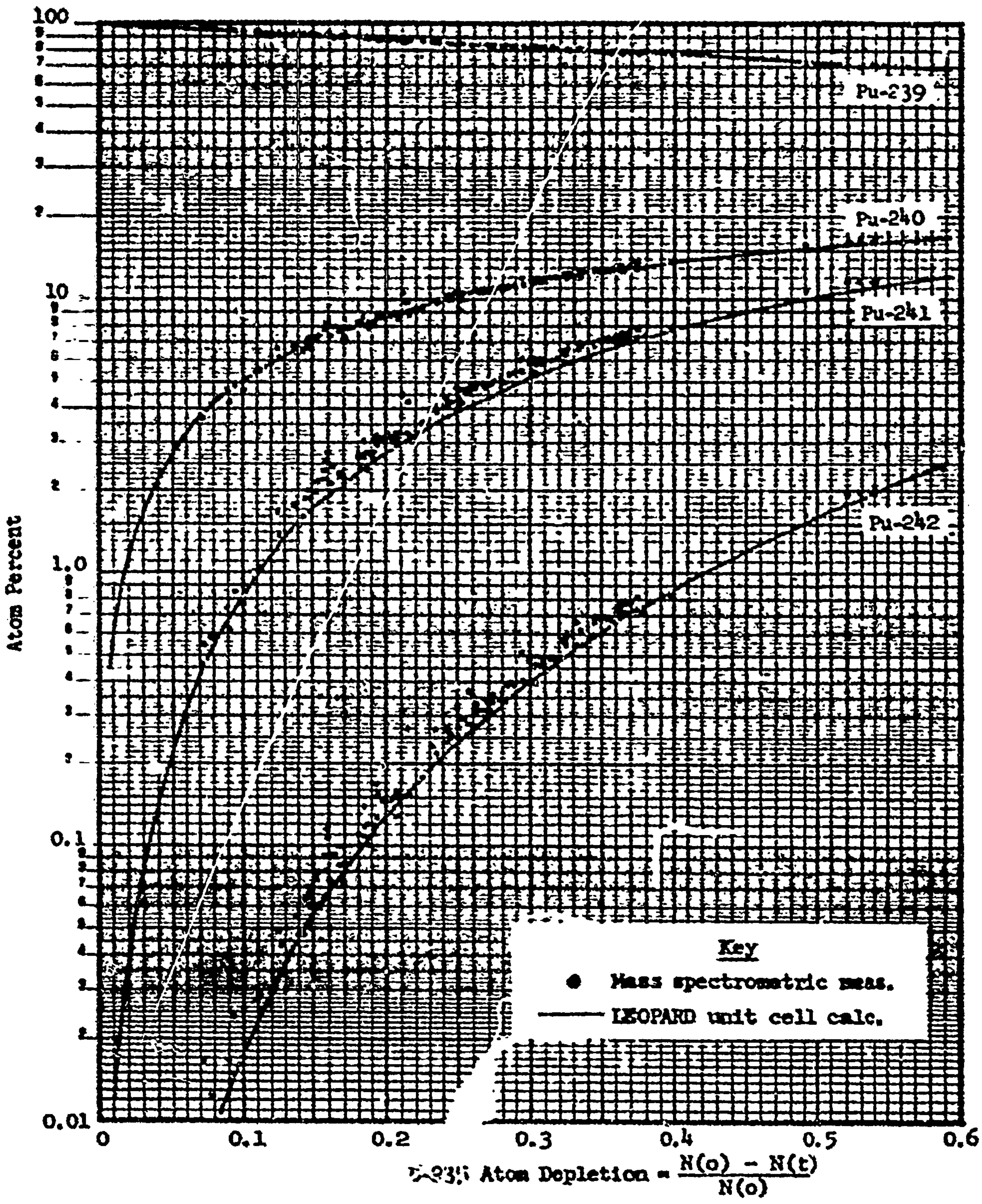


PLUTONIUM/URANIUM MASS RATIO AS A FUNCTION OF URANIUM-235 DEPLETION

FIG. 3.2.1-13

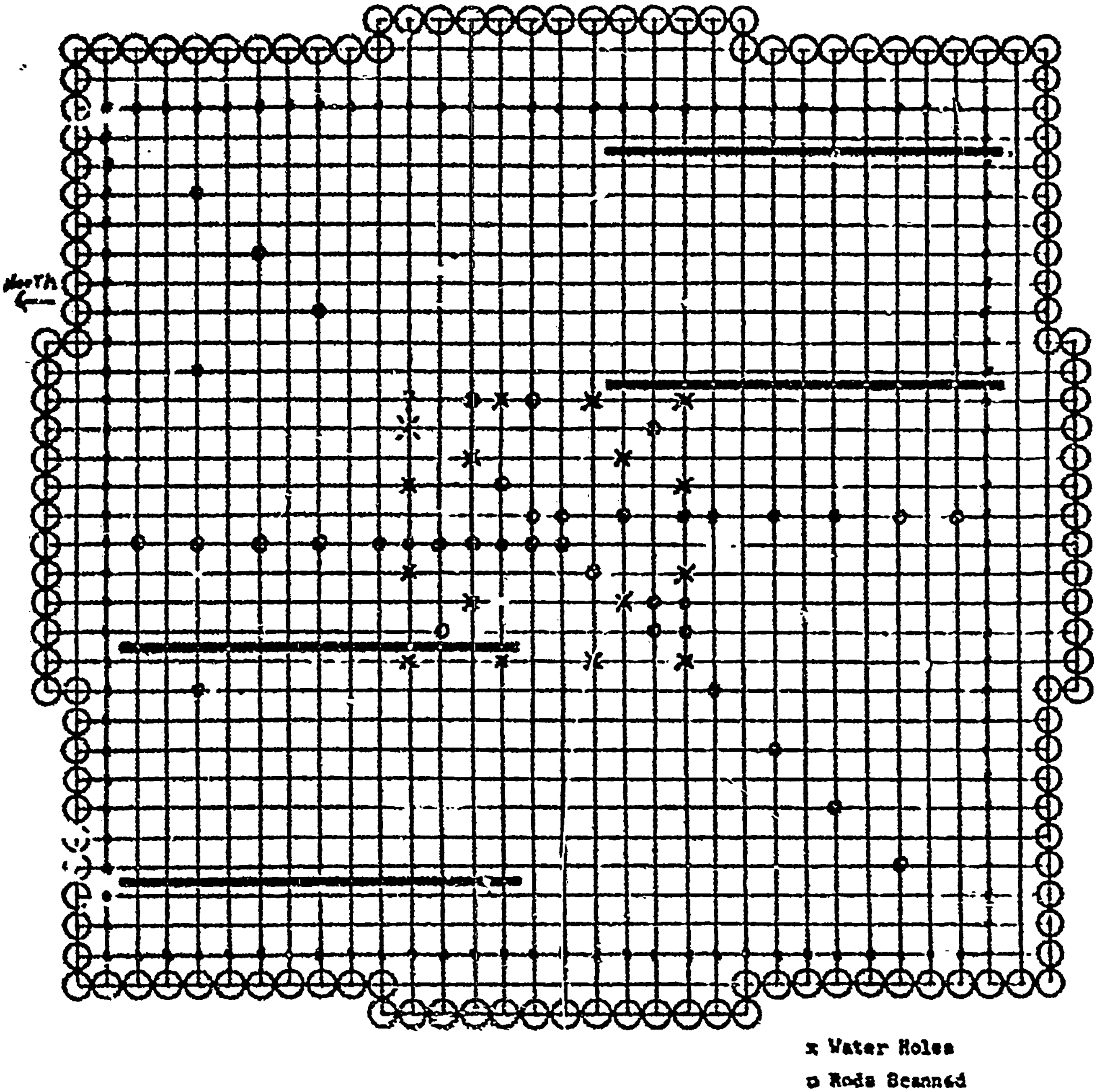


FRACTION OF PLUTONIUM-239 IN PLUTONIUM AS A FUNCTION OF URANIUM-235 DEPLETION FIG. 3.2.1-1



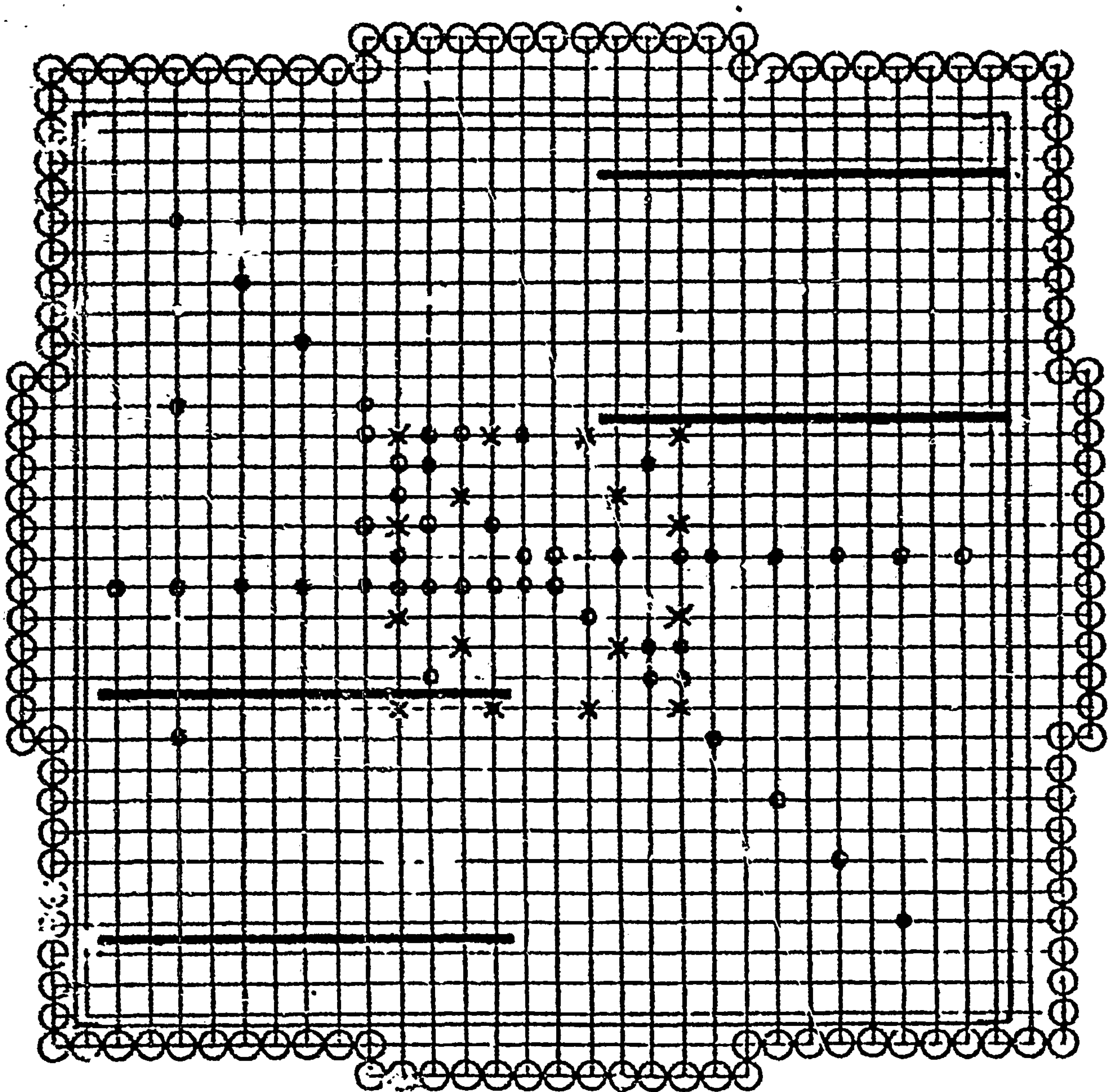
COMPOSITION OF PLUTONIUM AS A FUNCTION OF URANIUM-235 DEPLETION

FIGURE 3.2.1-15



0.484" Diameter, 0.6" Pitch

PLAN OF CRITICAL EXPERIMENT (UNBORATED CASE)
FIG. 3.2.1-16



X - Water Hole
O - Rod Scanned
Core: 30 x 30

0.484 " Diameter , 0.6 " Pitch

PLAN OF CRITICAL EXPERIMENT (BORATED CASE)
FIG. 3.2.1-17

(1.[illegible]) 1.[illegible] [illegible] 1.5[illegible]7 1.637	(1.[illegible]) 1.586	(1.6[illegible]) 1.629 1.630	(1.714) 1.689	(1.7[illegible]) 1.720 1.764 1.743	(1.570) 1.519 1.514	(1.[illegible])	(1.292) 1.305	(1.202)	(1.112) 1.080 1.115	(1.018)	(.922) .939 .[illegible]	(.835)	(.790) .847 .811
	(1.707) 1.692 1.718	(1.[illegible])	(1.[illegible]) 1.[illegible]29	Water Hole	(1.6[illegible]) 1.652								
		Water Hole	(1.894) 1.851 1.835	(1.776) 1.734 1.834 1.722	(1.549)								
			(1.757) 1.753 1.715	(1.723) 1.728 1.789 1.682	(1.512)								
				Water Hole	(1.549) 1.528								
					(1.386) 1.[illegible] 1.344								
							(1.043) 1.034 1.039						
									(.789) .795 .790				
											(.544) .573 .546		

Comparison of Experimental and Calculated Power Distribution
30 x 30 Core With 16 Water Holes - 479 ppm Boron in Water
(1/8th Core Symmetry - One Block Represents One Fuel Rod)

(XXXX) PDQ Result
XXXX Experimental Value

BORATED POWER DISTRIBUTION COMPARISON
FIG. 3.2.1-18

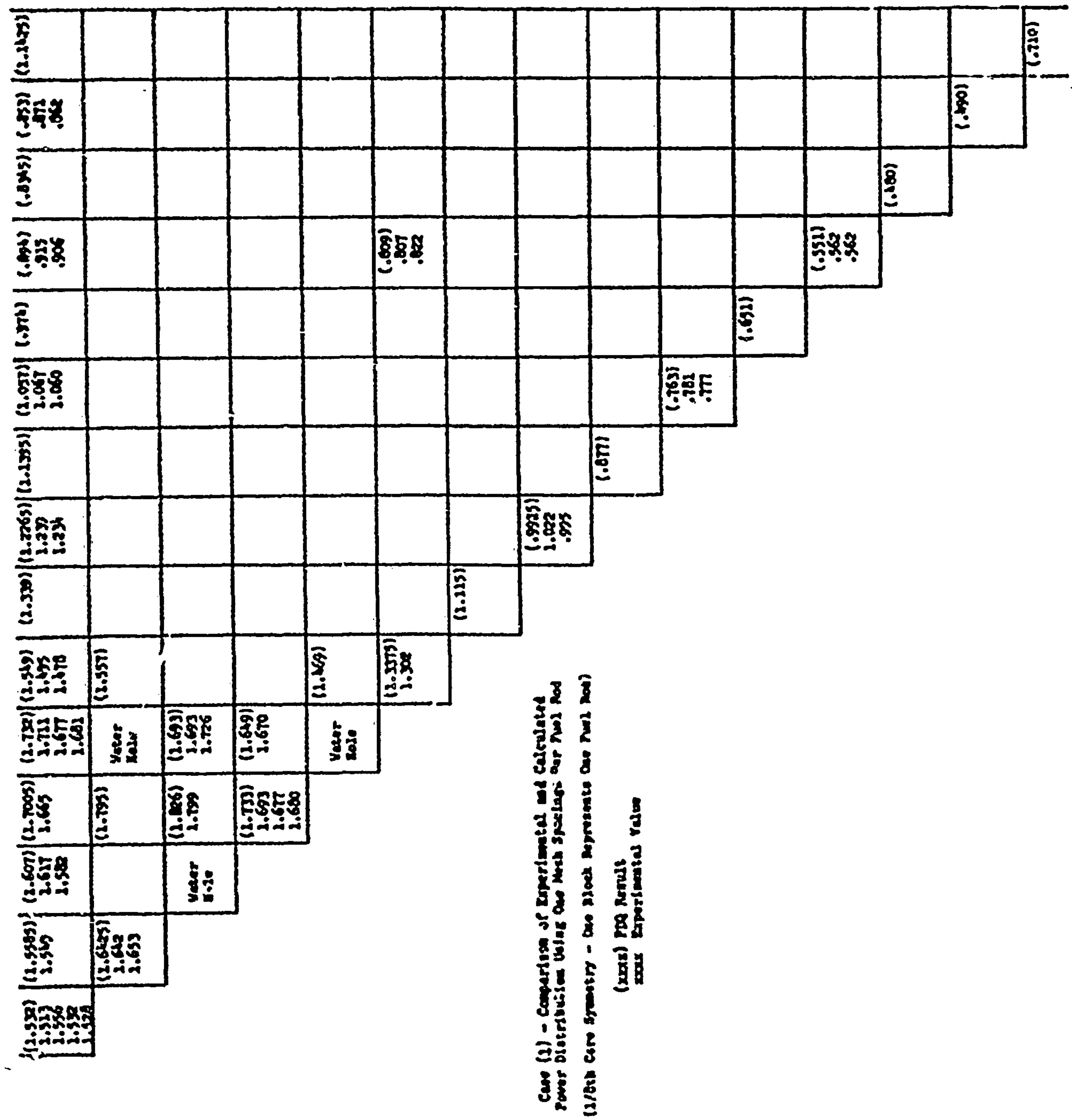
(1.525) 1.5036 [illegible] 1.528 1.521	(1.548) 1.542	(1.586) 1.610 1.574	(1.651) 1.657	(1.727) 1.703 1.664 1.673	(1.515) 1.488 1.472	(1.336)	(1.230) 1.233 1.228	(1.145)	(1.062) 1.0625 1.055	(.984)	(.897) .911 .908	(.833)	(.8460) .8675 .858	
	(1.637) 1.646 1.635		(1.829)	Water Hole	(1.601)									
		Water Hole	(1.842) 1.791	(1.722) 1.718 1.686										
			(1.704) 1.666 1.670 1.672	(1.689) 1.663										
				Water Hole	(1.494)									
					(1.330) 1.296						(.811) .803 .818			
						(1.119)								
							(.997) 1.017 .990							
								(.882)						
									(.766) .778 .774					
										(.653)				
											(.550) .559 .559			
												(.476)		
													(.480)	
														(.702)

Case (2) - Comparison of Experimental and Calculated Power Distribution Using Two Mesh Spacings Per Fuel Rod

(1/8th Core Symmetry - One Block Represents One Fuel Rod)

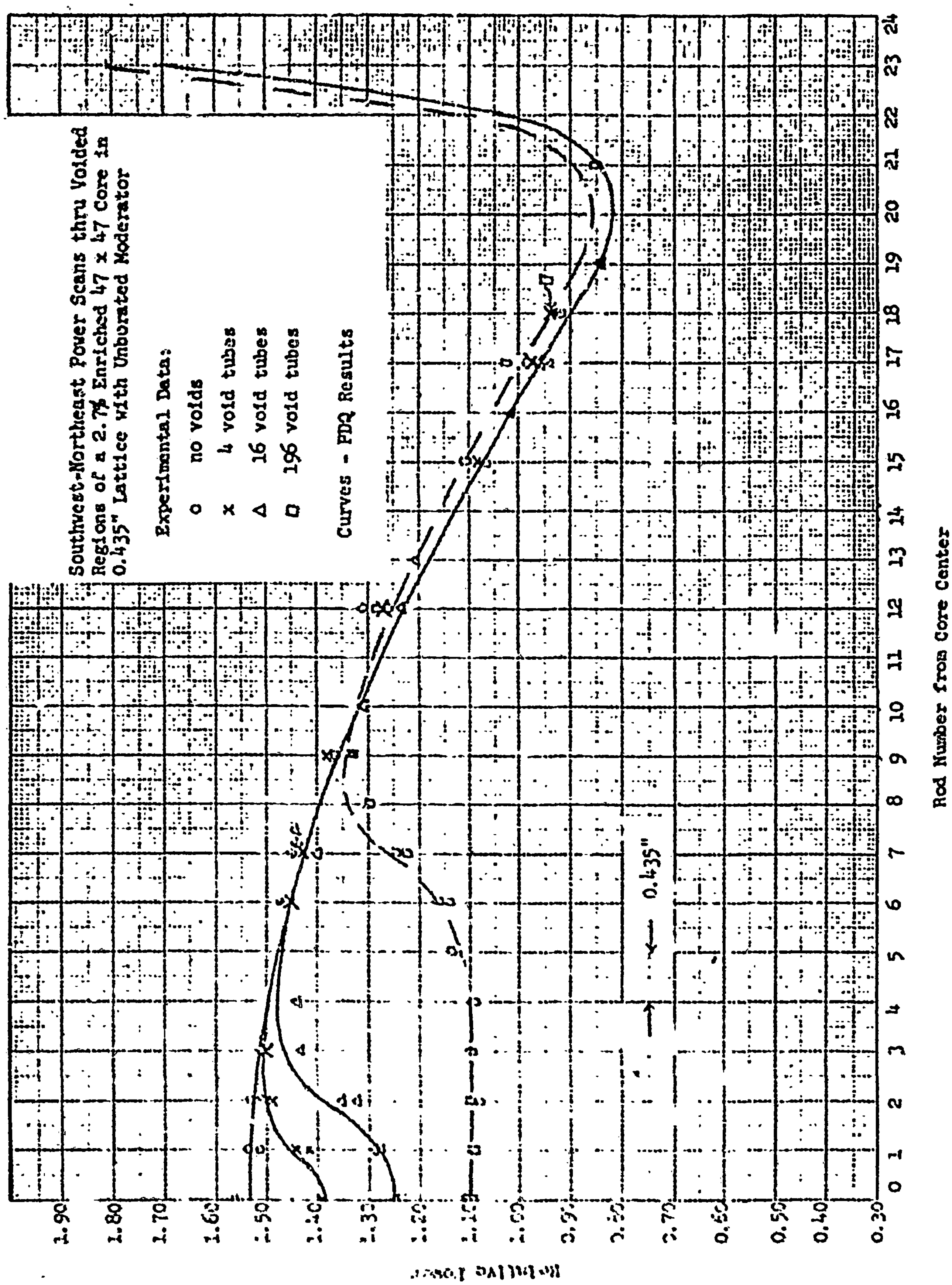
(xxxx) PDQ Result
xxxx Experimental Value

UNBORATED POWER DISTRIBUTION COMPARISON
FIG. 3.2.1-19



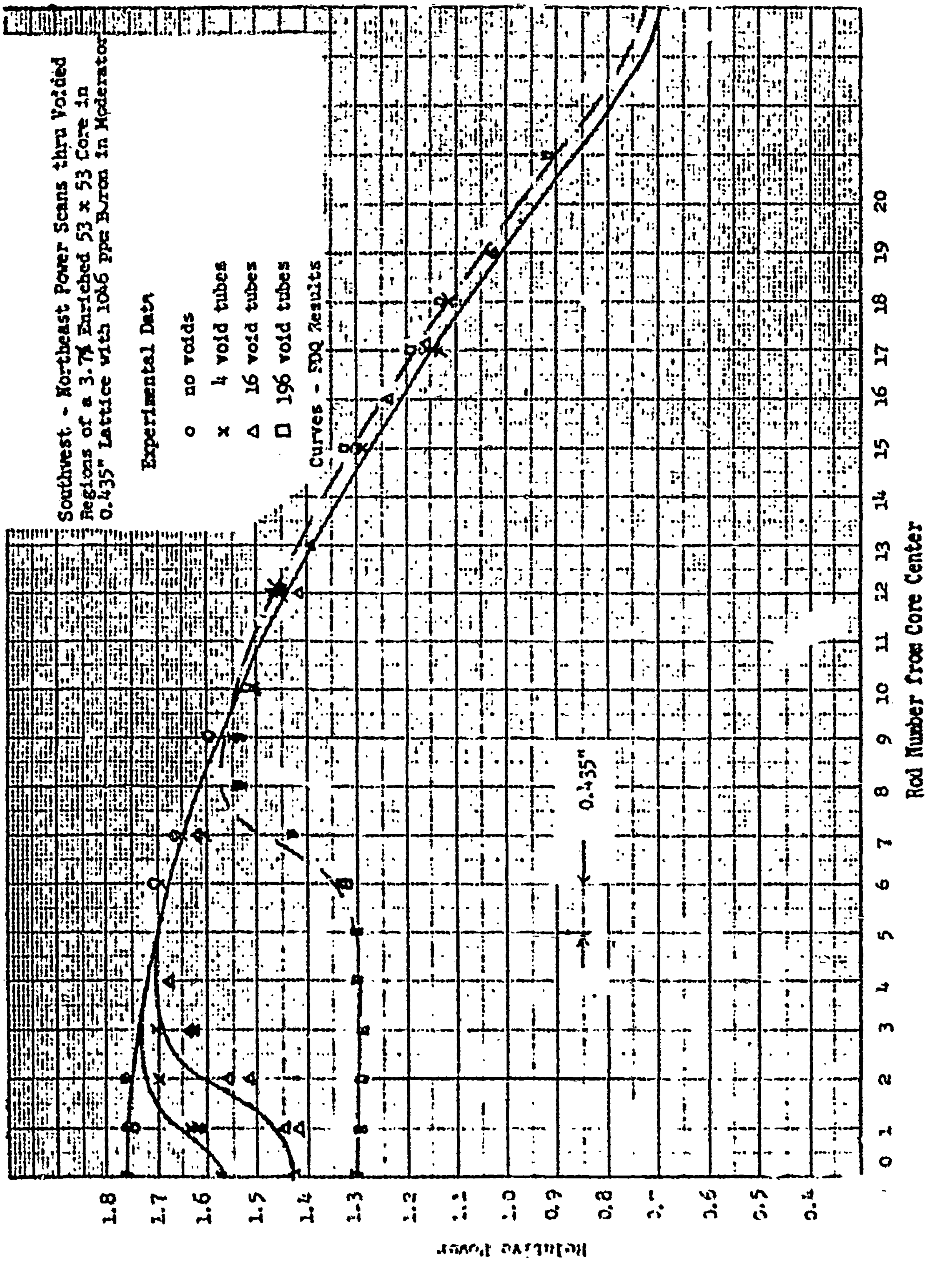
COMPARISON OF EXPERIMENTAL AND CALCULATED POWER DISTRIBUTION USING ONE MESH SPACING PER FUEL ROD

FIG. 3.2.1-20



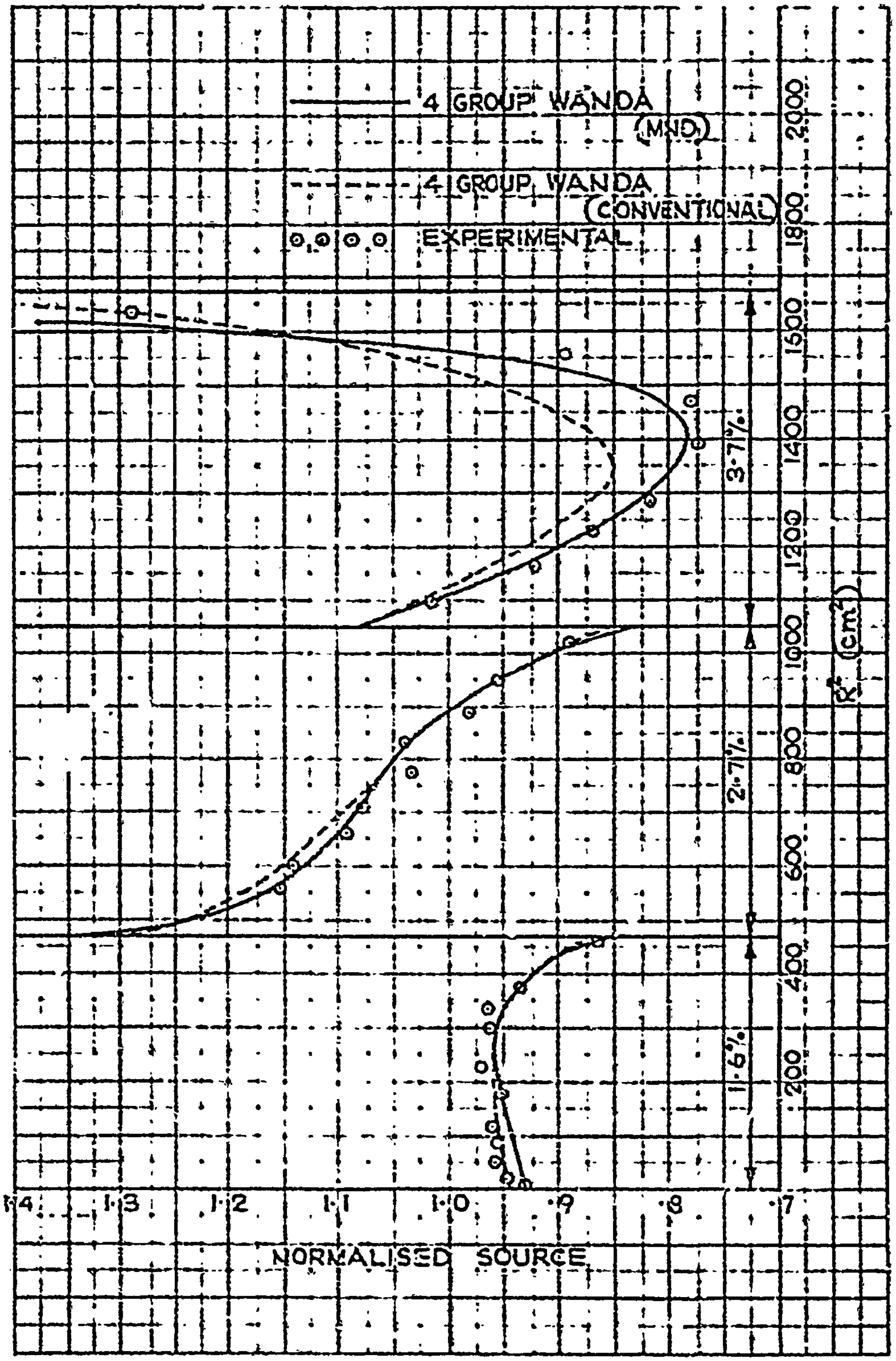
COMPARISON OF CALCULATED POWER DISTRIBUTION WITH EXPERIMENTAL POWER SCANS - UNBORATED CORE

FIG. 3.2.1-21



COMPARISON OF CALCULATED POWER DISTRIBUTION WITH EXPERIMENTAL POWER SCANS - BORATED CORE

FIG. 3.2.1-22



RADIAL FUEL ROD SCAN
FIG. 3.2.1-23

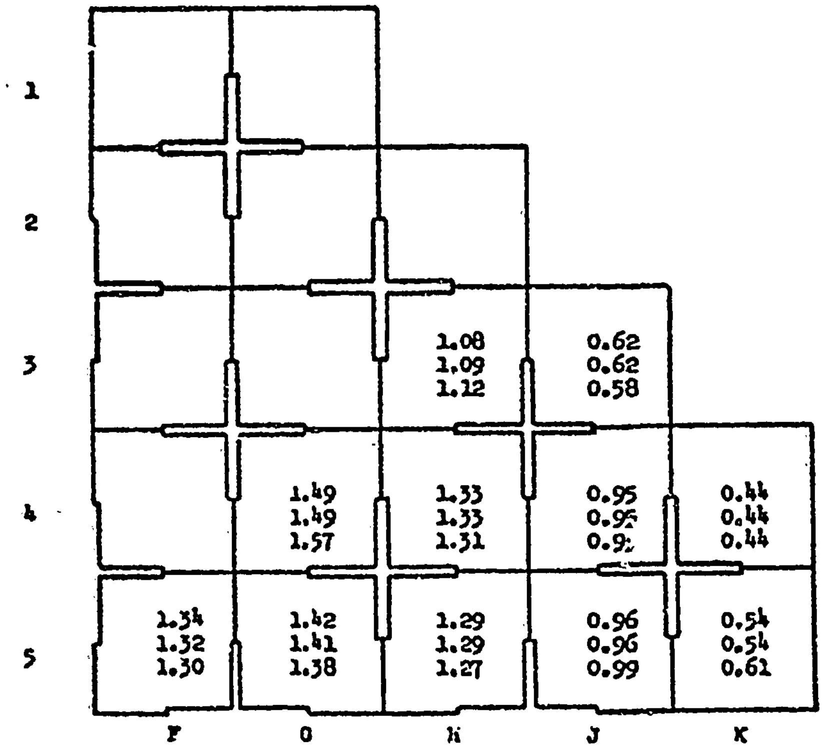
	F	G	H	J	K
1	0.60 0.59 0.56	0.44 0.43 0.40			
2	1.11 1.10 1.08	0.90 0.89 0.82	0.53 0.53 0.45		
3	1.38 1.38 1.45	1.19 1.18 1.22	0.91 0.91 1.06	0.52 0.52 0.48	
4	1.53 1.60 1.59	1.37 1.37 1.40	1.15 1.15 1.14	0.85 0.86 0.81	0.42 0.42 0.37
5	1.68 1.70 1.69	1.52 1.47 1.52	1.31 1.30 1.37	1.04 1.04 1.03	0.56 0.56 0.55

Note: The top value in each assembly is found from the control rod interchange TURBO, the centre value is from the TURBO with no interchange, and the bottom value is experimentally determined.

Analytical and Experimental Power Distribution for YANKEE Core 1. TURBO Time Step 14, 7400 EFPH - ¼ core shown.

YANKEE CORE 1 POWER DISTRIBUTION COMPARISON

FIG. 3.2.1-24



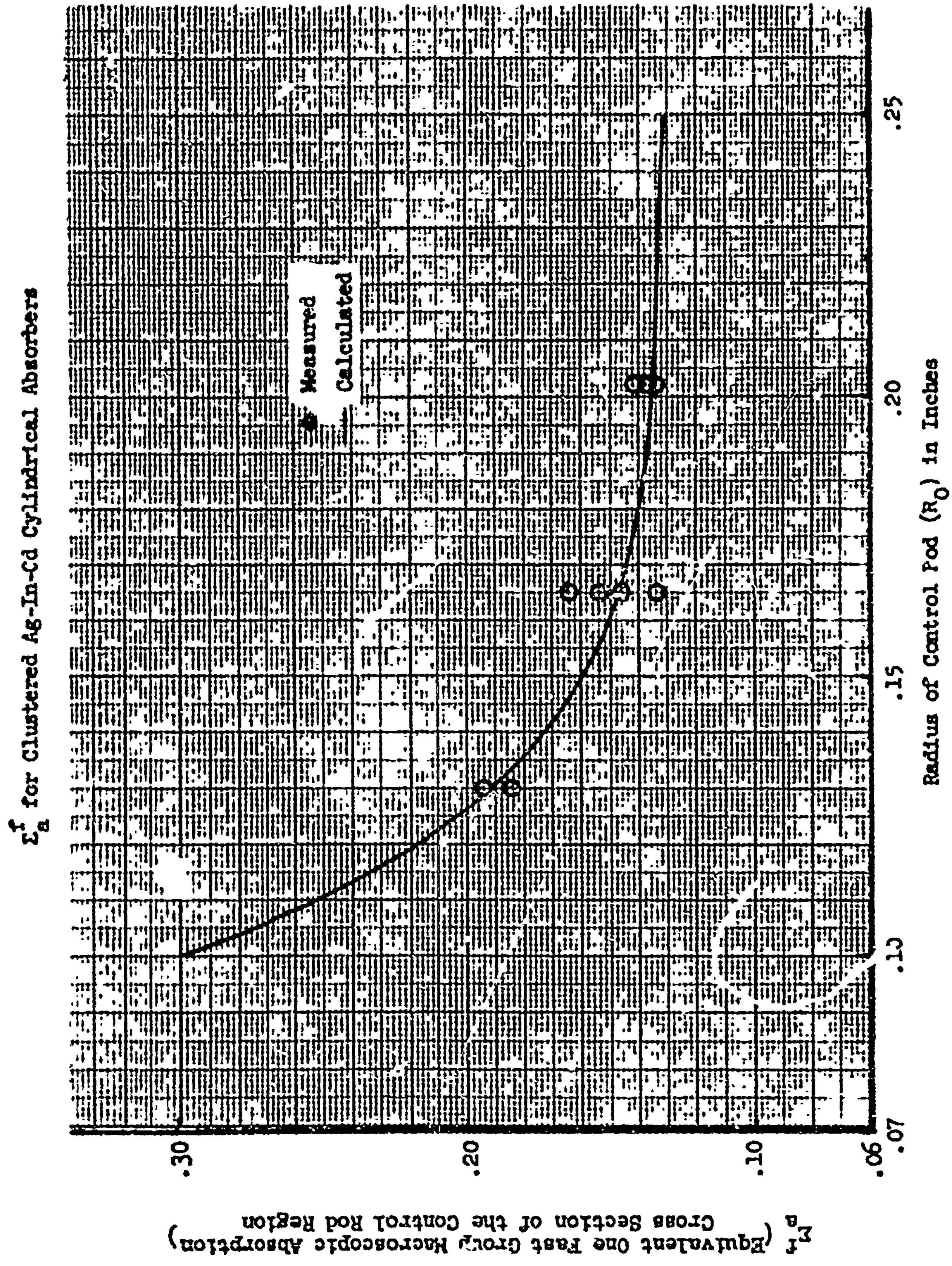
Notes:

1. The top value in each assembly is found from the control rod interchange TURBO, the centre value is from the TURBO with no interchange, and the bottom value is experimentally determined.

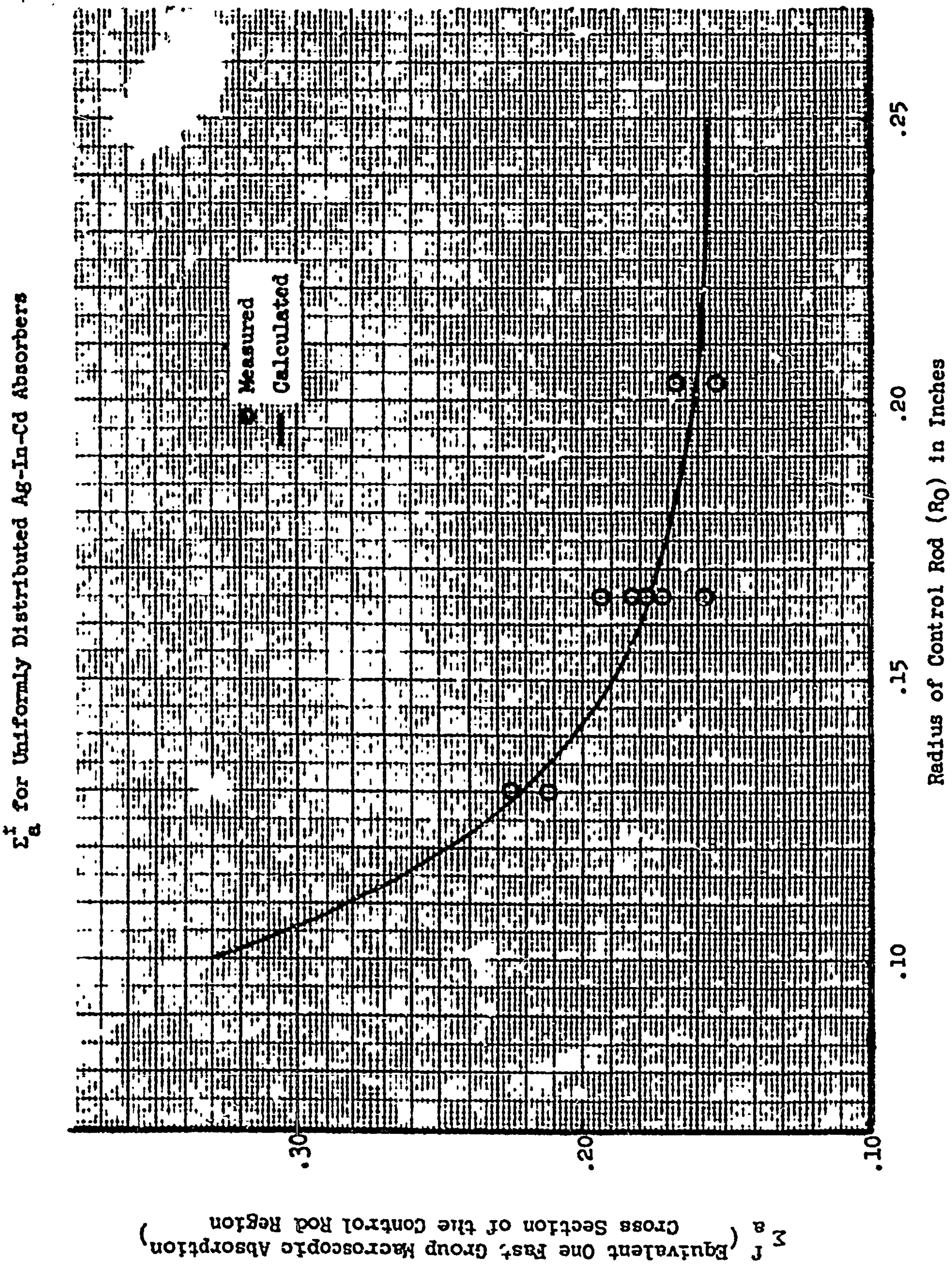
2. The burn-up distribution is defined by the ratio of the average burn-up in an assembly to the average in the quadrant. The numbers shown are the average of such values in symmetric assemblies.

Analytical and Experimental Burn-up Distributions for YANKEE Core 1. TURBO Time Step 14, 7400 EFPH - 1/4 core shown, Figures for 1/8 core.

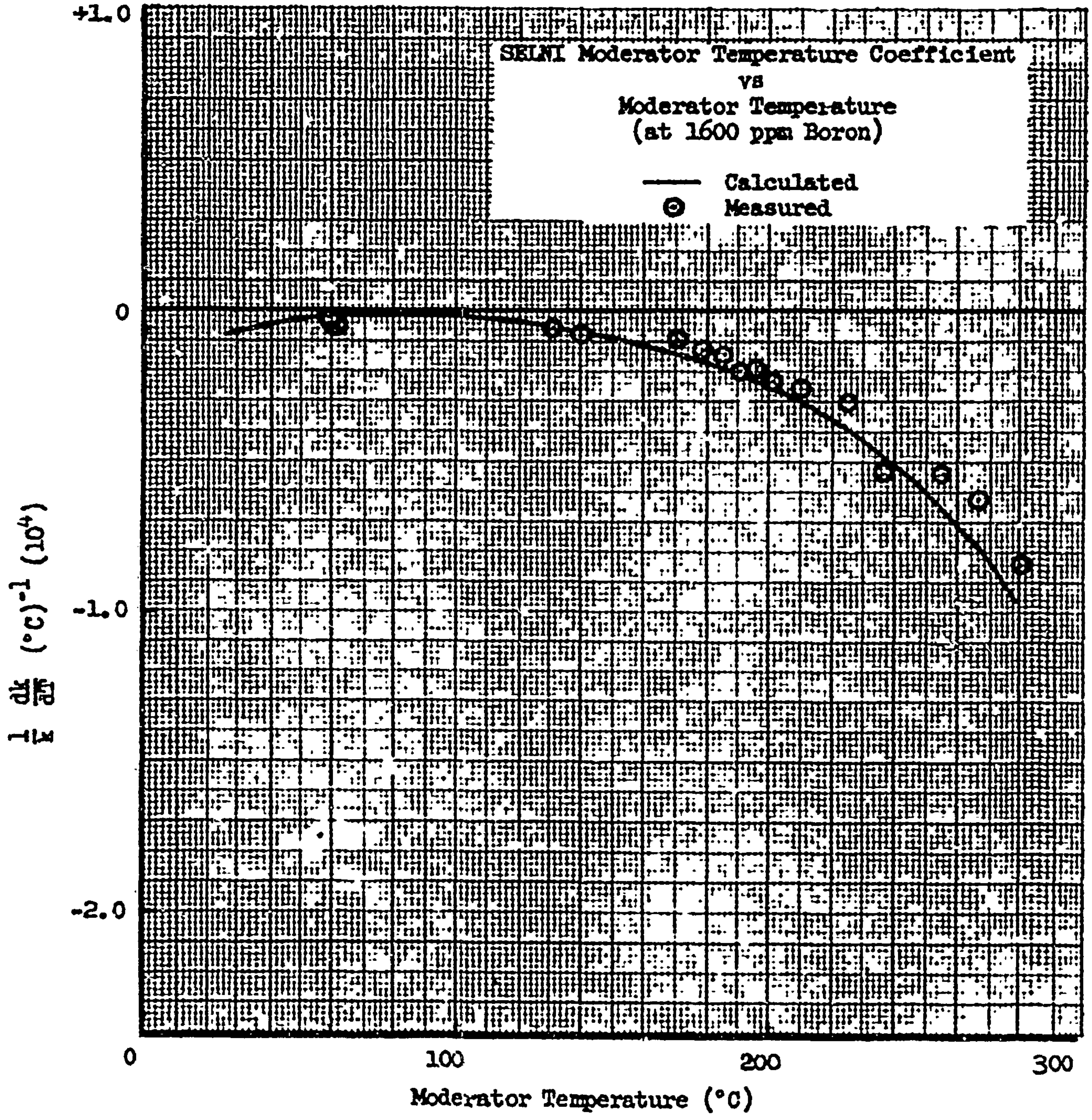
YANKEE CORE 1 BURNUP DISTRIBUTION COMPARISON
FIG. 3.2.1-25



FAST ABSORPTION OF CLUSTERED ABSORBERS
FIG. 3.2.1-26

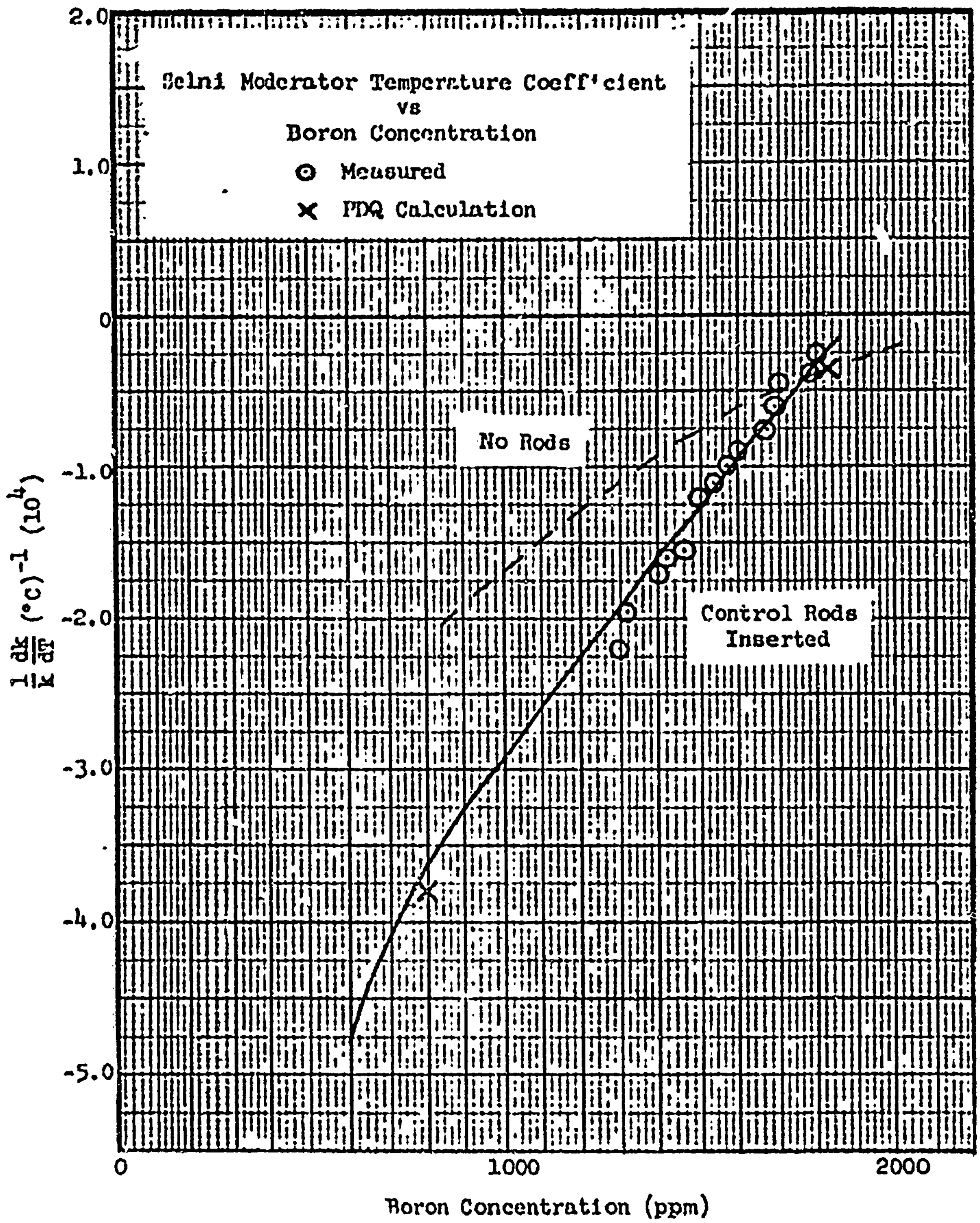


FAST ABSORPTION OF UNIFORMLY DISTRIBUTED ABSORBERS
FIG. 3.2.1-27



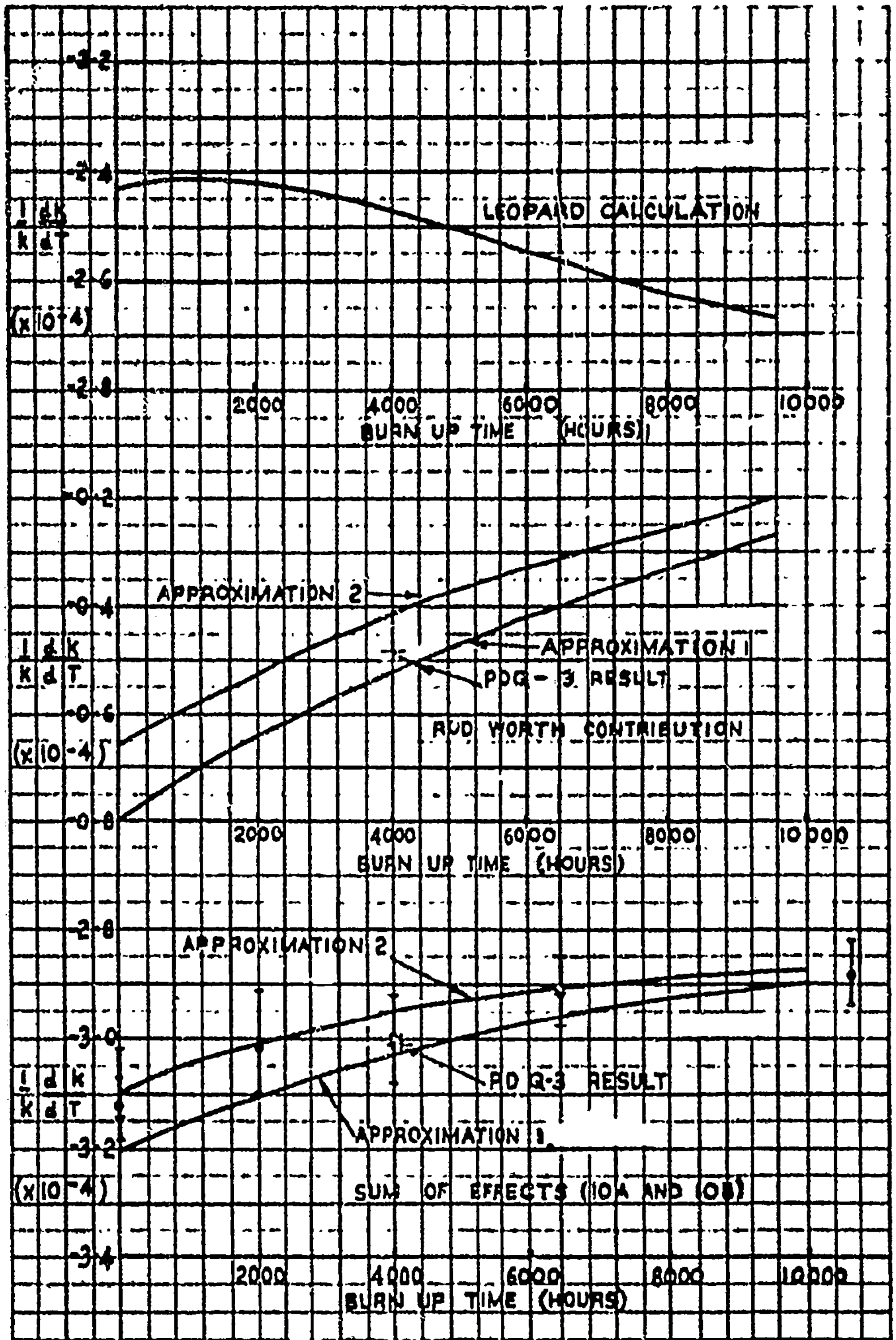
SELNI TEMPERATURE COEFFICIENT Vs. MODERATOR TEMPERATURE
(1600 PPM BORON)

FIG. 3.2.1-28



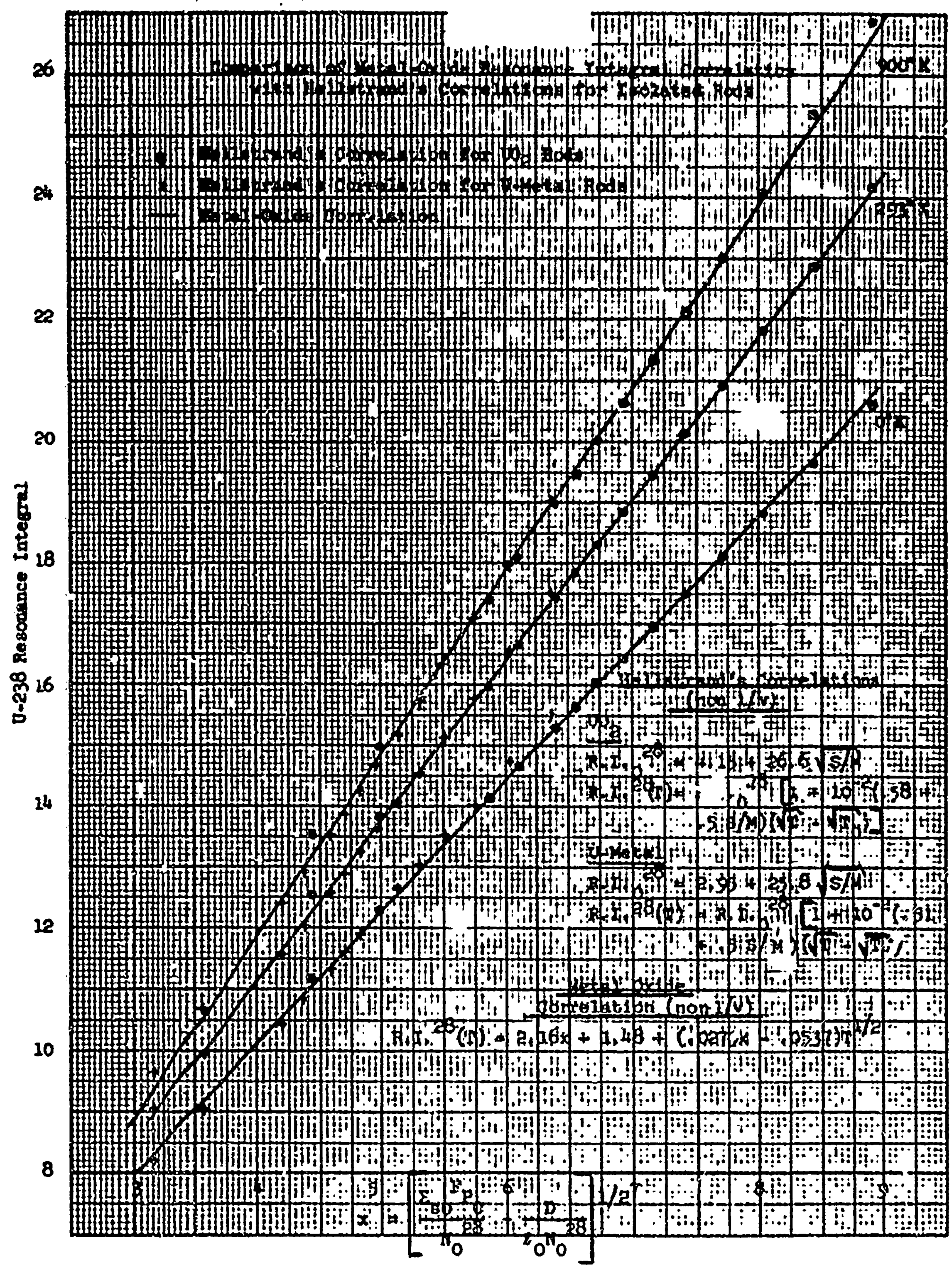
MODERATOR TEMPERATURE COEFFICIENT Vs. BORON CONCENTRATION

FIG. 3.2.1-29

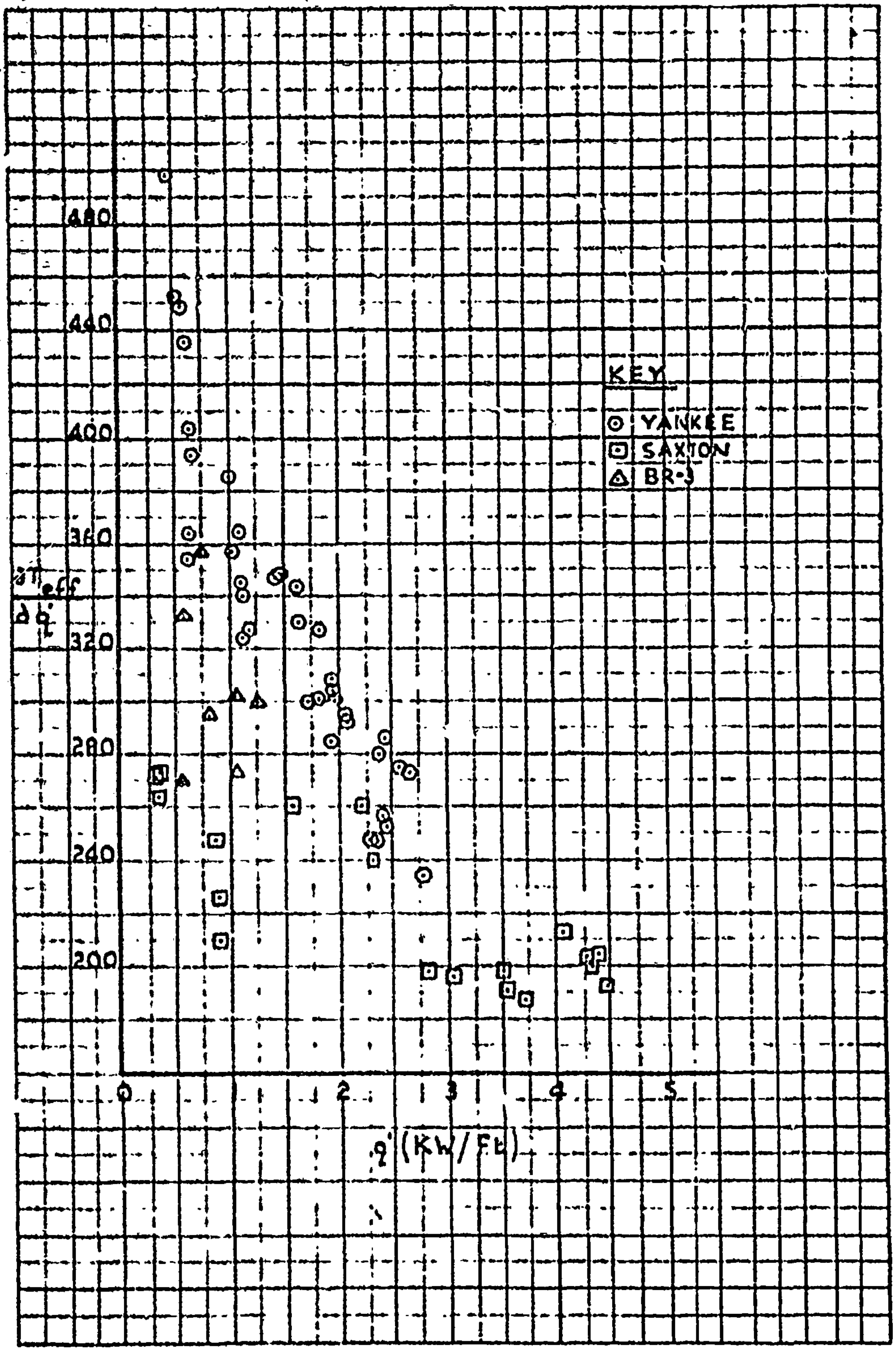


COMPARISON OF CALCULATED AND MEASURED MODERATOR TEMPERATURE COEFFICIENT Vs. BURNUP

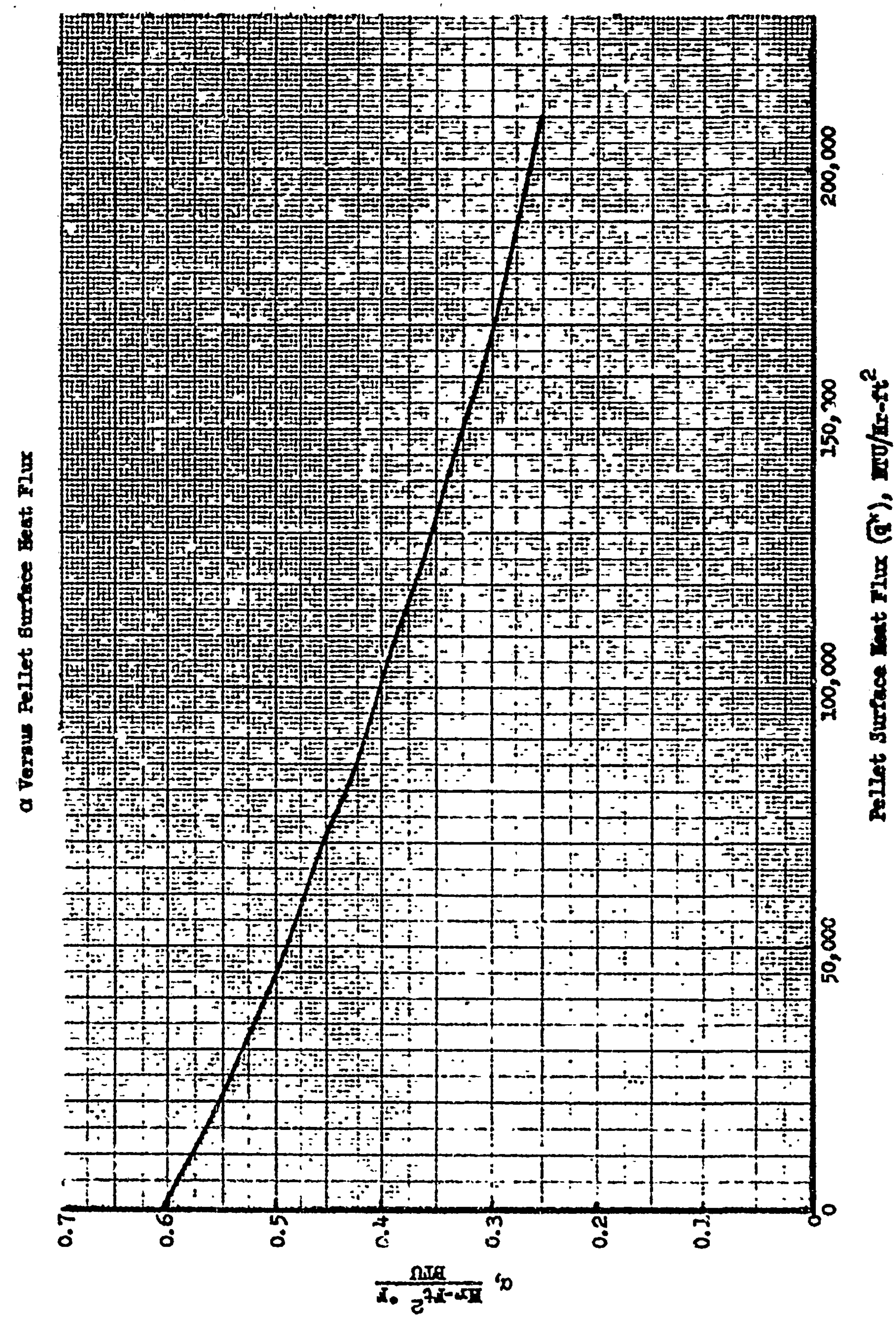
FIG. 3.2.1-30



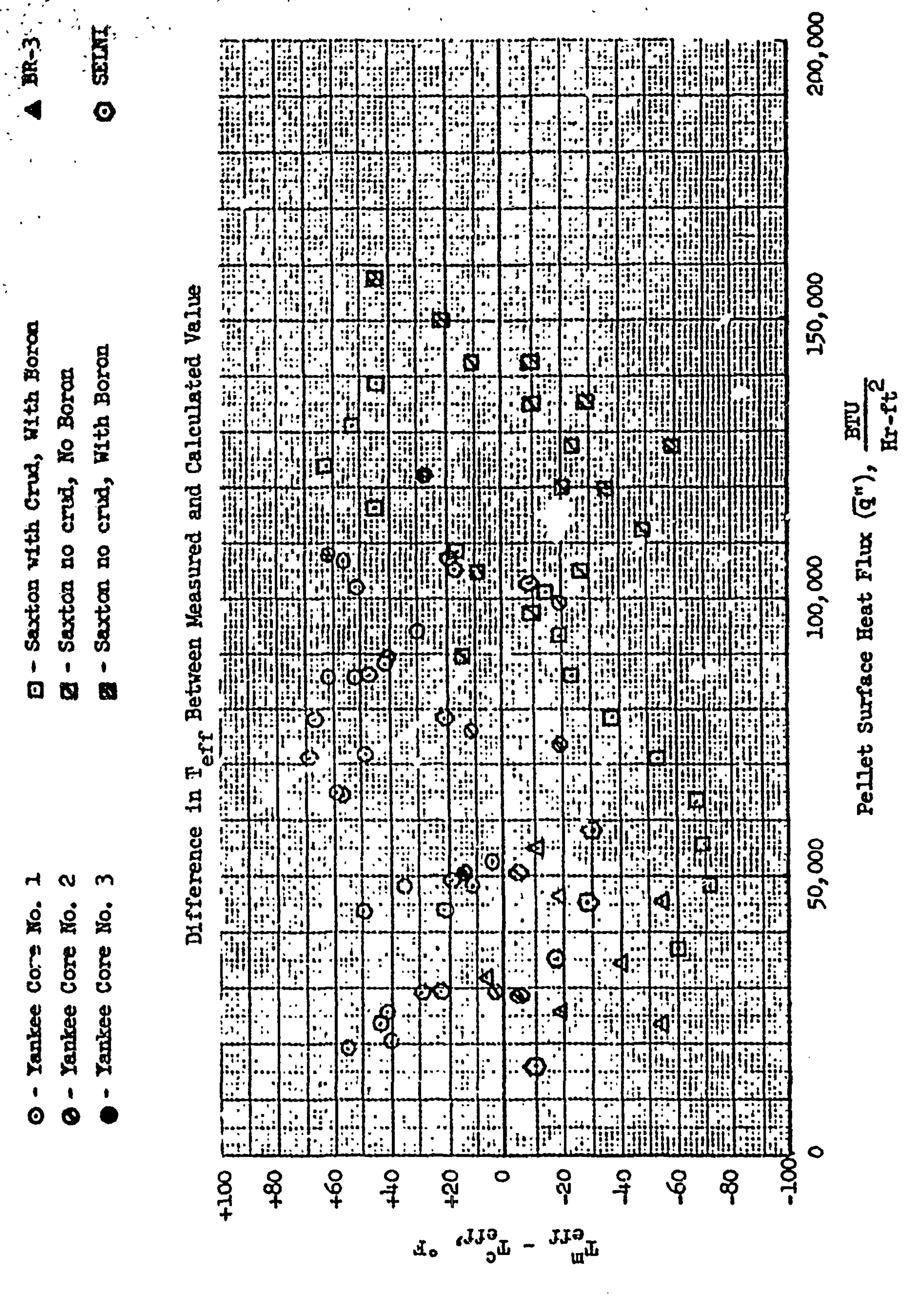
COMPARISON OF RESONANCE INTEGRAL CORRELATIONS
FIG. 3.2.1-31



FUEL TEMPERATURE CHANGES Vs. POWER DENSITY
FIG. 3.2.1-32



ALPHA Vs. HEAT FLUX
FIG. 3.2.1-33



COMPARISONS OF EFFECTIVE FUEL TEMPERATURE WITH CHANGING HEAT FLUX

FIG. 3.2.1-34

3.2.2 THERMAL AND HYDRAULIC DESIGN AND EVALUATION

Thermal and Hydraulic Characteristics of the Design

Thermal Data

Central Temperature of the Hot Pellet

The temperature distribution in the pellet is mainly a function of the uranium dioxide thermal conductivity and the local power density. The absolute value of the temperature distribution is affected by the cladding temperature and the thermal condu ce of the gap between the pellet and the cladding.

The occurrence of nucleate boiling maintains maximum cladding surface temperature below about 657°F at nominal system pressure. The contact conductance between the fuel pellet and cladding is a function of the contact pressure and the composition of the gas in the gap[1][2] and may be calculated by the following equation:

$$h = 0.6\ P + \frac{k}{f(14.4 \times 10^{-6})}$$

where

h is conductance in Btu/hr ft^2 °F

P is contact pressure in psi

k is the thermal conductivity of the gas mixture in the rod

f is the correction factor for the accommodation coefficient

This calculational procedure yields a conductance of approximately 1000 Btu/hr ft^2 °F when the pellet contacts the clad with zero contact pressure and gas composition is 75% fission gas and 25% air.

The thermal conductivity of uranium dioxide was evaluated from published results of recent work at ORNL[3], Chalk River[4], and WAPD.[5][6] The design curve for thermal conductivity is given in Figure 3.2.2-1. The section of the curve at temperatures between 0°F and 3000°F is based on the data of Godfrey, et al.[3]

The section of the curve between 3000°F and 5000°F was based on two factors:

i) In-pile observations of fuel melting dictate a positive temperature coefficient for conductivity above approximately 3000°F. The temperature dependence in this range should conform to an exponential curve since this reflects the most credible physical interpretation of the high temperature conductivity increase.

ii) The area under the recommended curve is such that the integral $\int k\, dt$ is equal to approximately 97 w/cm as given by Robertson, et al[4] and Duncan.[5] This value is based upon the interpretation of fuel melt radius as determined at Hanford[7] and Chalk River.[4]

Thermal conductivity can be represented best by the following equations:

Temperature Range - $0 \leq T \leq 1650°C$

$$k = \frac{40.4}{464 + T} + 1.32 \times 10^{-4} e^{1.88 \times 10^{-3} T}$$

Temperature Range - $1650°C \leq T \leq 2800°C$

$$k = 0.019 + 1.32 \times 10^{-4} e^{1.88 \times 10^{-3} T}$$

with k in w/cm°C for 95 per cent dense UO_2 and T in °C

Based upon the above considerations, the maximum central temperature of the hot pellet at steady state is shown in Table 3.2.2-1. This temperature is well below the melting temperatures of the irradiated UO_2 which is assumed to be about 4800°F.[8] The maximum central temperature of the hot pellet during an overpower of 112% is also shown in Table 3.2.2-1.

Westinghouse Experience With High Power Fuel Rods

Westinghouse experience with fuel rods operating at high power ratings has been summarized in Appendix A, Indian Point No. 2 Preliminary Safety Analysis Report (Docket 50-247) and in Appendix-Section IX of the Preliminary Safeguards Report for the Saxton Reactor Operating at 35 MWt (Docket 50-146). These reports present considerable statistical evidence of successful operation of high performance Zircaloy clad fuel rods in CVTR (1368 rods) and Shippingport Core I Blanket (94,920 rods). Since the date of these reports, a significant amount of additional information has been developed relating to the integrity of free standing Zircaloy clad oxide fuel rods at high power ratings. In addition, a comprehensive experimental program has been initiated to extend the operating experience to higher power and to higher exposures for many of these fuel rods. This information is summarized in Figure 3.2.2-2.

The figure shows that thirty Saxton Plutonium Project fuel rods have operated at a design peak power level of up to 18.5 kw/ft to a peak exposure of 27,800 MWD/MTM [Megawatt days per metric ton of metal (U = Pu)]. No failure have occurred with this fuel. The continuing irradiation of nine Saxton Plutonium Project assemblies (9 x 9 rod array) will achieve peak exposure of 37,000 MWD/MTM at design linear power ratings up to 18.5 kw/ft. In the Saxton overpower test, two selected fuel rods from the Saxton Plutonium Porject assemblies were removed after exposure of 18,000 MWD/MTM and inserted in a subassembly for short time irradiation at a design rating of 25 kw/ft. The subassembly operated entirely satisfactory at those conditions for test period of 10 days. The Saxton Plutonium Project will be extended by irradiating approximately 250 rods to peak burnups of about 45,000 MWD/MTM at design linear power levels ranging from 9.5 to 23.6 kw/ft.

The figure also shows that twenty-two Carolina - Virginia Tube Reactor (CVTR) rods have operated at design power levels up to 25 kw/ft. These are currently being examined and the results will be available by the end of 1968.

Another planned experimental program includes irradiation of four special assemblies (14 x 14 rod array) in the Zorita reactor at design linear power levels up to 20 kw/ft to an exposure of 25,000 MWD/MTU by February, 1970. This irradiation will be continued to about 50,000 MWD/MTU by May, 1971.

More detailed information about W experience with high power fuel rod bars has been provided in WCAP-7125[20].

Heat Flux Ratio and Data Correlation

Departure from Nucleate Boiling, (DNB), is predicted upon a combination of hydrodynamic and heat transfer phenomena and is affected by the local and upstream conditions including the flux distribution.

In reactor design, the heat flux associated with DNB and the location of DNB are both important. The magnitude of the local fuel rod temperature after DNB depends upon the axial location where DNB occurs. The W-3 DNB correlation [9], which has been utilized in this design, incorporates both local and system parameters in predicting the local DNB heat flux. This correlation includes the non-uniform flux effect, and the upstream effect which includes inlet enthalpy or length. The local DNB heat flux ratio (defined as the ratio of the DNB heat flux to the local heat flux) is indicative of the contingency available in the local heat flux without reaching DNB.

Objective of the W-3 DNB Correlation

The W-3 DNB correlation[9] has been developed to predict the DNB flux and the location of DNB equally well for uniform and an axially non-uniform heat flux distribution. This correlation replaces the preceding WAPD q" and AH DNB correlations published in the Nucleonics [10] May 1963, in order to eliminate the discontinuity of the latter at the saturation temperature and to provide a single unambiguous criterion of the design margin.

The sources of the data used in developing this correlation are:

WAPD-188	(1958)	CU-TR-No. 1 (NW-208)	(1964)
ASME Paper 62-WA-297	(1962)	CISE-R-90	(1964)
CISE-R-63	(1962)	DP-895	(1964)
ANL-6675	(1962)	AEEW-R-356	(1964)
GEAP-3766	(1962)	BAW-3238-7	(1965)
AEEW-R-213 and 309	(1963)	AE-RTL-778	(1965)
CISE-R-74	(1963)	AEEW-355	(1965)
CU-MPR-XIII	(1963)	EUR-2490.e	(1965)

The comparison of the measured to predicted DNB flux of this correlation is given in Figure 3.2.2-3. The local flux DNB ratio versus the probability of not reaching DNB is plotted in Figure 3.2.2-4. This plot indicates that with a DNBR of 1.3 the probability of not reaching DNB is 95% at a 95% confidence level.

Recent rod bundle data without mixing vanes agree very well with the predicted DNB flux as shown in Figure 3.2.2-5, and rod bundle data with mixing vanes (Figure 3.2.2-6) show on the average an 8% higher value of DNB heat flux than predicted by the W-3 DNB correlation.

It should be emphasized that the inlet subcooling effect of the W-3 correlation was obtained from both uniform and non-uniform data. The existence of an inlet subcooling effect has been demonstrated to be real and hence the actual subcooling should be used in the calculations. The W-3 correlation was developed from tests with flow in tubes and rectangular channels. Good agreement is obtained when the correlation is applied to test data for rod bundles.

Local Non-Uniform DNB Flux

The W-3 correlation gives the equivalent uniform DNB heat flux, $q''_{DNB,EU}$, for a given set of system and local conditions. The heat distribution upstream of the DNB point affects the value of the DNB flux. This influence is accounted for by the F-factor.[9] The non-uniform DNB heat flux, $q''_{DNB,N}$, is given by

$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F} \qquad (1)$$

Definition of DNB Ratio (DNBR)

The DNB heat flux ratio is defined as

$$DNBR = \frac{q''_{DNB,N}}{q''_{loc}} = \frac{q''_{DNB,EU}}{(F)(q''_{loc})} \qquad (2)$$

where q''_{loc} is the actual local heat flux.

The F-factor may be considered as a hot spot factor, applicable to DNB, due to the axial heat flux distribution. An alternate, although improper, DNB ratio could be defined as $\frac{q''_{DNB,EU}}{q''_{loc}}$ instead of $\frac{q''_{DNB,EU}}{(F)(q''_{loc})}$.

Since the F-factor at the minimum DNBR location is generally greater than unity, this alternate DNBR would be greater than the proper DNBR as defined by equation (2). Because this alternate DNBR does not consider the effects of the non-uniform flux distribution, it does not give the correct physical meaning to DNB and is therefore not used in the evaluation of DNB ratios.

Procedure for Using W-3 Correlation

In predicting the local DNB flux in a non-uniform heat flux channel, the following two steps are required:

i) The uniform DNB heat flux, $q''_{DNB,EU}$, is computed with the W-3 correlations using the specified local reactor conditions.

ii) This equivalent uniform heat flux is converted into corresponding non-uniform DNB heat flux, $q''_{DNB,N}$, for the non-uniform flux distribution in the reactor. This is accomplished by dividing the uniform DNB flux by the F-factor.[9] Since F is generally greater than unity $q''_{DNB,N}$ will be smaller than $q''_{DNB,EU}$.

To calculate the DNBR of a reactor channel, the values of $\frac{q''_{DNB,N}}{q''_{loc}}$ along the channel are evaluated and the minimum value is selected as the minimum DNBR incurred in that channel.

The W-3 correlation depends on both local and inlet enthalpies of the actual system fluid, and the upstream conditions are accommodated by the F-factor. Hence, the correlation provides a realistic evaluation of the safety margin on heat flux.

Film Boiling Heat Transfer Coefficient

Heat transfer after departure from nucleate boiling is conservatively assumed to be limited by film boiling immediately, and the period of transition boiling is neglected.

The correlation used to evaluate these film boiling heat transfer coefficients was developed by Tong, Sandberg and Bishop[11] and is shown in Figure 3.2.2-7.

$$\left(\frac{hD}{k}\right)_f = 0.0193 \left(\frac{DG}{\mu}\right)_f^{0.80} \left(\frac{C_p \mu}{k}\right)_f^{1.23} \left(\frac{\rho_g}{\rho_b}\right)^{0.68} \left(\frac{\rho_g}{\rho_\ell}\right)^{0.068}$$

where $\rho_b = \rho_g \alpha + \rho_\ell (1 - \alpha)$

and

C_p = heat capacity at constant pressure (Btu/lb-°F)

D = equivalent diameter of flow channel (ft)

h = heat transfer coefficient (Btu/hr-ft^2-°F)

G = mass flow rate (lb/hr-ft^2)

k = thermal conductivity (Btu/hr-ft-°F)

α = void fraction

ρ = density (lbs/ft^3)

μ = viscosity (lbs/ft-hr)

Subscripts:

g - Evaluation of the property at the saturated vapor condition

ℓ - Evaluation of the property at the saturated liquid condition

f - Evaluation of the property at the average film temperature

w - Evaluation of the property at the wall temperature

b - Evaluation of the property at the average bulk-fluid condition

The heat transfer correlation was developed for flow rates equal or greater than 0.8×10^6 lb/hr sq ft over a pressure range of 580 to 3190 psia, for qualities as high as 100 percent and heat flux from 0.1 to 0.65×10^6 Btu/hr sq ft.

Hot Channel Factors

The total hot channel factors for heat flux and enthalpy rise are defined as the maximum-to-core average ratios of these quantities. The heat flux factors consider the local maximum at a point (the "hot spot"), and the enthalpy rise factors involve the maximum integrated value along a channel (the "hot channel").

Definition of Engineering Hot Channel Factor:

Each of the total hot channel factors is the product of a nuclear hot channel factor describing the neutron flux distribution and an engineering hot channel factor to allow for variations from design conditions. The engineering hot channel factors account for the effects of flow conditions and fabrication tolerances and are made up of subfactors accounting for the influence of the variations of fuel pellet diameter, density and enrichment; fuel rod diameter; pitch and bowing; inlet flow distribution; flow redistribution; and flow mixing.

The enthalpy rise engineering hot channel factors are evaluated using the THINC codes[12]. These factors which are obtained by the THINC analysis will vary with the operating conditions. For this plant the engineering hot channel factors are 1.03 for F_q^E and an average value of 1.01 for $F_{\Delta H}^E$. The subfactors used in obtaining these values are described in the following paragraphs.

Heat Flux Engineering Subfactor, F_q^E

This subfactor, determined by statistically combining the tolerances for the fuel diameter, density, enrichment and the fuel rod diameter, pitch and bowing is 1.03. Measured manufacturing data from the first three Yankee cores, the SELNI core and Indian Point Core B show this factor is conservative in comparison to the value obtained for the probability limit of three standard deviations. Thus, it is expected that a statistical sampling of the fuel assemblies of this plant will also show this subfactor is conservative.

Enthalpy Rise Engineering Subfactor, $F_{\Delta H}^E$

Pellet Diameter, Density Enrichment and Fuel Rod Diameter, Pitch and Bowing:

Based on the applicable tolerances and consistent with the probability limit of three standard deviations for the measured Yankee, SELNI, SENA, SCE, Connecticut Yankee and Indian Point data, a value of 1.08 was selected for this subfactor.

Inlet Flow Maldistribution:

Studies performed on 1/7 scale hydraulic reactor models indicate that a conservative design basis is to consider a 5% reduction in the flow to the hot fuel assembly under isothermal conditions. This inlet flow reduction in the THINC analysis results in an increase of 1% in the hot channel enthalpy rise.

Flow Redistribution:

The flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow resistance in the channel due to the local or bulk boiling. A nominal value for this hot channel subfactor when evaluated by the THINC codes is 1.03, but in practice during the thermal and hydraulic analysis, the actual value for each case is calculated individually.

Flow Mixing:

Mixing vanes have been incorporated into the spacer grid design. These vanes induce flow mixing between the various flow channels in a fuel assembly and also between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavorable mechanical tolerances.

In the THINC analysis, the benefit of coolant mixing in all the subchannels in the hot assembly is considered and a mixing factor of approximately 0.90 is used to evaluate the enthalpy rise to the point of minimum DNB ratio.

The above subfactors are combined to obtain the total engineering hot channel factor for enthalpy rise of 1.01. Table 3.2.2-2 is a tabulation of the design engineering hot channel factors.

Pressure Drop and Hydraulic Forces

The total pressure loss across the reactor vessel, including the inlet and outlet nozzles, and the pressure drop across the core are listed in Table 3.2.2-1. These values include a 10% uncertainty factor.

The hydraulic forces are not sufficient to lift a rod control cluster during normal operation even if the rod cluster is detached from its coupling.

Thermal and Hydraulic Design Parameters

The thermal and hydraulic design parameters are given in Table 3.2.2-1.

Thermal and Hydraulic Evaluation

W-3 Equivalent Uniform Flux DNB Correlation

The equivalent uniform DNB flux $q'_{DNB,\ EU}$ is calculated from the W-3 equivalent uniform flux DNB correlation as follows:

$$\frac{q''_{DNB,\ EU}}{10^6} = [(2.022 - 0.0004302p) + (0.1722 - 0.0000984p)e^{(18.177 - 0.004129p)\chi}]$$

$$\times [1.037 + \frac{G}{10^6}(0.1484 - 1.596\ \chi + 0.1729\ \ \chi/\chi/)] \times [1.157 - \ \ 869\chi]$$

$$\times [0.2664 + 0.8357e^{-3.151D_e}] \times [0.8258 + 0.0007 \ldots (H_s \ \ \ \ _{in})] \quad (3)$$

The heat flux is in Btu/hr ft^2 and the units of the parameters are as listed below. The ranges of parameters of the data used in developing this correlation are:

System pressure, p = 1000 to 2300 psia
Mass velocity, G = 1.0×10^6 to 5.0×10^6 lb/hr ft^2
Equivalent diameter, D_e = 0.2 to 0.7 inches
Quality, X_{loc} = - 0.15 to + 0.15
Inlet enthalpy, $H_{in} \geq$ 400 Btu/lb
Length, L = 10 to 144 inches
$\dfrac{\text{Heated perimeter}}{\text{Wetted perimeter}}$ = 0.88 to 1.00

Geometries - circular tube, rectangular channel and rod bundles
Flux = Uniform and equivalent uniform flux converted from non-uniform data by using F-factor of Reference (9).

<u>Local Non-Uniform DNB Flux</u>

The local non-uniform $q''_{DNB,N}$ is calculated as follows:

$$q''_{DNB,N} = q''_{DNB,EU}/F$$

where

$$F = \frac{C}{q''_{\text{local at } \ell_{DNB}} \times (1-e^{-C\ell_{DNB}})} \int_0^{\ell_{DNB}} q''(z) e^{-C(\ell_{DNB} - z)} dz \quad (5)$$

$$C = 0.44 \frac{(1 - X_{DNB})^{7.9}}{(G/10^6)^{1.72}} \text{ inch}^{-1}$$

ℓ_{DNB} = distance from the inception of local boiling to the point of DNB.

z = distance from the inception of local boiling, measured in the direction of flow.

In determining the F-factor, the value of q''_{local} at ℓ_{DNB} in equation (5) was measured as $z = \ell_{DNB}$, the location where the DNB flux is calculated. For a uniform flux, F becomes unity so that $q''_{DNB,N}$ reduces to $q''_{DNB,EU}$ as expected. The comparisons of predictions by using W-3 correlations and the non-uniform DNB data obtained by B&W[13], Winfrith[14] and Fiat are given in Figure 3.2.2-8 and Figure 3.2.2-9. The criterion for determining the predicted location of DNB is to evaluate the ratio of the predicted DNB flux to the local heat flux along the length of the channel. The location of the minimum DNB ratio is considered to be location of DNB.

Application of the W-3 Correlation in Design

During steady state operation at the nominal design conditions, the DNB ratios are determined. Under other operating conditions, particularly overpower transients, more limiting conditions develop than those existing during steady state operation. The DNB correlations are sensitive to several parameters. In addition, thermal flux generated under transient conditions is also sensitive to many parameters. Therefore, for each case studied, a conservative combination of the significant parameters is used as an initial condition. These parameters include:

a) Reactor coolant system pressure
b) Reactor coolant system temperature
c) Reactor power (determined from secondary plant calorimetrics)
d) Core power distribution (hot channel factors).

For transient accident conditions where the power level, system pressure and core temperature may increase, the DNBR is limited to a minimum value of 1.30. The Reactor Control and Protection System is designed to prevent any credible combination of conditions from occurring which would result in a lower DNB ratio.

DNB Evaluation

A preliminary evaluation is made to predict the statistical number of fuel rods in the core that might reach DNB, both under normal operating conditions and under assumed overpower conditions. For this calculation, a convolution procedure is utilized in which the product of the number of fuel rods experiencing a given DNB ratio and the probability of reaching that DNB ratio is summed over the entire core.

Two cases were investigated using this method; one in which the nominal conditions of Table 3.2.2-1 were used and a second in which the coolant parameters were adjusted to give a minimum DNB ratio of 1.30 at 112% power. The results obtained for these cases are shown in Table 3.2.2-3.

Table 3.2.2-4 summarizes the results of a sensitivity study to show the effect of major parameters on the statistical number of fuel rods which may experience DNB, using the design axial power distribution and the design and best estimate radial power distributions as shown in Figure 3.2.2-10.

Effects of DNB on Neighboring Rods

Westinghouse has never observed DNB to occur in a group of neighboring rods in a rod bundle as a result of DNB in one rod in the bundle.

DNB With Physical Burnout

Westinghouse[21] has conducted DNB tests in a 25-rod bundle where physical burnout occurred with one rod. After this occurrence, the 25 rod test section was used for several days to obtain more DNB data from the other rods in the bundle. The burnout and deformation of the rod did not affect the performance of neighboring rods in the test section during the burnout or the validity of the subsequent DNB data points as predicted by the W-3 correlation. No occurrences of flow instability or other abnormal operation were observed.

DNB With Return to Nucleate Boiling

Additional DNB tests have been conducted by Westinghouse[22] in 19 and 21 rod bundles. In these tests, DNB without physical burnout was experienced more than once on single rods in the bundles for short periods of time. Each time, a reduction in power of approximately 10% was sufficient to re-establish nucleate boiling on the surface of the rod. During these and subsequent tests, no adverse effects were observed on this rod or any other rod in the bundle as a consequence of operating in DNB.

Hydrodynamic and Flow Power Coupled Instability

The interaction of hydrodynamic and spatial effects have been considered and it is concluded that a large margin exists between the design conditions and those for which an instability is possible.

It has been known for some time that heated channels in parallel can lead to flow instability. If substantial boiling takes place, periodic flow instabilities have been observed and, as long ago as 1938, Ledinegg[15] proposed a stability criterion on the basis of which the concept of inlet orificing has been developed to stabilize flow. More recent work[16-18] has demonstrated that periodic instabilities are possible which violate the Ledinegg criterion.

In normal flow channels with little or no boiling, the type of instability proposed by Ledinegg is not possible since it results primarily from the large changes in water density along the channel due to boiling. Moreover, the periodic instabilities examined by Quandt[16,17] and Meyer[18] are not exhibited in non-boiling channels of the type found in PWR cores.[19]

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TABLE 3.2.2-1

THERMAL AND HYDRAULIC DESIGN PARAMETERS

Total Heat Output, MWt	2758
Total Heat Output, Btu/hr	9413×10^6
Heat Generated in Fuel, %	97.4
Maximum Thermal Overpower, %	112
Nominal System Pressure, psia	2250
Hot Channel Factors	
Heat Flux	
Nuclear, F_q^N	3.12
Engineering, F_q^E	1.03
Total	3.21 10
Enthalpy Rise	
Nuclear, $F_{\Delta H}^N$	1.75
Coolant Flow	
Total Flow Rate, lbs/hr	136.3×10^6
Average Velocity Along Fuel Rods, ft/sec	15.4
Average Mass Velocity, lb/hr-ft^2	2.53×10^6
Coolant Temperature, °F	
Nominal Inlet	543.0
Average Rise in Vessel	53.0
Average Rise in Core	55.5
Average in Core	571
Average in Vessel	569.5
Nominal Outlet of Hot Channel	633.5
Heat Transfer	
Active Heat Transfer Surface Area, ft^2	52,200
Average Heat Flux, Btu/hr-ft^2	175,600
Maximum Heat Flux, Btu/hr-ft^2	567,300
Maximum Thermal Output, kw/ft	18.4
Maximum Clad Surface Temperature at Nominal Pressure, °F	657
Maximum Average Clad Temperature at Rated Power, °F	716
Fuel Central Temperatures for nominal fuel rod dimensions, °F	
Maximum at 100% Power	4090
Maximum at 112% Power	4380
DNB Ratio	
Minimum DNB Ratio at nominal operating conditions	2.00
Pressure Drop, psi	
Across Core	31.5
Across Vessel, including nozzles	50.0

TABLE 3.2.2-2

ENGINEERING HOT CHANNEL FACTORS

F_q^E	Pellet Diameter, Density Enrichment, and Eccentricity Rod Diameter, (Pitch and Bowing)	1.03
$F_{\Delta H}^E$	Pellet Diameter, Density, Enrichment Rod Diameter, Pitch and Bowing	1.08
	Inlet Flow Maldistribution	1.01
	Flow Redistribution	1.03
	Flow Mixing	0.90 *
	Resulting $F_{\Delta H}^E$	1.01

* To point of Minimum DNB ratio

TABLE 3.2.2-3

STATISTICAL NUMBER OF RODS THAT COULD EXPERIENCE DNB
USING THE W-3 DNB CORRELATION

% of Peak Power	% Core Area	DNB Ratio In Area	Statistical Number* of Rods in Area That Could Experience DNB
Rated Power (Design Power Distribution - Peak Radial Factor = 1.75)			
100-95	1.6	2.14	.040
95-90	3.4	2.29	.035
90-85	4.0	2.46	.016
85-80	5.3	2.64	.008
80-75	7.7	2.85	.005
			.104 rods total
112% Overpower Condition (Design Power Distribution, T_{inlet} = 559.1, P = 2200 ps:			
100-95	1.6	1.43	9.765
95-90	3.4	1.60	5.355
90-85	4.0	1.77	1.512
85-80	5.3	1.95	0.500
80-75	7.7	2.16	0.173
75-70	7.2	2.39	0.044
70-65	7.2	2.61	0.013
65-60	7.3	2.86	0.004
			17.366 rods total

* The statistical number of rods which could experience DNB takes into account the distribution of the experimental data from which the W-3 correlation was developed and the distribution of the power in the core.

TABLE 3.2.2-4

SENSITIVITY ANALYSIS

Effect of Varying the Power Distribution

Power Distribution	Power % of Rated	Tin	Pressure	Statistical Number* of Fuel Rods Which May Experience DNB
Design	112	543	2250	0.827
Best Estimate	112	543	2250	0.065

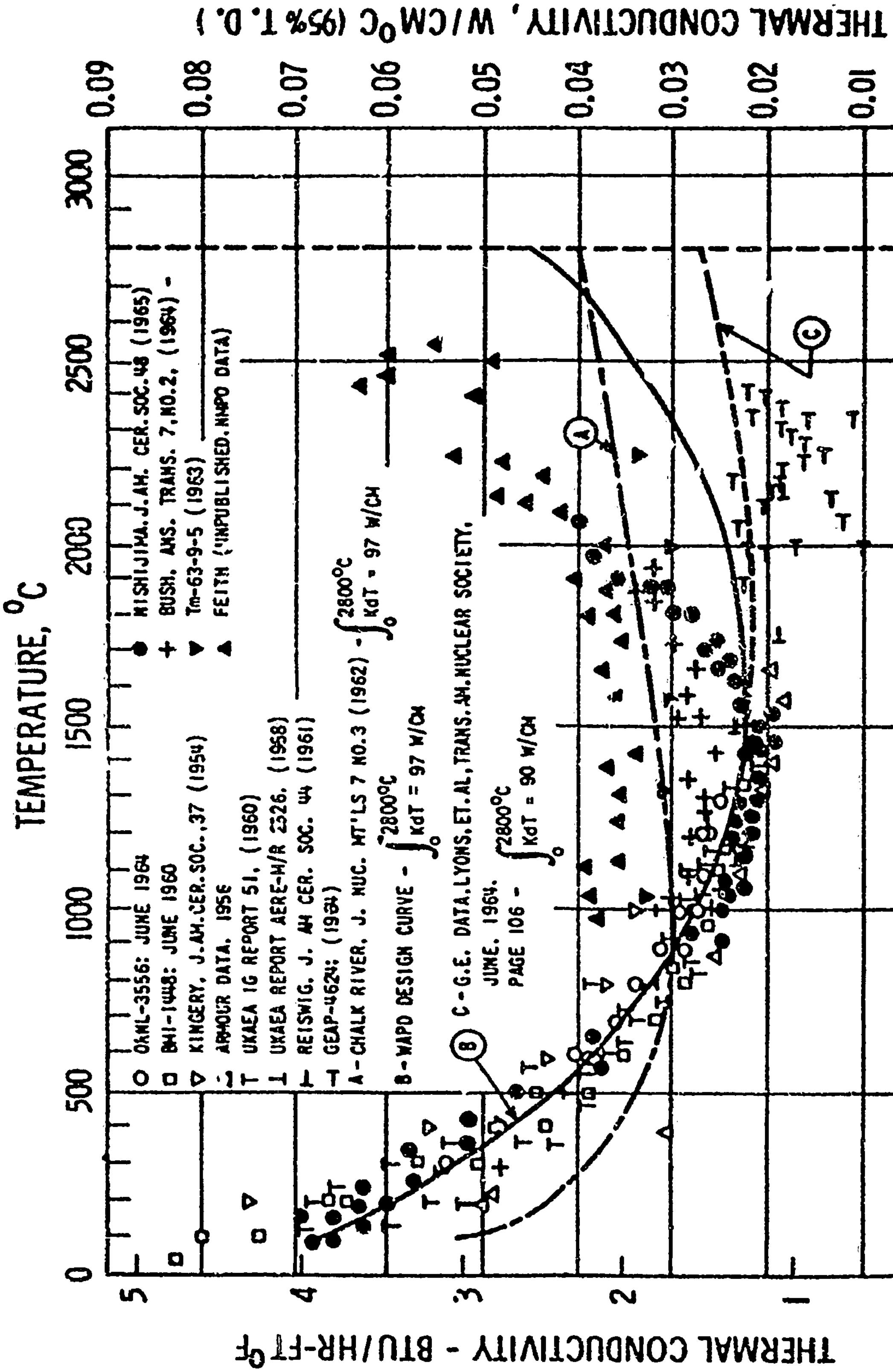
Effect of Varying Power Levels

Power Distribution	Power % of Rated	Tin	Pressure	Statistical Number* of Fuel Rods Which May Experience DNB
Best Estimate	100	543	2250	0.008
Best Estimate	112	543	2250	0.065

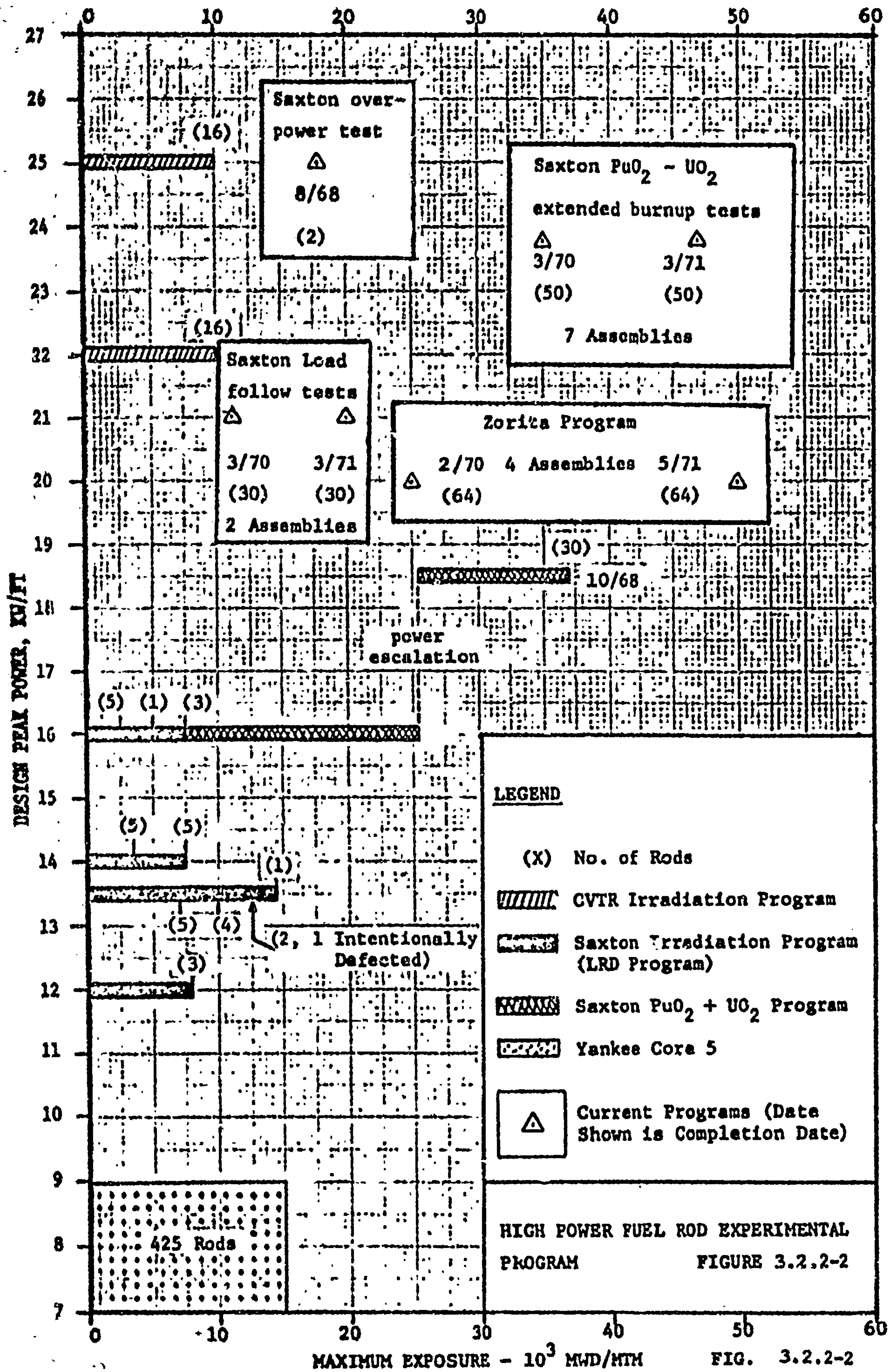
Effect of Varying Flow Rate at 112% Power

Power Distribution	Flow % of Rated	Tin	Pressure	Statistical Number* of Fuel Rods Which May Experience DNB
Best Estimate	100	543	2250	0.065
Best Estimate	95	543	2250	0.113
Best Estimate	90	543	2250	0.199

* The statistical number of rods which could experience DNB takes into account the distribution of the experimental data from which the W-3 DNB correlation was developed and the distribution of the power in the core.

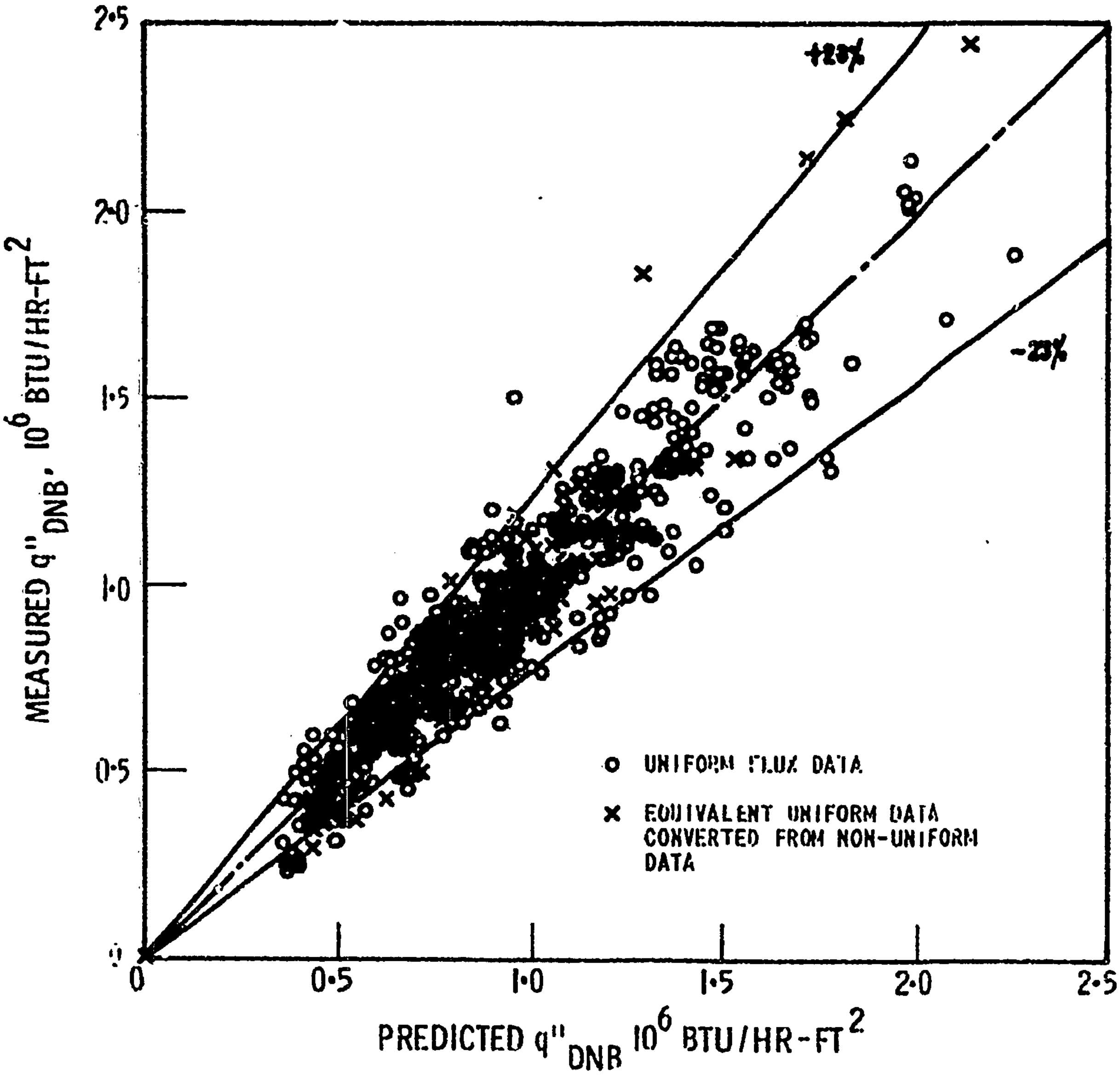


THERMAL CONDUCTIVITY OF URANIUM DIOXIDE
FIG. 3.2.2-1

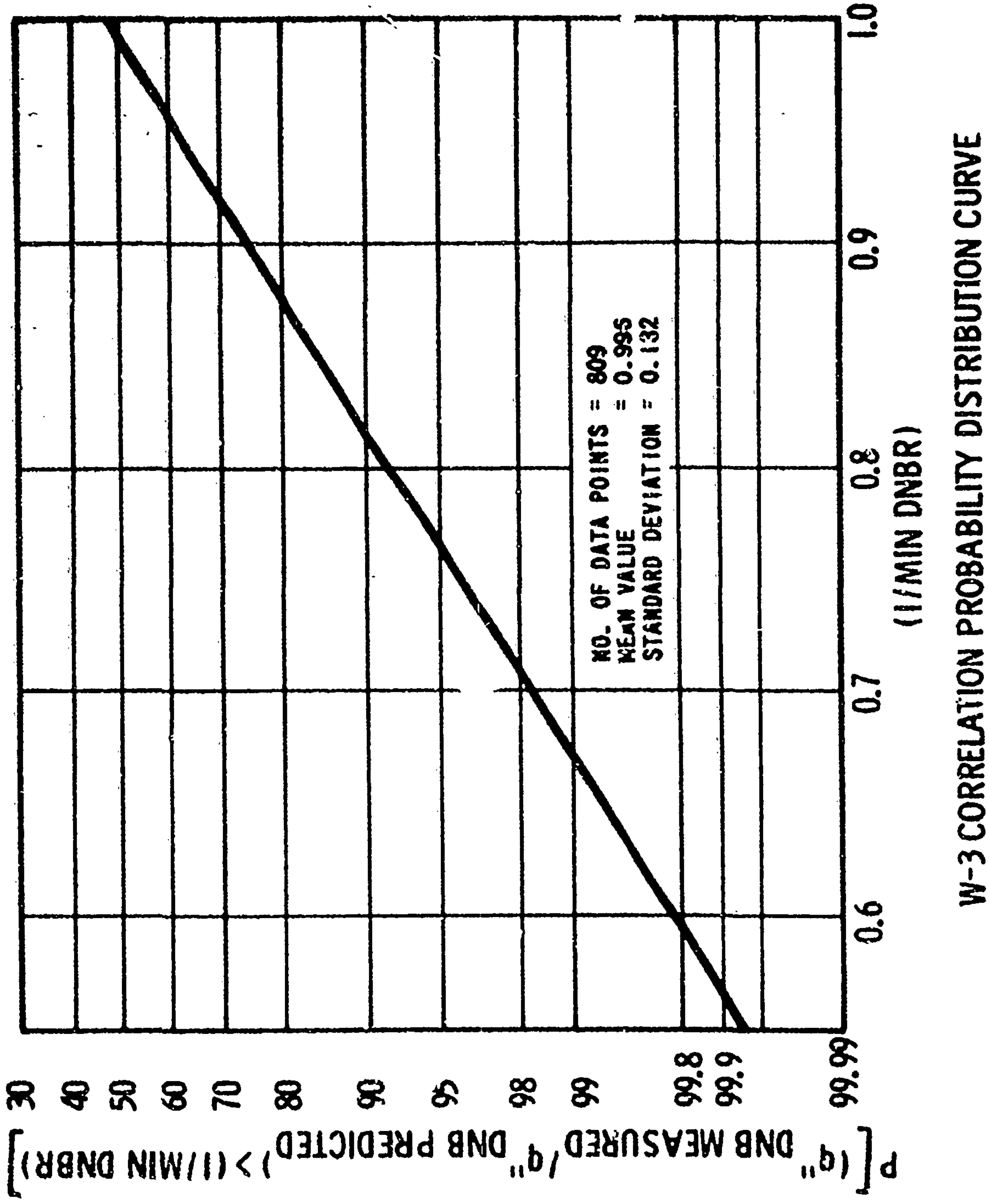


HIGH POWER FUEL ROD EXPERIMENTAL PROGRAM FIGURE 3.2.2-2

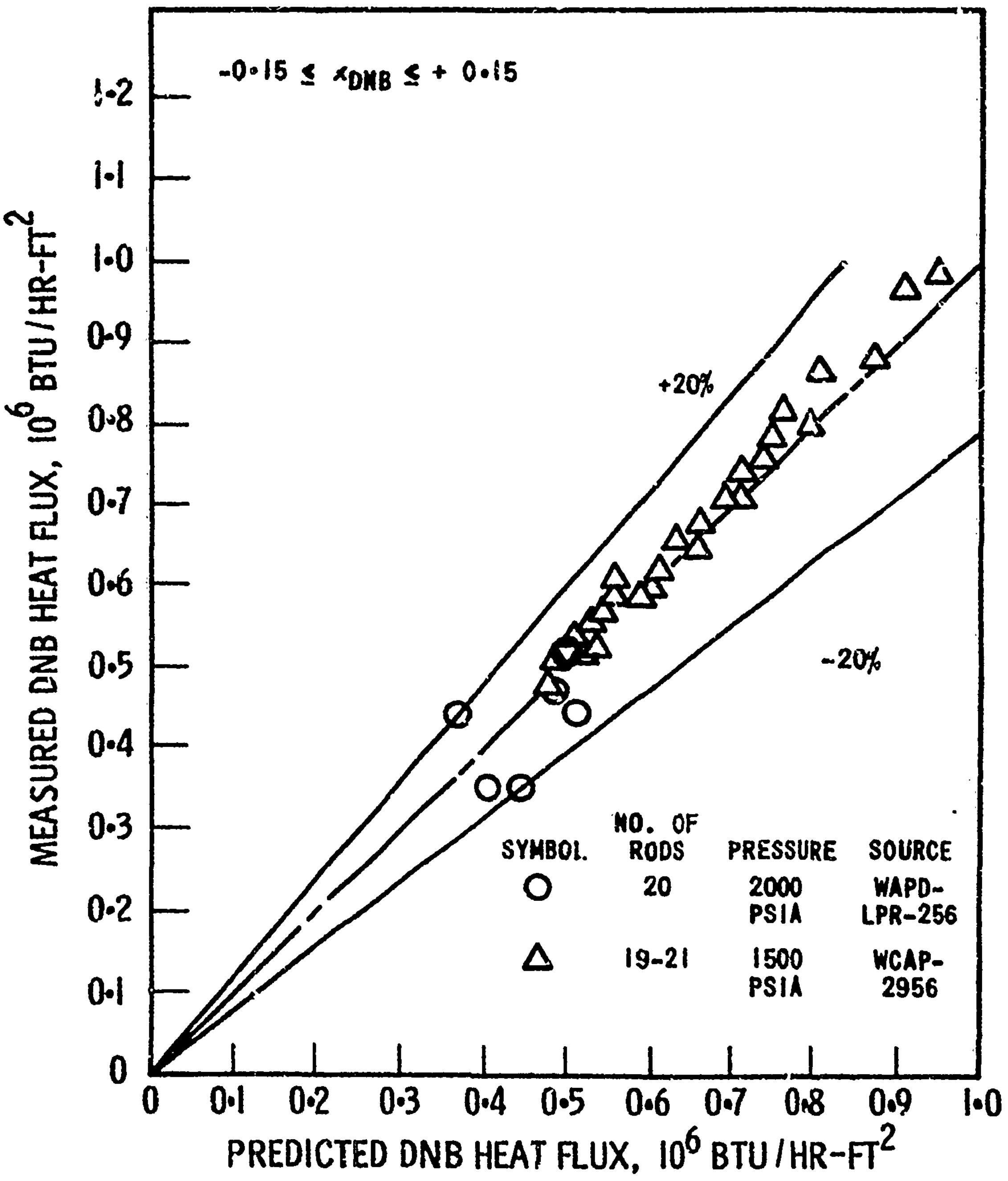
FIG. 3.2.2-2



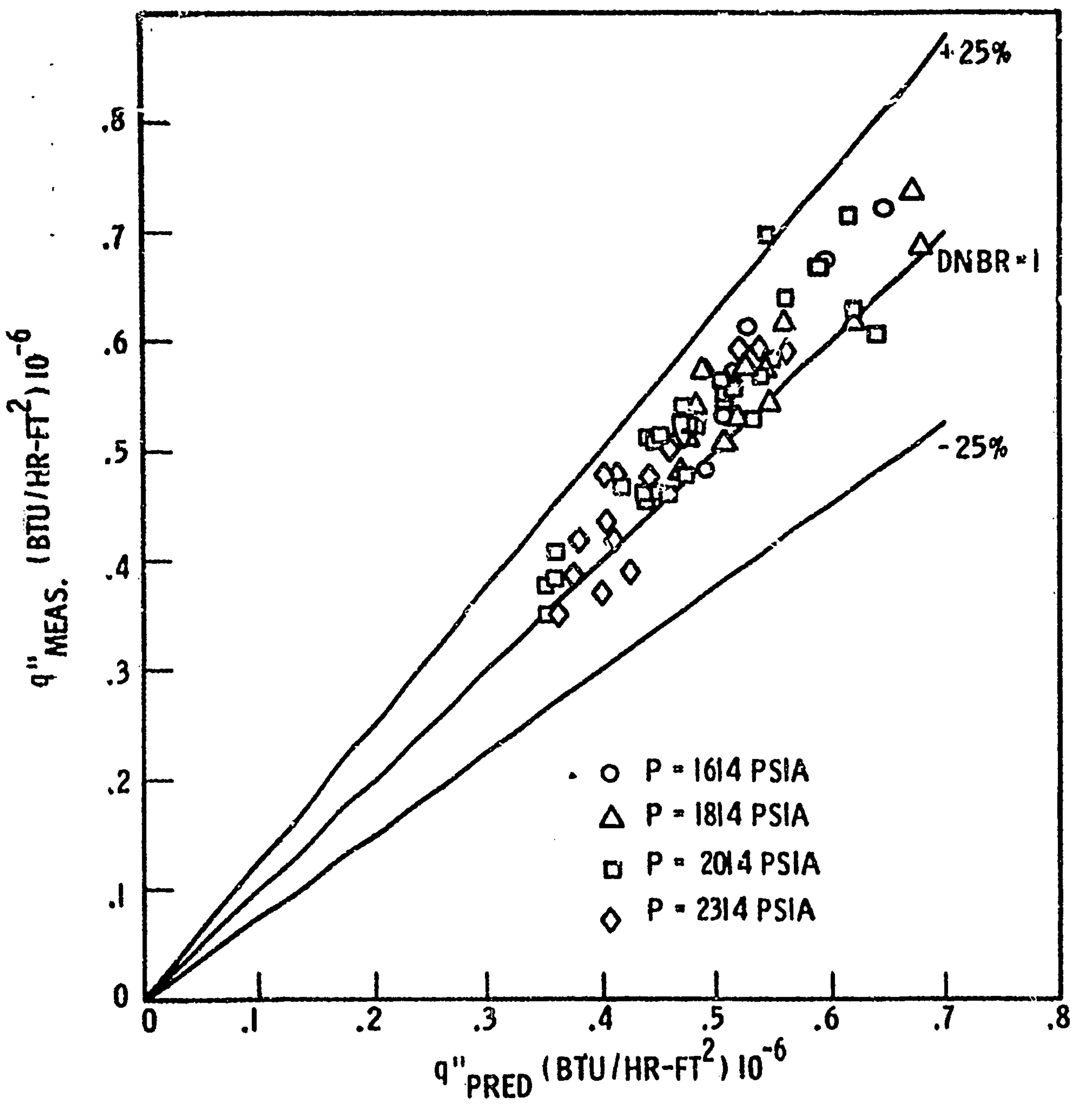
COMPARISON OF W-3 PREDICTION AND UNIFORM FLUX DATA
FIG. 3.2.2-3



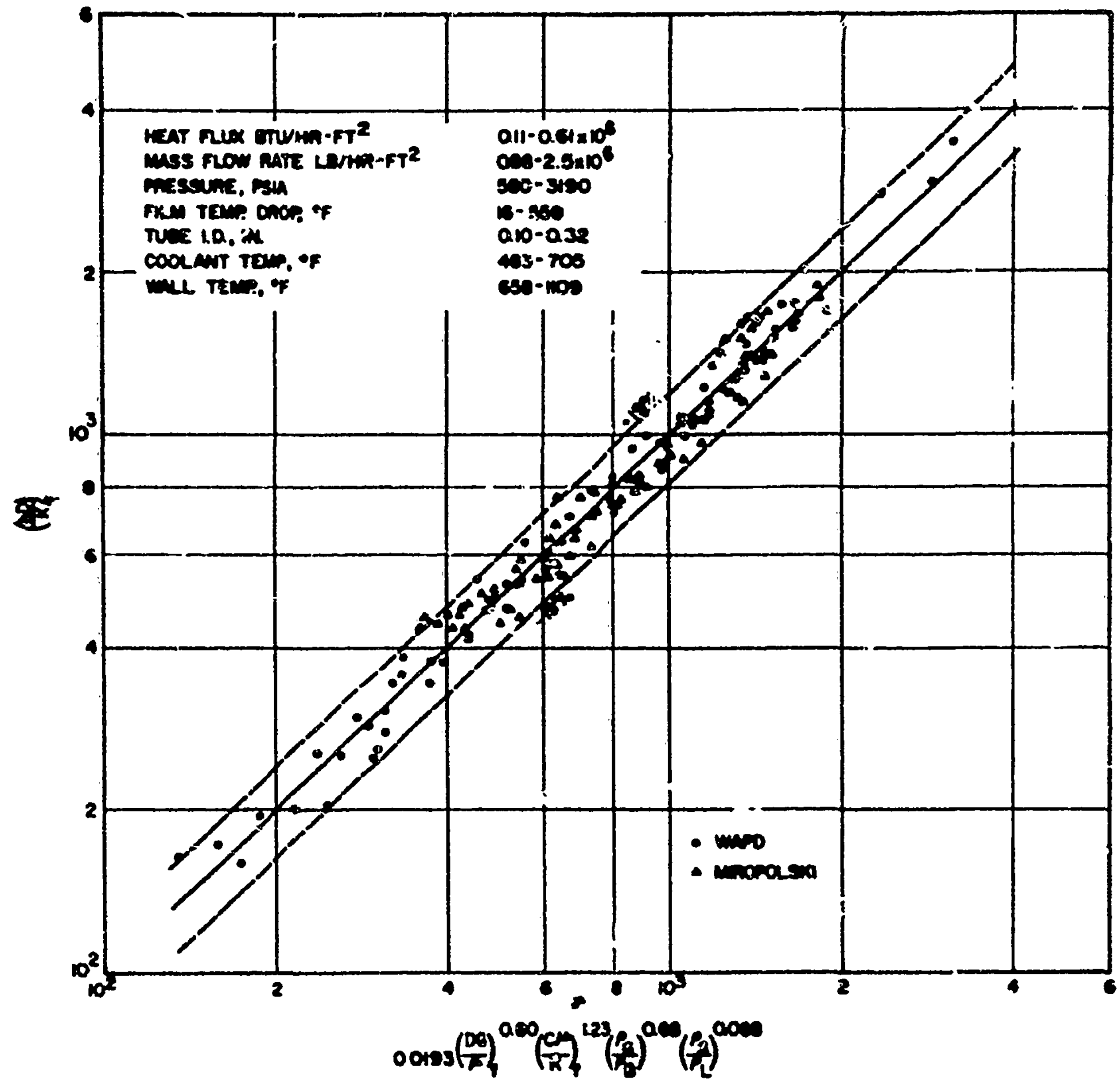
W-3 CORRELATION PROBABILITY DISTRIBUTION CURVE
FIG. 3.2.2-4



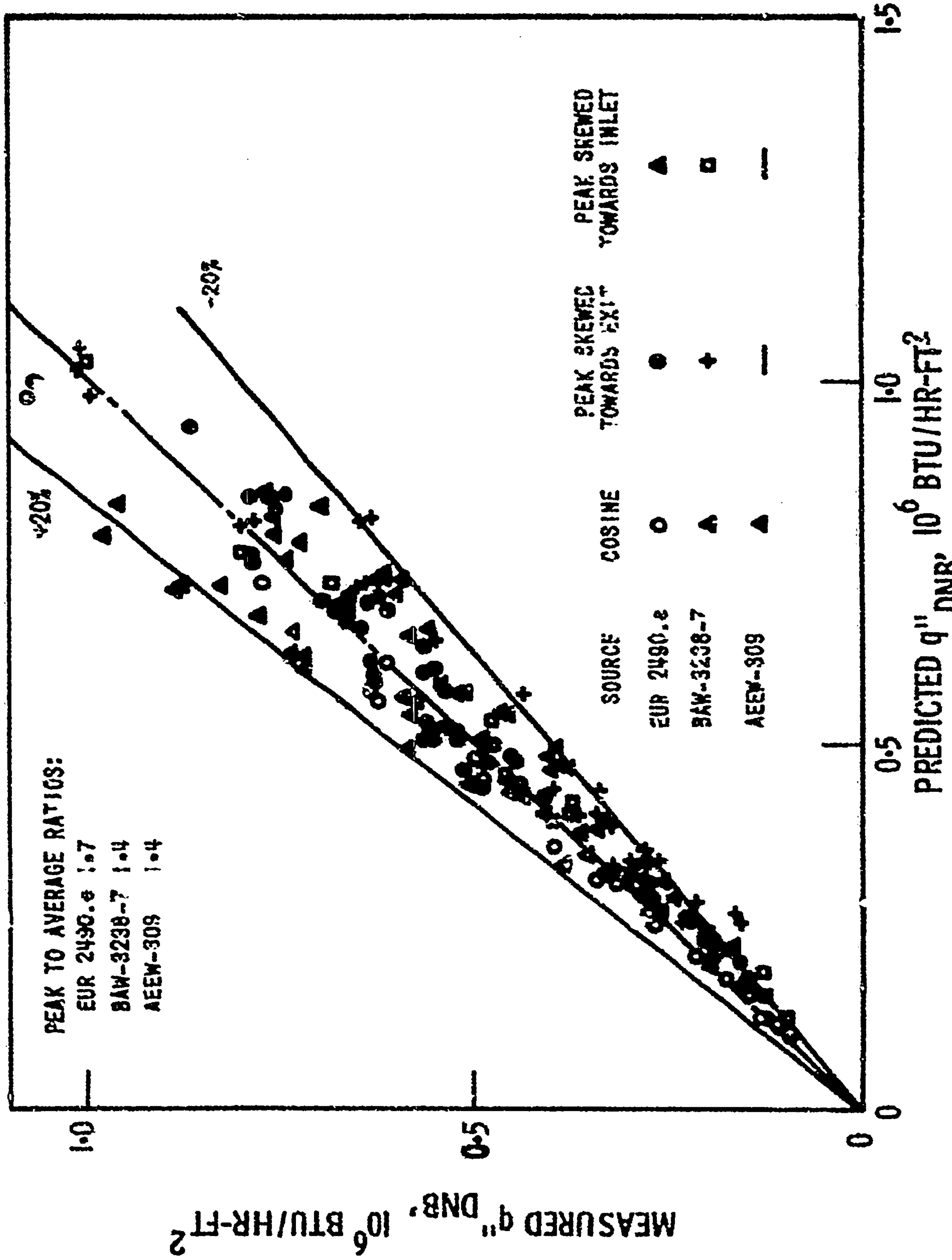
COMPARISON OF W-3 CORRELATION WITH ROD BUNDLE DNB DATA
(SIMPLE GRID WITHOUT MIXING VANE) FIG. 3.2.2-5



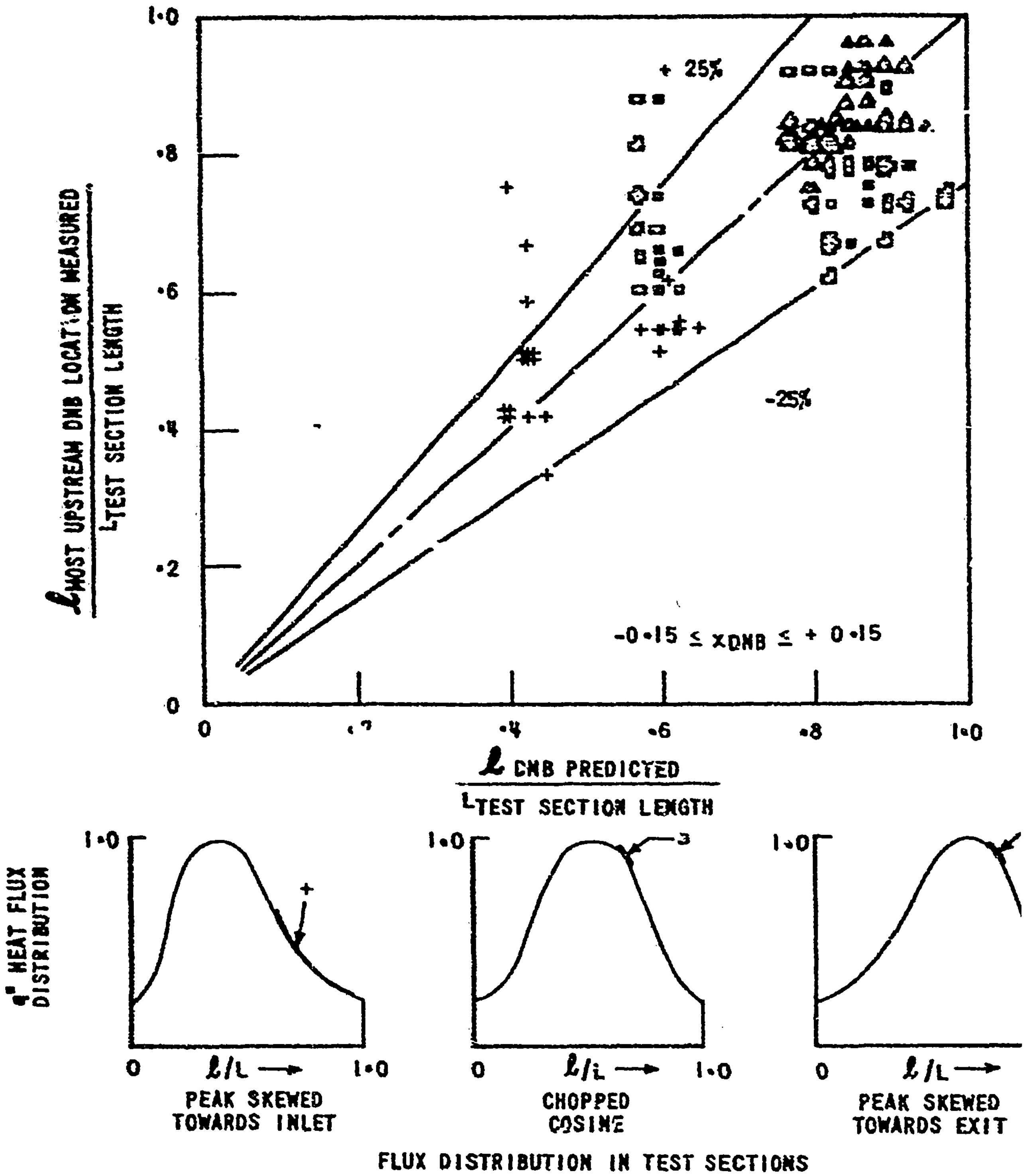
COMPARISON OF W-3 CORRELATION WITH ROD BUNDLE DNB DATA
(SIMPLE GRID WITH MIXING VANE) FIG. 3.2.2-8



STABLE FILM BOILING HEAT TRANSFER DATA AND CORRELATION
FIG. 3.[illegible]-7



COMPARISON OF W-3 PREDICTION AND NON-UNIFORM FLUX DATA
$(0.15 \leq X_{DNB} \leq + 0.15)$ FIG. 3.2.2-8



COMPARISON OF W-3 PREDICTION WITH MEASU
DNB LOCATION

FIG. 3.2.2-9

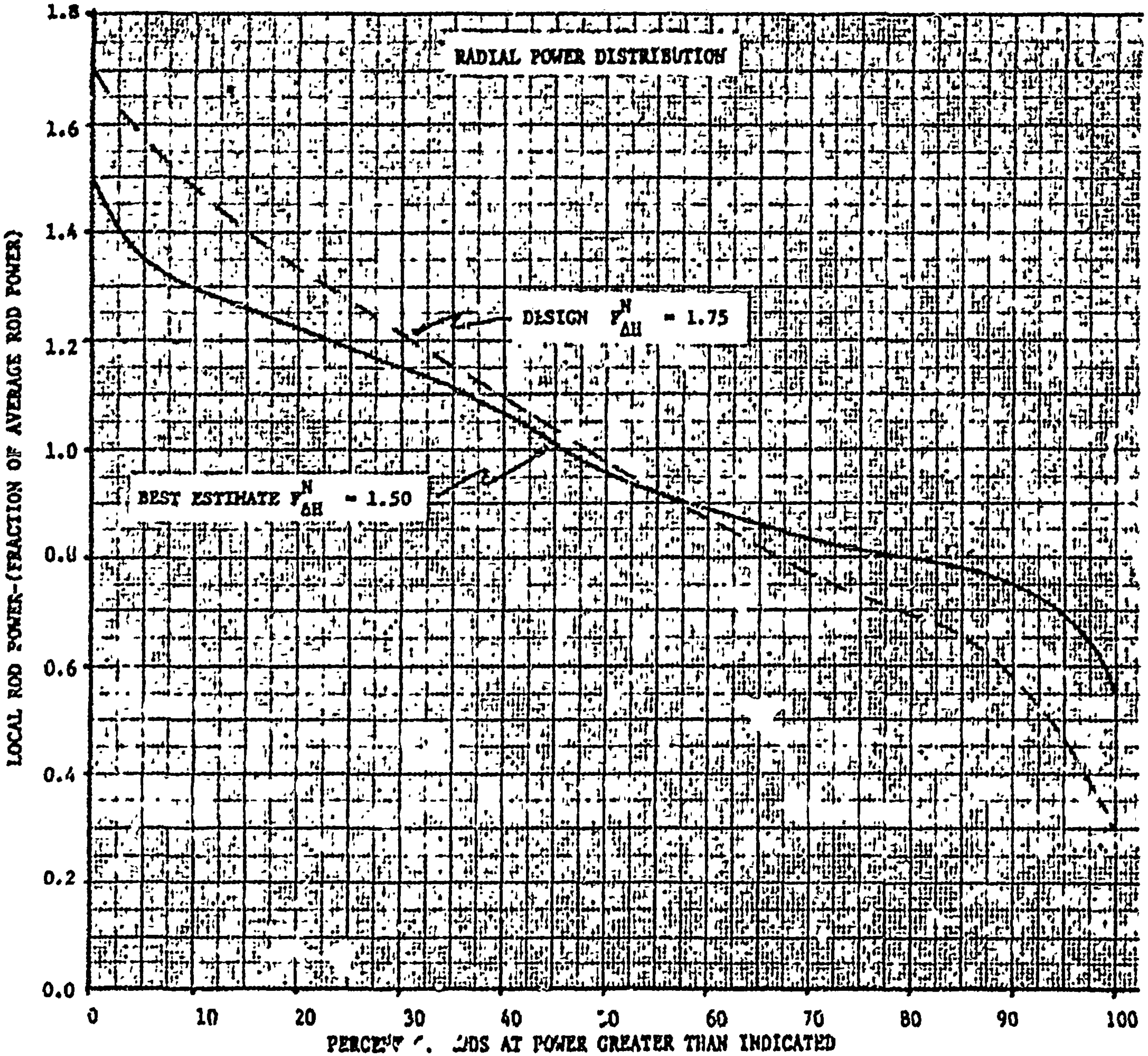


FIGURE 3.2.2-10

3.2.3 MECHANICAL DESIGN AND EVALUATION

The reactor core and reactor vessel internals are shown in cross-section in Figure 3.2.3-1 and in elevation in Figure 3.2.3-2. The core, consisting of the fuel assemblies, control rods, source rods, burnable poison rods, and plugging devices, provides and controls the heat source for the reactor operatio The internals, consisting of the upper and lower core support structure, are designed to support, align, and guide the core components, direct the coolant flow to and from the core components, and to support and guide the in-core instrumentation. A listing of the core mechanical design parameters is given in Table 3.2.3-1.

The fuel assemblies are arranged in a roughly circular cross-sectional pattern. The assemblies are all identical in configuration, but contain fuel of different enrichments depending on the location of the assembly within the core.

The fuel is in the form of slightly enriched uranium dioxide ceramic pellets. The pellets are stacked to an active height of 144 inches within Zircaloy-4 tubular cladding which is plugged and seal welded at the ends to encapsulate the fuel. The enrichments of the fuel for the various regions in the core are given in Table 3.2.3-1. Heat generated by the fuel is removed by demineralized light water which flows upward through the fuel assemblies and acts as both moderator and coolant.

The core is divided into regions of three different enrichments. The loading arrangement for the initial cycle is indicated on Figure 3.2.3-2. Refueling takes place generally in accordance with an inward loading schedule.

The control rods, designated as Rod Cluster Control Assemblies (RCCA), consist of groups of individual absorber rods which are held together by a spider at the top end and actuated as a group. In the inserted position, the absorber rods fit within hollow guide thimbles in the fuel assemblies. The guide thimbles are an integral part of the fuel assemblies and occupy locations within the regular fuel rod pattern where fuel rods have been deleted. In the withdrawn position, the absorber rods are guided and supported laterally by guide tubes which form an integral part of the upper core support structure. Figure 3.2.3-4 shows a typical rod cluster control assembly.

As shown in Figure 3.2.3-2, the fuel assemblies are positioned and supported vertically in the core between the upper and lower core plates. The core plates are provided with pins which index into closely fitting mating holes in the fuel assembly top and bottom nozzles. The pins maintain the fuel assembly alignment which permits free movement of the control rods from the fuel assembly into the guide tubes in the upper support structure without binding or restriction between the rods and their guide surfaces.

Operational or seismic loads imposed on the fuel assemblies are transmitted through the core plates to the upper and lower support structures and ultimately to the internals support ledge at the pressure vessel flange in the case of vertical loads or to the lower radial support and internals support ledge in the case of horizontal loads. The internals also provide a form fitting baffle surrounding the fuel assemblies which confines the upward flow of coolant in the core area to the fuel bearing region.

Reactor Internals

Design Description

The reactor internals are designed to support and orient the reactor core fuel assemblies and control rod assemblies, absorb the control rod dynamic loads and transmit these and other loads to the reactor vessel flange, provide a passageway for the reactor coolant, and support in-core instrumentation. The reactor internals are shown in Figure 3.2.3-2.

The internals are designed to withstand the forces due to weight, preload of fuel assemblies, control rod dynamic loading, vibration, and earthquake acceleration. These internals are analyzed in a manner similar to Connecticut Yankee, San Onofre, Zorita, Saxton and Yankee. Under the loading conditions, including conservative effects of design earthquake loading, the structure satisfies stress values prescribed in Section III, ASME Nuclear Vessel Code.

The reactor internals are equipped with bottom-mounted in-core instrumentation supports. These supports are designed to sustain the applicable loads outlined above.

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and thermal shield), the upper core support structure and the in-core instrumentation support structure.

Lower Core Support Structure

The major containment and support member of the reactor internals is the lower core support structure, shown in Figure 3.2.3-5. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the thermal shield, the intermediate diffuser plate and the bottom support plate which is welded to the core barrel. All the major material for this structure is Type 304 Stainless Steel. The core support structure is supported at its upper flange from a ledge in the reactor vessel head flange and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are axial baffle and former plates which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a 2" inch thick member through which the necessary flow distributor holes for each fuel assembly are machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the lower core support of the core barrel in order to provide stiffness and to transmit the core load to the lower core support. Intermediate between the support plate and lower core support plate is positioned a perforated plate to diffuse uniformly the coolant flowing into the core.

The one piece thermal shield is fixed to the core barrel at the top with rigid bolted connections. The bottom of the thermal shield is connected to the core barrel by means of axial flexures. This bottom support allows for differential axial growth of the shield with respect to the core barrel but restricts radial or horizontal movement of the bottom of the shield. Rectangular tubing in which vessel material samples can be inserted and irradiated during reactor operation, are welded to the thermal shield and extend to the top of the thermal shield. These samples are held in the rectangular tubing by a preloaded spring device at the top and bottom.

The lower core support structure and principally the core barrel serve to provide passageways and control for the coolant flow. Inlet coolant flow from the vessel inlet nozzles proceeds down the annulus between the core barrel and the vessel wall, flows on both sides of the thermal shield, and then into a plenum at the bottom of the vessel. It then turns and flows up through the lower support, passes through the intermediate diffuser plate and then through the lower core plate. The flow holes in the diffuser plate and the lower core are arranged to give a very uniform entrance flow distribution to the core. After passing through the core the coolant enters the area of the upper support structure and then flows generally radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles.

A small amount of water also flows between the baffle plates and core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum to provide cooling of the head. Both these flows eventually are directed into the upper support structure plenum and exit through the vessel outlet nozzles.

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell and partially through the lower support columns to the lower core support and thence through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell to be distributed to the lower radial support to the vessel wall, and to the core barrel flange. Transverse acceleration of the fuel assemblies is transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by a radial support type connection of the upper core plate to slab sided pins pressed into the core barrel.

The main radial support system of the core barrel is accomplished by "key" and "keyway" joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel block is welded to the vessel I.D. Another Inconel block is bolted to each of these blocks, and has a "keyway" geometry. Opposite each of these is a "key" which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity, and may be viewed as a beam fixed at the top and simply supported at the bottom.

Radial and axial expansions of the core barrel are accommodated but transverse movement of the core barrel is restricted by this design. With this system, cycle stresses in the internal structures are within the ASME Section III limits. This eliminates any possibility of failure of the core support.

In the event of downward vertical displacement of the internals, energy absorbing devices limit the displacement by contacting the vessel bottom head. The load is transferred through the energy devices of the internals.

The energy absorbers, cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Their number and design are determined so as to limit the forces imposed to less than yield. Assuming a downward vertical displacement, the potential energy of the system is absorbed mostly by the strain energy of the energy absorbing devices.

The free fall in the hot condition is on the order of 1/2 inch, and there is an additional strain displacement in the energy absorbing devices of approximately 3/4 inch. Alignment features in the internals prevent cocking of the internals structure during this postulated drop. The control rods are designed to provide assurance of control rod insertion capabilities under this assumed drop of internals condition. The drop distance of about 1-1/4 inch is not enough to cause the tips of the shutdown group of RCC assemblies to come out of the guide tubes in the fuel assemblies.

Upper Core Support Assembly

The upper core support assembly, shown in Figure 3.2.3-6, consists of the top support plate, deep beam sections, and upper core plate between which are contained 48 support columns and 61 guide tube assemblies. The support columns establish the spacing between the top support plate, deep beam sections, and the upper core plate and are fastened at top and bottom to these plates and beams. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. The guide tube assemblies, shown on Figure 3.2.3-7, sheath and guide the control rod drive shafts and control rods and provide no other mechanical functions. They are fastened to the top support plate and are guided by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the control rod shroud tube which is attached to the upper support plate and guide tube.

The upper core support assembly, which is removed as a unit during refueling operation, is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel which in turn engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 0°, 90°, 180°, and 270°. Four slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies and control rods is thereby assured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly and is compressed by the reactor vessel head flange.

Vertical loads from weight, earthquake acceleration, hydraulic loads and fuel assembly preload are transmitted through the upper core plate via the support columns to the deep beams and top support plate and then to the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the top support plate and upper core plate. The top support plate is particularly stiff to minimize deflection.

In-Core Instrumentation Support Structures

The in-core instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom.

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to in-line columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

In addition to the upper in-core instrumentation, there are reactor vessel bottom port columns which carry the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 144 inches and the trailing ends of the thimbles (at the seal line) are extracted approximately 15 feet during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and the conduits are provided at the seal line. During normal operation, the retractable thimbles are stationary and move only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal line is cleared for the retraction operation. Section 7.4 contains more information on the layout of the in-core instrumentation system.

The in-core instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence.

Evaluation of Core Barrel and Thermal Shield

The internals design is based on analysis, test and operational information. Troubles in previous Westinghouse PWR's have been evaluated and information derived has been considered in this design. For example, new Westinghouse uses a one-piece thermal shield which is attached rigidly to the core barrel at one end and flexured at the other. The early designs that malfunctioned were multi-piece thermal shields that rested on vessel lugs and were not rigidly attached at the top.

Early core barrel designs that have malfunctioned in service, now abandoned, employed threaded connections such as tie rods, joining the bottom support to the bottome of the core barrel, and a bolted connection that tied the core barrel to the upper barrel. The malfunctioning of core barrel designs in earlier service was believed to have been caused by the thermal shield which was oscillating, thus creating forces on the core barrel. Other forces were induced by unbalanced flow in the lower plenum of the reactor. In todya's RCC design there are no fuel followers to necessitate a large bottom plunum in the reactor. The elimination of these fuel followers enabled Westinghouse to build a shorter core barrel.

The Connecticut Yankee reactor and the Zorita reactor core barrels are of the same construction as the Indian Point reactor core barrel. Deflection measuring devices employed in the Connecticut Yankee reactor during the hot-functional test, and deflection and strain gages employed in the Zorita reactor during the hot-functional test have provided important information that has been used in the design of the present day internals, including that for Indian Point. When the Connecticut Yankee thermal shield was modified to the same design as for Southern California Edison, it, too, operated satisfactorily as was evidenced by the examination after the hot-functional test. After these hot-functional tests on all of these reactors, a careful inspection of the internals was provided. All the main structural welds were examined, nozzle interfaces were examined for any differential movement, upper core plate inside supports were examined, the thermal shield attachements to the core barrel including all lockwelds on the devices used to lock the bolt were checked, no malfunctions were found.

Substantial scale model testing was performed at APD. This included tests which involved a complete full scale fuel assembly which was operated at reactor flow, temperature and pressure conditions. Tests were run on a 1/7th scale model of the Indian Point reactor. Measurements taken from these tests indicate very little shield movement, on the order of a few mils when scaled up to Indian Point. Strain gage measurements taken on the core barrel also indicate very low stresses. Testing to determine thermal shield excitation due to inlet flow disturbances have been included. Information gathered from these tests was used in the design of the thermal shield and core barrel. It can be concluded from the testing program and the analyses with the experience gained that the design as employed on the Indian Point Plant is adequate.

In order to confirm the internals design, deflection gages will be mounted on the thermal shield top and bottom for the hot-functional test. Six such gages will be mounted in the top of the thermal shield equidistant between the fixed supports and eight located at the bottom, equidistant between the six flexures, and two next to flexure supports. The internals inspection, just before the hot-functional test, will include looking at mating bearing surfaces, main welds and welds that are used on bolt locking devices. At the conclusion of the hot-functional test, measurement readings will be taken from the deflectometers on the shield and the internals will be re-examined at all key areas for any evidence of malfunction.

Core Components

Design Description

Fuel Assembly

The overall configuration of the fuel assemblies is shown in Figures 3.2.3-8 and 3.2.3-9. The assemblies are square in cross-section, nominally 8.426 inches on a side, and have an overall height of 160.1 inches.

The fuel rods in a fuel assembly are arranged in a square array with 15 rod locations per side and a nominal centerline-to-centerline pitch of 0.563 inch between rods. Of the total possible 225 rod locations per assembly, 20 are occupied by guide thimbles for the RCC rods and one for in-core instrumentation. The remaining 204 locations contain fuel rods. In addition to fuel rods, a fuel assembly is composed of a top nozzle, a bottom nozzle, 9 grid assemblies, 20 absorber rod guide thimbles, and one instrumentation thimble.

The guide thimbles in conjunction with the grid assemblies and the top and bottom nozzles comprise the basic structural fuel assembly skeleton. The top and bottom ends of the guide thimbles are welded to the top and bottom nozzles respectively. The grid assemblies, in turn, are welded to the guide thimbles at each location along the height of the fuel assembly at which lateral support for the fuel rods is required. Within this skeletal framework the fuel rods are contained and supported and the rod-to-rod centerline spacing is maintained along the assembly.

Bottom Nozzle

The bottom nozzle is a square box-like structure which controls the coolant flow distribution to the fuel assembly and functions as the bottom structural element of the fuel assembly. The nozzle, which is square in cross-section, is fabricated from 304 stainless steel parts consisting of four side plates, 12 cross bars, and four pads or feet. The side plates are welded together at the corners to form a plenum for inlet coolant to the fuel assembly. The cross bars are welded at each end to the top edges of the side plates and function as the bottom end support for the fuel rods. The bottom support surface for the fuel assembly is formed by the four pads which are welded to the side plates in the corners.

Coolant flow to the fuel assembly is directed from the plenum in the bottom nozzle upward to the interior of the fuel assembly and to the chann' between assemblies through the slots between the cross bars and cutouts in the side plates. The cross bars are positioned laterally beneath the fuel rods and are sized so that the fuel rods, which are bottomed on the bars, cannot pass through the spacing clearance between them.

The RCC guide thimbles, which carry axial loads imposed on the assembly, are fastened to the bottom nozzle cross bars. These loads as well as the weight of the assembly are distributed through the nozzle to the lower core support plate. Indexing and positioning of the fuel assembly in the core is controlled through two holes in diagonally opposite pads which mate with locating pins in the lower core plate. Lateral loads imposed on the fuel assembly are also transferred to the core support structures through the locating pins.

Top Nozzle

The top nozzle is a box-like structure, which functions as the fuel assembly upper structural element and forms a plenum space where the heated fuel assembly discharge coolant is mixed and directed toward the flow holes in the upper core plate. The nozzle is comprised of an adaptor plate enclosure, top plate, two clamps, four leaf springs, and assorted hardware. All parts with the exception of the springs and their hold down bolts are constructed of Type 304 stainless steel. The springs are made from age hardenable Inconel 718 and the bolts from Inconel 600.

The adaptor plate is square in cross-section, and is perforated by machined slots to provide for coolant flow through the plate. At assembly, the top ends of the control guide thimbles are fitted through individual bored holes in the plate and welded to the plate around the circumference of each hole. Thus, the adaptor plate acts as the fuel assembly top end plate, and provides a means of distributing evenly among the guide thimbles any axial loads imposed on the fuel assemblies.

The nozzle enclosure is actually a square thin walled tubular shell which forms the plenum section of the top nozzle. The bottom end of the enclosure is pinned and welded to the periphery of the adaptor plate, and the top end is welded to the periphery of the top plate.

The top plate is square in cross-section with a square central hole. The hole allows clearance for the RCC absorber rods to pass through the nozzle into the guide thimbles in the fuel assembly and for coolant exit from the fuel assembly to the upper internals area. Two pads containing axial through-holes which are located on diametrically opposite corners of the top plate provide a means of positioning and aligning the top of the fuel assembly. As with the bottom nozzle, alignment pins in the upper core plate mate with the holes in the top nozzle plate.

Hold down forces of sufficient magnitude to oppose the hydraulic lifting forces on the fuel assembly are obtained by means of the leaf springs which are mounted on the top plate. The springs are fastened in pairs to the top plate at the two corners where alignment holes are not used and radiate out from the corners parallel to the sides of the plate. Fastening of each pair of springs is accomplished with a clamp which fits over the ends of the springs and two bolts (one per spring) which pass through the clamp and spring, and thread into the top plate. At assembly, the spring mounting bolts are torqued sufficiently to preload against the maximum spring load and then lockwelded to the clamp which is counter-bored to receive the bolt head.

The spring load is obtained through deflection of the spring by the upper core plate. The spring form is such that it projects above the fuel assembly and is depressed by the core plate when the internals are loaded into the reactor. The free end of the spring is bent downward and captured in a key slot in the top plate to guard against loose parts in the reactor in the event (however remote) of spring fracture. In addition, the fit between the spring and key slot and between the spring and its mating slot in the clamp are sized to prevent rotation of either end of the spring into the control rod path in the event of spring fracture.

In addition to its plenum and structural functions, the nozzle provides a protective housing for components which mate with the fuel assembly. In handling a fuel assembly with a control rod inserted, the control rod spider is contained within the nozzle. During operation in the reactor, the nozzle protects the absorber rods from coolant cross flows in the unsupported span between the fuel assembly adaptor plate and the end of the guide tube in the upper internals package. Plugging devices which fill the ends of the fuel assembly thimble tubes at unrodded core locations, and the spiders which support the source rods and burnable poison rods are all contained within the fuel top nozzle.

Guide Thimbles

The control rod guide thimbles in the fuel assembly provide guided channels for the absorber rods during insertion and withdrawal of the control rods. They are fabricated from a single piece of Type 304 stainless steel tubing, which is drawn to two different diameters. The larger inside diameter at the top (.515 inch) provides a relatively large annular area for rapid insertion during a reactor trip and to accommodate a small amount of upward cooling flow during normal operations. The bottom portion of the guide thimble is of reduced diameter (.454 in) to produce a dashpot action when the absorber rods near the end of travel in the guide thimbles during a reactor trip. The transition zone at the dashpot section is conical in shape so that there are no rapid changes in diameter in the tube.

Flow holes are provided just above the transition of the two diameters to permit the entrance of cooling water during normal operation, and to accommodate the outflow of water from the dashpot during reactor trip.

The dashpot is closed at the bottom by means of a welded end plug. The end plug has a bayonet extension which is welded to the bottom nozzle during fuel assembly fabrication.

The top ends of the guide thimbles are fitted through individual bored holes in the plate and welded to the plate around the circumference of each hole.

Grids

The spring clip grid assemblies consist of individual slotted straps which are assembled and interlocked in an "egg-crate" type arrangement and then furnace brazed to permanently join the straps at their points of intersection. Details such as spring fingers, support dimples, mixing vanes, and tabs are punched and formed in the individual straps prior to assembly.

Two types of grid assemblies are used in the fuel assembly. One type having mixing vanes which project from the edges of the straps into the coolant stream is used in the high heat region of the fuel assemblies for mixing of the coolant. A grid of this type is shown in Figure 3.2.3-10. Grids of the second type, located at the bottom and top ends of the assembly, are of the nonmixing type. They are similar to the mixing type with the exception that mixing vanes are not used on the internal straps.

The spacing between grids in shown on Figure 3.2.3-9. The variation in span lengths is the result of optimization of the thermal-hydraulic and structural parameters. The grids are spot welded to each guide thimble using weld tabs which are integrally formed on the grid strap details.

The outside straps on all grids contain mixing vanes which, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or loading and unloading the core. Additional small tabs on the outside straps and the irregular contour of the straps are also for this purpose.

Inconel 718 is chosen for the grid material because of its corrosion resistance and high strength properties. After the combined brazing and solution annealing temperature cycle, the grid material is age hardened to obtain the material strength necessary to develop the required grid spring forces.

Fuel Rods

The fuel rods consist of uranium dioxide ceramic pellets in slightly cold worked Zircaloy-4 tubing which is plugged and seal welded at the ends to encapsulate the fuel. Sufficient void volume and clearances are provided within the rod to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, and fuel swelling due to accumulated fission products without overstressing of the cladding or seal welds. Shifting of the fuel within the cladding is prevented during handling or shipping prior to core loading by a carbon steel helical compression spring which bears on the top of the fuel.

At assembly, the pellets are stacked in the cladding to the required fuel height. The compression spring is then inserted into the top end of the fuel and the end plugs pressed into the ends of the tube and welded. A hold-down force of approximately six times the weight of the fuel is obtained by compression of the spring between the top end plug and the top of the fuel pellet stack. The rod space between the pellets and clad contains air at one atmosphere.

The fuel pellets are in the form of a right circular cylinder and consist of slightly enriched uranium-dioxide powder which is compacted by cold pressing and sintering to the required density. The ends of each pellet are dished slightly to allow the greater axial expansion at the center of the pellets to be taken up within the pellets themselves and not in the overall fuel length.

For the first core, the ellets in the outer region have a density of approximately 9.97 gm cc (91% of theoretical density) while those in the two inner regions (checkerboard pattern, see Fig. 3.2.3-3) have densities of 10.08 and 10.30 gm/cc corresponding to 92% and 94% of theoretical density respectively. Lower pellet densities are used to compensate for the effects of the higher burnup which the fuel experiences in those regions.

A different fuel enrichment as listed in Table 3.2.3-1 is used for each of the three regions in the first core loading.

Identification of the fuel enrichment in each of the fuel rods is maintained by an identification mark on the fuel rod top end plug. This aids in ensuring that rods of the proper enrichment will be loaded into each fuel assembly. The identification numbers on the fuel assembly top nozzles will then maintain the enrichment identity and ensure that the assemblies with the correct enrichment are loaded into the proper core region.

Each assembly will be assigned a core loading position. A record will then be made of the core loading position, serial number and enrichment. During the core loading, two independent checks will be made to ensure that the actual loading position agrees with the position assigned.

During initial core loading and subsequent refueling operations, detailed handling and checkoff procedures, will be utilized throughout the sequence. The initial core will be loaded in accordance with the core loading diagram similar to Figure 3.2.3-3 which shows the location for each of the three enrichment types of fuel assemblies used in the loading together with the serial number of the assemblies in the region.

Rod Cluster Control Assemblies

The control rods or rod cluster control (RCC) assemblies each consist of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. These assemblies, one of which is shown in Figure 3.2.3-4 are provided to control the reactivity of the core under operating conditions. These assemblies are of two types, those with rods containing full length absorber material and those with rods containing a 36-inch length absorber section with the remainder of the absorber height filled with inert Al_2O_3 material. The number of each type of RCCA is specified in Table 3.2.3-1.

The absorber material used in the control rods is silver-indium-cadmium alloy which is essentially "black" to thermal neutrons and has sufficient additional resonance absorption to significantly increase its worth. The alloy is in the form of extruded single length rods which are sealed in stainless steel tubes to prevent the rods from coming in direct contact with the coolant.

The overall control rod length is such that when the assembly has been withdrawn through its full travel, the tip of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble. Prototype tests have shown that the RCC assemblies are very easily inserted and not subject to binding even under conditions of severe misalignment.

The spider assembly is in the form of a center hub with radial vanes supporting cylindrical fingers from which the absorber rods are suspended. Handling detents, and detents for connection to the drive shaft, are machined into the upper end of the hub. A spring pack is assembled into a skirt integral to the bottom of the hub to stop the RCC assembly and absorb the impact energy at the end of a trip insertion. The radial vanes are joined to the hub, and the fingers are joined to the vanes by furnace brazing. A centerpost which holds the spring pack and its retainer is threaded into the hub within the skirt and welded to prevent loosening in service. All components of the spider assembly are made from Type 304 stainless steel except for the springs which are Inconel X-750 alloy and the retainer which is of 17-4 Ph material.

The absorber rods are secured to the spider so as to assure troublefree service. The rods are first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins are welded in place. The end plug below the pin position is designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments.

In construction, the silver-indium-cadmium rods are inserted into cold-worked stainless steel tubing which is then sealed at the bottom and the top by welded end plugs. Sufficient diametral and end clearance are provided to accommodate relative thermal expansions and to limit the internal pressure to acceptable levels.

The bottom plugs are made bullet-nosed to reduce the hydraulic drag during a reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles. The upper plug is threaded for assembly to the spider and has a reduced end section to make the joint more flexible.

Stainless steel clad silver-indium-cadmium alloy absorber rods are resistant to radiation and thermal damage thereby ensuring their effectiveness under all operating conditions. Rods of similar design have been successfully used in the Saxton, SELNI and Indian Point 1 reactors.

Neutron Source Assemblies

Six neutron source assemblies will be utilized in the core. These will consist of two assemblies with four secondary source rods each, and four assemblies with one secondary source rod and one primary source rod each. The rods in each assembly will be fastened to a spider at the top end. The spider for the four secondary source rod assembly is similar to the RCCA spiders, while the latter assembly spider is similar to that of the burnable poison and plugging device assemblies.

In the core, the neutron source assemblies will be inserted into the RCC guide thimbles in fuel assemblies at unrodded locations. The location and orientation of each of the assemblies in the core is shown in Figure 3.2.3-11.

The primary and secondary source rods both utilize the same type of cladding material as the absorber rods (cold-worked type 304 stainless steel tubing, 0.432 in. O.D. with 0.019 in. thick walls). The secondary source rods contain Sb-Be pellets stacked to a height of 118.65 inches. The primary source rods contain capsules of Po-Be source material 6 inches long and Sb-Be pellet material to fill the remainder of the rod height. Design criteria similar to that for the fuel rods is used for the design of the source rods; i.e., the cladding is free standing, internal pressures are always less than reactor operating pressure, and internal gaps and clearances are provided to allow for differential expansions between the source material and cladding.

Plugging Devices

In order to limit bypass flow through the RCC guide thimbles in fuel assemblies which do not contain either control rods source assemblies, or burnable poison rods, the fuel assemblies at those locations are fitted with plugging devices. The plugging devices consist of a flat spider plate with short rods suspended from the bottom surface and a spring pack assembly and mixing device attached to the top surface. At installation in the core, the plugging devices fit with the fuel assembly top nozzles and rest on the adaptor plate. The short rods project into the upper ends of the thimble tubes to reduce the bypass flow area. The spring pack is compressed by the upper core plate when the upper internals package is lowered into place. Similar short rods are also used on the source assemblies to fill the ends of all vacant fuel assembly guide thimbles.

All components in the plugging device, except for the springs, are constructed from type 304 stainless steel. The springs (one per plugging device) are wound from an age hardenable nickel base alloy to obtain higher strength.

Burnable Poison Rods

The burnable poison rods are statically suspended and positioned in vacant RCC thimble tubes within the fuel assemblies at nonrodded core locations. The poison rods in each fuel assembly are grouped and attached together at the top end of the rods by a flat spider plate which fits with the fuel assembly top nozzle and rests on the top adaptor plate.

The spider plate (and the poison rods) are held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals package is lowered into the reactor. This ensures that the poison rods cannot be lifted out of the core by flow forces.

The poison rods consist of borated pyrex glass tubes contained within type 304 stainless steel tubular cladding which is plugged and seal welded at the ends to encapsulate the glass. The glass is also supported along the length of its inside diameter by a thin wall type 304 stainless steel tubular inner liner. A burnable poison rod is shown in longitudinal and transverse cross-sections in Figure 3.2.3-11.

The density of natural boron in glass is equal to 0.0866 gm/cc. This corresponds to 0.0429 gm of natural boron per centimeter of glass rod length. The total core inventory of natural boron contained in the rods is equal to 18.2 kg. The rods are designed in accordance with the standard fuel rod design criteria; i.e., the cladding is free standing at reactor operating pressures and temperatures and sufficient cold void volume is provided within the rods to limit internal pressures to less than the reactor operating pressure assuming total release of all helium generated in the glass as a result of the B_{10} (n,α) reaction. The large void volume required for the helium is obtained through the use of glass in tubular form which provides a central void along the length of the rods. The resulting clad stresses at temperature and pressure are given in WCAP 7113.[3]

Based on available data on properties of Pyrex glass and on nuclear and thermal calculations for the rods, gross swelling or cracking of the glass tubing is not expected during operation. Some minor creep of the glass at the hot spot on the inner surface of the tube is expected to occur but continues only until the glass comes into contact with the inner liner. The inner liner is provided to maintain the central void along the length of the glass and to prevent the glass from slumping or creeping into the void as a result of softening at the hot spot. The wall thickness of the inner liner is sized to provide adequate support in the event of slumping but to collapse locally before rupture of the exterior cladding if large volume changes due to swelling or cracking should possibly occur.

The top end of the inner liner is open to receive the helium which diffuses out of the glass.

To ensure the integrity of the burnable poison rods, the tubular cladding and end plugs are procurred to the same specifications and standard of quality as is used for stainless steel fuel rod cladding and end plugs in other Westinghouse plants. In addition, the end plug seal welds are checked for integrity by visual inspection and x-ray. The finished rods are helium leak checked.

Evaluation of Core Components

Fuel Evaluation

The fission gas release and the associated buildup of internal gas pressure in the fuel rods is calculated by the FIGHT code based on experimentally determined rates. The increase of internal pressure in the fuel rod due to this phenomena is included in the determination of the maximum cladding stresses at the end of core life when the fission product gap inventory is a maximum.

The maximum allowable strain in the cladding, considering the combined effects of internal fission gas pressure, external coolant pressure, fuel pellet swelling and clad creep is limited to less than 1 per cent throughout core life. The associated stresses are below the yield strength of the material under all normal operating conditions.

To assure that manufactured fuel rods meet a high standard of excellence from the standpoint of functional requirements, many inspections and tests are performed both on the raw material and the finished product. These tests and inspections include chemical analysis, tensile testing of fuel tubes, dimensional inspection, X-ray of both end plug welds, ultrasonic testing and helium leak tests.

In the event of cladding defects, the high resistance of uranium dioxide fuel pellets to attack by hot water protects against fuel deterioration or decrease in fuel integrity. Thermal stress in the pellets, while causing some fracture of the bulk material during temperature cycling, does not result in pulverization or gross void formation in the fuel matrix. As shown by operating experience and extensive experimental work in the industry, the thermal design parameters conservatively account for any changes in the thermal performance of the fuel element due to pellet fracture.

The consequences of a breach of cladding are greatly reduced by the ability of uranium dioxide to retain fission products including those which are gaseous or highly volatile. This retentiveness decreases with increasing temperature or fuel burnup, but remains a significant factor even at full power operating temperature in the maximum burnup element.

A survey of fuel elements behavior in high burnup uranium dioxide[1] indicates that for an initial uranium dioxide void volume, which is a function of the fuel density, it is possible to conservatively define the fuel swelling as a function of burnup. Since Region 3 will be retained through three cycles of reactor operation, and Region 2 through 2 cycles, the pellet density has been reduced from 94% to 92% in Region 2 and 91% in Region 3 to accommodate the effects of increased burnup.

The perforation of fuel rod cladding so as to release fission products or fuel material is directly related to cladding stress and strain under normal operating and overpower conditions. Design limits and damage limits (cladding perforation) in terms of stress and strain are as follows:

	Damage Limit	Design Limit
Stress	Ultimate strength 57,000 psi minimum	Yield strength - 45,000 psi minimum
Strain	1.7%	1.0%

The damage limits given above are minimum values. Actual damage limits depend upon neutron exposure and normal variation of material properties and would generally be greater than these minimum damage limits. For most of the fuel rod life the actual stresses and strains are considerably below the design limits. Thus, significant margins exist between actual operating conditions and the damage limits.

The other parameters having an influence on cladding stress and strain and the relationship of these parameters to the damage limits are as follows:

1. Internal gas pressure:

   An internal gas pressure of approximately [illegible],800 psi is required to produce cladding stresses equal to the damage limits under normal operating conditions. The maximum design internal pressure under nominal conditions is 2250 psia which is equal to the coolant pressure. The maximum internal gas pressure at beginning of life is [illegible]0 psia. The end of life internal gas pressure is dependent upon the fuel rod power history and will not exceed the design limit of 2250 psia.

2. Cladding temperature:

   The strength of the fuel cladding is temperature dependent. The minimum ultimate strength reduces to the design yield strength at an average cladding temperature of approximately 850°F. The maximum average cladding temperature during normal operating conditions is given in Table 3.2.2-1.

3. Burnup:

   Fuel burnup results in fuel swelling which produces cladding strain. The strain damage limit is not expected to be reached until the peak burnup reaches approximately 65,000 MWD/MTU. The maximum expected burnup in the fuel is approximately 50,000 MWD/MTU. A burnup of 65,000 MWD/MTU would cause the fission gas pressure to increase but will still be well below that required for the cladding stress to reach the damage limit. The design equilibrium first core average burnup is about 33,000 MWD/MTU.

4. Fuel temperature and kw/ft:

At zero burnup, cladding damage is calculated to occur at 31 kw/ft based upon cladding strain reaching the damage limit. At this power rating 17% of the pellet central region is expected to be in the molten condition.

Evaluation of Burnable Poison Rods

The burnable poison rods are positively positioned in the core inside RCC assembly guide thimbles and held down in place by attachment to a spider assembly compressed beneath the upper core plate and hence cannot be the source of any reactivity transient. Due to the low heat generation rate, and the conservative design of the poison rods, there is no possibility for release of the poison as a result of helium pressure or clad heating during accident transients including loss of coolant.

Two burnable poison rods of reduced length but similar in design to those to be used in the Indian Point Plant Unit II Reactor have been exposed to in-pile test conditions in the Saxton Test Reactor since October 1967.

A visual exami ation of the rod was made in early June 1968, and a visual and profilometer examination was made in July 30, 1968 after an exposure of 1900 effective full power hours (~25% B^{10} depletion). The rods were found to be in excellent condition and profilometry results showed no dimensio ariation from the original new condition.

An experimental verification of the reactivity worth calculations for pyrex glass tubing is presented in Appendix 3A.

Effects of Vibration and Thermal Cycling on Fuel Assemblies

Analyses of the effect of cyclic deflection of the fuel rods, grid spring fingers, RCC control rods, and burnable poison rods due to hydraulically induced vibrations and thermal cycling show that the design of the components is good for an infinite number of cycles.

In the case of the fuel rod grid spring support, the amplitude of a hydraulically induced motion of the fuel rod is extremely small (~.001), and the stress associated with the motion is significantly small (<100 psi). Likewise, the reactions at the grid spring due to the motion is much less than the preload spring force and contact is maintained between the fuel clad and the grid spring and dimples. Fatigue of the clad and fretting between the clad and the grid support is not anticipated.

The effect of thermal cycling on the grid-clad support is merely a slight relative movement between the grid contact surfaces and the clad, which is gradual in nature during heat-up and cool-down. Since the number of cycles of the occurrence is small over the life of a fuel assembly (~3 years), negligible wear of the mating parts is expected.

In-core operation of assemblies in the Yankee Rowe and Saxton reactors using similar clad support have verified the calculated conclusions. Additional test results under simulated reactor environment in the Westinghouse Reactor Evaluation Channel also support these conclusions.

The dynamic deflection of the full and part-length control rods and the burnable poison rods is limited by their fit with the inside diameter of either the upper portion of the guide thimble or the dashpot (.0765 in. diametral clearance at guide thimble; .0145 in. diametral clearance at the dashpot). With this limitation, the occurrence of truly cyclic motion is questionable. However, an assumed cyclic deflection through the available clearance gap results in an insignificantly low stress in either the clad tubing or in the flexture joint at the spider or retainer plate. The above consideration assumes the rods are supported as cantilevers from the spider, or the retainer plate in the case of the burnable poison rods.

A calculation, assuming the rods are supported by the surface of the dashpots and at the upper end by the spider or retainer, results in a similar conclusion.

Control Rod Drive Mechanism

a) Full Length Rods

Design Description

The control rod drive mechanisms are used for withdrawal and insertion of the rod cluster control assemblies into the reactor core and to provide sufficient holding power for stationary support.

Fast total insertion (reactor trip) is obtained by simply removing the electrical power allowing the rods to fall by gravity. Typical total insertion time is about 2 to 3 seconds.

The complete drive mechanism, shown in Figure 3.2.3-12, consists of the internal (latch) assembly, the pressure vessel, the operating coil stack, the drive shaft assembly, and the rod position indicator coil stack.

Each assembly is an independent unit which can be dismantled or assembled separately. Each mechanism pressure housing is threaded onto an adaptor on top of the reactor pressure vessel and seal welded. The operating drive assembly is connected to the control rod (directly below) by means of a grooved drive shaft. The upper section of the drive shaft is suspended from the working components of the drive mechanism. The drive shaft and control rod remain connected during reactor operation, including tripping of the rods.

Main coolant fills the pressure containing parts of the drive mechanism. All working components and the shaft are immersed in the main coolant and depend upon it for lubrication of sliding parts.

Three magnetic coils, which form a removable electrical unit and surround the rod drive pressure housing induce magnetic flux through the housing wall to operate the working components. They move two sets of latches which lift, lower and hold the grooved drive shaft.

The three magnets are turned on and off in a fixed sequence by solid-state switches for the full length rod assemblies.

The sequencing of the magnets produces step motion over the 144 inches of normal control rod travel.

The mechanism develops a lifting force approximately two times the static lifting load. Therefore, extra lift capacity is available for overcoming mechanical friction between the moving and the stationary parts. Gravity provides the drive force for rod insertion and the weight of the whole rod assembly is available to overcome any resistance.

The mechanisms are designed to operate in water at 650°F and 2485 psig. The temperature at the mechanism head adaptor will be much less than 650°F because it is located in a region where there is limited flow of water from the reactor core, while the pressure is the same as in the reactor pressure vessel.

A multi-conductor cable connects the mechanism operating coils to the 125 volt d-c power supply. The power supply is described in Section 7.3.2.

Latch Assembly

The latch assembly contains the working components which withdraw and insert the drive shaft and attached control rod. It is located within the pressure housing and consists of the pole pieces for three electromagnets. They actuate two sets of latches which engage the grooved portion of the drive shaft.

The upper set of latches move up or down to raise or lower the drive rod by 5/8 inch. The lower set of latches have a 1/16 inch axial movement to shift the weight of the control rod from the upper to the lower latches.

Pressure Vessel

The pressure vessel consists of the pressure housing and rod travel housing. The pressure housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel. It provides space for the drive shaft during its upward movement as the control rod is withdrawn from the core.

Operating Coil Stack

The operating coil stack is an independent unit which is installed on the drive mechanism by sliding it over the outside of the pressure housing. It rests on a pressure housing flange without any mechanical attachment and can be removed and installed while the reactor is pressurized.

The three operater coils are made of round copper wire which is insulated with a double layer of filament type glass yarn.

The design operating temperature of the coils is 200°C. Average coil temperature can be determined by resistance measurement. Forced air cooling along the outside of the coil stack maintains a coil casing temperature of approximately 120°C or lower.

Drive Shaft Assembly

The main function of the drive shaft is to connect the control rod to the mechanism latches. Grooves for engagement and lifting by the latches are located throughout the 144 in. of control rod travel. The grooves are spaced 5/8 inch apart to coincide with the mechanism step length and have 45° angle sides.

The drive shaft is attached to the control rod by the coupling. The coupling has two flexible arms which engage the grooves in the spider assembly.

A 1/4 inch diameter disconnect rod runs down the inside of the drive shaft. It utilizes a locking button at its lower end to lock the coupling and control rod. At its lower end, there is a disconnect assembly. For remote disconnection of the drive shaft assembly from the control rod, a button at the top of the drive rod actuates the connect/disconnect assembly.

During plant operation, the drive shaft assembly remains connected to the control rod at all times. It can be attached and removed from the control rod only when the reactor vessel head is removed.

Position Indicator Coil Stack

The position indicator coil stack slides over the rod travel housing section of the pressure vessel. It detects drive rod position by means of cylindrically wound differential transformer which span the normal length of the rod travel (144 inches).

Drive Mechanism Materials

All parts exposed to reactor coolant, such as the pressure vessel, latch assembly and drive rod, are made of metals which resist the corrosive action of the water.

Three types of metals are used exclusively: stainless steels, Inconel X, and cobalt based alloys. Wherever magnetic flux is carried by parts exposed to the main coolant, 400 series stainless steel is used. Cobalt based alloys are used for the pins, latch tips, and bearing surfaces.

Inconel X is used for the springs of both latch assemblies and 304 stainless steel is used for all pressure containment. Hard chrome plating provides wear surfaces on the sliding parts and prevents galling between mating parts (such as threads) during assembly.

Outside of the pressure vessel, where the metals are exposed only to the reactor plant container environment and cannot contaminate the main coolant, carbon and stainless steels are used. Carbon steel, because of its high permeability, is used for flux return paths around the operating coils. It is zinc-plated 0.001 inch thick to prevent corrosion.

Principles of Operation

The drive mechanisms shown schematically in Figure 3.2.3-13 with-draw and insert their respective control rods as electrical pulses are received by the operator coils.

ON and OFF sequence, repeated by switches in the power programmer causes either withdrawal or insertion of the control rod. Position of the control rod is indicated by the transformer action of the position indicator coil
13 stack surrounding the rod travel housing. The transformer output changes as the top of the ferromagnetic drive shaft assembly moves up the rod travel housing.

Generally, during plant operation, the drive mechanisms hold the control rods withdrawn from the core in a static position, and only one coil, either the movable gripper coil, or the stationary gripper coil is energized on each mechanism.

Control Rod Withdrawal: The control rod is withdrawn by repeating the following sequence:

(1) Movable Gripper Coil - ON

The movable gripper armature raises and swings the movable gripper latches into the drive shaft groove.

(2) Stationary Gripper Coil - OFF

Gravity causes the stationary gripper latches and armature to move downward until the load of the drive shaft is transferred to the movable gripper latches. Simultaneously, the stationary gripper latches swing out of the shaft groove.

(3) Lift Coil - ON

The 5/8 inch gap between the lift armature and the lift magnet pole closes and the drive rod raises one step length.

(4) Stationary Gripper Coil - ON

The stationary gripper armature raises and closes the gap below the stationary gripper magnetic pole, swing the stationary gripper latches into a drive shaft groove. The latches contact the shaft [illegible] 1/16 inch. The load is so transferred from [illegible] stationary gripper [illegible]

(5) Movable Gripper Coil - [illegible]

The movable gripper arm[illegible] [illegible]mature under the force of three springs and gravity. Three links, pinned to the movable gripper armature, swing the three movable gripper latches out of the groove.

(6) Lift Coil - OFF

The gap between the lift armature and the lift magnet pole opens. The movable gripper latches drop 5/8 inch to a position adjacent to the next groove.

Control Rod Insertion:

The sequence for control rod insertion is similar to that for control rod withdrawal:

(1) Lift Coil - ON

The movable gripper latches are raised to a position adjacent to a shaft groove.

(2) Movable Gripper Coil - ON

The movable gripper armature raises and swings the movable gripper latches into a groove.

(3) Stationary Gripper Coil - OFF

The stationary gripper armature moves downward and swings the stationary gripper latches out of the groove.

(4) Lift Coil - OFF

Gravity separates the lift armature from the lift magnet pole and the control rod drops down 5/8 inch.

(5) Stationary Gripper Coil - ON

(6) Movable Gripper Coil - OFF

The sequences described above are termed as one step or one cycle and the control rod moves 5/8 inch for each cycle. Each sequence can be repeated at a rate of up to 80 steps per minute and the control rods can therefore be withdrawn or inserted at a rate of up to 50 inches per minute.

Control Rod Tripping:

If power to the movable gripper coil is cut off, as for tripping, the combined weight of the drive shaft and the rod cluster control assembly is sufficient to move the latches out of the shaft groove. The control rod falls by gravity into the core. The tripping occurs as the magnetic field, holding the movable gripper armature against the lift magnet, collapses and the movable gripper armature is forced down by the weight acting upon the latches.

b) Part Length Rods

The part length control rod drive mechanisms are used to position the part length rod control cluster assemblies within the reactor core. The mechanisms drive the control clusters at a constant speed of fifteen inches per minute in either direction. Removal or loss of power causes the rods to stop all motion immediately.

The complete control rod drive mechanism (CRDM) consists of the internal rotor assembly, the pressure vessel, the drive shaft assembly, the external operating stator and the position detecting systems.

The internal rotor assembly is the operating center of the mechanism. The drive rod assembly is permanently attached to the rotor assembly and is disconnected with special tools only in the case of exceptional maintenance. During refueling, the drive rod is disconnected from the control rod and driven into the travel housing for storage. The rotor assembly is free to rotate in and held in place within the pressure vessel by ball bearing assemblies. Five free rotating "ball-nuts" are held captive in the lower cylindrical portion of the rotor and are canted to match the lead angle of the drive rod threads. As the internal rotor assembly rotates the ball nuts turn within the threads of the drive rod, translating vertical motion to it, much as a turning nut would cause a bolt to rise or fall in a slot which prevents the bolt from rotating. In the case of the drive mechanism, the rotational torque is taken through the RCCA in the core. It is not desirable to attempt to take this rotational torque within the mechanism itself, because of the possibility of imposing unknown, and potentially large, internal torques in the lead screw and RCCA due to "wind-up" during its travel. The normal torque on the RCCA is of the order of 1.5 ft. lbs. The maximum over torque would occur with the mechanism cold when a rod becomes stuck. This would be the motor stalling torque cold, and has a maximum value of approximately 12 ft. lbs. This torque is taken by the RCCA into its fuel assembly and cannot be transmitted to adjacent fuel assemblies. The torque is positively transmitted via fingers and slots in the coupling sleeve and the RCCA spider. The stator provides rotational energy to the rotor after first releasing the mechanism brake. The brake is a positive mechanical type which does not rely on friction, and consists of a spring loaded split cylinder, concentric with the drive rod and pinned to the rotor. It is released by energizing a motor coil which pulls the upper halves of the cylinder together, thereby releasing the lower clam shell-like brake. There are four springs in the brake, and the design is such that the breakage of a spring will not impair brake performance

If one or both brake arm pivot pins should fail, the brake will engage, whether or not the motor is energized.

As detailed previously the ball nuts are at all times in engagement with the screwthread on the drive shaft; they cannot be moved radially outwards, therefore, they can never release the rod and allow it to drop. The only means whereby the drive rod can move is by rotation of the cage holding the ball-nuts. The magnetic circuitry of the motor is such that energizing the windings simultaneously disengages the mechanical brake and permits free rotation of the armature, i.e., there is no separate electrical circuit for the brake. In operation, at least two of the six windings are energized by the sequencing to obtain the required motion. In the holding mode, two or three of the windings are energized thereby holding the final position of the armature and preventing rod motion even though the brake is disengaged.

Loss of power to the mechanism will engage the brake and prevent rod motion. The application of power to any or all of the windings will disengage the brake. However, the holding force created by a single winding is sufficient to overcome the rundown torque produced by the mechanism load. Therefore, the rod cannot move except under the control of the power supply.

The rotational energy is supplied in sequential pulses to the armature which rotate directionally 15° per pulse as controlled by the power supply.

The mechanism is capable of developing a minimum lifting force of approximately 700 pounds. Insertion force is limited to the weight of the drive train and control rod cluster. This is achieved by not providing a thrust bearing to take upward thrust, and allowing the rotor to rise on the lead-screw until it engages a vertical face stop, thereby stalling the motor without generating downthrust.

The part length rod mechanism are designed to operate in water at 650°F and 2485 psig. Flow from the reactor core is restricted by the natural configuration and also by thermal bushings in order to limit the operating temperature. This temperature restriction will increase the life of the stator and wear parts of the rotor.

The pressure housing consists of the rotor assembly housing and rod travel housing. It is designed and fabricated in accordance with Section III of the ASME B&PV Code.

The top of the rod travel housing is closed by a screwed plug which is sealed with a conoseal joint. The design of this area is such that any leakage of reactor fluid will produce only a vertical jet of water and thence a downward thrust.

Should a longitudinal split appear in the rod travel hous[illegible], the resultant jet will be retarded in exactly the same manner as that of the magnetic jack type mechanisms for full length control rods. The position indicating coils and their housings are identical for both mechanism designs.

The drive shaft assembly is of extremely conservative design. The material used is cold drawn 17-4 PH in the H 1100 condition. The maximum load on the assembly is only 270 lbs.

The function of the drive shaft is to connect the mechanism to the control rod with a coupling which has two flexible arms to engage the grooves in the spider body at the top of the control rod.

The coupling is essentially the same as that used on the normal magnetic jack control rod drive mechanisms: the principal difference is the provision of slots and fingers to transmit torque from the drive rod to the RCCA spider. The basic design is therefore well proven. During plant operation, the RCCA and drive shaft remain connected at all times, and the design incorporates several safety features to ensure this. First, a spring loaded spherical button in the coupling forces the flexible arms into the grooves in the RCCA spider. This button is unlocked by a disconnect rod which runs down inside the drive shaft. Second, if this button should fail, a stop nut on the end of the disconnect rod prevents the arms from flexing sufficiently to allow them to slip from the coupling grooves unless the disconnect rod is lifted. Third, a spring retaining ring prevents movement of the disconnect rod unless a special coupling/uncoupling tool is used. Because of the "non-scrammable" feature of the part-length control rod drive mechanisms, the drive shaft cannot be disconnected from the roller nut sleeve, and therefore is disconnected from the RCCA before removing the reactor head.

Positive indication of correct remote coupling and uncoupling is derived from the position of the top of the drive rod when the rod is driven into contact with the RCCA. If the rod is uncoupled, this "full-down" position is about 1-1/2 in. higher than that when coupled.

Should maintenance be required it is possible to remove the cooling air shroud, the position indicating coils, and the motor stator. It is further possible to remove all the working parts from the pressure housing without removing the pressure housing from the head adapter. All this may be accomplished with the reactor head in position on the reactor.

All parts exposed to the reactor coolant, such as the pressure housing, drive rod and rotor assembly, are made of materials which resist the corrosive action of the water.

Fuel Assembly and RCCA Mechanical Evaluation

To confirm the mechanical adequacy of the fuel assembly and full length RCCA assembly, functional test programs have been conducted on a full scale Indian Point No. 2 prototype 12 ft. canless fuel assembly and control rod. The prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 hours. The prototype mechanism accumulated 2,260,892 steps and 600 scrams. At the end of the test the CRDM was still operating satisfactorily. A correlation was developed to predict the amplitude of flow excited vibration of individual fuel rods and fuel assemblies. Inspection of the fuel assembly and drive line components did not reveal significant fretting. The wear of the absorber rods, fuel assembly guide thimbles, and upper guide tubes was minimal. The control rod free fall time against 125% of nominal flow was less than 1.5 seconds to the dashpot (10 ft. of travel). Additional tests had previously been made on a full scale San Onofre mock up version of the fuel assembly and control rods.[2]

Indian Point No. 2,1/7 Scale Mockup Tests

A 1/7 scale model of the Indian Point No. 2 internals was designed and built for hydraulic and mechanical testing. The tests provided information on stresse and displacements at selected locations on the structure due to static loads, flow induced loads, and electromagnetic shaker loads. Flow distribution and pressure drop information were obtained. Results of the static tests indicated that mean strains in the upper core support plate and upper support columns are below design limits. Strains and displacements measured in the model during flow tests verified that no damaging vibration levels were present. Additional information gained from the tests were the natural frequency and damping of the thermal shield and other components in air and water. Model response can be related to the full scale plant for most of the expected exciting phenomena, but across the board scaling is not possible. Specifically exciting phenomena which are strongly dependent on Reynolds number cannot be scaled. In areas where Reynolds number may be important, either (1) the measured vibration amplitudes were many times lower than a level that would be damaging, or (2) full scale vibration data has been obtained.

Loading and Handling Tests

Tests simulating the loading of the prototype fuel assembly into a core location have also been successfully conducted to determine that proper provisions had been made for guidance of the fuel assembly during refueling operation.

Axial and Lateral Bending Tests

In addition axial and lateral bending tests have been performed in order to simulate mechanical loading of the assembly during refueling operation. Although the maximum column load expected to be experienced in service is approximately 1000 lb. the fuel assembly was successfully loaded to 2200 lb. axially with no damage resulting. This information is also used in the design of fuel handling equipment to establish the limits for inadvertent axial loads during refueling.

Part Length Rod Research and Development.

Although the part length CRDM has a number of new features it is based on, and largely similar to, a proven design used on U.S. Navy reactors. The U.S. Navy mechanisms are designed and manufactured by the same company, and the same people, as are producing the present part length mechanisms. Thus the same skills and experience are used for both applications. Although details are not available for security reasons, it is known that a comprehensive series of tests have been performed on the U.S. Navy mechanisms, the experience from which will be incorporated in the part length mechanism design. Even so, further full scale tests will be carried out on the prototype of this mechanism to further assure its reliability and safety. Specifically, a full life test will be conducted, using the complete drive train and RCCA, at reactor pressure and temperature conditions.

REFERENCES, Section 3.2.3

1. Daniel, R. C., et al, "Effects of High Burnup on Zircaloy-Clad Bulk UO_2, Plate Fuel Element Samples, "WAPD-263, (September, 1965).

2. Large Closed Cycle Water Reactor Research and Development Program Quarterly Progress Reports for the Period January 1963 through June 1965 (WCAP-3738, 3739, 3743, 3750, 3269-2, 3269-2, 3269-5, 3269-6, 3269-12 and 3269-13).

3. WCAP 7113, "Use of Burnable Poison Rods in Westinghouse Pressurized Water Reactors," October 1967.

4. WCAP 7072 "Use of Part Length Absorber Rods in Westinghouse Pressurized Water Reactors,".

TABLE 3.2.3-1

CORE MECHANICAL DESIGN PARAMETERS [1]

Active Portion of the Core	
Equivalent Diameter, in.	132.7
Active Fuel Height, in.	144.0
Length-to-Diameter Ratio	1.09
Total Cross-Section Area, ft^2	96.06
Fuel Assemblies	
Number	193
Rod Array	15 x 15
Rods per Assembly	204 [2]
Rod Pitch, in.	0.563
Overall Dimensions	8.426 x 8.426
Fuel Weight, (as UO_2), pounds	216,600
Total Weight, pounds	276,000
Number of Grids per Assembly	9
Number of Guide Thimbles	20
Diameter of Guide Thimbles (upper part), in.	0.545 O.D.x0.515
Diameter of Guide Thimbles (lower part), in.	0.484 O.D.x0.454
Fuel Rods	
Number	39,372
Outside Diameter, in.	0.422
Diametral Gap, in.	0.0065
Clad Thickness, in.	0.0243
Clad Material	Zircaloy
Overall Length	149.7
Length of End Cap, overall, in.	0.688
Length of End Cap, inserted in rod, in.	0.250
Fuel Pellets	
Material	UO_2 sintered
Density (% of Theoretical)	
Region 1	94 (10.3 g/cc)
Region 2	92 (10.08 g/cc)
Region 3	91 (9.97 g/cc)
Feed Enrichments w/o	
Region 1	2.2
Region 2	2.7
Region 3	3.2
Diameter, in.	0.3669
Length, in.	0.600

TABLE 3.2.3-1 (Cont'd)

Rod Cluster Control Assemblies

Neutron Absorber	5% Cd, 15% In, 80% Ag
Cladding Material	Type 304 SS - Cold Worked
Clad Thickness, in.	0.019
Number of Clusters	
Full Length	53
Part Length	8
Number of Control Rods per Cluster	20
Length of Rod Control, in.	156.436 (overall)
	149.136 (insertion length)
Length of Absorber Section, in.	142.00 (full length)
	36.00 (part length)

Core Structure

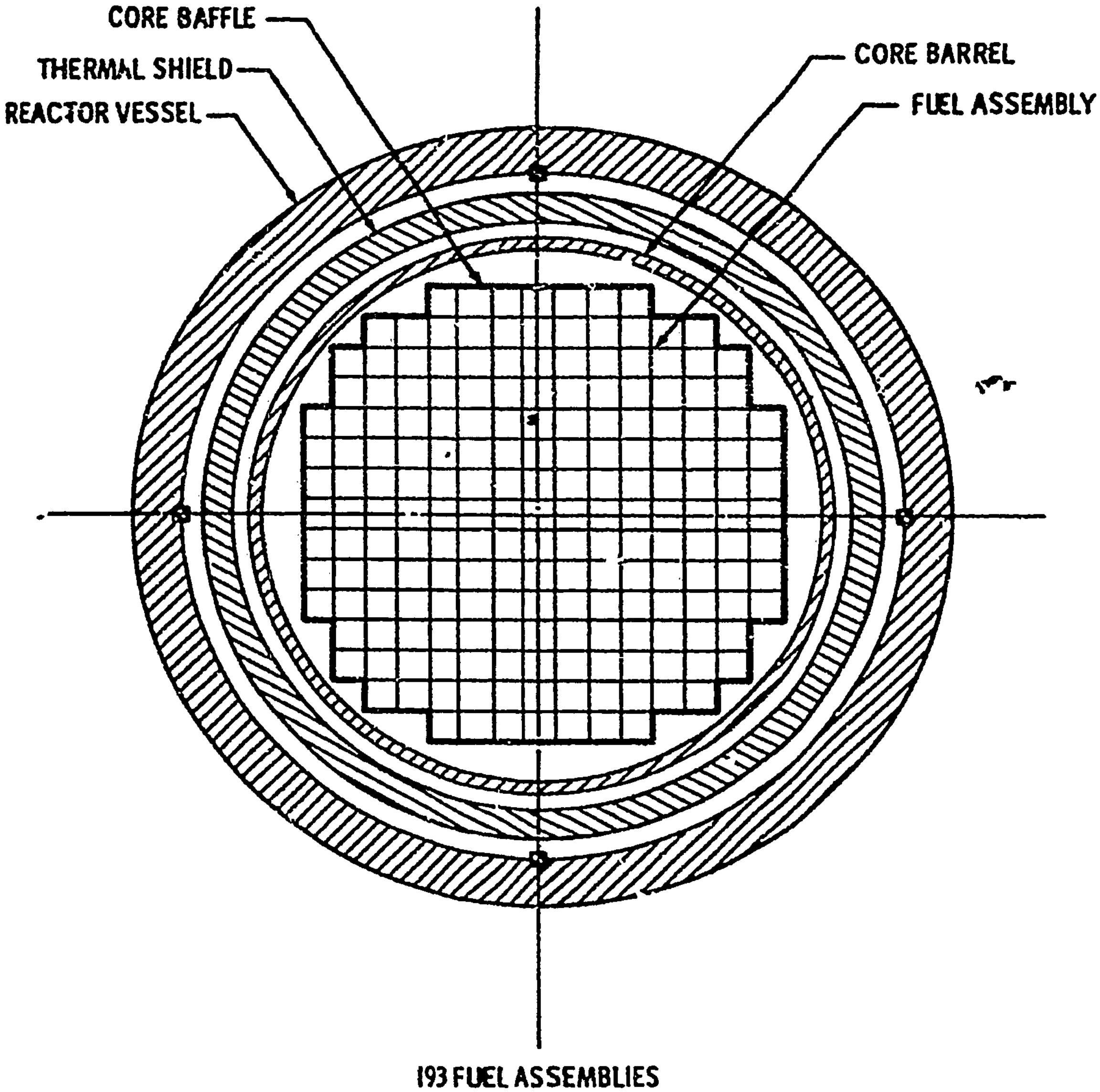
Core Barrel, in.	
I.D.	148.0
O.D.	152.5
Thermal Shield, in.	
I.D.	158.5
O.D.	164.0

Burnable Poison Rods

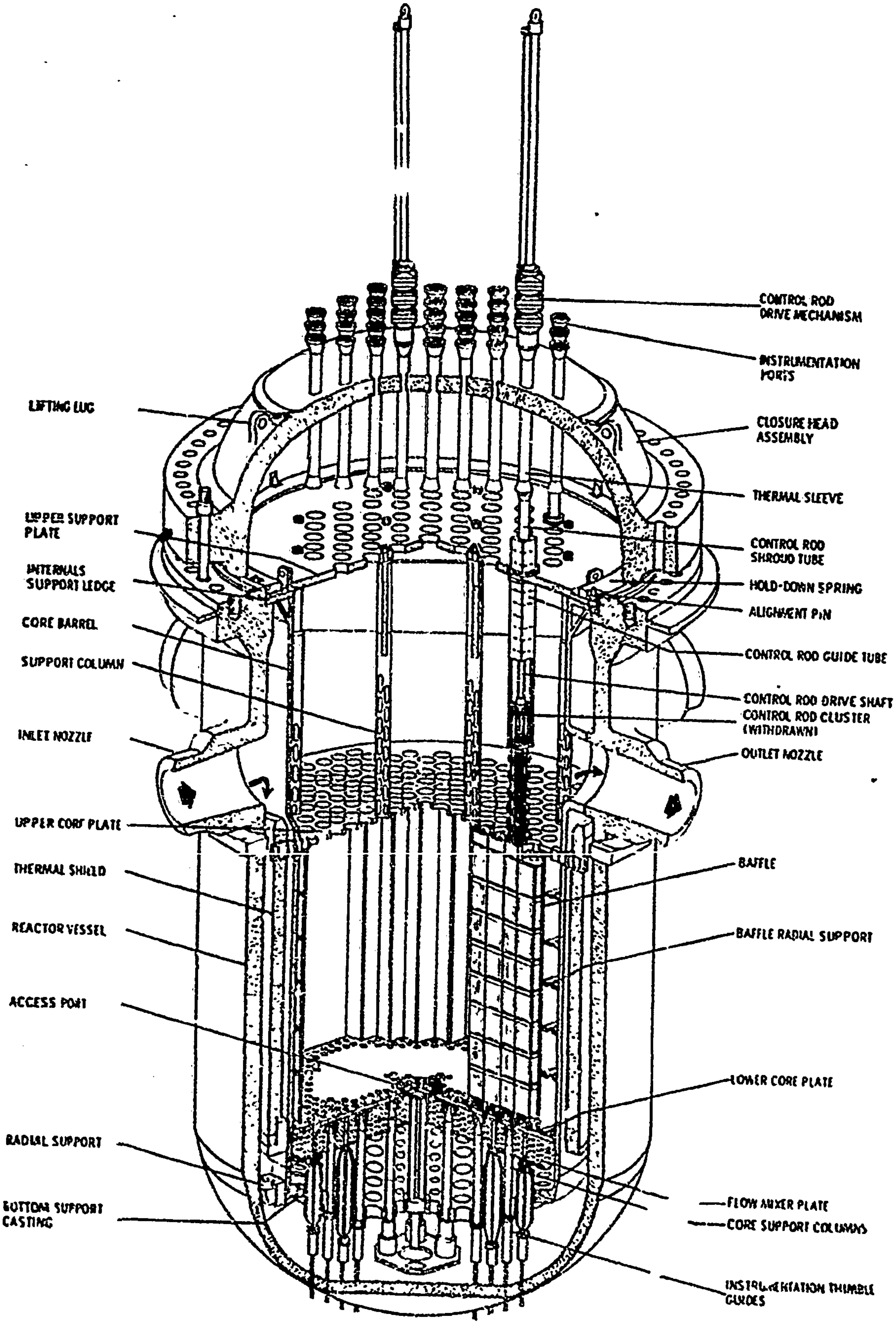
Number	1156
Material	Borosilicate Glass
Outside Diameter, in.	0.4395
Inner Tube, O.D. in.	0.2365
Clad Material	S.S.
Inner Tube Material	S.S.
Boron Loading (natural) gm/cm of glass rod	0.0429

(1) All dimensions are for cold conditions.

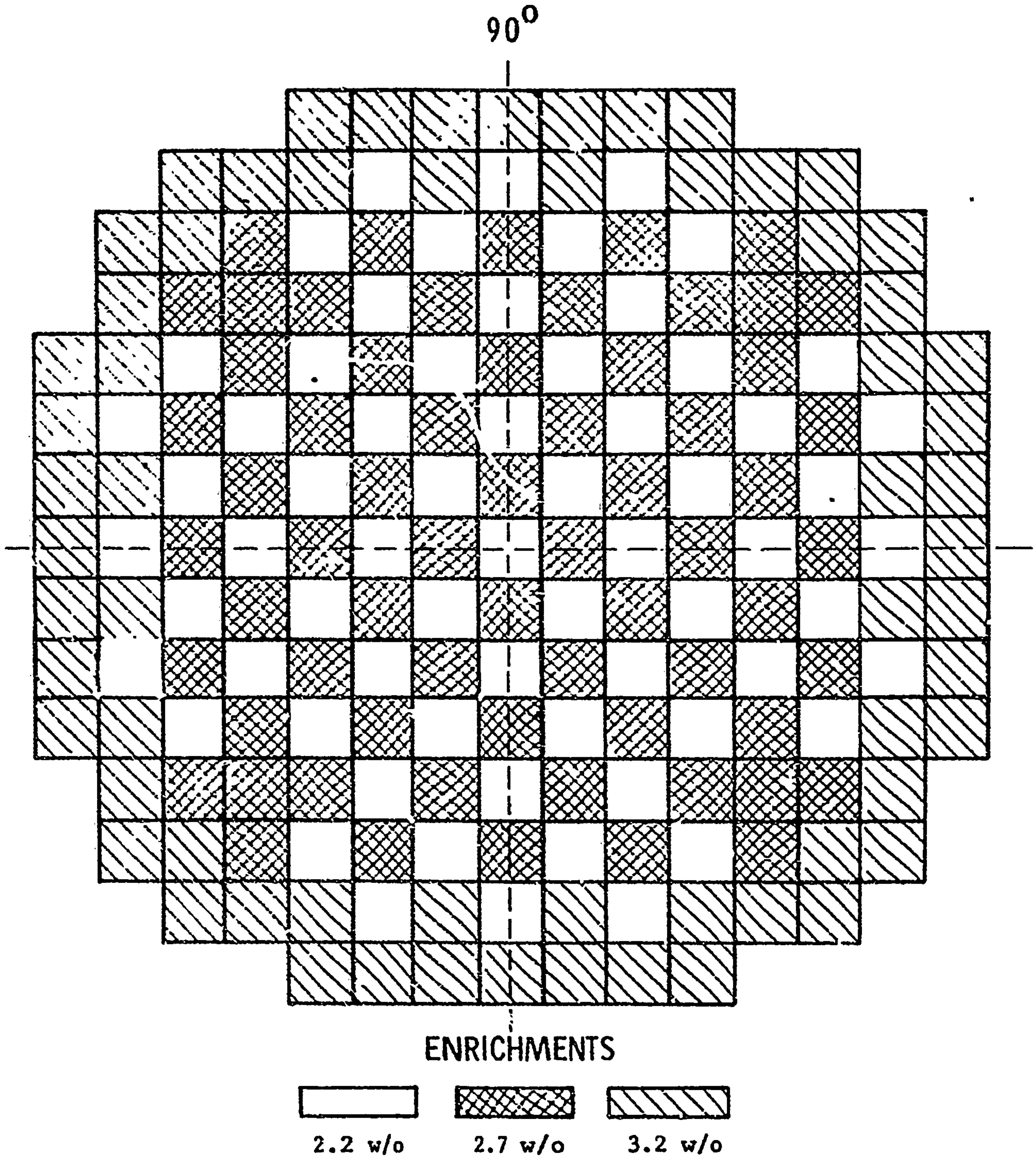
(2) Twenty-one rods are omitted: Twenty provide passage for control rods and one to contain in-core instrumentation.



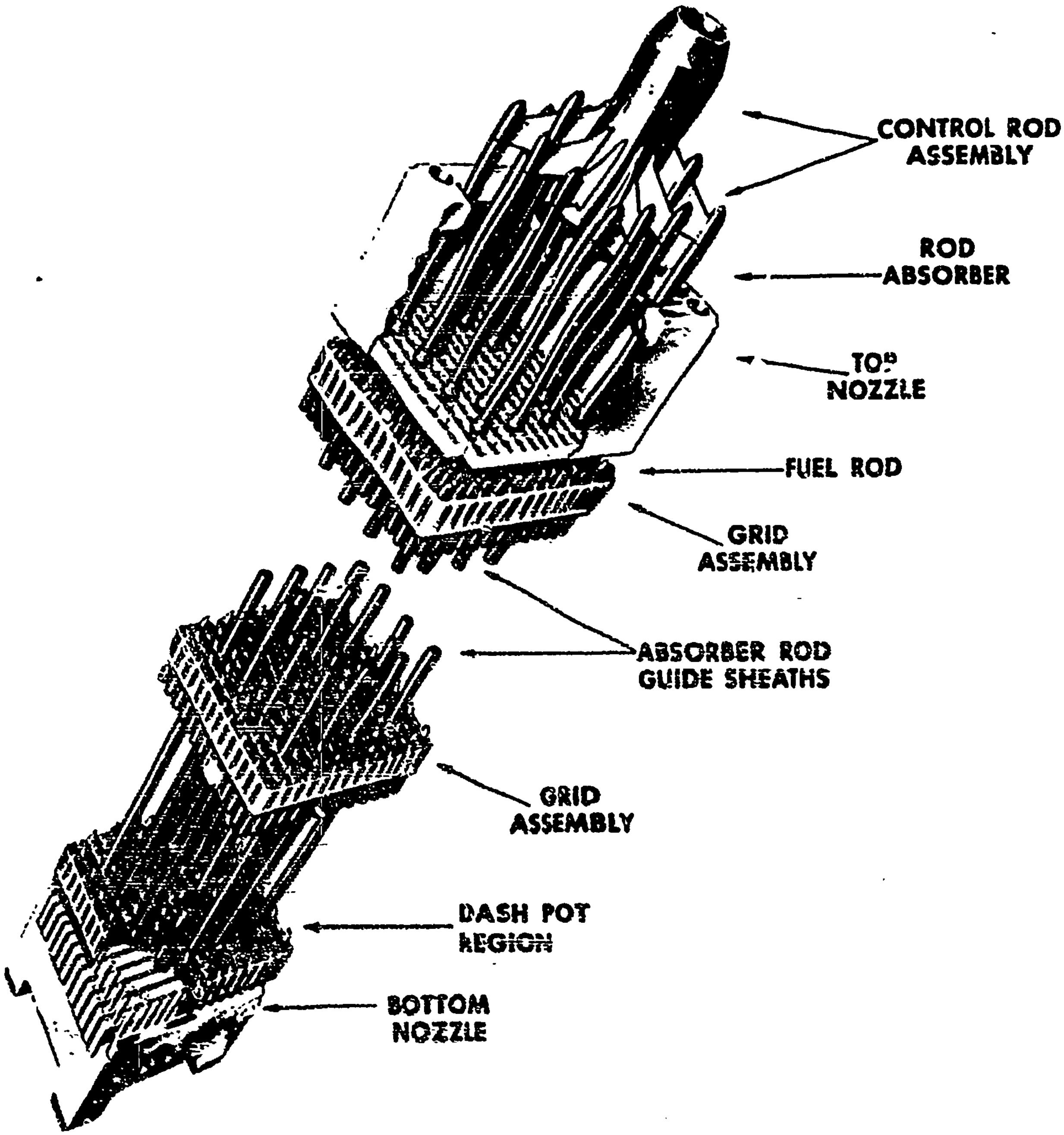
CORE CROSS SECTION
FIG. 3.2.3-1



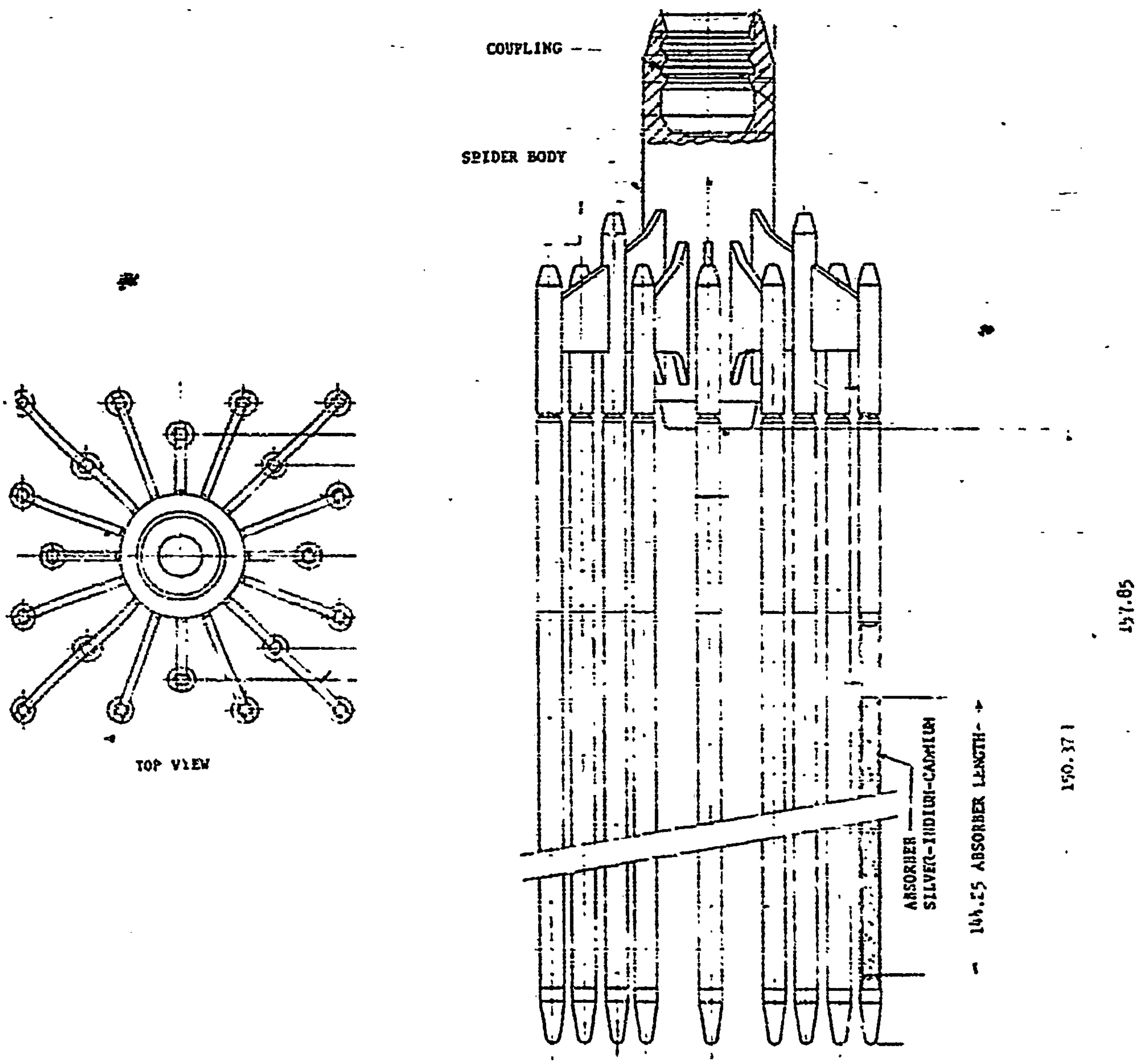
REACTOR VESSEL INTERNALS
FIGURE 3.2.3-2



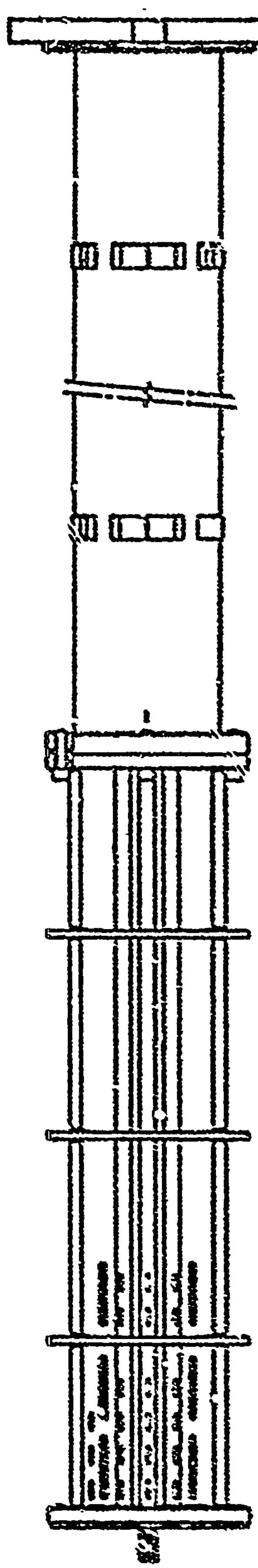
CORE LOADING ARRANGEMENT
FIG. 3.2.3-3

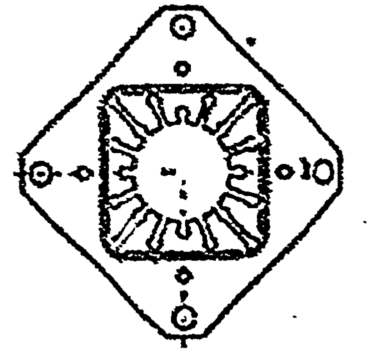


TYPICAL ROD CLUSTER CONTROL ASSEMBLY
FIGURE 3.2.3-4



ROD CONTROL CLUSTER ASSEMBLY OUTLINE
FIGURE 3.2.3-4a





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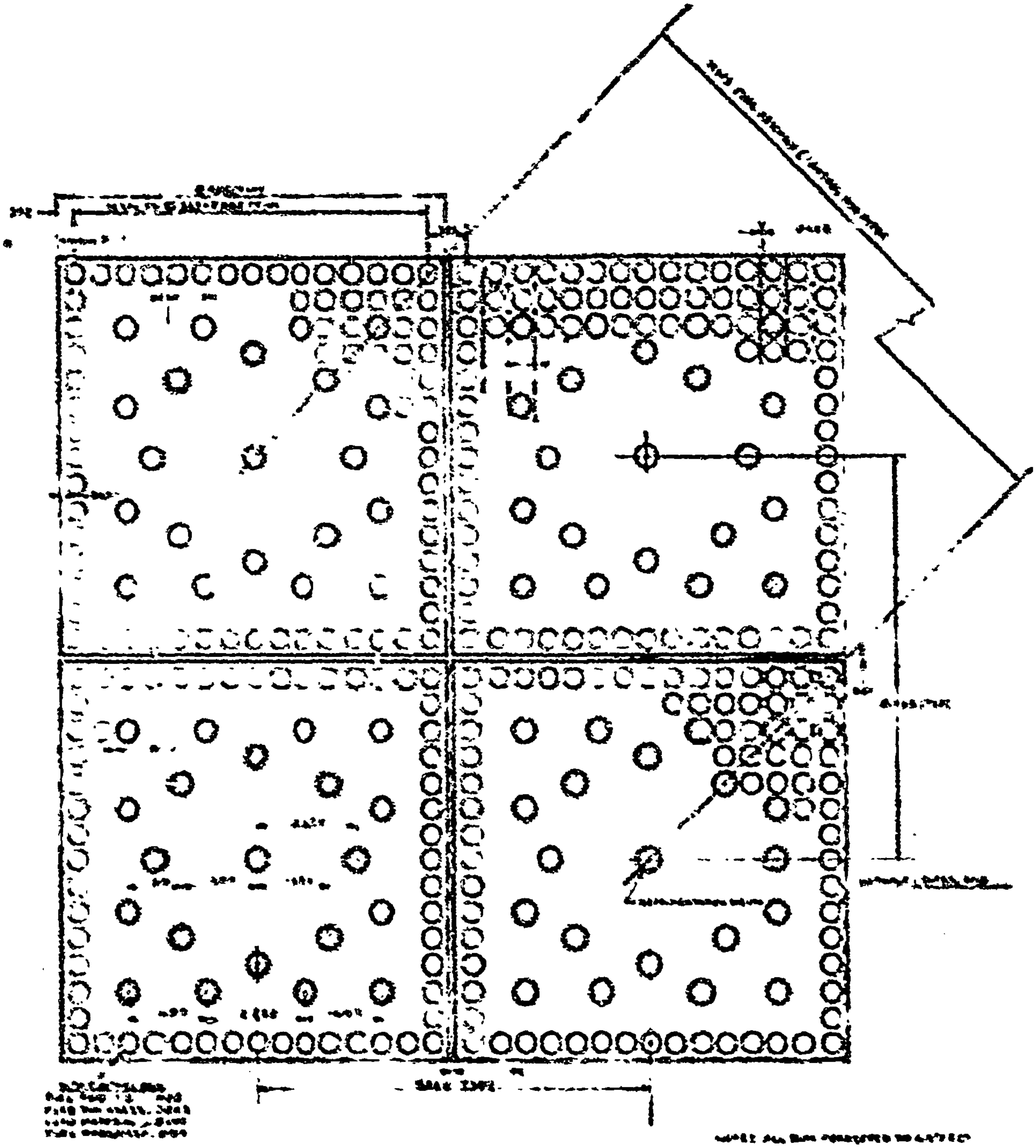
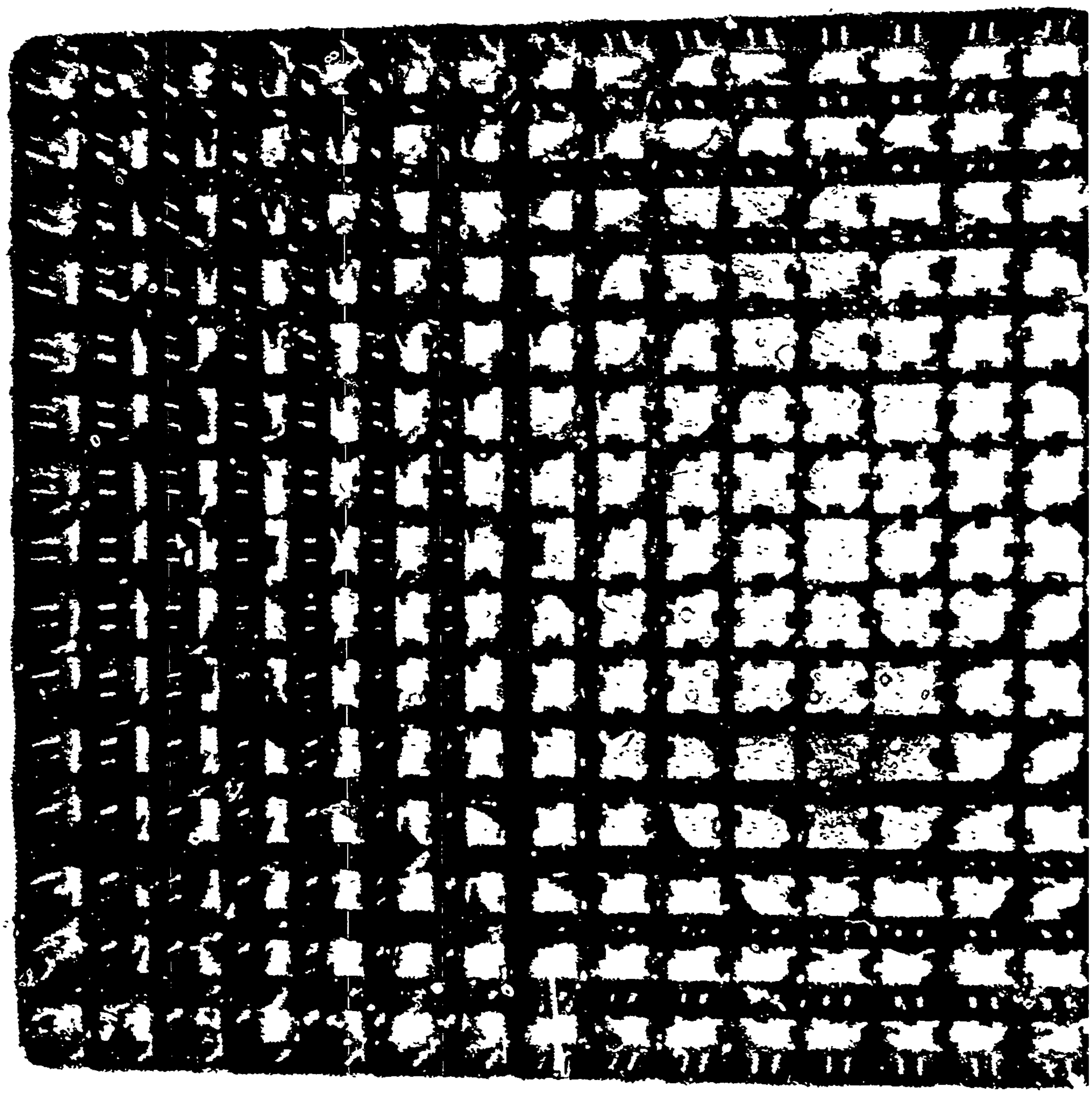
GUIDE TUBE ASSEMBLY
FIG. 3.2.3-7

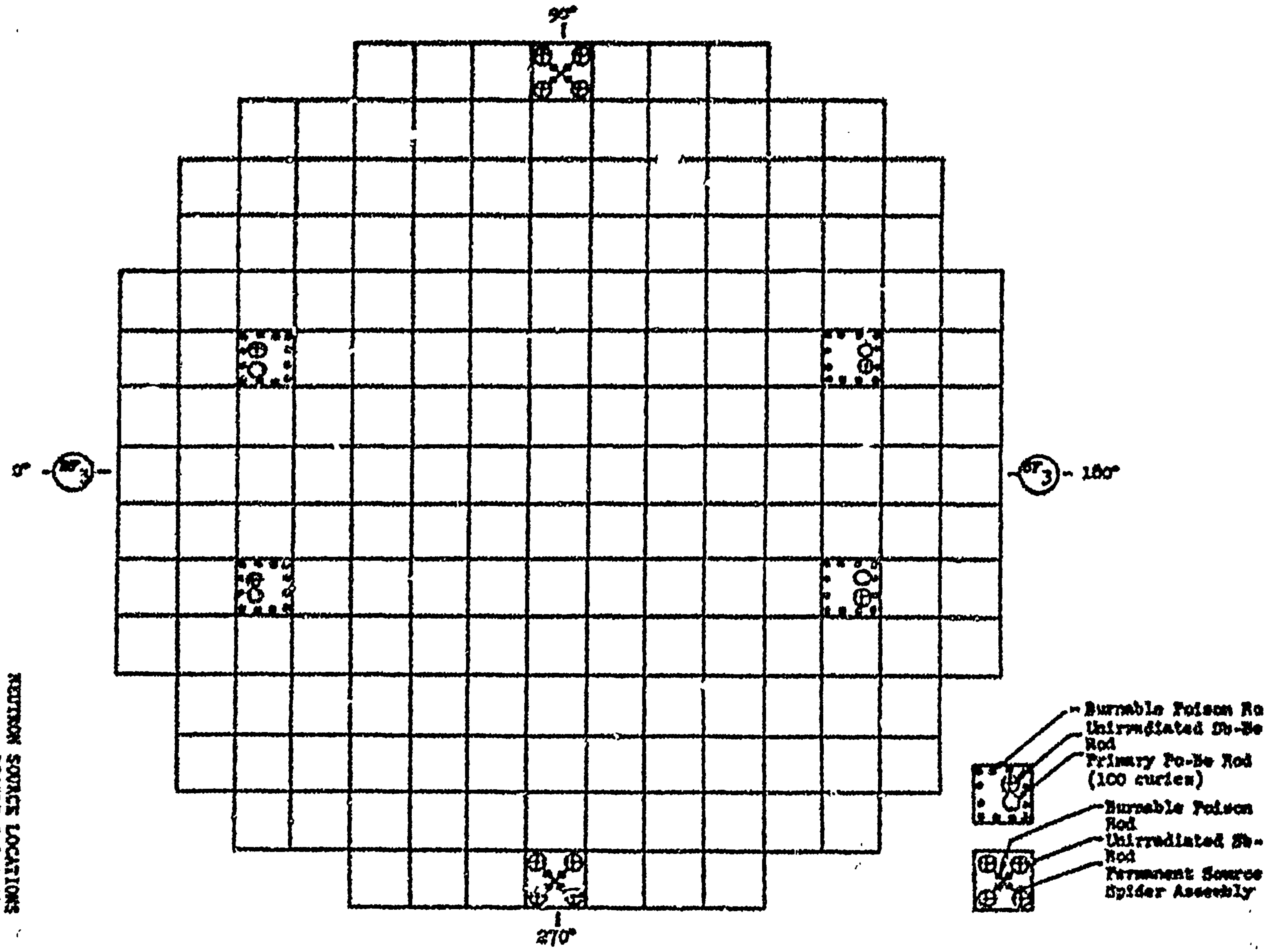


FUEL ASSEMBLY AND CONTROL CLUSTER CROSS SECTION
FIGURE 3.2.3-8



SPRING CLIP GRID ASSEMBL

FIGURE 3.2.3.10



NEUTRON SOURCE LOCATIONS
FIGURE 3.2.3-11

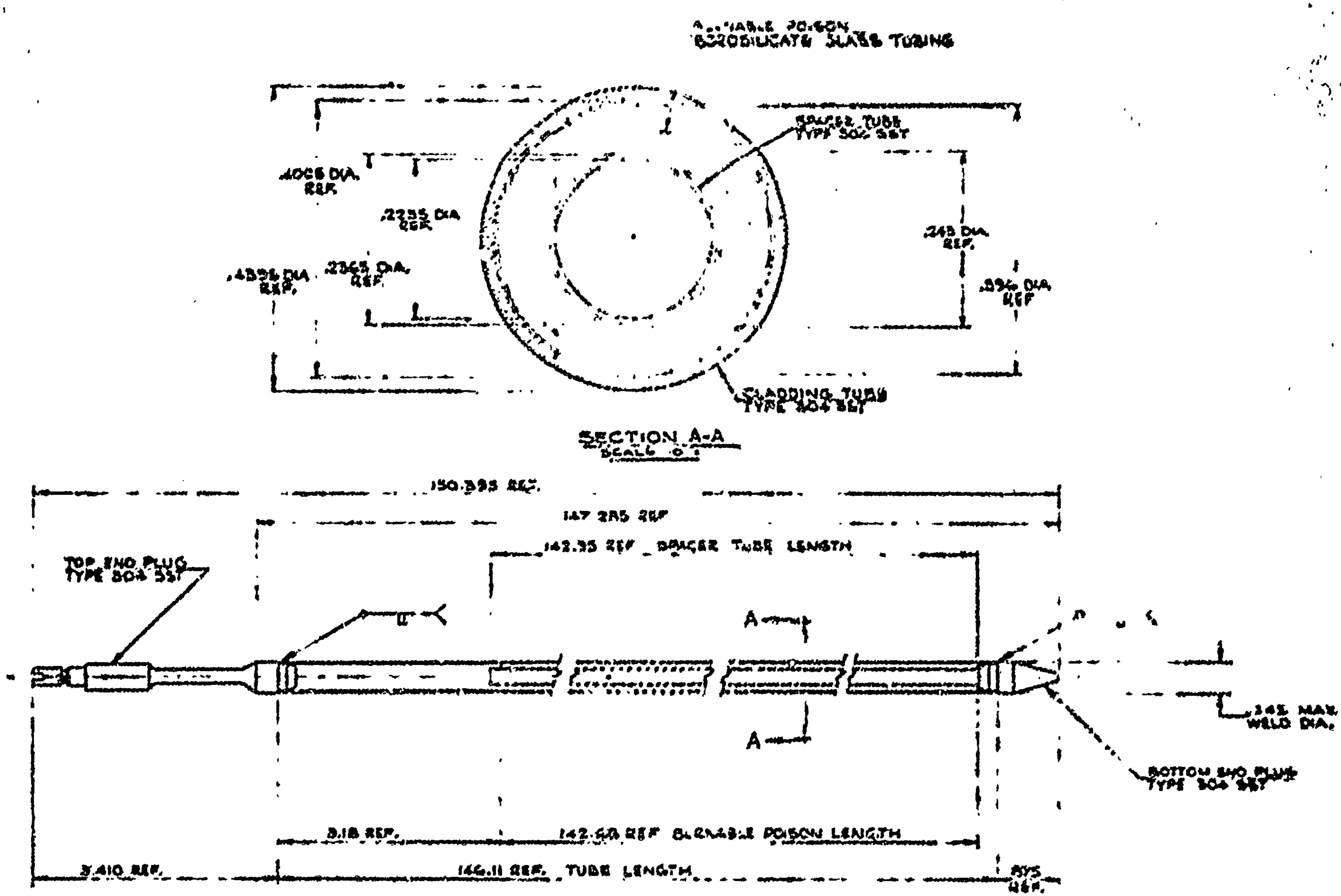
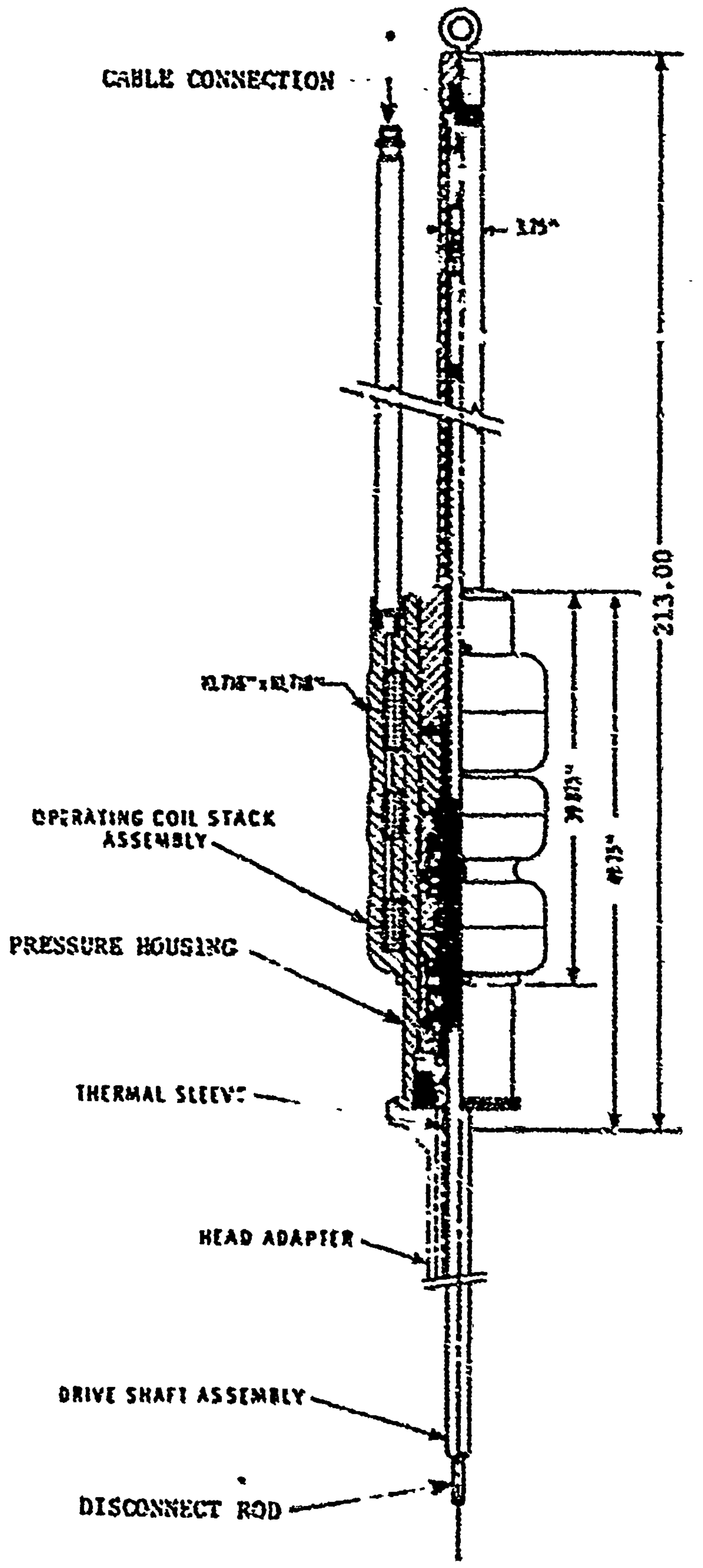


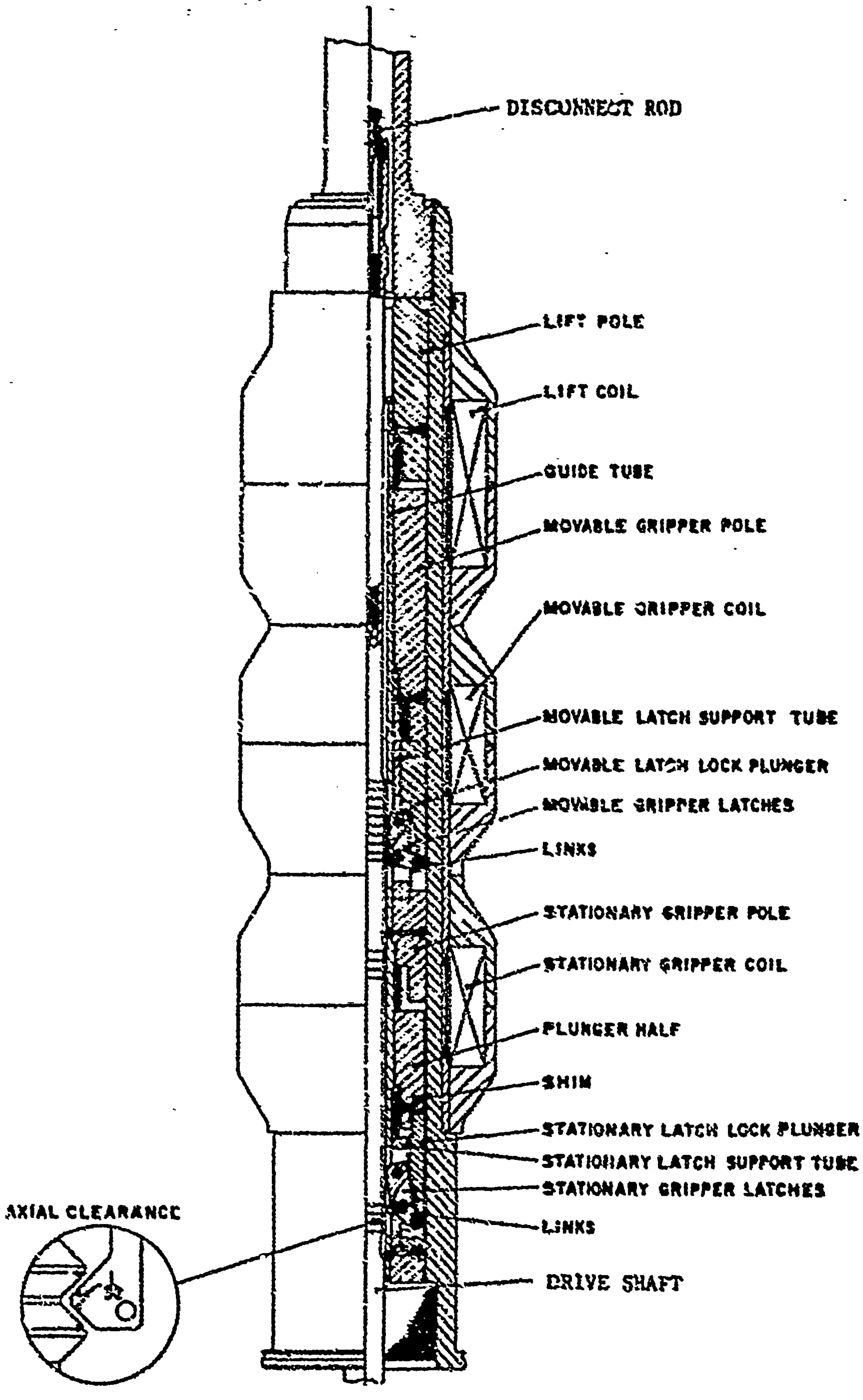
FIGURE 3.2.3-12

Burnable Poison



CONTROL ROD DRIVE MECHANISM ASSEMBLY

FIGURE 3.2.3-13



CONTROL ROD DRIVE MECHANISM SCHEMATIC
FIGURE 3.2.3-14

3.3 TESTS AND INSPECTIONS

3.3.1 REACTIVITY ANOMALIES

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burn-up. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated. The methods employed in calculating the reactivity of the core vs. burnup and the reactivity worth of boron vs. burnup are given in Section 3.2.1.

3.3.2 THERMAL AND HYDRAULIC TESTS AND INSPECTIONS

General hydraulic tests on models are used to confirm the design flow distributions and pressure drops (1,2). Fuel assemblies and control and drive mechanisms are also tested. On-site measurements are made to confirm the design flow rates.

Vessel and internals inspections are also reviewed to check such thermal and hydraulic design values as bypass flow. As part of startup physics testing a series of core power distribution measurements are made over the entire range of operation in terms of design control rod configuration by means of the core moveable detector system. These measurements are analyzed and the results compared with the analytical predictions upon which safety analysis is based with regard to both radial and axial power distribution. The design hot channel factors are used as criteria for acceptable results.

3.3.3 CORE COMPONENT TESTS AND INSPECTIONS

To ensure conformance of all materials, components and assemblies to the design requirements, a release point program is established with the assembly manufacture which requires upgrading of all raw materials, special processes, i.e., welding, heat treating, nondestructive testing, etc. and those characteristics of detail parts which directly affect the assembly and alignment of the reactor internals. The upgrading is accomplished by the issuance of an Inspection Release by quality control (QC) after conformance has been verified.

A resident QC representative performs a surveillance/audit program at the manufacturer's facility and witnesses the required tests and inspections and issues the inspection releases. An example is the radiographic examination of the welds joining core barrel shell courses.

Components and materials supplied by Westinghouse to the assembly manufacture are subjected to a similar program. Quality Control engineers develop inspection plans for all raw materials, components and assemblies. Each level of manufacturing is evaluated by a qualified inspector for conformance i.e. witnessing the ultrasonic testing of core plate raw material. Upon completion of specified events, all documentation is audited prior to releasing the material or component for further manufacturing. All documentation and inspection releases are maintained in the Quality Control central records section. All materials are traceable to the mill heat number.

In conclusion a set of "as built" dimensions are taken to verify conformance to the design requirements and assure proper fitup between the reactor internals and the reactor pressure vessel.

Fuel Quality Control

Quality Control philosophy is generally based on the following inspections being performed to a 95% confidence that no more than 5 defects per 100 will be allowed, unless otherwise noted, using either a hypergeometric function with zero defectives for small lots of the latest revision of Mil-105B for large lots. This confidence level has been based on past experience gained during the manufacturing of over 400 metric tons of uranium cores. The following inspections are included.

1) <u>Component Parts</u> - All parts received are inspected to a 95 x 5 confidence level. The characteristics inspected depends upon the component parts and includes dimensional, visual, check audits of test reports, material certification and non-destructive testing such as X-ray and ultrasonic. Westinghouse materials and component specifications specify in detail the inspection to be performed. Hydrostatic and mechanical properties tests are made where applicable.

   All material used in the manufacture of this core was accepted and released by Quality Control.

2) <u>Pellets</u> - Inspection is performed to a 95 x 5 level for the dimensional characteristics such as diameter, length and squareness of ends. Additional visual inspections are performed for cracks, chips and porosity, according to standards established at the beginning of production. These standards are based upon standards used in previous cores, which have in turn served as standards for over 50 million pellets manufactured and used in operating cores. Density is determined in terms of weight per unit length and is plotted on zone charts used in controlling the process. Chemical analyses are taken on a sample basis throughout pellet production.

3) Rod Inspection - Rod inspection consists of the following 100% non-destructive inspection and is based on the experience, specifications, procedures and standards established on previously manufactured and operated cores.

a) Leak Testing - Each rod is tested to a known leak using mass spectrometry with helium being the detectable gas. This is the system used previously on the leak test of over 300,000 rods.

b) X-ray - All fuel rod weld enclosures are X-rayed at 0° and 90° using weld correction forms. X-rays are taken in accord with ASTM E-142 using 2-2T as the basis of acceptance. (This is equivalent to a .010 defect).

c) Dimensional - All rods are dimensionally inspected prior to final release and upgrading. The requirements included such itmes as length, camber, and visual inspection.

This ensures that 100 per cent of the rod welds have been checked by several different techniques.

4) Rod Upgrading - The rods upon final inspection are upgraded and available for fuel assembly loading.

5) Assembly - Inspection consists of 100 per cent inspection for critical dimensions and for assembly envelope. In addition, sample (95 x 5) inspection will be performed for channel spacing to assure minimum gap on critical channels.

6) Other Inspection - The following inspection will be performed as part of routine inspection operation:

a) Tool and gage inspection and control including standardization to primary and secondary working standards. Tool inspection is performed at prescribed intervals on all serialized tools. Complete records are kept of calibration and condition of tools and are available upon request.

b) Check audit inspection of all inspection activities and records to assure that prescribed methods are followed and that all records are correct and properly maintained.

c) Surveillance of outside contractors, including approval of standards and methods are performed where necessary. However, all final acceptance is based upon inspection performed at the Westinghouse plant.

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, meticulous process control is exercised.

The UO_2 powder is received from the supplier in sealed containers, the contents of which are fully identified both by descriptive tagging and preselected color coding. A single enrichment only is received per shipment.

Upon receipt, an additional Westinghouse identification tag completely describing the contents is affixed to the containers before transfer to segregated powder storage, where containers of different enrichment powders are never mixed.

Powder withdrawal from storage can be made by one authorized group only who direct the powder to the correct pellet production line. All pellet production lines are physically separated from each other and pellets of only a single enrichment and density are produced in a given production line.

Finished pellets are placed on trays having the same color code as the powder containers and transferred to segregared storage racks. Physical barriers prevent mixing of pellets of different densities and enrichments in this storage area. Unused powder and substandard pellets to be reprocessed are returned to storage in the original color coded containers.

Loading of the pellets into the cladding is again accomplished in isolated production lines and again only one density and enrichment is loaded on a line at a time.

At the time of loading, the top fuel tube end plug identification character is checked with the density and enrichment identification of the color code of the pellet storage tray. After each fuel tube is seal welded, it is given the same color coding as has been carried throughout the previous processes. The fuel tube remains color coded until just prior to installation in the fuel assembly. The color coding and end plug identification character, therefore, provide a cross reference of the fuel contained in the fuel rods.

At the time of installation into an assembly, the color coding is removed. After the fuel rods are installed, an inspector verifies that all fuel rods in an assembly have the same end plug identification, and that the top nozzle to be used on the assembly carries the correct identification character describing the fuel enrichment and density for the core region being fabricated. The top nozzle identification then becomes the permanent description of the fuel contained in the assembly.

REFERENCES, SECTION 3.3

(1) G. Hetsroni, "Hydraulic Tests of the SanOnofre Reactor Model," WCAP-3269-8, 1964.

(2) G. Hetsroni, "Studies of the Connecticut-Yankee Hydraulic Model", WCAP-2761, 1965.

APPENDIX 3A

Experimental Verification of Calculations for Boron Burnable Poison Rods

A number of experiments were performed at the Westinghouse Reactor Evaluation Center to investigate the reactivity worth of pyrex glass tubing similar to that employed in the Indian Point II core as burnable poison rods. Several configurations with and without glass burnable poison rods and with fuel loadings representative of power reactors were tested. The reactor used was a rectangular core four feet high with 29 or 30 fuel rods on a side. In each case the water height was adjusted until the reactor was just critical.

Analyses were performed for each of the configurations measured to determine the adequacy of the methods used to calculate burnable poison rod worths in the design of the Indian Point core. The results of the calculations for the different experimental configurations are listed in Table 3A-1. In each case the eigenvalue should be compared to the appropriate reference eigenvalue (core with fuel only) to eliminate the systematic bias which appears in the clean core calculation.

The descrepancy between the eigenvalue calculated for the unpoisoned and poisoned cases has been related to the fractional error in the neutron current into the boron. This error is also given in Table 3A-1. The burnable poison rods used in Indian Point II correspond to the thick walled tubes and in these cases the agreement is generally better than 5%.

TABLE 3.A-1

REACTIVITY DATA OF GLASS RODS ENCASED IN .020" SS .395"ID, .435" OD

NOMINAL GLASS ROD DATA: SOLID ROD: .396" OD, L = 48" : THICK WALL ID = .228", OD = .375", L = 48"
THIN WALL ID = .316", OD = .394", L - 48"

Case	Configuration	Loading	Just Crit. Water Ht cm	Calculated K_{eff} for Critical Core	Error In Poison Absorption %
1	No Inserts - Clean	25^2 = 625	64.10		
2	One Solid Glass Rod	25^2-1 = 624	70.00		
3	No Inserts Clean Core	29^2 = 841	43.25	.996915*	
4	Uniform Thin Wall Glass	29^2-25 = 816	69.61	.994633	4.6
5	Uniform Thick Wall Glass	29^2-25 = 816	83.81	.9948[illegible]	3.5
6	Central Assembly Pattern - 72 Water Holes	29^2-72 = 769	39.52	1.000085	
7	Central Assembly Pattern - 36 Thin Wall Glass 36 Water Holes	29^2-72 = 769	83.76	.996162	1.4
8	Central Assembly Pattern - 36 Thick Wall Glass 36 Water Holes	29^2-72 = 769	114.95	.996392	0.8
9	4 Water Holes Array	29^2-16 = 825	42.25		
10	No Inserts - Clean	30^2 = 900	40.59	.996596*	

*Reference Case

TABLE 3A-1 (Continued)

Case	Configuration	Loading	Just Crit. Water Ht cm	Calculated K_{eff} for Critical Core	Error in Poison Absorption, %
11	Uniform Water Holes	30^2-36 = 864	38.91	.99777	
12	Uniform Voids (SS Cladding)	30^2-36 = 864	42.62		
13	Uniform Thin Wall Glass Tubes	30^2-36 = 864	67.18	.993915	4.8
14	Uniform Thick Wall Glass Tubes	30^2-36 = 864	82.56	.994875	2.5
15	Uniform - .260" Ag-In-Cd	30^2-32 = 868	97.34		
16	Alternate Thin Wall Glass Tubes	30^2-18 = 882	49.42	.994963	6.2
17	Alternate Thick Wall Glass Tubes	30^2-18 = 882	51.90	.994965	5.1
18	Alternate Solid Glass Rods - Bare	30^2-18 = 882	54.91	.995925	1.8
19	Alternate Pattern - Ag-In-Cd (.260)	30^2-18 = 882	53.89		
20	Alternate Pattern - Ag-In-Cd (.330)	30^2-18 = 882	59.94		
21	Four Assembly Pattern - Water Holes	30^2-80 = 820	37.13	.999352	
22	Four Assembly Pattern - 40 Thick Wall Glass 40 Water Holes	30^2-80 = 820	68.73	.997573	1.8

3. Continuous monitoring - and appropriate alarm functions - of both axial and diametral power tilts, using signals from the eight out-of-core ion chambers, with additional information provided by the core exit thermocouples and moveable in-core flux detectors.

4. Provisions of part-length control rods as power shaping devices and automatic trip set point reduction with excessive axial power tilts (top-to-bottom offset). Since the core is expected to be X-Y stable, automatic protection against diametral transients is not required. However, an alarm function is provided to alert the operator to the existence of such tilts before a limiting value on diametral power tilt is reached.*

5. Control rod cluster malpositioning even under the most limiting case will not lead to a DNBR 1.3 at operating conditions. Means for detecting such a misalignment are also provided.

II. Spatial Xenon Stability

A. Axial Xenon Stability

The potential existence of axial power distribution anomalies due to xenon redistribution have been reported in WCAP-7208[6]. Results of these studies have shown that the reactor will be unstable toward xenon oscillations in this dimension; consequently, power shaping devices (i.e., part-length control rods) and automatic protection (i.e., trip set point reduction with excessive axial power imbalance) are provided. Operating philosophy and procedures for monitoring and controlling axial power anomalies have been described in WCAP-7208[6]. The primary means

---

* Stability toward diametral oscillations will have been verified at startup (with corrective action in the unlikely event these prove necessary). As burnup progresses, the reactor becomes increasingly stable toward diametral oscillations due to the decreasing soluble boron concentration and hence the continuously increasing moderator temperature coefficient feedback effect.

of detecting axial power distortions will be by means of the out-of-core ion chambers with appropriate operator display signals. Tests in the Connecticut Yankee reactor (see WCAP-9010[7] & 7407-L[5]) have verified the capability of these out-of-core ion chambers to detect significant axial power imbalances.

B. Diametral Xenon Stability

Results of the analytical investigations (primarily three-dimensional transient analyses reported in WCAP-7407-L[5] and WCAP-3680-22)[3] indicate that the Indian Point #2 reactor will be stable toward diametral xenon oscillations; consequently, X-Y control rods are not required. Comparison with experimental results in the Connecticut Yankee reactor tend to confirm the validity of the less conservative calculations (see WCAP-7407-L[5], Figure 3-1). Despite the expected absence of diametral xenon instability, a test will be performed at startup to demonstrate that artificially induced diametral oscillations decrease in amplitude as a function of time. Furthermore, extensive monitoring with appropriate display and alarm function is provided to alert the operators in the event a diametral power tilt should develop in the course of reactor operation. Consequently, no automatic safety protection against diametral xenon instability is required.

C. Analytical Techniques

In assessing potential power distribution anomalies arising from spatial xenon redistribution, primary reliance has been placed on time-dependent two-group diffusion calculations in three-dimensions including pointwise feedback effects due to coolant density and fuel pellet temperature changes. Means of incorporating the reactivity feedback effects are described in WCAP-3680-21 & 22[2,3] using semi-empirically fitted expressions whose coefficients were determined by other calculations (e.g., LEOPARD). In some cases, survey calculations were performed in one or two dimensions using both digital and modal techniques (see WCAP-3680-20),(1) to indicate trends and to identify the significance and relative importance of the various contributing parameters.

In performing three-dimensional time dependent stability analyses, standard design techniques (i.e., the LEOPARD Code) were used to compute the effect of the various feedback parameters on local reactivity. These results were fitted by a semi-empirical expression as described in Section 2.2 and 3.3 of WCAP-3680-21(2). These analytical fits, with appropriate coefficients as determined from LEOPARD type calculations, were then used in the three-dimensional spatial power calculations, which included coupled thermal hydraulic effects.

D. Instrumentation and Control

Instrumentation and appropriate display is provided to assure that reactor will be maintained within thermal limits (design hot nuclear channel factors) in the presence of power distribution anomalies caused by time dependent xenon redistribution. Primary reliance placed on the eight out-of-core ion chambers supplemented by information derived from the core exit thermocouples and from the movable in-core fission chambers.

The operator will have the out-of-core detector information available, backed up by the core exit thermocouples and the movable in-core detector readouts. The following out-of-core detector information is provided for the operator to alert him to the existence of any core instabilities, axial or diametral:

a) Four indicators, which indicate the difference between the top and bottom detectors. These signals will initiate runback and alarm.

b) Eight indicators which read out the individual currents of the four top and four bottom detectors.

c) One alarm for the four top detectors, when the maximum to average flux is exceeded.

d) One alarm for the four bottom detectors, when the maximum to average flux is exceeded.

e) Four 2 pen recorders - 2 detectors at 180° are on the same recorder.

f) Two 2 pen recorders - total current, i.e., combined top and bottom detector outputs.

g) One total current deviation alarm, i.e., when any one top and bottom total current deviates by a pre-set amount from the other three total current outputs, the operator is alerted to this condition.

With these indications and alarms, the operator has many cross-checks and comparisons available to him. Failure of one top or bottom detector will provide the operator with instant indication and alarm. The out-of-core detectors, backed by the core exit thermocouples and the movable in-core detectors, provide more than adequate information so that fixed in-core flux detectors are not required for operation or safety. Operation with one out-of-core ion chamber out of service does not compromise the safety of the plant.

III. Control Rod Positioning

Normal control rod operations have been described in Section 3.0 of the Indian Point Unit No. 2 FSAR. A deviation in the position of one or more control clusters relative to the position of the control bank can potentially lead to:

(i) Asymmetric fuel depletion,

(ii) Reduction in shutdown margin, or

(iii) Reduction in DNB margin.

Rod misalignment is not a safety problem which requires automatic protection because (i) asymmetric fuel depletion could possibly lead to unacceptable power distributions, but only if the condition were to persist for many hundreds of hours, (ii) misalignment of sufficient magnitude to consume the standard 1% Δk shutdown is not possible, because it would require an entire control bank to be several feet below the desired position; the complete misalignment of a single control cluster will reduce trip reactivity by not more than 0.2% Δk; and, (iii) misalignment of a single control cluster by as much as the entire height of the core with the most pessimistic xenon spatial distribution will not result in a DNBR less than 1.30 at operating conditions. Deviation of 15 inches will not result in a power distribution worse than design.

Misalignment of a rod is most limiting when the last control group (which may be partly inserted at full power) is fully inserted but one cluster is full-out. It has been shown for Indian Point 2 (see WCAP-7407-L,[5] Westinghouse Proprietary) that this case cannot lead to DNBR less than 1.3 at operating conditions even with the worst possible xenon distribution and the control bank (less one cluster) fully inserted.

Each control cluster has its own Position Indicator Channel, with a resolution of at least as good as ± 7.5 inches. Using the worst arrangement of such errors, the displacement of any control cluster from its bank could not exceed 15 inches without detection. A misalignment of 15 inches of any rod in any situation cannot cause design power distribution limits to be exceeded.

The Rod Position Indication System is the primary source of rod position information, but additional means, viz., ex-core ion chambers, thermocouples, and movable in-core fission chambers, are available.

Except for the central control rod cluster, a power tilt will result from any significant control rod misalignment and such a power tilt would be detected by the out-of-core ion chambers. In addition, the core exit thermocouples (as indicated in WCAP-7407-L[5] and 9010)[7] can be used to indicate the existance of a misalignment of the central control rod cluster as well as the location of a misaligned asymmetric control rod cluster. The movable in-core fission chamber system can also be used to detect and/or investigate a suspected control rod malpositioning.

REFERENCES

1. C. C. Poncelet and A. H. Christie, Xenon-Induced Spatial Instabilities in Large Pressurized Water Reactors, USAEC Report WCAP-3680-20, Westinghouse Electric Corp., March 1968.

2. F. B. Skogen and A. F. McFarlane, Control Procedures for Xenon-Induced X-Y Instabilities in Large Pressurized Water Reactors, USAEC Report WCAP-3680-21, Westinghouse Electric Corp., Feb., 1969.

3. F. B. Skogen and A. F. McFarlane, Xenon-Induced Spatial Instabilities in Three-Dimensions, USAEC Report WCAP-3680-22, Westinghouse Electric Corp., September 1969.

4. A. M. Christie et al., Control of Xenon Instabilities in Large Pressurized Water Reactors, Summary Report, USAEC Report WCAP-3680-23, Westinghouse Electric Corp., September 1969.

5. R. F. Barry et al., Power Maldistribution Investigations, Report WCAP-7407-L (Proprietary Class 2), Westinghouse Electric Corp.

6. Westinghouse Electric Corp., Power Distribution Control of Westinghouse Pressurized Water Reactors, Report WCAP-7208 (APD Proprietary Class 2), September 1968.

7. R. J. Johnson, Connecticut-Yankee Tests on Detection of Power Maldistribution, Report WCAP-9010 (NES Proprietary Class 2), Westinghouse Electric Corp., February 1969.

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4. REACTOR COOLANT SYSTEM

The Reactor Coolant System, shown in the Flow Diagram, Figure 4.2-1, consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and a steam generator. The system also includes a pressurizer, pressurizer relief tank, connecting piping, and instrumentation necessary for operational control.

4.1 DESIGN BASES

4.1.1 PERFORMANCE OBJECTIVES

The Reactor Coolant System transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator. Demineralized light water is circulated at the flow rate and temperature consistent with achieving the reactor core thermo-hydraulic performance presented in Section 3. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber used in chemical shim control.

The Reactor Coolant System provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits to acceptable values its uncontrolled release to the secondary system and to other parts of the plant under conditions of either normal or abnormal reactor behavior. During transient operation the systems heat capacity attenuates thermal transients generated by the core or extracted by the steam generators. The Reactor Coolant System accommodates coolant volume changes within the protection system criteria.

By appropriate selection of the inertia of the reactor coolant pumps, the thermal-hydraulic effects are reduced to a safe level during the pump coastdown which would result from a loss-of-flow situation. The layout of the system assures the natural circulation capability following a loss of flow to permit decay heat removal without overheating the core. Part of the systems piping is used by the Safety Injection System to deliver cooling water to the core during a loss-of-coolant accident.

4.1.2 GENERAL DESIGN CRITERIA

General design criteria which apply to the Reactor Coolant System are given below.

Quality Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required. (GDC 1)

The Reactor Coolant System is of primary importance with respect to its safety function in protecting the health and safety of the public.

Quality standards of material selection, design, fabrication and inspection conform to the applicable provisions of recognized codes and good nuclear practice (Section 4.1.7). Details of the quality assurance programs, test procedures and inspection acceptance levels are given in Sections 4.3.1 and 4.5. Particular emphasis is placed on the assurance of quality of the reactor vessel to obtain material whose properties are uniformly within tolerances appropriate to the application of the design methods of the code.

Performance Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design. (GDC 2)

All piping, components and supporting structures of the Reactor Coolant System are designed as Class I equipment i.e. they are capable of withstanding:

(a) The design seismic ground acceleration within code allowable working stresses.

(b) The maximum potential seismic ground acceleration acting in the horizontal and vertical direction simultaneously with no loss of function.

Details are given in Section 4.1.3.

The Reactor Coolant System is located in the containment whose design, in addition to being a Class I structure, also considers accidents or other applicable natural phenomena. Details of the containment design are given in Section 5.

Records Requirements

Criterion: The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public. (GDC 5)

Records of the design, of the major Reactor Coolant System components and the related engineered safety features components are maintained in the offices of Consolidated Edison and will be there throughout the life of the plant.

Records of fabrication are maintained in the manufacturers' plants as required by the appropriate Code, or other requirements pending submittal to Westinghouse or Consolidated Edison. They are available at any time to Consolidated Edison throughout the life of the plant. Construction records are available at the construction site and in the offices of Consolidated Edison where they will be retained for the life of the plant.

Missile Protection

Criterion: Adequate protection for those engineered safety features, the failures of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40)

The dynamic effects during blowdown following a loss-of-coolant accident are evaluated in the detailed layout and design of the high pressure equipment and barriers which afford missile protection. Support structures are designed with consideration given to fluid and mechanical thrust loadings.

The steam generators are supported, guided and restrained in a manner which prevents rupture of the steam side of a generator, the steam lines and the feedwater piping as a result of forces created by a Reactor Coolant System pipe rupture. These supports, guides and restraints also prevent rupture of the primary side of a steam generator as a result of forces created by a steam or feedwater line rupture.

The mechanical consequences of a pipe rupture are restricted by design such that the functional capability of the engineered safety features is not impaired.

4.1.3 PRINCIPAL DESIGN CRITERIA

The criteria which apply solely to the Reactor Coolant System are given below.

Reactor Coolant Pressure Boundary

Criterion: The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. (GDC 9)

The Reactor Coolant System in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and maintain the stresses within applicable code stress limits.

Fabrication of the components which constitute the pressure retaining boundary of the Reactor Coolant System is carried out in strict accordance with the applicable codes. In addition there are areas where equipment specifications for Reactor Coolant System components go beyond the applicable codes. Details are given in Section 4.5.1.

The materials of construction of the pressure retaining boundary of the Reactor Coolant System are protected by control of coolant chemistry from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime.

System conditions resulting from anticipated transients or malfunctions are monitored and appropriate action is automatically initiated to maintain the required cooling capability and to limit system conditions so that continued safe operation is possible.

The system is protected from overpressure by means of pressure relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code.

Isolatable sections of the system are provided with overpressure relieving devices discharging to closed systems such that the system code allowable relief pressure within the protected section is not exceeded.

Monitoring Reactor Coolant Leakage

Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (GDC 16)

Positive indications in the control room of leakage of coolant from the Reactor Coolant System to the containment are provided by equipment which permits continuous monitoring of containment air activity and humidity, and of runoff from the condensate collecting pans under the cooling coils of the containment air recirculation units. This equipment provides indication of normal background which is indicative of a basic level of leakage from primary systems and components. Any increase in the observed parameters is an indication of change within the containment, and the equipment provided is capable of monitoring this change. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, condensate runoff and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

Further details are supplied in Section 4.2.7.

Reactor Coolant Pressure Boundary Capability

Criterion: The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition. (GDC 33)

The reactor coolant boundary is shown to be capable of accommodating without further rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection. Details of this analysis are provided in Section 14.

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution, only the rod cluster control assemblies in the controlling groups are inserted in the core at power, and at full power these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to the operator to assure that this condition is met.

By using the flexibility in the selection of control rod groupings, radial locations and position as a function of load, the design limits the maximum fuel temperature for the highest worth ejected rod to a value which precludes any resultant damage to the primary system pressure boundary, i.e., gross fuel dispersion in the coolant and possible excessive pressure surges.

The failure of a rod mechanism housing causing a rod cluster to be rapidly ejected from the core is evaluated as a theoretical, though not a credible accident. While limited fuel damage could result from this hypothetical event, the fission products are confined to the Reactor Coolant System and the reactor containment. The environmental consequences of rod ejection are less severe than from the hypothetical loss of coolant, for which public health and safety is shown to be adequately protected. Reference is made to Section 14.

Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

Criterion: The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes. (GDC 34)

The reactor coolant pressure boundary is designed to reduce to an acceptable level the probability of a rapidly propagating type failure.

In the core region of the reactor vessel it is expected that the notch toughness of the material will change as a result of fast neutron exposure. This change is evidenced as a shift in the Nil Ductility Transition Temperature (NDTT) which is factored into the operating procedures in such a manner that full operating pressure is not obtained until the affected vessel material is above the Design Transition Temperature (DTT) and in the ductile material region. The pressure during startup and shutdown at the temperature below NDTT is maintained below the threshold of concern for safe operation.

The DTT is a minimum of NDTT plus 60°F and dictates the procedures to be followed in the hydrostatic test and in station operations to avoid excessive cold stress. The value of the DTT is increased during the life of the plant as required by the expected shift in NDTT, and as confirmed by the experimental data obtained from irradiated specimens of reactor vessel materials during the plant lifetime. Further details are given in Section 4.1.6.

All pressure - containing components of the reactor coolant system are designed, fabricated, inspected and tested in conformance with the applicable codes. Further details are given in Section 4.1.7.

Reactor Coolant Pressure Boundary Surveillance

Criterion: Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided. (GDC 36)

The design of the reactor vessel and its arrangement in the system provides the capability for accessibility during service life to the entire internal surfaces of the vessel and certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete.

Monitoring of the Nil Ductility Transition Temperature properties of the core region plates forgings, weldments and associated heat treated zones are performed in accordance with ASTM E185 (Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors). Samples of reactor vessel plate materials are retained and catalogued in case future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The fracture mechanics specimens are the Wedge Opening Loading (WOL) type specimens. The observed shifts in NDTT of the core region materials with irradiation will be used to confirm the calculated limits to startup and shutdown transients.

To define permissible operating conditions below DTT, a pressure range is established which is bounded by a lower limit for pump operation and an upper limit which satisfies reactor vessel stress criteria. To allow for thermal stresses during heatup or cooldown of the reactor vessel, an equivalent pressure limit is defined to compensate for thermal stress as a function of rate of change of coolant temperature. Since the normal operating temperature of the reactor vessel is well above the maximum expected DTT, brittle fracture during normal operation is not considered to be a credible mode of failure.

4.1.4 DESIGN CHARACTERISTICS

Design Pressure

The Reactor Coolant System design and operating pressure together with the safety, power relief and pressurizer spray valves set points, and the protection system set point pressures are listed in Table 4.1-1. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics. The design pressures and data for the respective system components are listed in Tables 4.1-2 through 4.1-6. Table 4.1-7 gives the design pressure drop of the system components.

Design Temperature

The design temperature for each component is selected to be above the maximum coolant temperature in that component under all normal and anticipated transient load conditions. The design and operating temperatures of the respective system components are listed in Tables 4.1-2 through 4.1-6.

Seismic Loads

The seismic loading conditions are established by the "design earthquake" and "maximum potential earthquake". The former is selected to be typical of the largest probable ground motion based on the site seismic history. The latter is selected to be the largest potential ground motion at the site based on seismic and geological factors and their uncertainties.

For the "design earthquake" loading condition, the nuclear steam supply system is designed to be capable of continued safe operation. Therefore, for this loading condition critical structures and equipment needed for this purpose are required to operate within normal design limits. The aseismic design for the "maximum potential earthquake" is intended to provide a margin in design that assures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that the Reactor Coolant System components do not lose their capability to perform their safety function. This has come to be referred to as the "no-loss-of-function" criteria and the loading condition as the "no-loss-of-function earthquake" loading condition.

The criteria adopted for allowable stresses and stress intensities in vessels and piping subjected to normal loads plus seismic loads are defined in Appendix A.

These criteria assure the integrity of the Reactor Coolant System under seismic loading.

For the combination of normal and design earthquake loadings, the stresses in the support structures are kept within the limits of the applicable codes.

For the combination of normal and no-loss-of-function earthquake loadings the stresses in the support structures are limited to values as necessary to assure their integrity and to maintain the stresses in the Reactor Coolant System components within the allowable limits as previously established.

4.1.5 CYCLIC LOADS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trip, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes and the buses thereof are given in Table 4.1-8. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 4.4.1. The cycles are estimated for equipment design purposes (40 year life) and are not intended to be an accurate representation of actual transients or actual operating experience. For example the number of cycles for plant heatup and cooldown at 100°F per hour was selected as a conservative estimate based on an evaluation of the expected requirements. The resulting number, which averages five heatup and cooldown cycles per year, could be increased significantly; however, it is the intent to represent a conservative realistic number rather than the maximum allowed by the design.

Although loss of flow and loss of load transients are not included in the tabulation since the tabulation is only intended to represent normal design transients, the effect of these transients have been analytically evaluated and are included in the fatigue analysis for primary system components.

Over the range from 15% full power up to and including but not exceeding 100% of full power, the Reactor Coolant System and its components are designed to accommodate 10% of full power step changes in plant load and 5% of full power per minute ramp changes without reactor trip. The

Reactor Coolant System will accept a complete loss of load from full power with reactor trip. In addition, the turbine bypass and steam dump system makes it possible to accept a step load decrease of 50% of full power without reactor trip.

4.1.6 SERVICE LIFE

The service life of Reactor Coolant System pressure components depends upon the end-of-life material radiation damage, unit operational thermal cycles, quality manufacturing standards, environmental protection, and adherence to established operating procedures.

The reactor vessel is the only component of the Reactor Coolant System which is exposed to a significant level of neutron irradiation and it is therefore the only component which is subject to material radiation damage effects.

The NDTT shift of the vessel material and welds, due to radiation damage effects is monitored by a radiation damage surveillance program which conforms with ASTM-18 standards.

Reactor vessel design is based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material, as a result of operations such as leak testing and plant heatup and cooldown.

To establish the service life of the Reactor Coolant System components as required by the ASME (Section III) Boiler and Pressure Vessel Code for Class "A" vessels, the unit operating conditions have been established for the 40 year design life. These operating conditions include the cyclic application of pressure loadings and thermal transients.

The number of thermal and loading cycles used for design purposes are listed in Table 4.1-8.

4.1.7 CODES AND CLASSIFICATIONS

All pressure-containing components of the Reactor Coolant System are designed, fabricated, inspected and tested in conformance with the applicable codes listed in Table 4.1.9.

The Reactor Coolant System is classified as Class I for seismic design, requiring that there will be no loss of function of such equipment in the event of the assumed maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously, when combined with the primary steady state stresses.

TABLE 4.1-1

REACTOR COOLANT SYSTEM PRESSURE SETTINGS

	Pressure, psig
Design Pressure	2485
Operating Pressure (at pressurizer)	2235
Safety Valves	2485
Power Relief Valves	2335
Pressurizer Spray Valves (open)	2260
High Pressure Trip	2385
High Pressure Alarm	2335
Low Pressure Trip	1700
Low Pressure Alarm	2135
Hydrostatic Test Pressure	3110

TABLE 4.1-2

REACTOR VESSEL DESIGN DATA

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure, psig	3110
Design Temperature, °F	650
Overall Height of Vessel and Closure Head, ft-in. (Bottom Head O.D. to top of Control Rod Mechanism Housing)	43-9 11/16
Water Volume, (with core and internals in place, ft^3	4647
Thickness of Insulation, min., in.	3
Number of Reactor Closure Head Studs	54
Diameter of Reactor Closure Head Studs, in.	7
ID of Flange, in.	167 1/16
OD of Flange, in.	205
ID at shell, in.	173
Inlet Nozzle ID, in.	27 1/2
Outlet Nozzle, ID, in.	29
Clad Thickness, min., in.	5/32
Lower Head Thickness, min., in.	8 5/16
Vessel Belt-Line Thickness, min., in.	8 5/8
Closure Head Thickness, in.	7
Reactor Coolant Inlet Temperature, °F	554.8
Reactor Coolant Outlet Temperature, °F	612.6
Reactor Coolant Flow, lb/hr	1.34×10^8

TABLE 4.1-3

PRESSURIZER & PRESSURIZER RELIEF TANK DESIGN DATA

Pressurizer

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3110
Design/Operating Temperature, °F	680/653
Water Volume, Full Power ft^3*	1080
Steam Volume, Full Power ft^3	720
Surge Line Nozzle Diameter, in/Pipe Schedule	14/Sch 140
Shell ID, in/Calculated Minimum Shell Thickness, in	84/4.1
Minimum Clad Thickness, in.	0.188
Electric Heaters Capacity, kw	1800
Heatup rate of Pressurizer using Heaters only, °F/hr	55 (approximately)

* 60% of net internal volume

Power Relief Valves	
Number	2
Set Pressure (open), psig	2335
Capacity, lb/hr Saturated steam/valve	179,000
Safety Valves	
Number	3
Set Pressure, psig	2485
Capacity, lb/hr Saturated steam/valve	408,000

Pressurizer Relief Tank

Design pressure, psig	100
Rupture Disc Release Pressure, psig	100
Design temperature, °F	340
Normal water temperature, °F	Containment Ambient
Total volume, Ft^3	1800
Rupture Disc Relief Capacity, lb/hr	1.224×10^6

TABLE 4.1-4

STEAM GENERATOR DESIGN DATA

Number of Steam Generators		4
Design Pressure, Reactor Coolant/Steam, psig		2485/1085
Reactor Coolant Hydrostatic Test Pressure (tube side-cold), psig		3110
Design Temperature, Reactor Coolant/Steam, °F		650/600
Reactor Coolant Flow, lb/hr		33.5×10^6
Total Heat Transfer Surface Area, ft^2		44,430
Heat Transferred, Btu/hr		2631×10^6
Steam Conditions at Full Load, Outlet Nozzle:		
Steam Flow, lb/hr		3.315×10^6
Steam Temperature, °F		513.8
Steam Pressure, psig		755
Feedwater Temperature, °F		427.2
Overall Height, ft-in.		63 - 1.625
Shell OD, upper/lower, in.		166/127.5
Shell Thickness, upper/lower, in		3.5/2.63
Number of U-tubes		3260
U-tube Diameter, in		0.875
Tube Wall Thickness, (average), in		0.050
Number of manways/ID, in		3/16
Number of handholes/ID, in		2/6
	2758 MWt	Zero Power
Reactor Coolant Water Volume, ft^3	924	924
Primary Side Fluid Heat Content, Btu	24.18×10^6	23.62×10^6
Secondary Side Water Volume, ft^3	1613	2447
Secondary Side Steam Volume, ft^3	2966	2132
Secondary Side Fluid Heat Content, Btu	44.4×10^6	67.4×10^6

TABLE 4.1-5

REACTOR COOLANT PUMPS DESIGN DATA

Number of Pumps	4
Design Pressure/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3110
Design Temperature (casing), °F	650
RPM at Nameplate Rating	1189
Suction Temperature, °F	555
Net Positive Suction Head, ft	170
Developed Head, ft	272
Capacity, gpm	89,700
Seal Water Injection, gpm	8
Seal Water Return, gpm	3
Pump Discharge Nozzle ID, in	27 1/2
Pump Suction Nozzle, ID, in.	31
Overall Unit Height, ft.	28.38
Water Volume, ft^3	192
Pump-Motor Moment of Inertia, $lb\text{-}ft^2$	82,000
Motor Data:	
Type	AC Induction Single Speed, Air Cooled
Voltage	6600
Insultation Class	B Thermalastic Epoxy
Phase	3
Frequency, cps	60
Starting	
Current, amp	2950
Input (hot reactor coolant), kw	4250
Input (cold reactor coolant), kw	5600
Power, HP (nameplate)	6000

TABLE 4.1-6

REACTOR COOLANT PIPING DESIGN DATA

Reactor Inlet Piping, ID, in.	27 1/2
Reactor Inlet Piping, nominal thickness, in.	2.375
Reactor Outlet Piping, ID, in.	29
Reactor Oulet Piping, nominal thickness, in.	2.50
Coolant Pump Suction Piping, I.D., in.	31
Coolant Pump Suction Piping, nominal thickness, in.	2.625
Pressurizer Surge Line Piping, I.D., in.	14
Pressurizer Surge Line Piping, nominal thickness, in.	1.25
Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure, (cold) psig	3110
Design Temperature, °F	650
Design Temperature, (pressurizer surge line,) °F	680
Water Volume, (all 4 loops including surge line) ft^3	1156

TABLE 4.1-7

REACTOR COOLANT SYSTEM DESIGN PRESSURE DROP

	Pressure Drop, psi
Across Pump Discharge Leg	1.1
Across Vessel, including nozzles	46.7
Across Hot Leg	1.3
Across Steam Generator	32.3
Across Pump Suction Leg	3.0
Total Pressure Drop	84.4

x

TABLE 4.1-8

THERMAL AND LOADING CYCLES

Transient Condition	Design Cycles†
1. plant heatup at 100°F per hour	200 (5/yr*)
2. plant cooldown at 100°F per hour	200 (5/year)
3. plant loading at 5% of full power per minute	14,500 (1/day)
4. plant unloading at 5% of full power per minute	14,500 (1/day)
5. Step load increase of 10% of full power (but not to exceed full power)	2,000 (1/week)
6. step load decrease of 10% of full power	2,000 (1/week)
7. step load decrease of 50% of full power	200 (5/year)
8. reactor trip	400 (10/year)
9. hydrostatic test at 3110 psig pressure, 100°F temperature	5 (pre-operational)
10. hydrostatic test at 2485 psig pressure and 400°F temperature	40 (post-operational)

11. steady state fluctuations - the reactor coolant average temperature for purposes of design is assumed to increase and decrease a maximum of 6°F in one minute. The corresponding reactor coolant pressure variation is less than 100 psig. It is assumed that an infinite number of such fluctuations will occur

† Estimated for equipment design purposes (40-year life) and not intended to be an accurate representation of actual transients or to reflect actual operating experience.

* This transient includes pressurizing to 2235 psig.

TABLE 4.1-9

REACTOR COOLANT SYSTEM - CODE REQUIREMENTS

Component	Codes
Reactor Vessel	ASME III* Class A
Rod Drive Mechanism Housing	ASME III* Class A
Steam Generators	
Tube side	ASME III* Class A
Shell side***	ASME III* Class C
Reactor Coolant Pump Volute	ASME III* Class A
Pressurizer	ASME III* Class A
Pressurizer Relief Tank	ASME III* Class C
Pressurizer Safety Valves	ASME III*
Reactor Coolant Piping	USAS B31.1**
System Valves, Fittings and Piping	USAS B31.1**

* ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

** USAS B31.1 Code for Pressure Piping

*** The shell side of the steam generator conforms to the requirements for Class A vessels and is so stamped as permitted under the rules of Section III.

4.2 SYSTEM DESIGN AND OPERATION

4.2.1 GENERAL DESCRIPTION

The Reactor Coolant System consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator, a pump, loop piping, and instrumentation. The pressurizer surge line is connected to one of the loops. Auxiliary system piping connections into the reactor coolant piping are provided as necessary. A flow diagram of the system is shown in Figure 4.2-1, and a schematic flow diagram denoting principle parameters under normal steady state full power operating conditions is shown on Figure 4.2-2.

The containment boundary shown on the flow diagram indicates those major components which are to be located inside the containment. The intersection of a process line with this boundary indicates a functional penetration.

Reactor Coolant System design data are listed in Table 4.1-2 through 4.1-6. A power level of 100% rated output for 80% of the time is considered an estimate of ideal operation over the service life of the system.

Pressure in the system is controlled by the pressurizer, where water and steam pressure is maintained through the use of electrical heaters and sprays. Steam can either be formed by the heaters, or condensed by a pressurizer spray to minimize pressure variations due to contraction and expansion of the coolant. Instrumentation used in the pressure control system is described in Section 7. Spring-loaded steam safety valves and power-operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank, where the discharged steam is condensed and cooled by mixing with water.

4.2.2 COMPONENTS

Reactor Vessel

The reactor vessel is cylindrical in shape with a hemispherical bottom and a flanged and gasketed removable upper head. The vessel is designed in accordance with Section III (Nuclear Vessels) of ASME Boiler and Pressure Vessel Code. Figure 4.2-3 is a schematic of the reactor vessel The materials of construction of the reactor vessel are given in Table 4.2-1.

Coolant enters the reactor vessel through inlet nozzles in a plane just below the vessel flange and above the core. The coolant flows downward through the annular space between the vessel wall and the core barrel into a plenum at the bottom of the vessel where it reverses direction. Approximately ninety-five per cent of the total coolant flow is effective for heat removal from the core. The remainder of the flow includes the flow through the RCC guide thimbles, the leakage across the outlet nozzles, and the flow deflected into the head of the vessel for cooling the upper flange. All the coolant is united and mixed in the upper plenum, and the mixed coolant stream then flows out of the vessel through exit nozzles located on the same plane as the inlet nozzles.

A one-piece thermal shield, concentric with the reactor core, is located between the core barrel and the reactor vessel. It is attached to the core barrel. The shield, which is cooled by the coolant on its downward pass, protects the vessel by attenuating much of the gamma radiation and some of the fast neutrons which escape from the core. This shield minimizes thermal stresses in the vessel which result from heat generated by the absorption of gamma energy. This protection is further described in Section 3.2.3.

Fourty-eight core instrumentation nozzles are located on the lower head.

The reactor closure head and the reactor vessel flange are joined by fifty-four 7 in. diameter studs. Two metallic O-rings seal the reactor vessel when the reactor closure head is bolted in place. A leakoff connection is provided between the two O-rings to monitor leakage across the inner O-ring. In addition, a leak off connection is also provided beyond the outer O-ring seal.

The vessel is insulated with metallic reflective-type insulation supported from the nozzles. Insulation panels are provided for the reactor closure head which are supported on the refueling seal ledge and vent shroud support rings.

The reactor vessel internals are designed to direct the coolant flow, support the reactor core, and guide the control rods in the withdrawn position. The reactor vessel contains the core support assembly, upper plenum assembly, fuel assemblies, control cluster assemblies, surveillance specimens, and in-core instrumentation.

Surveillance specimens made from reactor vessel steel are located between the reactor vessel wall and the thermal shield. These specimens will be examined at selected intervals to evaluate reactor vessel material NDTT changes as described in Section 4.5.1.

The reactor internals are described in detail in Section 3.2.3 and the general arrangement of the reactor vessel and internals is shown in Figure 3.2.3-2.

Reactor vessel design data is listed in Table 4.1-2.

Pressurizer

The general arrangement of the pressurizer is shown in Figure 4.2-4, and the design characteristics are listed in Table 4.1-3.

The pressurizer maintains the required reactor coolant pressure during steady-state operation, limits the pressure changes caused by coolant thermal expansion and contraction during normal load transients, and prevents the pressure in the Reactor Coolant System from exceeding the design pressure.

The pressurizer contains replaceable direct immersion heaters, multiple safety and relief valves, a spray nozzle and interconnecting piping, valves and instrumentation. The electric heaters located in the lower section of the vessel regulate the Reactor Coolant System pressure by keeping the water and steam in the pressurizer at saturation temperature. The heaters are capable of raising the temperature of the pressurizer and contents at approximately 55°F/hr during startup of the reactor.

The pressurizer is designed to accommodate positive and negative surges caused by load transients. The surge line which is attached to the bottom of the pressurizer connects the pressurizer to the hot leg of a reactor coolant loop. During a positive surge, caused by a decrease in plant load, the spray system, which is fed from the cold leg of a coolant loop, condenses steam in the vessel to prevent the pressurizer pressure from reaching the set point of the power operated relief valves. The spray valves on the pressurizer are power operated. In addition, the spray valves can be operated manually by a switch in the control room. A small continuous spray flow is provided to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excess cooling of the spray piping.

During a negative pressure surge, caused by an increase in plant load, flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. Heaters are also energized on high water level during positive surges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop.

The pressurizer is constructed of low alloy steel with internal surfaces clad with austenitic stainless steel. The heaters are sheathed in austenitic stainless steel.

The pressurizer vessel surge nozzle is protected from thermal shock by a thermal sleeve. A thermal sleeve also protects the pressurizer spray nozzle connection.

Steam Generators

Each loop contains a vertical shell and U-tube steam generator. A steam generator of this type is shown in Figure 4.2-5. Principal design parameters are listed in Table 4.1-4. The steam generators are designed and manufactured in accordance with Section III (Nuclear Vessels) of the ASME Boiler and Pressure Vessel Code.

Reactor coolant enters the inlet side of the channel head at the bottom of the steam generator through the inlet nozzle, flows through the U-tubes to an outlet channel and leaves the generator through another bottom nozzle. The inlet and outlet channels are separated by a partition. Manways are provided to permit access to the U-tubes and moisture separating equipment. The procedure for plugging a defective steam generator tube is described in Appendix 4C.

Feedwater to the steam generator enters just above the top of the U-tubes through a feedwater ring. The water flows downward through an annulus between the tube wrapper and the shell and then upward through the tube bundle where part of it is converted to steam.

The steam-water mixture from the tube bundle passes through a steam swirl vane assembly which imparts a centrifugal motion to the mixture and separates the water particles from the steam. The water spills over the edge of the swirl vane housing and combines with the feedwater for another pass through the tube bundle.

The steam rises through additional separators which limit the moisture content of the steam to one fourth of one per cent or less under all design load conditions.

The steam generator is constructed primarily of carbon steel. The heat transfer tubes are Inconel. The interior surfaces of the channel heads and nozzles are clad with austenitic stainless steel, and the side of the tube sheet in contact with the reactor coolant is clad with Inconel. The tube to tube sheet joint is welded.

Reactor Coolant Pumps

Each reactor coolant loop contains a vertical single stage centrifugal pump which employs a controlled leakage seal assembly. A view of a controlled leakage pump is shown in Figure 4.2-6 and the principal design parameters for the pumps are listed in Table 4.1-5. The reactor coolant pump estimated performance and NPSH characteristics are shown in Figure 4.2-7. The performance characteristic is common to all of the higher specific speed centrifugal pumps and the 'knee' at about 45% design flow introduces no operational restrictions, since the pumps operate at full speed.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the impeller, discharged through passages in the diffuser and out through a discharge nozzle in the side of the casing. The motor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pumps in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, a second seal which directs the controlled leakage out of the pump, and a third seal which minimizes the leakage of water and vapor from the pump into the containment atmosphere.

A portion of the high pressure water flow from the charging pumps is injected into the reactor coolant pump between the impeller and the controlled leakage seal. Part of the flow enters the Reactor Coolant System through a labyrinth seal in the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A very small amount which leaks through the second seal is also collected and removed from the pump.

Component cooling water is supplied to the motor bearing oil coolers and the thermal barrier cooling coil.

The squirrel cage induction motor driving the pump is air cooled and has oil lubricated thrust and radial bearings. A water lubricated bearing provides radial support for the pump shaft.

An extensive test program has been conducted for several years to develop the controlled leakage shaft seal for pressurized water reactor applications. Long term tests have been conducted on less than full scale prototype seals as well as on full size seals. At the time of initial operation of this plant, operating experience with large size, controlled leakage shaft seal pumps will be available from plants such as San Onofre and Connecticut Yankee.

Pressurizer Relief Tank

Principal design parameters of the pressurizer relief tank are given in Table 4.1-3. It is shown on Figure 4.2-8.

Steam and water discharge from the power relief and safety valves pass to the pressurizer relief tank which is partially filled with water at or near ambient containment conditions. The tank normally contains water in a predominantly nitrogen atmosphere. Steam is discharged under the water level to condense and cool by mixing with the water. The tank is equipped with a spray and a drain to the Waste Disposal System which are operated to cool the tank following a discharge.

The tank size is based on the requirement to condense and cool a discharge equivalent to 110 per cent of the full power pressurizer steam volume.

The tank is protected against a discharge exceeding the design value by two rupture discs which discharge into the reactor containment. The rupture discs on the relief tank have a combined relief capacity equal to the combined capacity of the pressurizer safety valves. The tank design pressure (and the rupture discs setting) is twice the calculated pressure resulting from the maximum safety valve discharge described above. This margin is to prevent deformation of the disc. The tank and rupture disc holder are also designed for full vacuum to prevent tank collapse if the tank contents cool without nitrogen being added.

The discharge piping from the safety and relief valves to the relief tank is sufficiently large to prevent backpressure at the safety valves from exceeding 20 per cent of the set point pressure at full flow.

The pressurizer relief tank, by means of its connection to the Waste Disposal System, provides a means for removing any non-condensable gases from the Reactor Coolant System which might collect in the pressurizer vessel.

The tank is constructed of carbon steel with a corrosion resistant coating on the internal surface.

Piping

The general arrangement of the reactor coolant system piping is shown on the plant layout drawings in Section 1. Piping design data are presented in Table 4.1-6.

The austenitic stainless steel reactor coolant piping and fittings which make up the loops are 29 in. ID in the hot legs, 27-1/2 in. ID in the cold legs and 31 in. ID between each loop's steam generator outlet and its reactor coolant pump suction. The pressurizer relief line, which connects the pressurizer safety and relief valves' outlets to the inlet nozzle flange on the pressurizer relief tank, is constructed of carbon steel.

Smaller piping, including the pressurizer surge and spray lines, drains and connections to other systems are austenitic stainless steel. All piping connections are welded except for flanged connections at the pressurizer relief tank and at the relief and safety valves.

Thermal sleeves are installed at the following locations where high thermal stresses could otherwise develop due to rapid changes in fluid temperature during normal operational transients:

a) Return lines from the residual heat removal loop (safety injection lines).
b) Both ends of the pressurizer surge line.
c) Pressurizer spray line connection to the pressurizer.
d) Charging lines and auxiliary charging line connections.

Valves

All valve surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials. Connections to stainless steel piping are welded.

Valves that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection.

Component Supports

The support structures for the Reactor Coolant System Components are described in Appendix 4B.

4.2.3 PRESSURE-RELIEVING DEVICES

The Reactor Coolant System is protected against overpressure by control and protective circuits such as the high pressure trip and by code relief valves connected to the top head of the pressurizer. The relief valves discharge into the pressurizer relief tank which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Figure 4.2-1, and the valve design parameters are given in Table 4.1-3. Valve sizes are determined as indicated in Section 4.3.4.

Power-operated relief valves and code safety valves are provided to protect against pressure surges which are beyond the pressure limiting capacity of the pressurizer spray.

The pressurizer relief tank is protected against a steam discharge exceeding the design pressure value by two rupture discs which discharge into the reactor containment. The rupture disc relief conditions are given in Table 4.1-3.

4.2.4 PROTECTION AGAINST PROLIFERATION OF DYNAMIC EFFECTS

Engineered Safety Features and associated systems are protected from loss of function due to dynamic effects and missiles which might result from a loss-of-coolant accident. Protection is provided by missile shielding and/or segregation of redundant components. This is discussed in detail in Section 6.

The Reactor Coolant System is surrounded by concrete shield walls. These walls provide shielding to permit access into the containment during full power operation for inspection and maintenance of miscellaneous equipment. These shielding walls also provide missile protection for the containment liner plate.

The concrete deck over the Reactor Coolant System also provides for shielding and missile damage protection.

Lateral bracing is provided near the steam generator upper tube sheet elevation to resist lateral loads, including those resulting from seismic forces and pipe rupture forces. Additional bracing is provided at a lower elevation to resist pipe rupture loads.

Missile protection afforded by the arrangement of the Reactor Coolant System is illustrated in the containment structure drawings which are given in Section 5.

4.2.5 MATERIALS OF CONSTRUCTION

Each of the materials used in the Reactor Coolant System is selected for the expected environment and service conditions. The major component materials are listed in Table 4.2-1.

All reactor coolant system materials which are exposed to the coolant are corrosion-resistant. They consist of stainless steels and Inconel, and they are chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant. The chemical composition of the reactor coolant is maintained within the specification given in Table 4.2.2. Reactor coolant chemistry is further discussed in Section 4.2.8.

The water in the secondary side of the steam generators is held within the chemistry specification given in Table 4.2-3 to control deposits and corrosion inside the steam generators.

The phenomena of stress-corrosion cracking and corrosion fatigue are not generally encountered unless a specific combination of conditions is present. The necessary conditions are a susceptible alloy, an aggressive environment, a stress, and time.

It is characteristic of stress corrosion that combinations of alloy and environment which result in cracking are usually quite specific. Environments which have been shown to cause stress-corrosion cracking of stainless steels are free alkalinity in the presence of a concentrating mechanism and the presence of chlorides and free oxygen. With regard to the former, experience has shown that deposition of chemicals on the surface of tubes can occur in a steam blanketed area within a steam generator. In the presence of this environment, stress-corrosion cracking can occur in stainless steels having the nominal residual stresses resulting from normal manufacturing procedures. However, the steam generator contains Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel Alloy has excellent resistance to general and pitting-type corrosion in severe operating water conditions.

Considerable experience with Inconel in steam generator and heat exchanger applications has been accumulated in the industry. Since 1962, widespread adoption of Inconel for steam generator tubes in nuclear stations is evident: as for example, Connecticut-Yankee; San Onofre; PM-1, Sundance; PM-3A, McMurdo Sound; CVTR; NPD, and Hanford N-Reactor. For these plants, over 150,000 hours of trouble-free service has been accumulated. In none of these plants has there been any evidence of steam generator tube leakage. Materials with lead traces in the overall composition were present in the secondary side of the referenced plants. The use of lead in the materials of the secondary side of this plant has been minimized to the practical limit of that occurring as trace elements in metallurgical alloys and as such is insignificant.

All external insulation of Reactor Coolant System components is compatible with the component materials. The cylindrical shell exterior and closure flanges to the reactor vessel are insulated with metallic reflective insulation. The closure head is insulated with low halide-content insulating material. All other external corrosion-resistant surfaces in the Reactor Coolant System are insulated with low or halide-free insulating material as required.

The reactor vessel plate or forging material opposite the core is purchased to a specified Charpy V-notch test result of 30 ft-lb or greater at a corresponding nil-ductility transition temperature (NDTT) of 40°F or less, and the material is tested to verify conformity to specified requirements and to determine the actual NDTT value. In addition, this plate is 100 per cent volumetrically inspected by ultrasonic test using both longitudinal and shear wave methods.

The remaining material in the reactor vessel, and other Reactor Coolant System components, meets the appropriate design code requirements and specific component function.

The reactor vessel material is heat-treated specifically to obtain good notch-ductility which ensures a low NDTT, and thereby gives assurance that the finished vessel can be initially hydrostatically tested and operated near room temperature within the restrictions of NDTT + 60°F. The stress limits established for the reactor vessel are dependent upon the temperature at which the stresses are applied. As a result of fast neutron irradiation in the region of the core, the material properties will change, including an increase in the NDTT. An initial maximum value of NDTT of 40°F has been established during fabrication in this region.

The techniques used to measure and predict the integrated fast neutron ($E > 1$ Mev) fluxes at the sample locations are described in Appendix 4A. The calculation method used to obtain the maximum neutron ($E > 1$ Mev) exposure of the reactor vessel is identical to that described for the irradiation samples. Since the neutron spectra at the samples and vessel inner surface are identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of reactor vessel for some later stage in plant life. The maximum exposure of the vessel will be obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

The maximum integrated fast neutron ($E > 1$ Mev) exposure of the vessel is computed to be 2.4×10^{19} n/cm^2 for 40 years operation at 80 per cent load factor.

The calculated neutron exposure exceeds the value of 0.85×10^{19} n/cm^2 (E > 1 Mev) reported in the First Supplement to the Preliminary Facility Description and Safety Analysis Report. The reasons for the increase are:

a) Core design considerations leading to the adoption of a radial power distribution which includes an increased energy generation in the peripheral assemblies of the core; and

b) The associated effect on the azimuthal variation of fast neutron fluxes at the reactor vessel inner surface.

The predicted NDTT shift for an integrated fast neutron (E > 1 Mev) exposure of 2.4×10^{19} n/cm^2 is 240°F, the value obtained from the curve shown in Figure 4.2-9 for 550°F irradiation.

To evaluate the NDTT shift of welds, heat affected zones and base material for the vessel, test coupons of these material types have been included in the reactor vessel surveillance program described in Section 4.5.1.

The methods used to measure the initial NDTT of the reactor vessel base plate material are given in Appendix 4A.

4.2.6 MAXIMUM HEATING AND COOLING RATES

The reactor system operating cycles used for design purposes are given in Table 4.1-8 and described in Section 4.1.5. The normal system heating and cooling rate is 50°F/hr. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate, starting with a minimum water level, of 55°F/hr. This rate takes into account the small continuous spray flow provided to maintain the pressurizer liquid homogeneous with the coolant.

The fastest cooldown rates which result from the hypothetical case of a break of a main steam line are discussed in Section 14.

4.2.7 LEAKAGE

The existence of leakage from the Reactor Coolant System to the containment regardless of the source of leakage, is detected by one or more of the following conditions:

a) Two radiation sensitive instruments provide the capability for detection of leakage from the Reactor Coolant System. The containment air particulate monitor is quite sensitive to low leak rates. The containment radiogas monitor is much less sensitive but can be used as a backup to the air particulate monitor.

b) A third instrument used in leak detection is the humidity detector. This provides a means of measuring overall leakage from all water and steam systems within the containment but furnishes a less sensitive measure. The humidity monitoring method provides backup to the radiation monitoring methods.

c) A leakage detection system is included which determines leakage losses from all water and steam systems within the containment including that from the Reactor Coolant System. This system collects and measures moisture condensed from the containment atmosphere by the cooling coils of the main recirculation units. It relies on the principle that all leakages up to sizes permissible with continued plant operation will be evaporated into the containment atmosphere. This system provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary.

d) An increase in the amount of coolant makeup water which is required to maintain normal level in the pressurizer, or an increase in containment sump level are less sensitive menas of detecting leakage.

Leakage detection methods are described in detail and evaluated in Section 6.

Leakage Prevention

Reactor Coolant System components are manufactured to exacting specifications which exceed normal code requirements (as outlined in Section 4.1.6). In addition, because of the welded construction of the Reactor Coolant System and the extensive non-destructive testing to which it is subjected (as outlined in Section 4.5), it is considered that leakage through metal surfaces or welded joints is very unlikely.

However, some leakage from the Reactor Coolant System is permitted by the reactor coolant pump seals. Also, all sealed joints are potential sources of leakage even though the most appropriate sealing device is selected in each case. Thus, because of the large number of joints and the difficulty of assuring complete freedom from leakage in each case, a small integrated leakage is considered acceptable. Leakage from the reactor through its head flange will leak-off between the double O-ring seal and actuate an alarm in the control room.

Locating Leaks

Experience has shown that hydrostatic testing is successful in locating leaks in a pressure containing system.

Methods of leak location which can be used during plant shutdown include visual observation for escaping steam or water or for the presence of boric acid crystals near the leak. The boric acid crystals are transported outside the Reactor Coolant System in the leaking fluid and deposited by the evaporation process. Portable sonic detectors sensitive to ultrasonic frequencies provide another means for locating small leaks.

4.2.8 WATER CHEMISTRY

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of reactor coolant system surfaces.

All materials exposed to reactor coolant are corrosion resistant. Periodic analyses of the coolant chemical composition are performed to monitor the adherence of the system to the reactor coolant water quality listed in Table 4.2-2. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume System and Sampling System which are described in Section 9.

4.2.9 REACTOR COOLANT FLOW MEASUREMENTS

Elbow taps are used in the primary coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. The correlation between flow reduction and elbow tap read out has been well established by the following equation; $\frac{\Delta P}{\Delta P_o} = \left(\frac{\omega}{\omega_o}\right)^{2.0}$, where ΔP_o is the referenced pressure differential 10
with the corresponding referenced flow rate ω_o and ΔP is the pressure differential with the corresponding referenced flow rate ω. The full flow reference point is established during initial plant startup. The low flow trip point is then established by extrapolating along the correlation curve. The technique has been well established in providing core protection against low coolant flow in Westinghouse PWR plants. The expected absolute accuracy of the channel is within $\pm$ 10% and field results have shown the repeatability of the trip point to be within $\pm$ 1%. The analysis of the loss of flow transient presented in Section 14.1.6 assumes instrumentation error of $\pm$ 3%.

TABLE 4.2-1

MATERIALS OF CONSTRUCTION OF THE REACTOR COOLANT SYSTEM COMPONENTS

Component	Section	Materials
Reactor Vessel	Pressure Plate	SA-302, Gr. B
	Shell & Nozzle Forgings	A-508 Class 2
	Cladding, Stainless Weld Rod	Type 304 equivalent
	Thermal Shield and Internals	A-240, Type 304
	Insulation	Stainless Steel, Aluminum
Steam Generator	Pressure Plate	SA-302, Gr. B
	Cladding, Stainless Weld Rod	Type 304 equivalent
	Cladding for Tube Sheets	Inconel
	Tubes	SB-163
	Channel Head Castings	SA-216 WCC
Pressurizer	Shell	SA-302 Gr. B
	Heads	SA-216 WCC
	External Plate	SA-302, Gr. B
	Cladding, Stainless	Type 304 equivalent
	Internal Plate	SA-240 Type 304
	Internal Piping	SA-376 Type 316
Pressurizer Relief Tank	Shell	A-285 Gr. C
	Heads	A-285 Gr.C
	Internal surface coating	Amercoat 55 system

TABLE 4.2-1 (Continued)

Component	Section	Material
Piping	Pipes	A-376 Types 304 and 316
	Fittings	A-351, CF8M
	Nozzles	A-182 Type F316
Pump	Shaft	Type 304
	Impeller	A-351, CF8
	Casing	A-351, CF8M
Valves	Pressure Containing Parts	A-351, CF8 and CF8M and A-182 Type F316

TABLE 4.2-2

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Electrical Conductivity	Determined by the concentration of boric acid and alkali present. Expected range is < 1 to 40 μMhos/cm at 25°C.
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration) at 25°C.
Oxygen, ppm, max.	0.1
Chloride, ppm, max.	0.15
Fluoride, ppm, max.	0.1
Hydrogen, cc (STP)/kg H_2O	25-35
Total Suspended Solids, ppm, max.	1.0
pH Control Agent (Li^7OH)	0.3×10^{-4} to 3.2×10^{-4} molal strong base alkali equivalent to 0.22 to 2.2 ppm Li^7)
Boric Acid as ppm B	Variable from 0 to ∿ 3000

TABLE 4.2-3

STEAM GENERATOR WATER (STEAM-SIDE) CHEMISTRY SPECIFICATION

Total Dissolved Solids, max.	125 ppm
Total Suspended Solids, max.	5 ppm
pH (normal operation), 25°C	9.5 to 10.5
Free Caustic	Zero
Dissolved Oxygen	Essentially Zero (less than 0.005 ppm)
Chlorides, Max.	75 ppm

References for Figure 4.2-9

RADIATION INDUCED INCREASE IN TRANSITION TEMPERATURE FOR A302B STEEL

	References	Material	Temp. °F	Neutron Exposure n/cm^2 (>1 Mev)	ΔNDT °F
1.	NRL Report 6160 Page 12	SA302B	450	5×10^{18}	140
2.	NRL Report 6160 Page 12	SA302B	550	5×10^{18}	65
3.	NRL Report 6160 Page 13	SA302B	490	1.4×10^{19}	200
4.	ASTM-STP 341 Page 226	SA302B	550	6×10^{17}	30**
5.	ASTM-STP 341 Page 226	SA302B	550	6×10^{17}	45
6.	ASTM-STP 341 Page 226	SA302B	550	8×10^{18}	85**
7.	ASTM-STP 341 Page 226	SA302B	550	8×10^{18}	100
8.	ASTM-STP 341 Page 226	SA302B	550	1.5×10^{19}	130**
9.	ASTM-STP 341 Page 226	SA302B	550	1.5×10^{19}	140
10.	NRL Report 6160 Page 6	All Steels	<450	Various	Various
11.	Nuclear Science & Engineering [illegible]-38 (1964)	SA302B	<450	Various	Various
12.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	55[illegible]	3×10^{19}	120

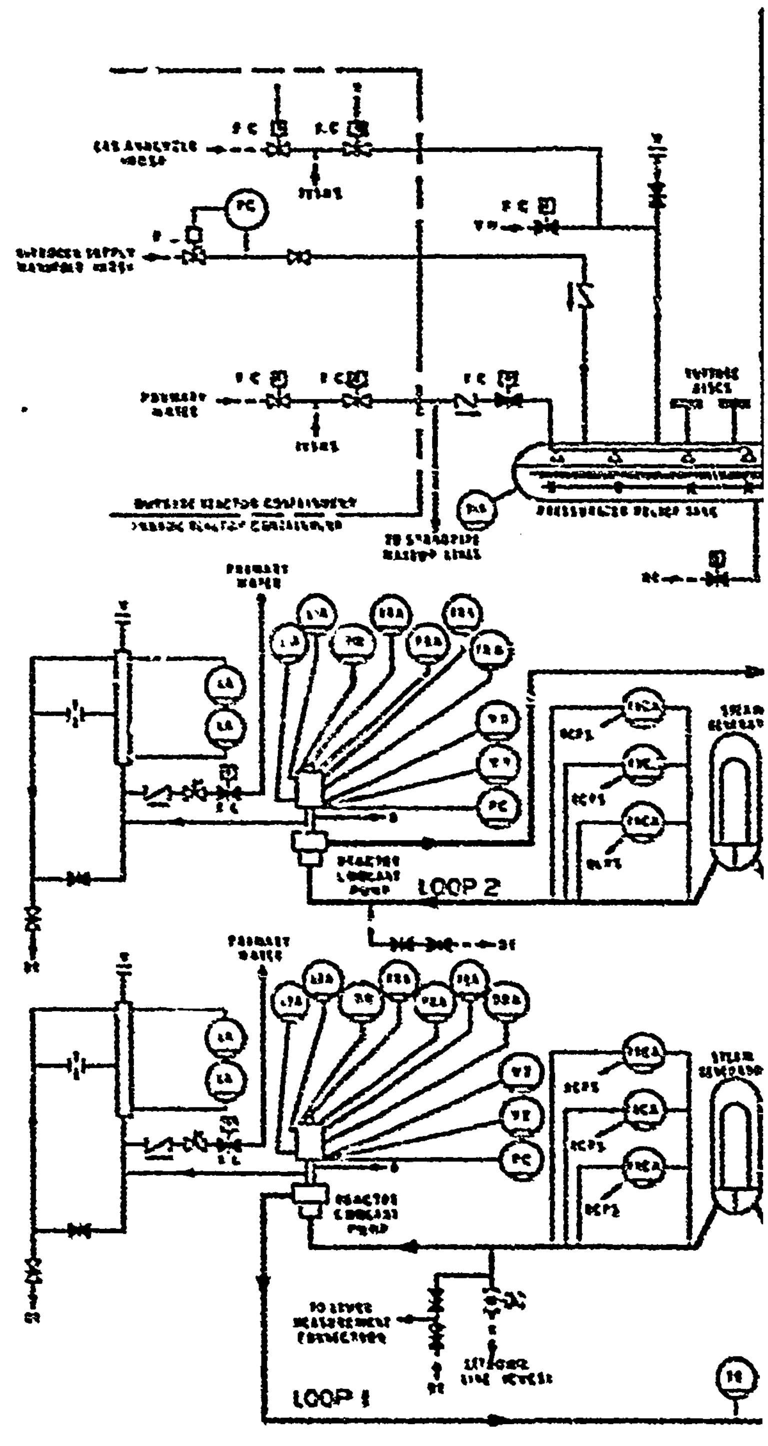
**Transverse Specimens

References (Continued)

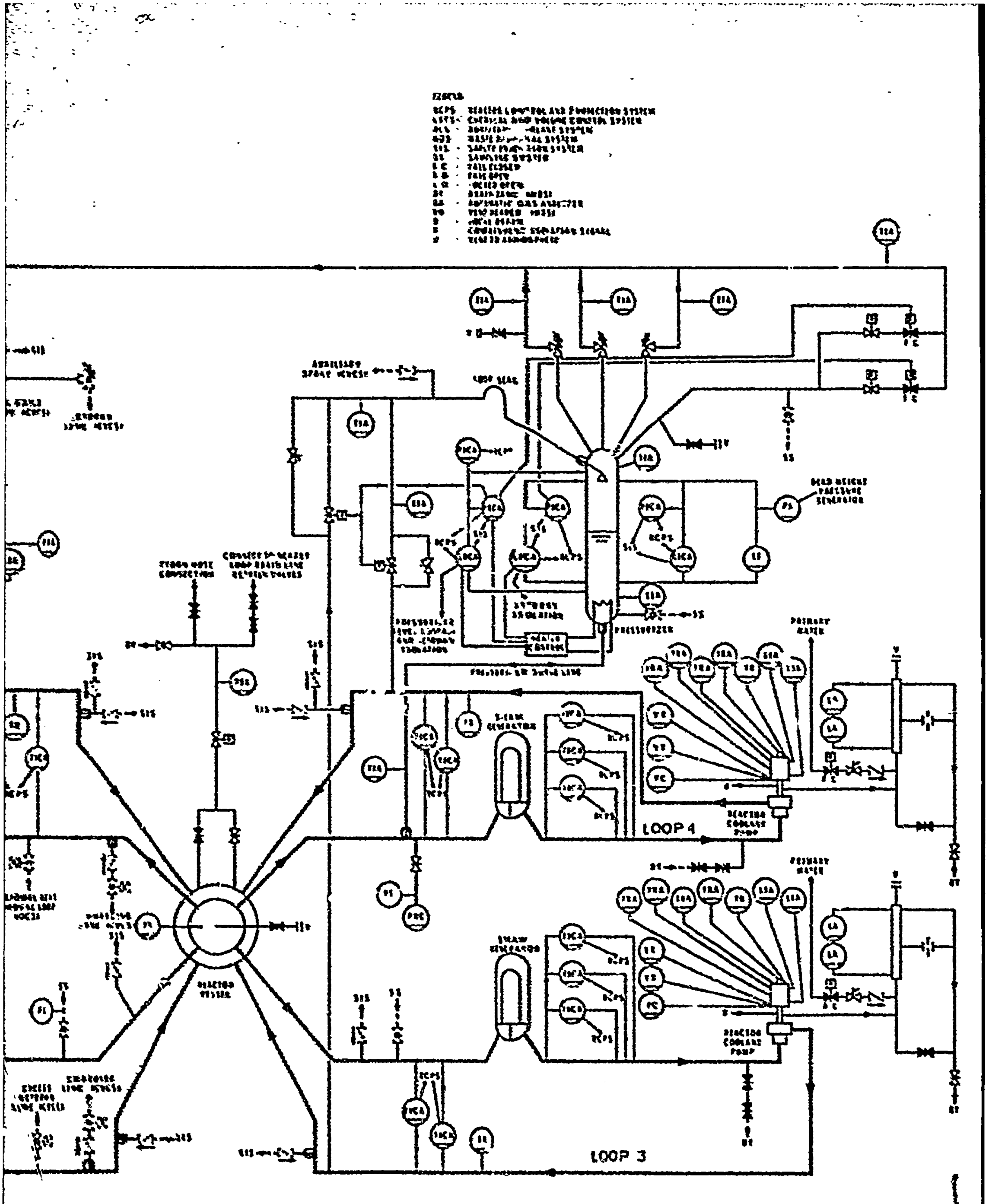
	References	Material	Temp. °F	Neutron Exposure n/cm^2 (>1 Mev)	ΔNDT °F
13.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	135
14.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	140
15.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	170
16.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	205
17.	NRL Report 6179 Page 9	SA302B	475-540	5×10^{19}	225
18.	NRL Report 6179 Page 9	SA302B	475-540	7×10^{19}	260
19.	NRL Report 6179 Page 9	SA302B	475-540	9×10^{19}	310
20.	NRL Report 6179 Page 9	SA302B	475-540	5×10^{19}	320
21.	NRL Report 6160 Page 15	SA302B	540*	4×10^{19}	200
22.	NRL Report 6160 Page 15	SA302B	540*	3×10^{19}	165
23.	Private Communication with NRL	SA302B	550	3.8×10^{18}	160

References (Continued)

	References	Material	Temp. °F	Neutron Exposure n/cm² (>1 Mev)	ΔNDT °F
24.	Progress Report No. 1, "Irradiation Tests on Reactor Pressure Vessel Steels in Br-3 Reactor Facilities" August, 1965	SA302B	~525	5.4×10^{18}	54
25.	"	SA302B	~525	1.2×10^{19}	96
26.	Progress Report No. 1, "Irradiation Tests on Reactor Pressure Vessel Steels in Br-3 Reactor Facilities August, 1965	SA302B	~600	9.5×10^{19}	260
27.	"	SA302B	~600	2×10^{20}	360

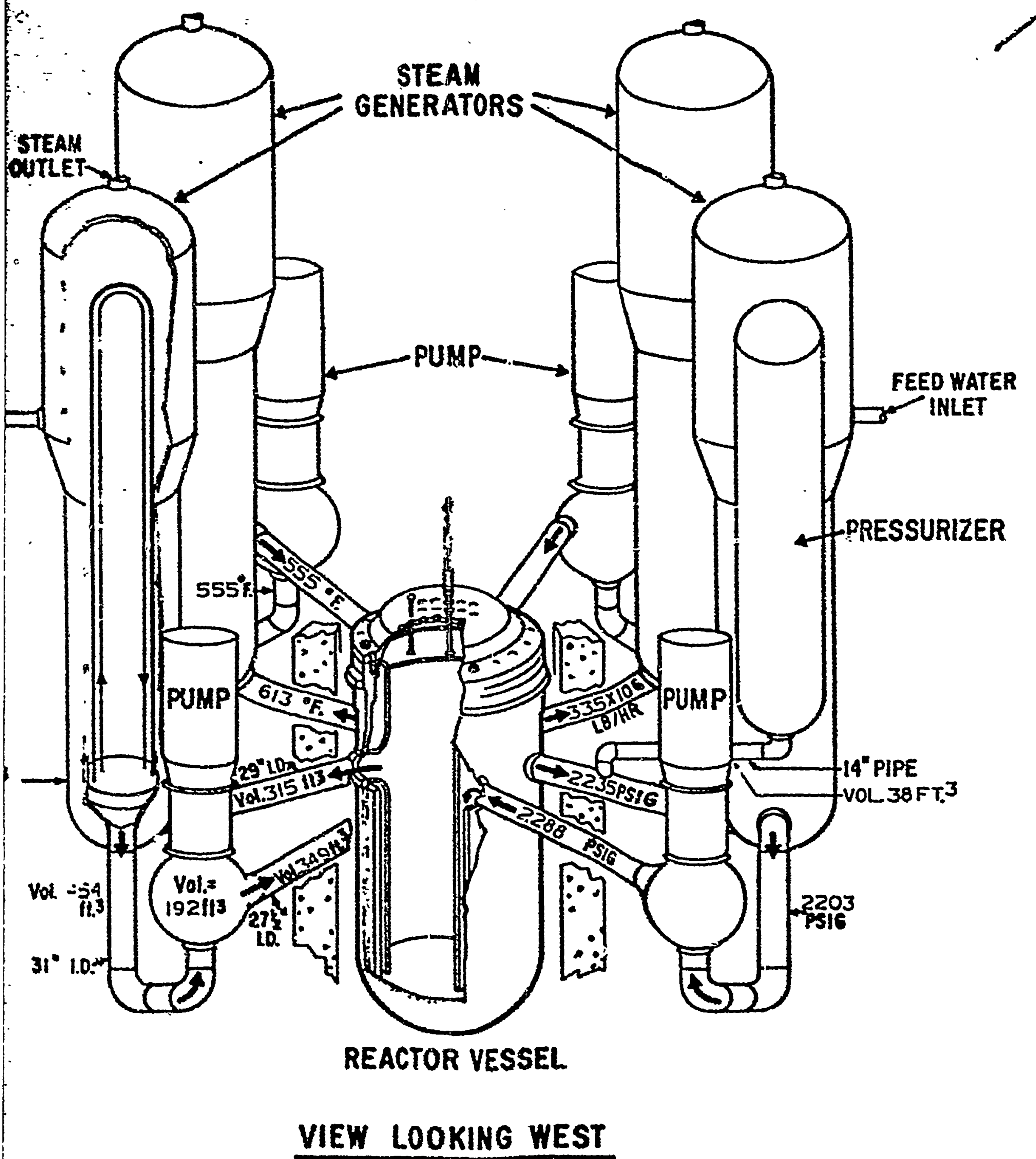


PC
LOOP 2
PC
LOOP 1

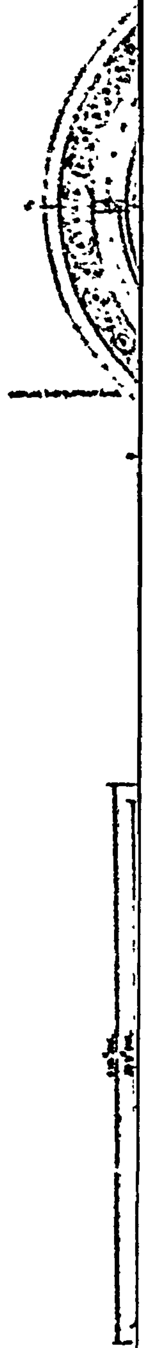


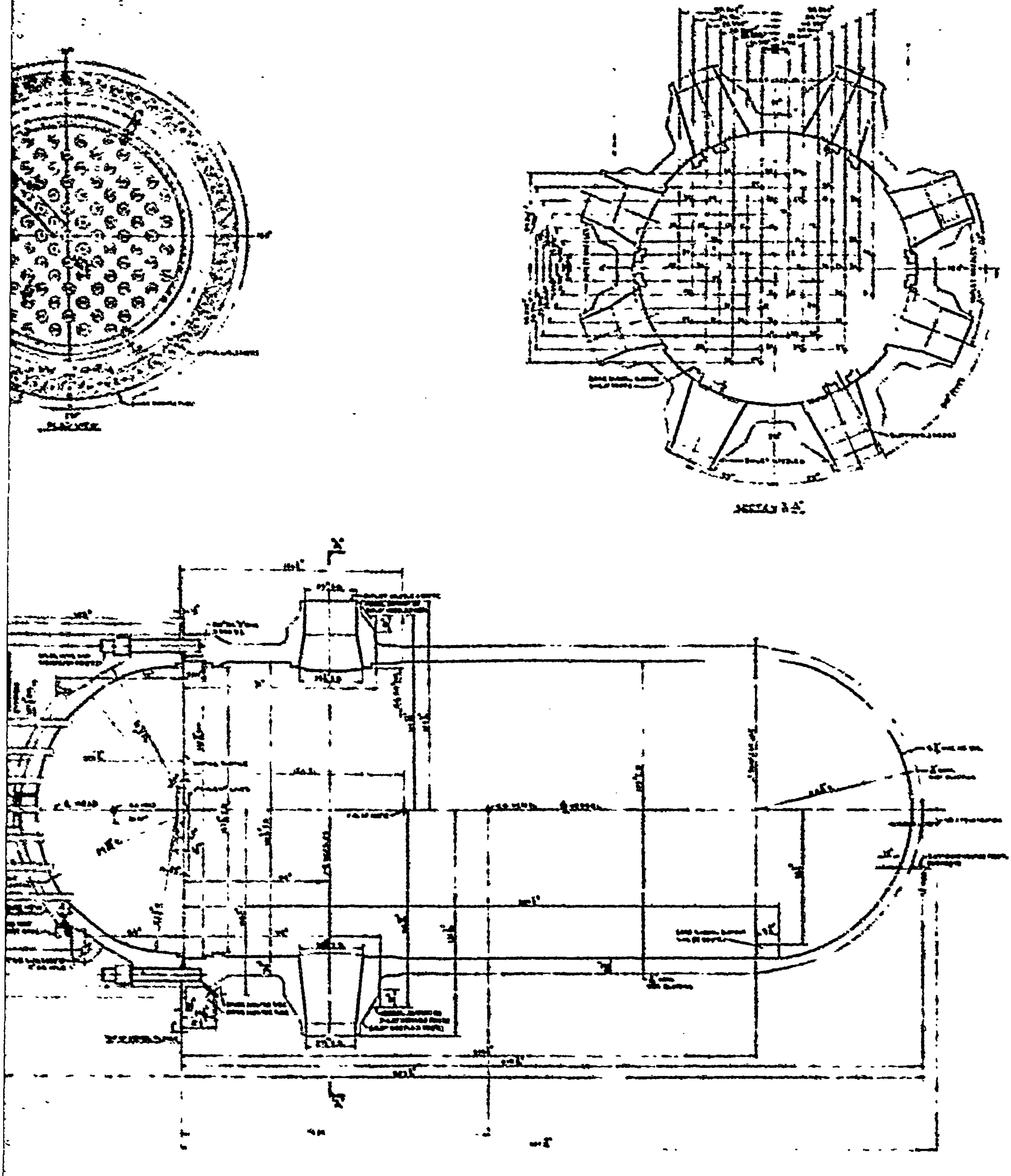
Reactor Coolant System Process Flow Diagram
Figure 4.2-1

SECONDARY SIDE WATER VOL. 1613
SECONDARY SIDE STEAM VOL. 2966
REACTOR COOLANT VOL. 935

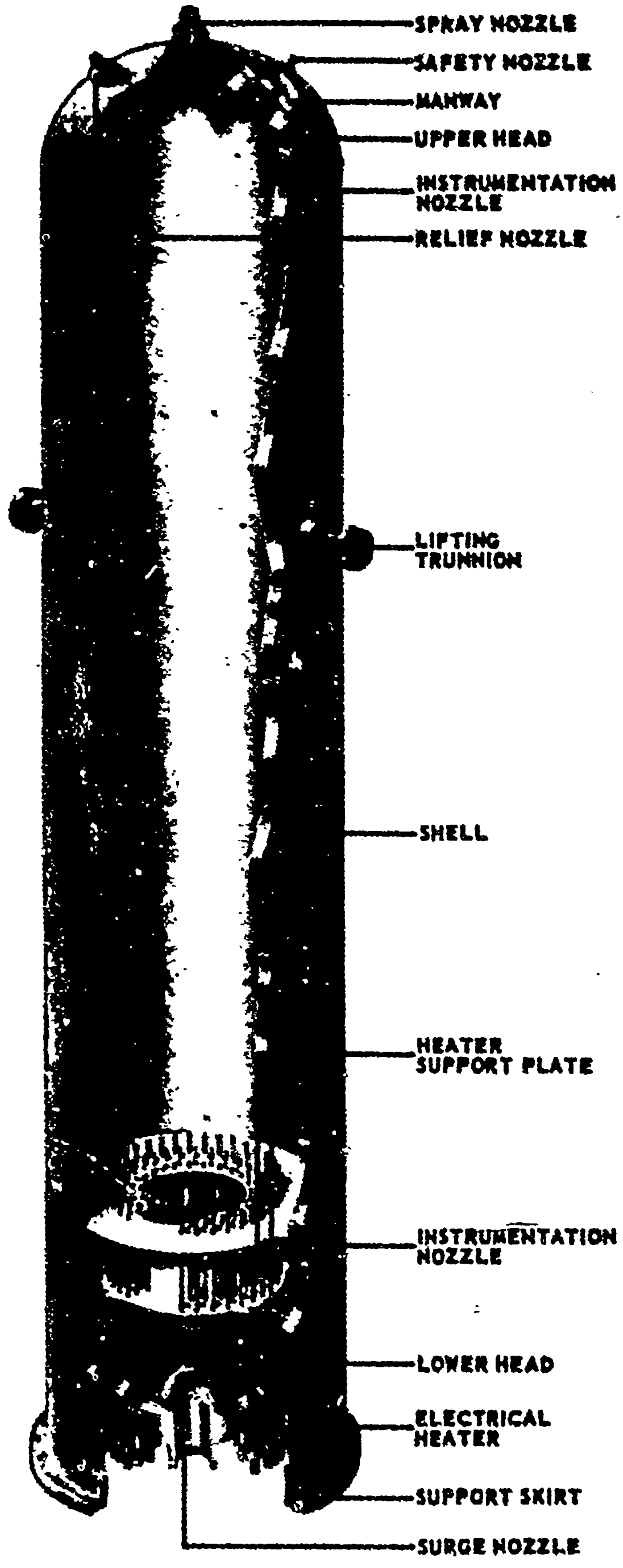


Reactor Coolant System Schematic Flow Diagram
Figure 4.2-2

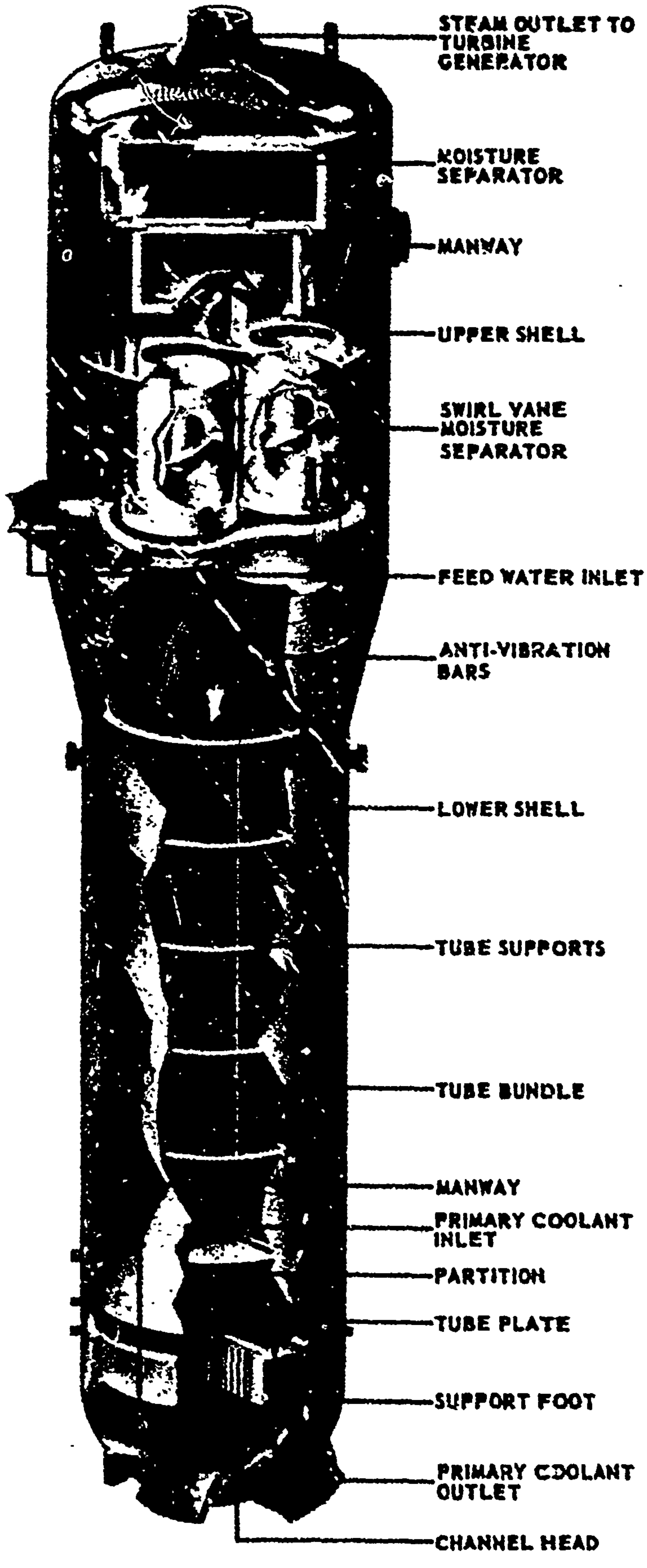




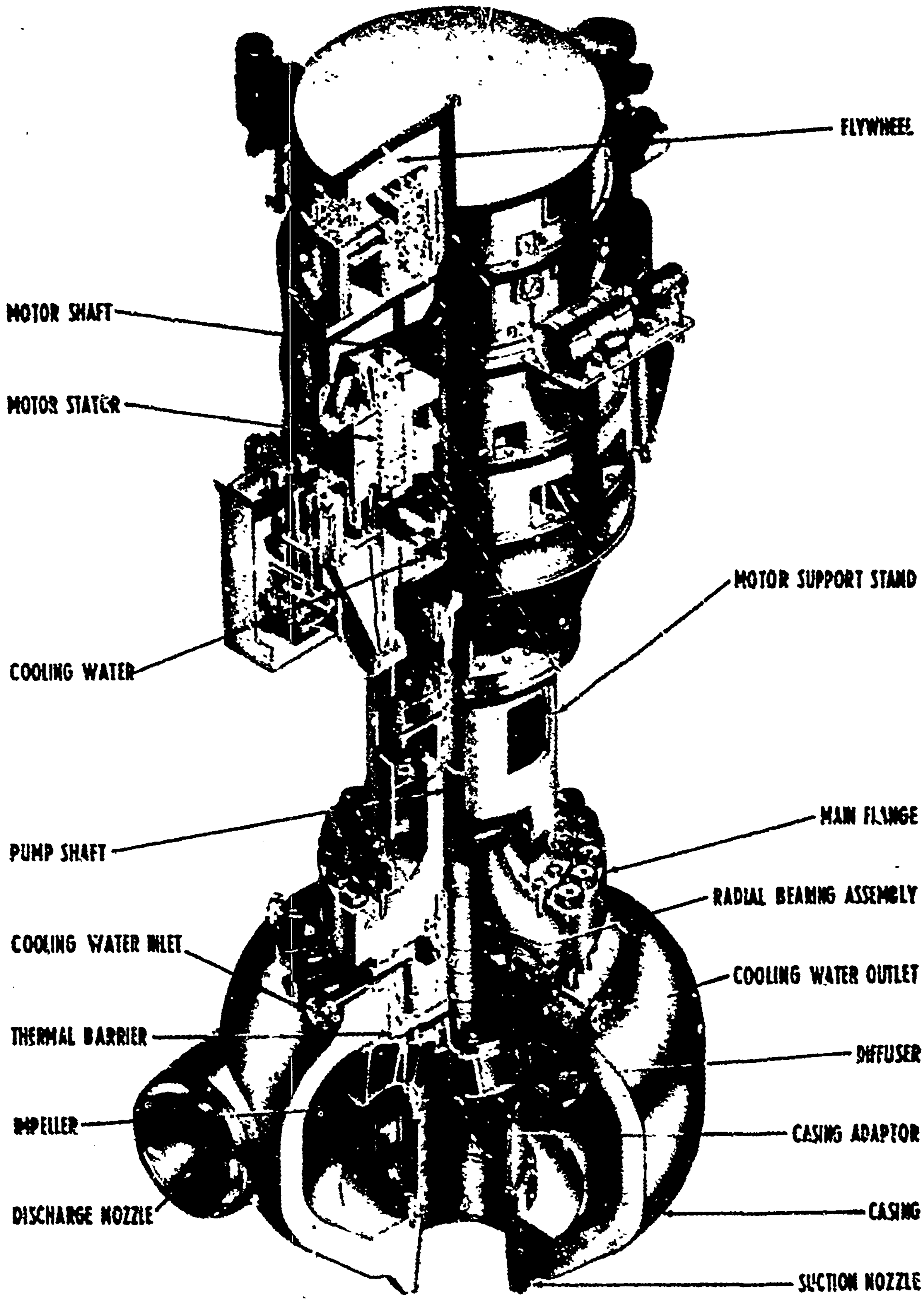
REACTOR VESSEL
Figure 4.2-3



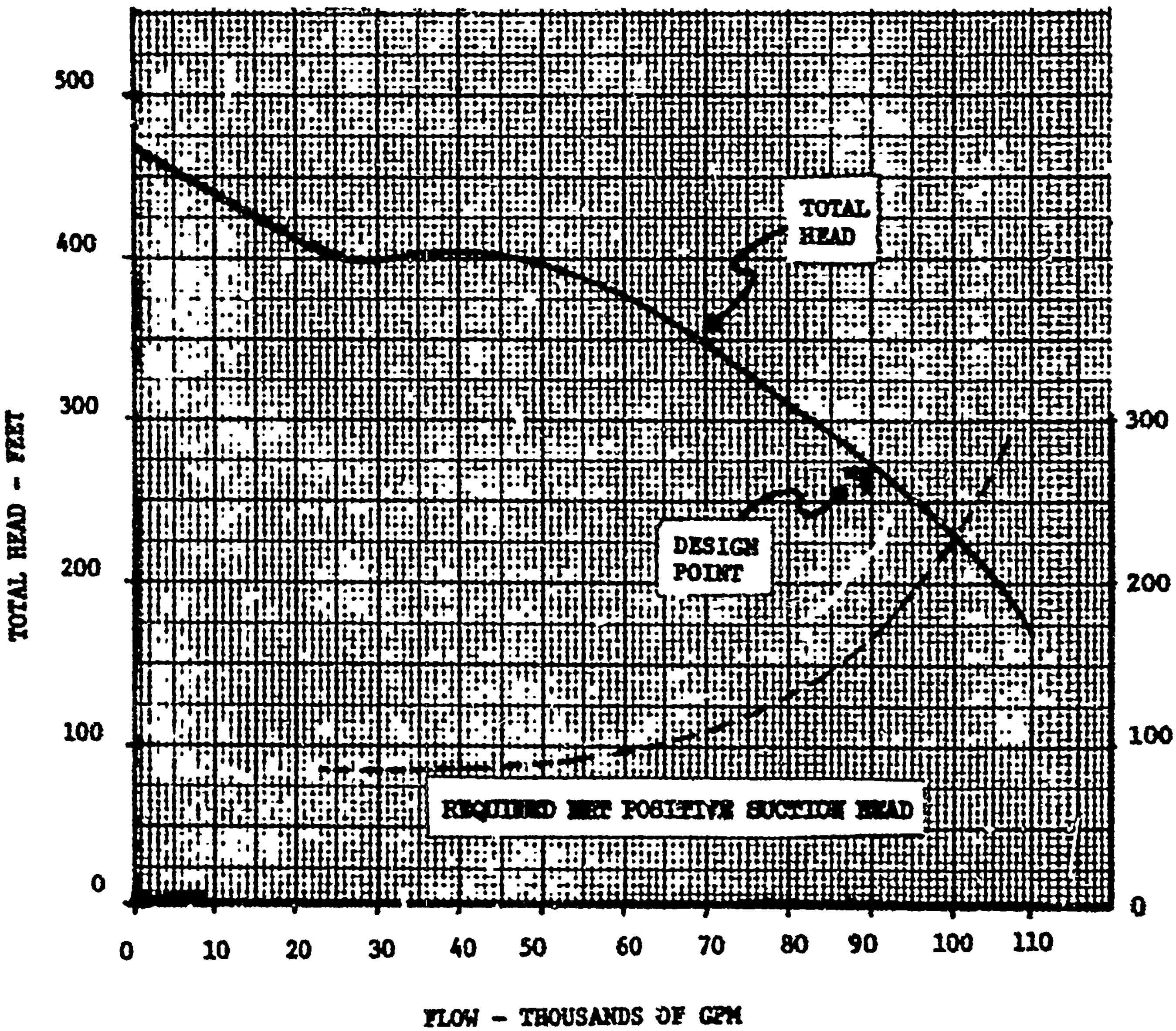
PRESSURIZER
FIGURE 4.2-4



STEAM GENERATOR
FIGURE 4.2-5

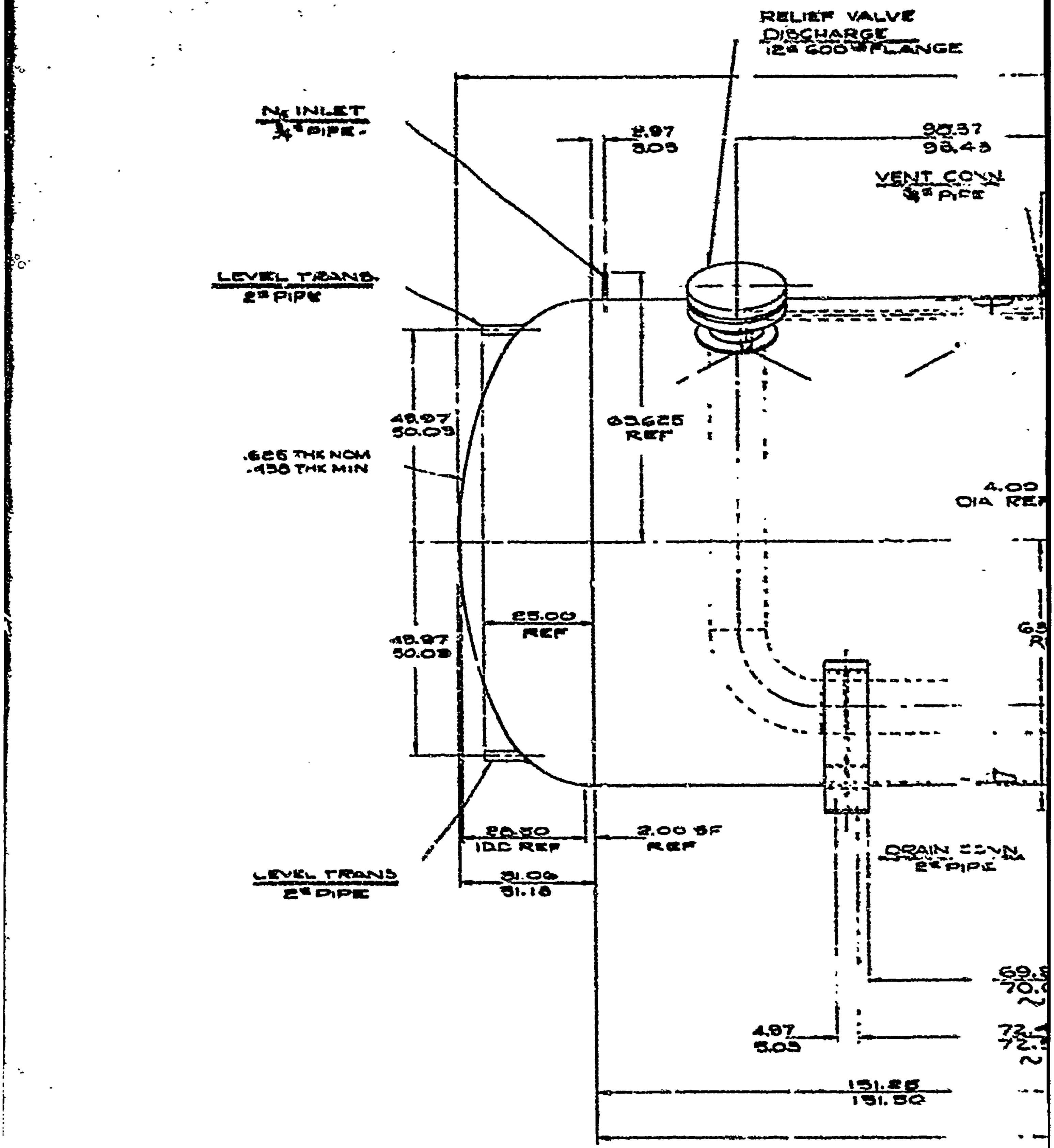


REACTOR COOLANT PUMP
FIGURE 4.2-6

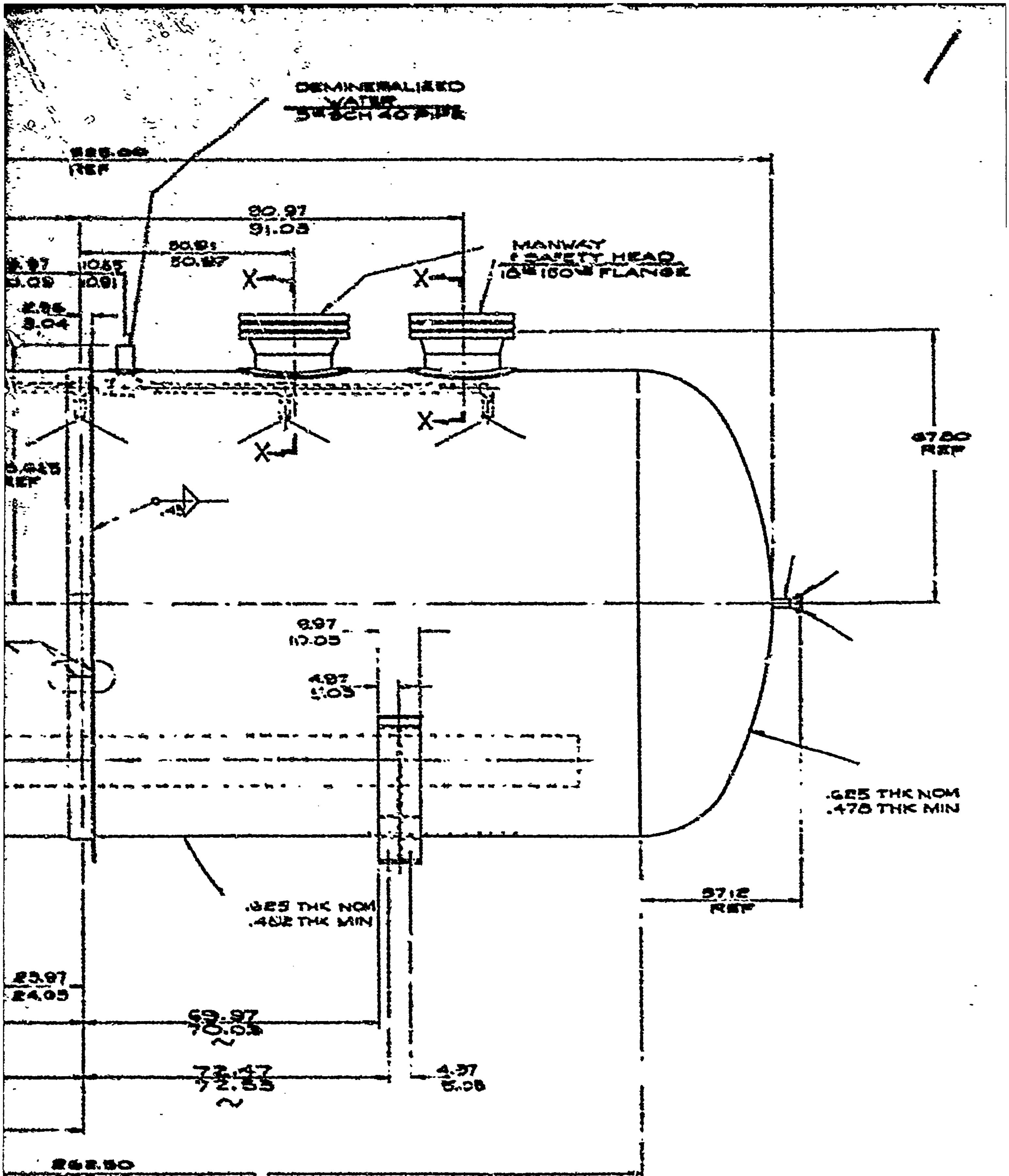


REACTOR COOLANT PUMP ESTIMATED PERFORMANCE CHARACTERISTIC

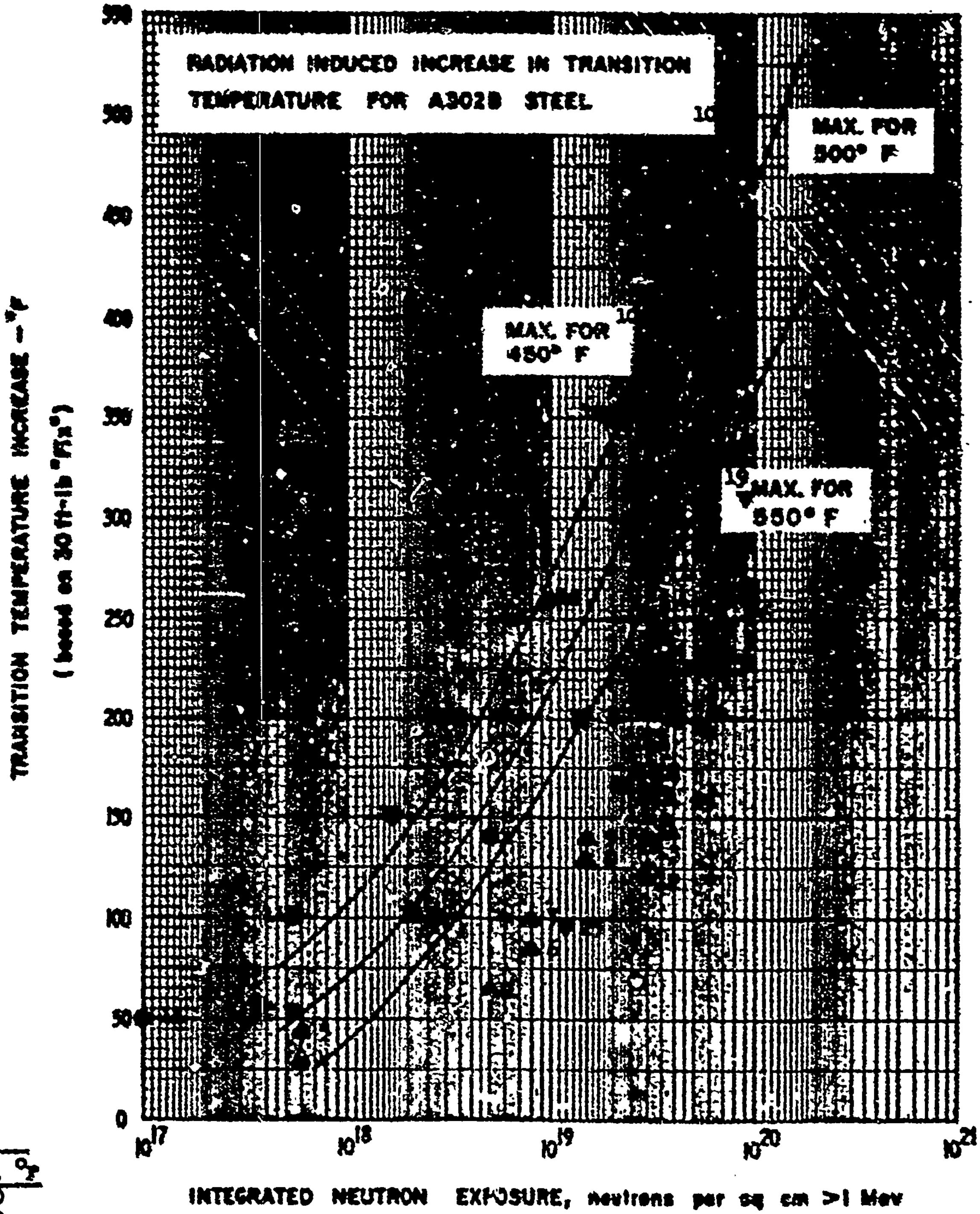
FIGURE 4.2-7



RELIEF VALVE
DISCHARGE
12" 600# FLANGE
N2 INLET
3/4" PIPE
2.97
3.03
93.37
93.43
VENT CONN
3/4" PIPE
LEVEL TRANS.
2" PIPE
49.97
50.03
63.625
REF
.625 THK NOM
.438 THK MIN
4.00
DIA REF
25.00
REF
49.97
50.03
26.50
IDD REF
2.00 SF
REF
DRAIN CONN
2" PIPE
LEVEL TRANS
2" PIPE
31.06
31.18
4.97
5.03
151.25
151.50



Pressurizer Relief Tank
Figure 4.2-8



Code

Temp. °F
● 450
■ 490
▲ 550
▼ 475 to 540
⊙ 600
Numbers 1 through 27 (see attached sheets)

RADIATION INDUCED INCREASE IN TRANSITION TEMPERATURE FOR A 302-B STEEL

FIGURE 4.2-9

4.3 SYSTEM DESIGN EVALUATION

4.3.1 SAFETY FACTORS

The safety of the reactor vessel and all other Reactor Coolant System pressure containing components and piping is dependent on several major factors including design and stress analysis, material selection and fabrication, quality control and operations control.

Reactor Vessel

A stress evaluation of the reactor vessel has been carried out in accordance with the rules of Section III of the ASME Nuclear Vessel Code. The evaluation demonstrates that stress levels are within the stress limits of the Code. Table 4.3-1 presents a summary of the results of the stress evaluation.

A summary of fatigue usage factors for components of the reactor v   -l is given in Table 4.3-2.

The cycles specified for the fatigue analysis are the results of an evaluation of the expected plant operation coupled with experience from nuclear power plants now in service, such as Yankee-Rowe. These cycles include five heatup and cooldown cycles per year, a conservative selection when the vessel may not complete more than one cycle per year during normal operation.

The vessel design pressure is 2485 psig while the normal operating pressure will be 2235 psig. The resulting operating membrane stress is therefore amply below the code allowable membrane stress to account for operating pressure transients.

To preclude the possibility of brittle failure the stresses allowed in the vessel when the operating temperature is below DTT are:

1. At DTT; a maximum stress of 20% yield
2. From DTT to DTT minus 200°F; a maximum stress decreasing from 20% to 10% yield
3. Below DTT minus 200°F; a maximum stress of 10% yield

These limits are based on the data reported by Kibara and Masubichi (Effect of Residual Stress on Brittle Fracture, April 1959, Welding Journal Volume 38) and Robertson (Propagation from Brittle Fracture in Steel, Journal of the Iron and Steel Institute, 1953), which show that if the stresses are maintained within the above limits, brittle fracture does not occur. These stress limits are maintained by prescribing operating procedures which rely upon administrative pressure and temperature control during heatup and cooldown as described in ASME Paper No. 63-WA-100, "Reactor Vessel Design Considering Radiation Effects", L. Porse.

The actual shift in NDTT will be established periodically during plant operation by testing of vessel material samples which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. To compensate for any increase in the NDTT caused by irradiation, the limits given in the plant operating manual on the pressure-temperature relationship are periodically changed to stay within the stress limits, which are stated above, during heatup and cooldown.

The vessel closure contains fifty-four 7-inch studs. The stud material has a minimum yield strength of 104,400 psi at design temperature. The membrane stress in the studs when they are at the steady state operational condition is approximately 40,000 psi. This means that twenty-one of the fifty-four studs have the capability of withstanding the hydrostatic end load on the vessel head without the membrane stress exceeding yield strength of the stud material at design temperature.

The normal operating temperature always exceeds even the highest anticipated DTT during the life of the plant. Thus the emphasis of conservative operation is placed on heatup and cooldown because long term irradiation of the vessel raises the DTT and thereby limits the heatup or cooldown rates. The conservatism in setting up the temperature-pressure relationship limits stated above are:

1. Us   stress concentration factor of 4 on assumed flaws in calc`ating the stresses.
2. Use of nominal yield of material instead of actual yield.
3. Neglecting the increase in yield strength resulting from radiation effects.

As part of the plant operator training program Westinghouse instructs supervisory and operating personnel in reactor vessel design, fabrication and testing as well as present and future precautions necessary for pressure testing and operating modes. The need for record keeping is stressed, such records boing helpful for future summation of time at power level and temperature which tends to influence the irradiated properties of the material in the core region. These instructions are incorporated in the operating manuals.

Steam Generators

Claculations confirm that the steam generator tube sheet will withstand the loading (which is a quasi-static rather than a shock loading) by loss of reactor coolant. The maximum primary membrane plus primary bending stress in the tube sheet under these conditions is 23,600 psi. This is well below ASME Section III yield strength of 41,400 psi at 650°F. Because the pressure in the primary channel head could drop to zero under the condition postulated, no damage will result to the channel head.

The rupture of primary or secondary piping has been assumed to impose a maximum pressure differential of 2250 psi across the tubes and tube sheet from the primary side or a maximum pressure differential of 1100 psi across the tubes and tube sheet from the secondary side, respectively. Under these conditions there is no rupture of the primary to secondary boundary (tubes and tube sheet). This criterion prevents any violation of the containment boundary.

To meet this criterion, it has been established that under the postulated accident conditions, where a primary to secondary side differential pressure of 2250 psia exists, the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through the tube sheet thickness, does not exceed 90% of the material yield stress at the operating temperature; in addition the primary membrane plus primary bending stress in the tube sheet ligaments, averaged across the ligament width at the tube sheet surface location giving maximum stress, do not exceed 135% of the material yield stress at the operating temperature. This criterion is felt to be applicable to abnormal operating circumstances in that it is consistant with the ASME, Nuclear Pressure Vessel Code, Section III rules, Para. N-714, 2 for hydrotest limitations.

An examination of stresses under these conditions show that for the case of a 2485 psig maximum tube sheet pressure differential the stresses are within acceptable limits. These stresses together with the corresponding stress limits are given in Table 4.3-3.

The tubes have been designed to the requirements (including stress limitations) of Section III for normal operation, assuming 1550 psig as the normal operating pressure differential. Hence, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

No significant corrosion of the Inconel tubing is expected during the lifetime of the plant. The corrosion rate reported in Reference (1)* show "worst case" rates of 15.9 mg/dm^2 in the 2000 hour test under steam generator operating conditions. Conversion of this rate to a 40 year plant life gives a corrosion loss of less than 1.5×10^{-3} inches which is insignificant compared to the nominal tube wall thickness of 0.050 inches.

In the case of a primary pressure loss accident, the secondary-primary pressure differential can reach 1100 psig. This pressure differential is less than the primary-secondary design pressure differential (1520 psi) for normal operating conditions. Hence, no stresses in excess of those covered in Section III rules for normal operation are experienced on the tube sheet for this

*Reference (1) W. E. Berry and F. W. Fink, "The Corrosion of Inconel in High Temperature Water," Batelle Memorial Institute, April, 1958.

accident case. For the tubes, actual pressure tests of 3/4 in. O.D./.058 inch wall Inconel tubing show collapse under external pressure of 5700-5900 psi. Extrapolating this data to 7/8 in. O.D./.050 inch wall tubes, collapse would occur at about 2630 psi at 650°F. This gives a factor of safety of 2.4 against collapse under the 1100 psig accidental application of external pressure to tubes. A check of the ASME Section VIII design curves for Iron-Chromium-Nickel Steel cylinders under external pressure indicates a predicted collapse pressure for the tubes of 2310 psi, which is close to the extrapolated value for the experimental results.

Consideration has been given to the superimposed effects of secondary side pressure loss and the maximum potential earthquake loading. The fluid dynamic forces on the internal components affecting the primary-secondary boundary (tubes) has been considered as well. For this condition, the criterion is that no rupture of primary to secondary boundary (tubes and tube sheet) occurs.

For the case of the tube sheet, the maximum hypothetical earthquake loading will contribute an equivalent static pressure loading over the tube sheet of less than 10 psi (for vertical shock). Such an increase is small when compared to the pressure differentials (up to 2485 psig) for which the tube sheet is designed. Under horizontal shock loading of the maximum hypothetical earthquake the stresses are less than those for 1.0g gravity loading experienced in a horizontal position, which the design can readily accept.

The fluid dynamic forces on the internals under secondary steam break accident conditions indicate, in the most severe case, that the tubes are adequate to constrain the motion of the baffle plate with some plastic deformation but boundary integrity is maintained.

The ratio of the allowable stresses (based on an allowable membrane stress of 0.9 of the nominal yield stress of the material) to the computed stresses are summarized in Table 4.3-4.

4.3.2 RELIANCE ON INTERCONNECTED SYSTEMS

The principal heat removal systems which are interconnected with the Reactor Coolant System are the Steam and Power Conversion, the Safety Injection and Residual Heat Removal Systems. The Reactor Coolant System is dependent upon the steam generators, and the steam, feedwater, and condensate systems for decay heat removal from normal operating conditions to a reactor coolant temperature of approximately 350°F. The layout of the system ensures the natural circulation capability to permit adequate core cooling following a loss of all main reactor coolant pumps.

The flow diagram of the Steam and Power Conversion System is shown on Figure 10-1. In the event that the condensers are not available to receive the steam generated by residual heat, the water stored in the feedwater system may be pumped into the steam generators and the resultant steam vented to the atmosphere. The auxiliary feedwater system will supply water to the steam generators in the event that the main feedwater pumps are inoperative.

The Safety Injection System is described in Section 6. The Residual Heat Removal System is described in Section 9.

4.3.3 SYSTEM INTEGRITY

A complete stress analysis which reflects consideration of all design loadings detailed in the design specification has been prepared by the manufacturer. The analysis shows that the reactor vessel, steam generator, pump casing and pressurizer comply with the stress limits of Section III of the ASME Code. A similar analysis of the piping shows that it complies with the stress limits of the applicable USAS Code.

As part of the design control on materials, Charpy V-notch toughness test curves are run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generator and pressurizer to provide assurance for hydrotesting and operation in the ductile region at all times. In addition, dropweight tests are performed on the reactor vessel plate material.

As an assurance of system integrity, all components in the system are hydrotested at 3110 psig prior to initial operation.

4.3.4 OVERPRESSURE PROTECTION

The reactor coolant system is protected against overpressure by safety valves located on the top of the pressurizer. The safety valves on the pressurizer are sized to prevent system pressure from exceeding the design pressure by more than 10 per cent, in accordance with Section III of the ASME Boiler and Pressure Vessel Code. The capacity of the pressurizer safety valves is determined from considerations of: (1) the reactor protective system, and (2) accident or transient conditions which may potentially cause overpressure.

The combined capacity of the safety valves is equal to or greater than the maximum surge rate resulting from complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve setting.

Details of the analysis are reported in Section 14.1.8. Experience has shown that the safety valve capacity so determined is adequate for all the other transients as the results of Section 14.1 show.

4.3.5 SYSTEM INCIDENT POTENTIAL

The potential of the Reactor Coolant System as a cause of accidents is evaluated by investigating the consequences of certain credible types of components and control failures as discussed in Sections 14.1, and 14.2. Reactor coolant pipe rupture is evaluated in Section 14.3.

4.3.6 REDUNDANCY

Each loop of the Reactor Coolant System contains a steam generator and a reactor coolant pump. Operation at reduced reactor power is possible with one loop out of service. For added reliability, power to the reactor coolant pumps is normally supplied by electrically separated buses as shown in Figure 8-1.

TABLE 4.3-1

SUMMARY OF PRIMARY PLUS SECONDARY STRESS INTENSITY
FOR COMPONENTS OF THE REACTOR VESSEL

Area	Stress Intensity (psi)	Allowable Stress 3 Sm (psi) (Operating Temperature)
Control Rod Housing	55,300	69,900
Head Flange	50,400	80,100
Vessel Flange	45,350	80,100
Closure Studs	95,870	110,200
Primary Nozzles - Inlet	48,400	80,100
Outlet	54,060	80,100
Core Support pad	40,800	69,900
Bottom head to shell	34,100	80,100
Bottom instrumentation	53,900	69,900
Nozzle belt to shell	32,600	80,100

TABLE 4.3-2

SUMMARY OF CUMULATIVE FATIGUE USAGE FACTORS FOR COMPONENTS OF THE REACTOR VESSEL

Item	Usage Factor*
Control rod housing	0.06
Head Flange	0.015
Vessel Flange	0.005
Stud bolts	0.310
Primary nozzles - Inlet	0.020
Outlet	0.028
Core support pad (lateral)	0.015
Bot. head to shell	0.003
Bot. instrumentation	0.142
Nozzle Belt to shell	0.0015

* As defined in Section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

TABLE 4.3-3

STRESSES DUE TO MAXIMUM STEAM GENERATOR TUBE SHEET PRESSURE DIFFERENTIAL (2485 PSIG)

Stress	(668°F) Computed Value	Allowable Value
Primary Membrane Stress	23,300 psi	37,000 psi (.9 Sy)
Primary Membrane plus Primary Bending Stress	53,000 psi	55,600 psi (1.35 Sy)

In addition to the foregoing evaluation, elasto-plastic limit analysis of the tube sheet-head-shell combination indicates a limit pressure of 3400 psi at operating conditions, giving a safety factor of 1.36 for the abnormal condition.

TABLE 4.3-4

RATIO OF ALLOWABLE STRESSES TO COMPUTED STRESSES
FOR A STEAM GENERATOR TUBE
SHEET PRESSURE DIFFERENTIAL OF 2485 PSIG

Component Part	Stress Ratio
Channel head	1.35
Channel head-tube sheet joint	1.63
Tubes	1.20
Tube sheet	
Max. Avg. Ligament	1.04
Effective Ligament	1.58

4.4 SAFETY LIMITS AND CONDITIONS

4.4.1 SYSTEM HEATUP AND COOLDOWN RATES

Operating limits for the Reactor Coolant System with respect to heatup and cooldown rates are defined in the Technical Specification.

The stress level of material in the reactor vessel, or in other Reactor Coolant System components, is a combination of stresses caused by internal pressures and by thermal gradients. The latter are significant as they may result from a rate of change of reactor coolant temperature. Operating restrictions are imposed to limit the combined stresses to 20% of minimum yield stress when at the Design Transition Temperature (DTT). The DTT is defined as the initial Nil Ductility Transition Temperature (NDTT) plus the increase in NDTT due to irradiation experienced plus 60°F. This stress limit (20% of Y.S.) is reduced linearly to a value of 10% of yield at a temperature 200°F below DTT. Curves are incorporated in the plant operating manual which define the operating limits for initial operation and for end of life operation. To establish the latter an adjustment is made for the maximum expected NDTT shift (240°F) which the reactor vessel material will experience because of the fast neutron dose it will receive. The predicted shift will be verified by the surveillance program testing. The limits for initial operation are used to define operational limitations and these curves are periodically updated to reflect irradiation exposure of the vessel and the results of the surveillance program.

4.4.2 REACTOR COOLANT ACTIVITY LIMITS

Release of activity into the reactor coolant in itself does not constitute a hazard. Activity in the coolant could constitute a hazard only if the reactor coolant system barrier is breached, and then only if the coolant contains excessive amounts of activity which could be released to the environment. The plant systems are designed for operation with activity in the reactor coolant systems corresponding to 1 per cent fuel defect. The waste gas system is designed such that rupture of a gas decay tank, following a refueling shutdown wherein the gaseous activity is removed from the reactor coolant to the waste gas tanks for decay, will not result in offsite whole body exposure in excess of 0.5 rem. In the event of a steam generator tube rupture high activity level at the condenser air ejector exhaust will divert the activity discharge back to the containment. These accidents are analyzed in Section 14.2. The reactor coolant system operational activity limit is defined in the Technical Specifications.

4.4.3 MAXIMUM PRESSURE

The Reactor Coolant System serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the Reactor Coolant System is the primary barrier against the uncontrolled release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is assured. Thus, the safety limit of 2735 psig (110% of design pressure) has been established. This represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section III. Reactor Coolant System pressure settings are given in Table 4.1-1.

4.4.4 SYSTEM MINIMUM OPERATING CONDITIONS

Minimum Operating Conditions for the Reactor Coolant System for all phases of operation are given in the Technical Specifications.

4.5 INSPECTIONS AND TESTS

4.5.1 REACTOR COOLANT SYSTEM INSPECTION

Non-destructive Inspection of Material and Components Prior to Operation

Table 4.5-1 summarizes the quality assurance program for all Reactor Coolant System components. In this table all of the non-destructive tests and inspections which are required by Westinghouse specifications on Reactor Coolant System components and materials are specified for each component. All tests required by the applicable codes are included in this table. Westinghouse requirements, which are more stringent in some areas than those requirements specified in the applicable codes, are also included. The fabrication and quality control techniques used in the fabrication of the reactor coolant system are equivalent to those used for the reactor vessel.

The procedures by which it is ensured that the required level of quality assurance is achieved are described in Appendix B.

Westinghouse requires, as part of its reactor vessel specification, that certain special tests which are not specified by the applicable codes be performed. These tests are listed below:

1) Ultrasonic Testing - Westinghouse requires that a 100% volumetric ultrasonic test of reactor vessel plate for both shear wave and longitudinal wave be performed. Section III Class A vessel plates were required by code to receive only a longitudinal wave ultrasonic test on a 9 in. x 9 in. grid. The 100% volumetric ultrasonic test is a severe requirement, but it assures that the plate is of the highest quality.

2) Radiation Surveillance Program - In the surveillance programs, the evaluation of the radiation damage is based on pre- and post-

irradiation testing of Charpy V-notch, tensile and wedge opening loading (WOL) fracture mechanism type. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach, and are in accordance with ASTM E185, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors."

The reactor vessel surveillance programs use eight specimen capsules which are located about 3 inches from the vessel wall directly opposite the center portion of the core.

The capsules can be removed when the vessel head is removed. The capsules contain reactor vessel steel specimens from the shell plates or forgings located in the core region of the reactor and associated weld metal and heat affected zone metal. In addition, correlation monitors made from fully documented specimens of SA302 Grade B material obtained through Subcommittee II of ASTM Committee E10 Radioisotopes and Radiation Effects are inserted in the capsules. The eight capsules will contain at least 27 tensile specimens, 256 Charpy V-notch specimens (which will include weld metal and heat affected zone material) and 42 WOL specimens. Dosimeters including p[illegible] Ni, Al-Co, (0.15% Co), Cd shielded Al-Co, CdO shielded Np-237 and CdO shielded U-238 are placed in the impact specimens, tensile specimens or filler blocks drilled to contain the dosimeters. The dosimeters permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low melting alloys are included to monitor temperature of the specimens. The specimens are enclosed in a tight fitting stainless steel sheath to prevent corrosion:

The tentative schedule for removal of capsules is as follows:

Capsule	Estimated Exposure Time
1	Replacement of 1st region of core
2	Replacement of 2nd region of core
3	Replacement of 4th region of core
4	10 years
5	15 years
6	20 years
7 and 8	Extra capsules for complementary or duplicate testing or additional exposure

Irradiation of the specimens will be higher than the irradiation of the vessel because of the specimens are located in the vicinity of the core corners and are closer to the core than the vessel itself. Since these specimens will experience higher irradiation and are actual samples from the materials used in the vessel, the NDTT measurements will be representative of the vessel at a later time in life. Data from fracture toughness samples (WOL) are expected to provide additional information for use in determining allowable stresses for irradiated material.

Table 4.5-1 summarizes the quality assurance program with regard to inspections performed on primary system components. In addition to the inspections shown in Table 4.5-1, there are those which the equipment supplier performs to confirm the adequacy of material he receives, and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator are governed by ASME Code requirements. The inspection procedures and acceptance standards required on pipe materials and piping fabrication are governed by USAS B31.1 and Westinghouse requirements and are equivalent to those performed on ASME coded vessels.

Procedures for performing the examinations are consistent with those established in the ASME Code Section III and are reviewed by qualified Westinghouse engineers. These procedures have been developed to provide the highest assurance of quality material and fabrication. They consider not only the size of the flaws, but equally as important, how the material is fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the surfaces most subject to damage as a result of the heat treating, rolling, forging, forming and fabricating processes, receive a 100% surface inspection by Magnetic Particle or Liquid Penetrant Testing after all these operations are completed, although flaws in plates are inherently laminations in the center. All reactor coolant plate material is subject to shear as well as longitudinal ultrasonic testing to give maximum assurance of quality. All forgings receive the same inspection. In addition, 100% of the material volume is covered in these tests as an added assurance over the grid basis required in the code.

Westinghouse Quality Control engineers monitor the supplier's work, witnessing key inspections not only in the supplier's shop but in the shops of subvendors of the major forgings and plate material. Normal surveillance includes verification of records of material, physical and chemical properties, review of radiographs, performance of required tests and qualification of supplier personnel. An independent surveillance of the conformance to the fabrication and installation specifications and the quality control requirements of, amongst other things, the reactor coolant system components, is carried out by the United States Testing Company for Consolidated Edison.

Equipment specifications for fabrication require that suppliers submit the manufacturing procedures (welding, heat treating, etc.) to Westinghouse where they are reviewed by qualified Westinghouse engineers. This also is done on the field fabrication procedures to assure that installation welds are of equal quality.

Consolidated Edison engineers witness the hydrostatic test of the reactor vessel.

Field erection and field welding of the reactor coolant system are performed such as to permit exact fit-up of the 31" I.D. closure pipe subassemblies between the steam generator and the reactor coolant pump. After installation of the pump casing and the steam generator, measurements are taken of the pipe length required to close the loop. Based on these measurements, the 31" I.D. closure pipe subassembly is properly machined and then erected and field welded to the pump suction nozzle and to the steam generator exit nozzle.

Cleaning of RCS piping and equipment is accomplished before and/or during erection of various equipment. Stainless steel piping is cleaned in sections as specific portions of the systems are erected. Pipe and units large enough to permit entry by personnel are cleaned by locally applying approved solvents (Stoddard solvent, acetone and alcohol), and demineralized water, and by using a rotary disc sander or 18-8 wire brush to remove all trapped foreign particles. Standards for final physical and chemical cleanliness are defined in Table 1.3.1-1.

Section III of the ASME B&PV Code requires that nozzles carrying significant external loads shall be attached to the shell by full penetration welds. This requirement has been carried out in the reactor coolant piping, where all auxiliary pipe connections to the reactor coolant loop are made using full penetration welds.

The Reactor Coolant System components are welded under procedures which require the use of both preheat and post-heat. Preheat requirements, non-mandatory under Code rules, are performed on all weldments, including P1 and P3 materials which are the materials of construction in the reactor vessel, pressurizer and steam generators. Preheat and post-heat of weldments both serve a common purpose: the production of tough, ductile metallurgical structures in the completed weldment. Preheating, produces tough ductile welds by minimizing the formation of hard zones; whereas post-heating achieves this by tempering any hard zones which may have formed due to rapid cooling.

In-Service Inspection

During the design phase of the Reactor Coolant System, careful consideration is given to provide access for both visual and non-destructive in-service inspection of primary loop components. The following components and areas are available for visual and/or non-destructive inspection.

1) Reactor Vessel - The entire inside surface.
2) Reactor Vessel Nozzles - The entire inside surface.
3) Closure Head - The entire inside and outside surface.
4) Reactor Vessel Studs, Nuts and Washers.
5) Field Welds between the Reactor Vessel , Steam Generators, and Reactor Coolant Pumps and the Main Coolant Piping.
6) Reactor Internals
7) Reactor Vessel Flange Seal Surface
8) Fuel Assemblies
9) Rod Cluster Control Assemblies
10) Control Rod Drive Shafts
11) Control Rod Drive Mechanism Assemblies
12) Main Coolant Pipe External Surfaces (except for the five foot penetration of the primary shield)
13) Steam Generator - The external surface, the internal surfaces of the steam drum, and channel head.
14) Pressurizer - The internal and external surfaces.
15) Reactor Coolant Pump - The external surfaces, motor and impeller.

The design considerations which have been incorporated into the primary system design to permit the above inspections are as follows:

1) All reactor internals are completely removable. The tools and storage space required to permit these inspections are provided.
2) The closure head is stored dry on the reactor operating deck during refueling to facilitate direct visual inspection.

3) All reactor vessel studs, nuts and washers are removed to dry storage during refueling.

4) Removable plugs are provided in the primary shield just above the coolant nozzles, and the insulation covering the nozzle welds is readily removable.

5) Access holes are provided in the lower internals barrel flange to allow remote access to the reactor vessel internal surfaces between the flange and the nozzles without removal of the internals.

6) A removable plug is provided in the lower core support plate to allow access for inspection of the bottom head without removal of the lower internals.

7) The storage stand provided for storage of the internals allows for inspection access to both the inside and outside of the internals.

8) The station provided for changeout of control rod clusters from one fuel assembly to another is specially designed to allow inspection of both fuel assemblies and control rod clusters.

9) The control rod mechanism is specially designed to allow removal of the mechanism assembly from the reactor vessel head.

10) Manways are provided in the steam generator, steam drum and channel head to allow access for internal inspection.

11) A manway is provided in the pressurizer top head to allow access for internal inspection.

12) All insulation on primary system component areas required to be inspected is removable.

The use of conventional non-destructive, direct visual and remote visual test techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps have been incorporated into the design and manufacturing procedures in preparation for non-destructive test techniques which may be available in the future. These are:

1) Shop ultrasonic examinations are performed on all internally clad surfaces to an acceptance and repair standard to assure an adequate cladding bond to allow later ultrasonic testing of the base metal. Size of cladding bonding defect allowed is 3/4 inch.

2) The design of the reactor vessel shell in the core area is a clean, uncluttered cylindrical surface to permit future positioning of test equipment without obstruction.

3) Reactor Vessel Post-Operational Ultrasonic Testing - During the manufacturing stage, selected areas of the reactor vessel are ultrasonic tested and mapped to facilitate the in-service inspection program.

   The area selected for ultrasonic testing mapping include:

   a) Vessel flange radius, including the vessel flange to upper shell weld
   b) Middle shell course
   c) Lower shell course above the radial core supports
   d) Exterior surface of the closure head from the flange knuckle to the cooling shroud
   e) Nozzle to upper shell weld
   f) Middle shell to lower shell weld
   g) Upper shell to middle shell weld

   The pre-operational ultrasonic testing of these areas is performed on individual components at any stage of manufacturing where the clad components or sub-assemblies have undergone an interstage stress relief operation.

Various tests are currently underway to determine the effect of cladding surface finish on ultrasonic inspectability of vessel material.

The exact procedures for in-service inspection as detailed in the Technical Specifications are as follows:

REACTOR COOLANT SYSTEM IN-SERVICE INSPECTION

Applicability

Applies to preoperational and in-service structural surveillance of the reactor vessel and reactor coolant system boundary.

Objective

To assure the continued integrity of the reactor coolant system boundary.

Specifications

1. Prior to initial plant operation, an ultrasonic survey shall be made of reactor vessel shell welds, vessel nozzles, vessel flange welds, piping system butt welds and major welds on the pressurizer, steam generator, coolant piping and components to establish preoperational system integrity and establish baseline data.

2. The inspection interval shall be 10 years.

3. Postoperational nondestructive inspections listed in Table 4.5-2 shall be performed as specified. The results obtained from compliance with this specification shall be evaluated after 5 years and the conclusions of this evaluation shall be reviewed with the AEC.

4. The structural integrity of the reactor coolant system boundary shall be maintained at the level required by the original acceptance standards

throughout the life of the plant. Any evidence as a result of the inspections listed in Table 4.5-2, that defects have initiated or grown shall be investigated, including evaluation of comparable areas of the reactor coolant system.

5. The following definitions shall apply to the inspection methods to be employed in Table 4.5-2:

   (1) UT - Volumetric examination using ultrasonic techniques.

   (2) RT - Volumetric examination using radiography.

   (3) PT - Surface examination using liquid penetrant methods.

   (4) V - Visual examination by direct vision or by means of remote viewing devices.

   (5) IV - Indirect visual examination performed during periods when the Reactor Coolant System is subjected to hydrostatic test pressures.

6. Detailed records of each inspection shall be maintained to allow comparison and evaluation of future inspections.

TABLE 4.5-1

REACTOR COOLANT SYSTEM QUALITY ASSURANCE PROGRAM

Component		RT*	UT*	PT*	MT*	ET*
1.	Steam Generator					
	1.1 Tube Sheet					
	1.1.1 Forging		yes (+)		yes	
	1.1.2 Cladding		yes	yes (++)		
	1.2 Channel Head					
	1.2.1 Casting	yes			yes	
	1.2.2 Cladding			yes		
	1.3 Secondary Shell & Head					
	1.3.1 Plates		yes			
	1.4 Tubes		yes			yes
	1.5 Nozzles (forgings)		yes		yes	
	1.6 Weldments					
	1.6.1 Shell, longitudinal	yes			yes	
	1.6.2 Shell, circumferential	yes			yes	
	1.6.3 Cladding (Channel Head-Tube Sheet joint cladding restoration)			yes		
	1.6.4 Steam and Feedwater Nozzle to shell	yes			yes	
	1.6.5 Support brackets				yes	
	1.6.6 Tube to tube sheet			yes		
	1.6.7 Instrument connections (primary and secondary)				yes	
	1.6.8 Temporary attachments after removal				yes	
	1.6.9 After hydrostatic test (all welds and complete channel head)				yes	
	1.6.10 Nozzle safe ends (if forgings)	yes		yes		
	1.6.11 Nozzle safe ends (if weld deposit)			yes		
2.	Pressurizer					
	2.1 Heads					
	2.1.1 Casting	yes			yes	
	2.1.2 Cladding			yes		
	2.2 Shell					
	2.2.1 Plates		yes		yes	
	2.2.2 Cladding			yes		
	2.3 Heaters					
	2.3.1 Tubing (++++)		yes	yes		
	2.3.2 Centering of element	yes				
	2.4 Nozzle		yes	yes		

(+) Flat Surfaces Only
(++) Weld Deposite Areas Only

TABLE 4.5.1 (Continued)

Component	RT*	UT*	PT*	MT*	ET*
2.5 Weldments					
2.5.1 Shell, longitudinal	yes			yes	
2.5.2 Shell, circumferential	yes			yes	
2.5.3 Cladding			yes		
2.5.4 Nozzle Safe End (if forging)	yes		yes		
2.5.5 Nozzle Safe End (if weld deposit)			yes		
2.5.6 Instrument Connections			yes		
2.5.7 Support Skirt				yes	
2.5.8 Temporary Attachments after removal				yes	
2.5.9 All welds and cast heads after hydrostatic test				yes	
2.6 Final Assembly					
2.6.1 All accessible weld surfaces after hydrostatic test				yes	
3. Primary Coolant Piping					
3.1 Fittings (Castings)	yes		yes		
3.2 Fittings (Forgings)		yes	yes		
3.3 Pipe†		yes	yes		
3.4 Weldments					
3.4.1 Circumferential	yes		yes		
3.4.2 Nozzle to run pipe (no RT for nozzles less than 3 inches)	yes		yes		
3.4.3 Instrument connections			yes		
4. Pumps					
4.1 Casting	yes		yes		
4.2 Forgings		yes	yes		
4.2.1 Main Shaft		yes	yes		
4.2.2 Main Studs		yes	yes		
4.2.3 Flywheel (Rolled Plate)		yes			
4.3 Weldments					
4.3.1 Circumferential	yes		yes		
4.3.2 Instrument connections			yes		
5. Reactor Vessel					
5.1 Forgings					
5.1.1 Flanges		yes		yes	
5.1.2 Studs		yes		yes	
5.1.3 Head Adapters		yes	yes		
5.1.4 Head Adapter Tube		yes	yes		
5.1.5 Instrumentation Tube		yes	yes		
5.1.6 Main Nozzles		yes		yes	
5.1.7 Nozzle Safe Ends (If forging is employed)		yes	yes		

† except pressurizer surge line – UT only

TABLE 4.5.1 (Continued)

Component	RT*	UT*	PT*	MT*	ET*
5.2 Plates		yes		yes	
5.3 Weldments					
5.3.1 Main Seam	yes			yes	
5.3.2 CRD Head Adapter Connection			yes		
5.3.3 Instrumentation tube connection			yes		
5.3.4 Main nozzles	yes			yes	
5.3.5 Cladding		yes(+++)	yes		
5.3.6 Nozzle-safe ends (If forging)	yes		yes		
5.3.7 Nozzle safe ends (If weld deposite)	yes		yes		
5.3.8 Head adaptor forging to head adaptor tube	yes		yes		
5.3.9 All welds after hydrotest				yes	
6. Valves					
6.1 Castings	yes		yes		
6.2 Forgings (No UT for valves two inch and smaller)		yes	yes		

* RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle
ET - Eddy Current

(+) Flat Surfaces Only
(++) Weld Deposit Areas Only
(+++) UT of Clad Bond-toBase Metal
(++++) Or a UT and ET

TABLE 4.5-2

INSERVICE INSPECTION REQUIREMENTS FOR INDIAN POINT NO. 2

Component and Examination Area	Examination Method	Extent and Frequency of Examination (1)
A. Reactor Vessel (2)		
1. Closure studs and nuts	V and UT	The examination performed during each inspection interval shall cumulatively cover the entire number of studs and nuts.
2. Cladding - Six patches (each 36 square inches) evenly distributed in accessible sections of the vessel shell	V	The examination performed during each inspection interval shall include all of the patch areas subject to examination.
3. Instrument penetrations at bottom of vessel	IV	The extent of examination shall include twenty-five percent of the penetrations in the vessel during the inspection interval.
4. Primary nozzle to safe-end welds and safe-end to reactor coolant pipe welds	V	The individual examination performed during each inspection shall cover 100 percent of the circumference of the safe-end welds. All of the safe-end welds shall be examined during the inspection interval.

TABLE 4.5-2 (Continued)

Component and Examination Area	Examination Method	Extent and Frequency of Examination (1)
5. Interior surfaces, internals and integrally welded internal supports	V	The examination of interior vessel surfaces, internals, and the space below the reactor core, which are made accessible for examination by the removal of components during normal refueling outages shall be performed during each refueling period. Where access to the space below the reactor core during normal refueling outages precludes inspection of this space, at least one examination, at or near the end of each inspection interval, shall be conducted under conditions which enable inspection.
B. Reactor Vessel Head		
1. Head to flange circumferential weld	UT	The individual examinations performed during each inspection interval shall cumulatively cover 100 percent of the weld.
2. Control rod drive and instrument penetrations	IV	The examinations performed during each inspection interval shall cumulatively cover at least 25 percent of the total number of penetrations.
3. Control rod drive housing pressure containing welds	V	The examination performed during each inspection interval shall cumutalively cover at least 25 percent of the total number of such housings.

TABLE 4.5-2 (Continued)

Component and Examination Area	Examination Method	Extent and Frequency of Examination (1)
4. Cladding - six patches (each 36 square inches) evently distributed in the closure head	V and PT	Same as A-2.
C. Pressurizer (3)		
1. Longitudinal and circumferential welds	V and UT	The examination performed during each inspection interval shall cover at least 10 percent of the length of each longitudinal weld and 5 percent of the length of each circumferential weld.
2. Heater connections	IV	The extent of examination shall include 25 percent of the penetrations during the inspection interval.
3. Pressure retaining bolting under 2 inches in diameter	V and UT	The extent of the examination performed during each inspection interval shall cumulatively cover the entire number of bolts, studs and nuts.
4. Interior cladding - one patch (36 square inches) near the manway	V	The examination of the patch may be performed at or near the end of the inspection interval.

TABLE 4.5-2 (Continued)

Component and Examination Area	Examination Method	Extent and Frequency of Examination (1)
5. Nozzle to vessel welds	IV	The extent of examination shall cover all of the accessible nozzles during each refueling outage.
6. Integrally welded vessel supports	UT	The examination of the vessel support skirt during each inspection interval shall be, at least 10 percent of the lineal feet of welding to the vessel.
D. Steam Generators		
1. Tube sheet to head weld	V and UT	The examination performed during each inspection interval shall cover at least 10 percent of the length of the weld.
2. Pressure retaining bolting under 2 inches in diameter	V and UT	Same as C-3.
3. Integrally welded vessel supports	V and PT	The examination performed during each inspection interval shall be, at least, 10 percent of the lineal feet of welding to the vessel.
4. Interior cladding - one patch (36 square inches) near each manway	V	The examination of the patches may be performed at or near the end of inspection interval.

TABLE 4.5-2 (Continued)

Component and Examination Area	Examination Method	Extent and Frequency of Examination (1)
E. CVCS Regenerative Heat Exchanger		
1. Longitudinal and circumferential welds, including tube sheet to head and tube sheet to shell welds	V and UT	The examinations performed during each inspection interval shall cover at least 10 percent of the length of each longitudinal weld and 5% of the length of each circumferential weld.
2. Supports and hangers	V	The examination performed during each inspection interval shall cumulatively cover all support members and structures.
F. Piping Pressure Boundary		
1. Circumferential and longitudinal pipe welds (excluding sampling and instrumentation piping, and thermowells, and all other piping 2 inches in diameter and smaller).	V and UT	The examination performed during each inspection interval shall cumulatively cover 25 percent of the total number of circumferential joints, selectively distributed within the system boundary. In the case of longitudinal joints, the examination shall include one foot of weld from the intersection with the circumferential weld selected for examation.

TABLE 4.3-2 (Continued)

Component and Examination Area	Examination Method	Extent and Frequency of Examination (1)
2. Sampling and instrumentation piping welds, and thermowells	V and IV	Same as F-1.
3. Pressure retaining bolting under 2 inches in diameter	V and UT*	Same as C-3.
4. Integrally welded supports	UT or PT and V	The examination required to be performed during each inspection interval shall cumulatively cover 25 percent of the total number of integrally welded supports within the system boundary.
5. Piping supports and hangers	V	Same as E-2.
G. Pump Pressure Boundary		
1. Pump casing welds	UT or RT and V	The examination performed during each inspection interval shall include 100 percent of the pressure containing welds in, at least, one pump (with pressure containing welds) in each group of pumps performing similar functions in the system. The examination of the welds may be performed at or near the end of the inspection interval.

* UT down to and including 1-7/8 inches in diameter only

TABLE 4.5-7 (Continued)

Component and Examination Area	Examination Method	Extent and Frequency of Examination (1)
2. Pump casings	V	The internal surfaces of one disassembled pump, (with or without pressure containing welds) in each of the group of pumps performing similar functions in the system shall be visually examined during each inspection interval. The examinations of pump casings may be performed at or near the end of the inspection and may be performed on the same pump selected for the volumetric examination of pressure containing welds.
3. Pressure retaining bolting 2" and larger in diameter	V and UT	The examination performed during each inspection interval shall cumulatively cover the entire number of bolts and studs. Examination of bushings, threads and ligaments in base material of flanges may be performed from a face of the flange and are required to be examined only when the connection is disassembled.
4. Pressure retaining bolting under 2" in diameter	V and UT*	Same as C-3.
5. Integrally welded supports	UT or PT and V	Same as F-4.
6. Supports and hangers	V	Same as E-2.

* UT down to and including 1-7/8 inches in diameter only.

TABLE 4.5-2 (Continued)

Component and Examination Area	Examination Method	Extent and Frequency of Examination (1)
H. Valve Pressure Boundary		
1. Valve body welds on valves 3" and over in nominal pipe size	V and UT	The examinations performed during each inspection interval shall include 100 percent of the pressure containing welds in at least one valve (with pressure containing welds), in each group of valves of the same construction design, manufacturing method and manufacturer, performing similar functions in the system. The examination may be performed at or near the end of each inspection interval.
2. Valve bodies on valves 3" and over in nominal pipe size	V	The internal surfaces of one disassembled valve (with or without pressure containing welds) in each of the group of valves of the same construction design, manufacturing method, manufacturer and performing similar functions in the system shall be examined during each inspection interval. The examination of the valve bodies may be performed on the same valves selected for volumetric examination of the pressure containing welds.

Component and Examination Area	Examination Method	Extent and Frequency of Examination (1)
3. Pressure retaining bolting under 2" in diameter	V and UT*	Same as C-3.
4. Integrally welded supports	UT or PT and V	Same as F-4.
5. Supports and hangers	V	Same as E-2.
I. Excess Letdown Heat Exchanger		
1. Longitidinal and circumferential welds including shell to flange weld	V and UT	Same as E-1.
2. Pressure retaining bolting under 2" in diameter	V	Same as C-3
3. Supports and hangers	V	Same as E-2.
J. Reactor Vessel Irradiation Specimens	charpy V-notch, tensile and wedge opening loading (WOL)	Capsule 1 Replacement of 1st region of core Capsule 2 Replacement of 2nd region of core Capsule 3 Replacement of 4th region of core Capsule 4 End of the 10th year of operation Capsule 5 End of the 15th year of operation Capsule 6 End of the 20th year of operation
K. Primary Pump Flywheel	V and UT	The flywheel shall be visually examined at the first refueling. At the fourth refueling the outside surface shall be examined by ultrasonic methods.

* UT down to and including 1-7/8 inches in diameter only.

NOTES

1. With the exception of those components or areas for which the examination may be deferred to the end of the inspection interval, at least 25 percent of the required examinations shall have been completed by the expiration of one-third of the inspection interval (with credit for no more than 33 1/3 percent if additional examinations are completed) and at least 50 percent shall have been completed by the expiration of two-thirds of the inspection interval (with credit for no more than 66 2/3 percent). The remaining required examinations shall be completed by the end of the inspection interval. Successive inspections shall meet the requirements of paragraph ISI-243 of the ASME Rules for In-Service Inspection of Nuclear Reactor Coolant Systems.

2. Examination of certain reactor vessel areas such as longitudinal and circumferential shell welds, vessel to flange weld, primary nozzle to vessel welds, etc., require the use of equipment and techniques which have not as yet been fully developed. In anticipation that such equipment and techniques will be developed within the next several years to the point of practical application, a preoperational ultrasonic survey of these areas within the reactor vessel will be made to establish baseline data.

3. Examination of the pressurizer longitudinal and circumferential welds will be performed on accessible portions of the pressurizer shell. Approximately 50 percent of the shell is enclosed in a biological and missile shield and is therefore not accessible for examination.

APPENDIX 4A
DETERMINATION OF REACTOR PRESSURE VESSEL NDTT

1. MEASUREMENT OF INTEGRATED FAST NEUTRON (E >1.0 MEV) FLUX AT THE IRRADIATION SAMPLES

The spectrum of neutron fluxes at the irradiation samples is obtained from the multigroup diffusion code PIMG [1]. Dosimeters include CdO shielded U-238, Np-237, Co Al, Cu, Ni, Cd shielding C Al, and Fe from specimens will be contained in the capsule assemblies.

The procedure for measurement of fast neutron flux by the FE^{54} (n,p) Mn^{54} reaction is described below. The measurement technique for the other dosimeters, which are sensitive to different portions of the neutron spectrum, is similar.

The product of this reaction Mn^{54} has a half life of 314 days and emits gamma rays of 0.84 Mev energy which are easily detected using a Na I scintillator. In irradiated steel samples, chemical separation of the Mn^{54} may be performed to ensure freedom from interfering activities. This separation is simple and very effective, yielding sources of very pure Mn^{54} activity. In some samples all the interferences may be corrected for by gamma spectrometric methods without any chemical separation. The count data is used to give the specific activity of Mn^{54} per gram of iron. Because of the relatively long half life of Mn^{54} the flux may be calculated for irradiation periods up to about two years. Beyond this time the dosimeter begins to reflect the later stages of the irradiation. Calculation of total does is from flux and integrated power output. The burnout of the Mn^{54} produced is not significant until the thermal flux is about 10^{14} neutrons cm^{-2} sec^{-1}.

The analysis of the sample requires that two steps be completed, one the measurement of Mn^{54} disintegration rate per unit mass of sample and second measurement of the iron content of the sample. Having completed these analyses, the calculation of the flux is as follows:

For an irradiation the activity of any activation product is given by:

$$A = \phi \sigma N (1 - e^{-\lambda t_i}) e^{-\lambda t_d} \qquad (1)$$

where ϕ is the neutron flux
σ the cross-section
N number of target atoms
λ decay constant of product
t_i irradiation time
t_d decay time from end of irradiation to counting time

Then for a power reactor operating at various power levels over some long period we should allow for flux changes by dividing the exposure period into several parts and normalizing the flux in each part as that fraction of full power represented. Then for τ periods:

$$A = \phi_m \sigma N \sum_{1}^{\tau} (1-e^{-\lambda t_{i_n}}) e^{-\lambda t_{d_n}} F_n \qquad (2)$$

Where ϕ is flux at maximum power

t_{i_n} irradiation time in n^{th} period

t_{d_n} cooling time from end of n^{th} period

F_n flux normalizing factor which is

$$\frac{\text{actual power output in } n^{th} \text{ period}}{\text{maximum possible in } n^{th} \text{ period}}$$

If now we write

$$\phi_m / N = C \sum_{1}^{55} \phi_{PIMG}(E,r) \cdot \sigma_{F_e}(E) \qquad (3)$$

where the right hand side of equation (3) is the sum of the products of PIMG fluxes and the Fe^{54} (n,p) Mn^{54} cross section[2] averaged over the PIMG energy groups, then the measured neutron flux (E > 1 Mev) is given by

$$\phi\ (E > 1\ \text{Mev}) = C \sum_{E=1.0}^{10\ \text{Mev}} \phi_{PIMG}(E,r) \qquad (4)$$

where C is a constant

The error involved in the measurement of the specific activity of the dosimeter after irradiation is estimated to be ± 5%.

2. CALCULATION OF INTEGRATED FAST NEUTRON (E > 1.0 MEV) FLUX AT THE IRRADIATION SAMPLES

The method to be described herein is an approximation to the ideal 3 dimensional neutron transport solution but correlations between its predictions and measurements on samples irradiated in the Yankee and Saxton cores indicate good agreement.

The spectrum of neutron fluxes at the capsule location is obtained from the one dimensional multigroup diffusion code PIMG[1] for the array of annular shields surrounding a cylindrical core of infinite height. The cylindrical core has a cross sectional area equal to that of the actual core. The radial source distribution chosen for the core represents the expected average over the life of the plant. The magnitude of the neutron fluxes generated by the PIMG Code, which does not treat transport effects, is adjusted by application of a spatial correction factor. This factor is the ratio of the fast neutron dose rate calculated by the SPIC-1[3] code for an all water medium surrounding a typical Westinghouse PWR to the fast neutron dose rate obtained by P1MG in the identical geometry. The SPIC-1 fast neutron dose rate calculation uses an empirical fast neutron attenuation kernel in the form of a linear combination of single exponentials which has been fitted to the experimental fast neutron dose rate distribution in pure water.

The axial and azimuthal variations of neutron flux at the capsule location are determined separately. The axial distribution is expressed as the ratio of the normalized results of two calculations using PDQ4,[4] a two dimensional 4 group (r,z) diffusion code. In the first of these an infinitely high equivalent cylindrical core with a fission neutron source strength S_I, per unit height is surrounded by an all water medium containing the capsule location. In the second, the finite height is surrounded by an all water medium. The fixed source option of the PDQ4 code is selected so that the axial variation of source strength in the core represents a good approximation to the average over the core life. The radial distribution is identical to that chosen for P1MG. The ratio,

$$\frac{\phi\,(E,r,z)_F}{S_F} \times \frac{S_I}{\phi\,(E,r)_I}$$

where subscripts F and I denote finite and infinite core representations respectively, is the required axial correction term.

The azimuthal distributions of neutron fluxes at the sample location are derived from a comparison of the results of the two dimensional 4 group (x,y) code PDQ3[5] and the one dimensional 4 group diffusion program AIM-5[6]. In the PDQ3 calculation the core, whose shape can be specified exactly, is surrounded by an all water medium. The radial and azimuthal source distributions in the core are both reasonable approximations to the averages expected during the core life. The radial source distribution in the AIM-5 calculation, in which the equivalent cylindrical core is surrounded by an all water medium, is identical to that chosen for PIMG.

The product of,

1) The spatial corrected PIMG results,

2) Axial correction term, and

3) Azimuthal [illegible]ection term,

defines the three dimensional variation of neutron flux at the sample locations.

The technique described above overpredicts Saxton measurements by 30 per cent and the Yankee measured values by 14 per cent. In both reactors the measured results are averages for a set of specimens in a capsule located outside the thermal shield opposite a core corner. It has also given excellent agreement with measured data reported for the PM2A reactor. Based on the above evidence, it is concluded that the PIMG calculation, corrected as described, is conservative by approximately 20 per cent.

3. MEASUREMENT OF THE INITIAL NDTT OF THE REACTOR PRESSURE VESSEL BASE PLATE AND FORGINGS MATERIAL

The unirradiated or inital NDTT of pressure vessel base plate and forgings material is presently measured by two methods. These methods are the drop weight test per ASTM E208 and the Charpy V-notch impact test (Type A) per ASTM E23. The NDTT is defined in ASTM E208 as "the temperature at which a specimen is broken in a series of tests in which duplicate no break performance occurs at 10°F higher temperature". Using the Charpy V notch test, the NDTT is defined as the temperature at which the energy required to break the specimen is a certain "fixed" value. For SA 302B and A508 Class 2 steel the ASME _II Table N-421 specifies an energy value of 30 ft-lb. This value is based on a correlation with the drop weight test and is referred to as the "30 ft-lb-fix". A curve of the temperature versus energy absorbed in breaking the specimen is plotted. To obtain this curve, 15 tests are performed which include three tests at five different temparatures. The intersection of the energy versus temperature curve with the 30 ft-lb ordinate is designated as the NDTT.

The available data indicates differences as great as 40 degrees between curves plotted through the minimum and average values respectively. The determination of NDTT from the average curve is considered representative of the material and is consistent with procedures as specified in ASTM E23. In assessing the NDTT shift due to irradiation, the translation of the average curve is used.

As part of the Westinghouse surveillance program referred to above, Charpy V-impact tests, tensile tests, and fracture mechanico specimens are taken from the core region plates and forgings, and core region weldments including heat-affected zone material. The test locations are similar to those used in the tests by the fabricator at the plate mill.

The uncertainties of measurement of the NDTT of base plate are:

1) Differences in Charpy V-notch foot pound values at a given temperature between specimens.

2) -iation of impact properties through plate thi_kness.

The fracture toughness technology for pressure vessels and correlation with service failures based on Charpy V-notch impact data are based on the averaging of data. The Charpy V-notch 30 ft-lb "fix" temperature is based on multiple tests by the material supplier, the fabricator, and by Westinghouse as part of the surveillance program. The average of sets of three specimens at each test temperature is used in determining each of five data points (total of 15 specimens). In the review of available data, differences of 0°F to approximately 40°F have been observed in comparing curves plotted through the minimum and average values respectively. The value of NDTT derived from the average curve is judged to be representative of the material because of the averaging of at least 15 data points, consistent with the specified procedures of ASTM E23. In the case of the assessment of NDTT shift due to fast neutron flux, the displacement of transition curves is measured. The selection of maximum, minimum or average curves for this assessment is not significant since like curves would be used.

There are quantitative differences between the NDTT measurements at the surface, 1/4 thickness or the center of a plate. Differences in NDTT between 1/4T and the center in heavy plates have been observed to vary from improvement in the NDTT to increases up to 85°F. The NDTT at the surface has been measured to be as much as 85°F lower than at 1/4T.

The 1/4T location is considered conservative since the enhanced metallurgical properties of the surface are not used for the determination of NDTT. In addition, the limiting NDTT for the reactor vessel after operation will be based on the NDTT shift due to irradiation. Since the fast neutron dose is highest at the inner surface, usage of the 1/4T NDTT criterion is conservative.

Data is being accumulated on the variation of NDTT across heavy section steels at WAPD. Similarly, the Pressure Vessel Research Committee is sponsoring an evaluation of properties of pressure vessel steels in plates 6 to 12 inches thick. Preliminary data has shown NDTT differences between 1/4T and center of less than 20°F. The present criterion of using NDTT + 60°F at the 1/4T location without taking advantage of the enhanced properties at the surface of reactor vessel plates is conservative.

To assess any possible uncertainties in the consideration of NDTT shift for welds heat affected zone, and base metal, test specimens of these three "material types" has been included in the reactor vessel surveillance program.

APPENDIX 4B

SUPPORT STRUCTURES FOR REACTOR COOLANT SYSTEM COMPONENTS

The Reactor Coolant System Components and their supports are designed as Class I seismic components as discussed in Appendix A.

Reactor Vessel

The reactor vessel support structure consists of a circular box section ring girder, fabricated of carbon steel plates. The bottom flange of the girder is in continuous contact (except for openings for neutron detectors) with a non-yielding concrete foundation.

The reactor vessel has four supports located at alternate nozzles. Each support bears on a support shoe, which is fastened to the support structure. The support shoe is a structural member that transmits the support loads to the supporting structure. The support shoe is designed to restrain vertical, lateral and rotational movement of the reactor vessel, but allows for thermal growth by permitting radial sliding at each support on bearing plates.

Steam Generators

The steam generators are supported within a caged structural system consisting of four connected columns all welded together, fabricated of carbon steel members, with provisions for limited movement of the structure in a horizontal direction to accommodate piping expansion with a system of "Lubrite" plates, hydraulic snubbers, guides and stops. The "Lubrite" plates, hydraulic snubbers, guides and stops are designed as rigid support to resist the action of seismic and pipe break loads.

Reactor Coolant Pump

Each reactor coolant pump is supported on a three-legged structural system consisting of three connected columns fabricated of carbon steel members, structural sections and pipe. Provisions for limited movement of the structure in any horizontal direction to accommodate piping expansion is accomplished with a sliding "Lubrite" base plate arrangement and a system of tie rods and anchor bolts which restrain the structure from movement beyond the calculated limits.

Pressurizer

Pressurizer is supported on a free-standing structural system, consisting of six connected columns fabricated of carbon steel members, all welded together and secured at the base by anchor bolts.

Piping

The reactor coolant piping layout is designed on the basis of providing "floating" supports for the steam generator and reactor coolant pump in order to absorb the thermal expansion from the fixed or anchored reactor vessel. A comprehensive thermal analysis is performed to assure that stresses induced by linear thermal expansion are within code limit.

APPENDIX 4C

PROCEDURE FOR PLUGGING A TUBE IN A STEAM GENERATOR

The procedure for plugging a tube in a steam generator is as follows:

1. The reactor is shutdown and taken to cold shutdown condition i.e., both primary and secondary sides are depressurized and cold. Decay heat is removed via the residual heat removal system.

2. The reactor coolant level is lowered until the level is between the bottom of the steam generator and the hot leg elbow, thus maintaining the remainder of the hot leg between the elbow and the vessel full of water.

   The residual heat removal suction line is connected to the hot leg of loop 2 and the return line is connected to the cold legs of all four loops. Thus, lowering of water to this level does not affect operation of the residual heat removal system.

3. The steam generator is entered via the two manways, one on either side of the channel head partition plate. Prior to the performance of any work, the area around the steam generator is monitored to determine the radiation level. In the event of high radiation levels, biological shielding is installed around the coolant channel head, and portable respiratory apparatus is used if required.

4. Temporary membranes are placed over the inlet and outlet reactor coolant legs to the steam generator, to prevent any debris from entering the reactor coolant system.

5. The leaking tube is located and plugged.

6. The temporary membranes are removed and the manway covers are replaced thus resealing the system.

7. The reactor coolant level is raised to its normal cold shutdown level, and the air which has been introduced into the steam generator is vented in the normal manner i.e., in the same way as following a refueling shutdown.

APPENDIX 4D

SENSITIZED STAINLESS STEEL

Introduction

Westinghouse had evaluated the use of sensitized stainless steel for reactor components in pressurized water reactors. The results of this evaluation are summarized in WCAP 7477-L (Westinghouse proprietary) which covers the nature of sensitization, conditions leading to stress corrosion and associated problems with both sensitized and non-sensitized stainless steel. The results of extensive testing and service experience that justify the use of stainless steel in the sensitized condition for components in Westinghouse PWR Systems is presented in the report.

Stainless steel is subject to stress corrosion, and must not be exposed to certain environments which will cause cracking. Chlorides and fluorides are the most important contaminants, although oxygen, low pH, and elevated temperature generally must also be present to cause cracking. When subjected to environments that cause cracking, the cracks are usually intergranular in sensitized stainless steel, whereas they are usually transgranular in stainless steel which is not sensitized.

The stainless steel safe ends on the reactor vessel, pressurizer, and steam generator nozzles may become somewhat sensitized during stress relief of the vessel. The Post Weld Heat Treatment (PWHT) temperatures and minimum time are consistent with ASME Section III requirements. The degree of sensitization of the safe ends varies from plant to plant, depending on the materials used and the detailed processing performed by the various vendors. For Indian Point Unit No. 2, the specific design and construction practices are discussed in the following sections. The outer diameter and inner diameter safe ends of the reactor vessel were overlayed with Type 308L and Inconel weld me[illegible]l to eliminate any question of intergranual attack in areas where there is [illegible]ed accessibility for in-service inspection and plant maintanence. There is complete accessibility to the remaining RCS components. The pre-operational inspection of the RCS components provides assurance that there is no stress corrosion cracking of sensitized stainless steel.

Reactor Coolant System Nozzle Safe Ends

1. Reactor Vessel Primary Nozzle Safe Ends

A. Method of Fabrication (See Figure 4D-1)

(1) Wrought Stainless Steel - Type 316 Forging welded to A-508 nozzle with Inconel weld metal. Attached prior to final P.W.H.T.

(2) Forging was overlayed on I.D. and O.D. with Type 308 L stainless and Inconel weld metal. This was performed in the field after the primary coolant piping was attached to the nozzles.

B. Inspection

(1) Forging Safe Ends were examined by U.T. and P.T. at Combustion Engineering using Sec. III acceptance standards.

(2) Weld overlay of the I.D. and O.D. surfaces was examined by U.T. and P.T. The acceptance standards are shown below:

Ultrasonic Acceptance Standards

Each discontinuity that produces a response equal to or exceeding the calibration reference line and is 1/2 inch or greater in length shall be considered rejectable and shall be removed.

Discontinuities that produce a response equal to or greater than the calibration reference line and exceed 1/4 inch but are less than 1/2 inch in length shall be considered acceptable if separated by a minimum distance of 2 inches from similar discontinuities.

Each discontinuity that produces a response between 50 and 200 percent of the calibration reference line and exceed 1 inch but are not more than 1-1/2 inch in length shall be acceptable if separated by a minimum distance of 2 inches from similar indications.

Penetrant Inspection Acceptance Standards

1. Examination of welds by liquid penetrant methods shall be made over an area including the welds and base metal extending for at least .5 inch on each side of weld.

2. Surfaces examined by fluid penetrant methods shall be free of laps, fissures, cracks, other linear indications.

3. Weld Area and Adjacent Wrought Type Base Metal(s) - In any 6 inch length of weld and adjacent base metal examined, there shall be no indications greater than .062 in. in maximum dimension, nor shall there be more than six indications whose sum of maximum dimension specified herein. Any 6 inch length of weld shall be interpreted to denote the 6 in. length selected in the least favorable location with respect to the discontinuities disclosed by the inspection test. All surfaces examined shall be free of linearly disposed indications of four or more indications in a line and each separated by 1/16 inch or less, edge to edge.

4. Weld Area and Adjacent Cast Type Base Metal(s) -In any 6 inch length of weld examined, there shall be no indications greater than those defined in 3 above. The adjacent cast base metal shall be free of random indications in excess of those shown in the following table for a distance of not less than .5 inch, from toe(s) of weld:

Size of Indications, In.	Number per Sq. Inch
> 1/8	None
> 1/16 < 2/8	2
< 1/16	10

5. All surfaces examined shall be free of linearly disposed indications of four or more indications in a line and each separated by 1/16 in. or less, edge to edge. Rounded indications are those which are circular or elliptical with the length less than twice the width.

2. Steam Generator Primary Nozzle Safe Ends (See Figure 4D-2)

A. Method of Fabrication

Weld metal buttering applied to carbon steel (A-216 Casting) nozzles prior to final P.W.H.T. Stainless weld metal for the first layer is Type 309 and for the balance is Type 308L.

B. Inspection

Buttered safe ends were examined by P.T. and R.T. using ASME PV&B Code Sec. III acceptance standards.

3. Pressurizer (See Figure 4D-3)

A. Method of Fabrication

Wrought stainless steel pipe or forgings welded to carbon steel (A-216 Casting) nozzles with Type 309 weld metal before PWHT. The surge nozzle safe end is fabricated from SA-312 pipe, Type 316 and the spray, relief, and safety nozzle safe ends from SA-182 forgings, Type 316.

B. Inspection

Wrought material was examined by U.T. and P.T. using Sec. III acceptance standards.

Reactor Coolant System Construction

All primary piping and fittings have been given a solution annealing treatment consisting of heating to 1900 - 1950F, holding 1 hour per inch of thickness and water quenching. This assures that the material will not be sensitized.

Main coolant pipe welds are of Type 308 or 316 stainless steel. Welding was performed by the manual metal arc process after the root pass was completed using an insert followed by three layers using the manual gas shielded tungsten arc process. The maximum energy input possible with the manual metal arc process is on the order of 20,000 joules per linear inch of weld. With the large heat sink available in this thick walled pipe (2.375 to 3.00"), and the interpass temperature control of 350°F maximum, there will be no sensitization of the solution treated pipe during welding.

Venting provisions have been made at high points throughout the Reactor Coolant System to relieve entrapped air when the system is filled and pressurized. Principally, vents are installed on the reactor vessel head, the pressurizer, and the reactor coolant pumps. Additional vents are available on the control rod drive mechanisms, on instruments, and on a number of connecting pipes. For normal venting of the reactor coolant system, only the principal venting points are utilized. The amount of oxygen which could be trapped in the remaining small volumes becomes negligible as the system is pressurized and the oxygen is scavenged by the hydrazine specifically added for this purpose prior to operation. During operation the oxygen levels are kept low consistent with water chemistry requirements as described in the Technical Specifications.

Reactor Coolant System Operational Stresses

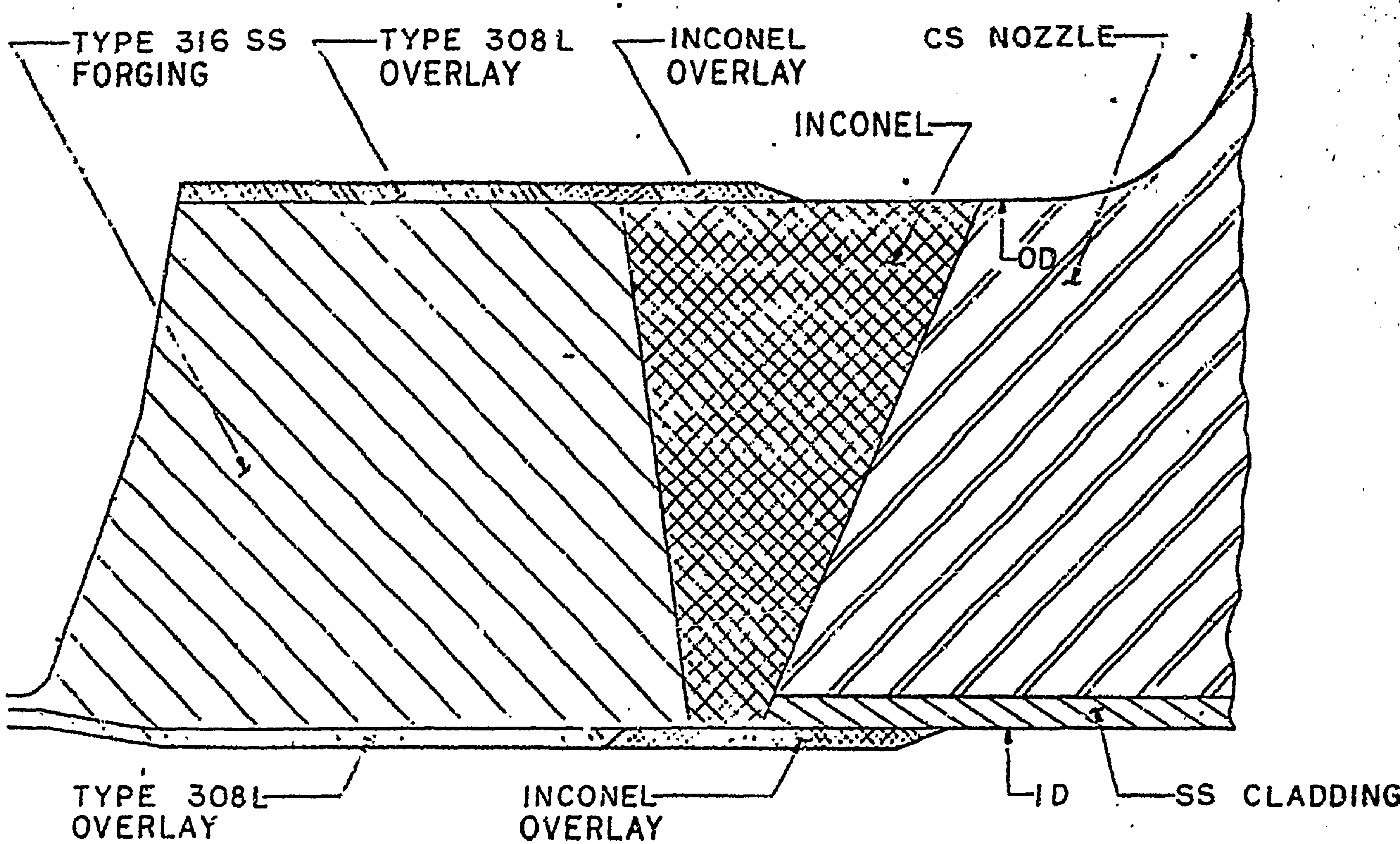
To avoid unusual stresses in areas where nozzle safe ends are joined to the piping, precautions are taken to eliminate unnecessary stresses due to erection of the various components of the Reactor Coolant System. The primary coolant system piping closure pieces are two pipe fitting subassemblies located between the steam generator and the primary coolant pump. The 40 degree elbow of the loop piping is first installed on the steam generator outlet nozzles. Then the gap to be closed by the closure pieces is physically measured between the 40 degree elbow outlet and the inlet nozzle of the pump. These measured dimensions for each individual loop are compensated and adjusted for the expected field weld shrinkage. The resulting net true dimensions are then transmitted to the pipe shop fabricator who prepares the final closure pipe subassemblies for each primary coolant loop. Upon welding these specially dimensioned pipe subassemblies in place, the primary coolant system closure is accomplished for each loop in a condition which is free from cold spring.

As a precaution that the behavior of the Reactor Coolant System during operating conditions is as predicted, measurements are made during incremental temperature increases during the hot functional test. The measurements are made to check the movement of the components at temperature and pressure to insure interferences are not present. The data taken during the test are compared with the flexibility analysis predictions. If some deviation from the predicted clearances is observed, the occurrences will be evaluated. The evaluation of the test data will determine the necessary action, if any, required before power operation to alleviate any potential overstressed areas - particularly at the nozzle safe ends.

In Service Inspection Capability

As a final check on the adequacy of the precautions taken to avoid any Reactor Coolant System failure as a result of severely sensitized stainless steel, a post-operational inspection plan has been developed for the nozzle safe ends within the Reactor Coolant System Boundary. The pressurizer and steam generator stainless steel safe ends which were subjected to the furnace atmosphere during final stress relief are accessible for visual, surface and volumetric inspection upon removal of the insulation at each safe-end. The reactor vessel safe-ends which were subjected to the furnace atmosphere are accessible for limited inspection by removal of the special access plugs provided in the primary concrete just above each nozzle. Upon removal of these plugs and the insulation on the safe end, approximately 120° of the top segment of the safe-ends are accessible for direct visual, surface and remote volumetric inspection.

When specially designed devices for remote ultrasonic inspection and applicable procedures are developed, and when metallurgical considerations indicate that this type of inspection is appropriate and necessary, such inspections will be accomplished utilizing the internal access to the reactor vessel safe ends and the limited external access.



PRIMARY NOZZLE
COMBUSTION ENGINEERING REACTOR VESSEL

Figure 4D-1

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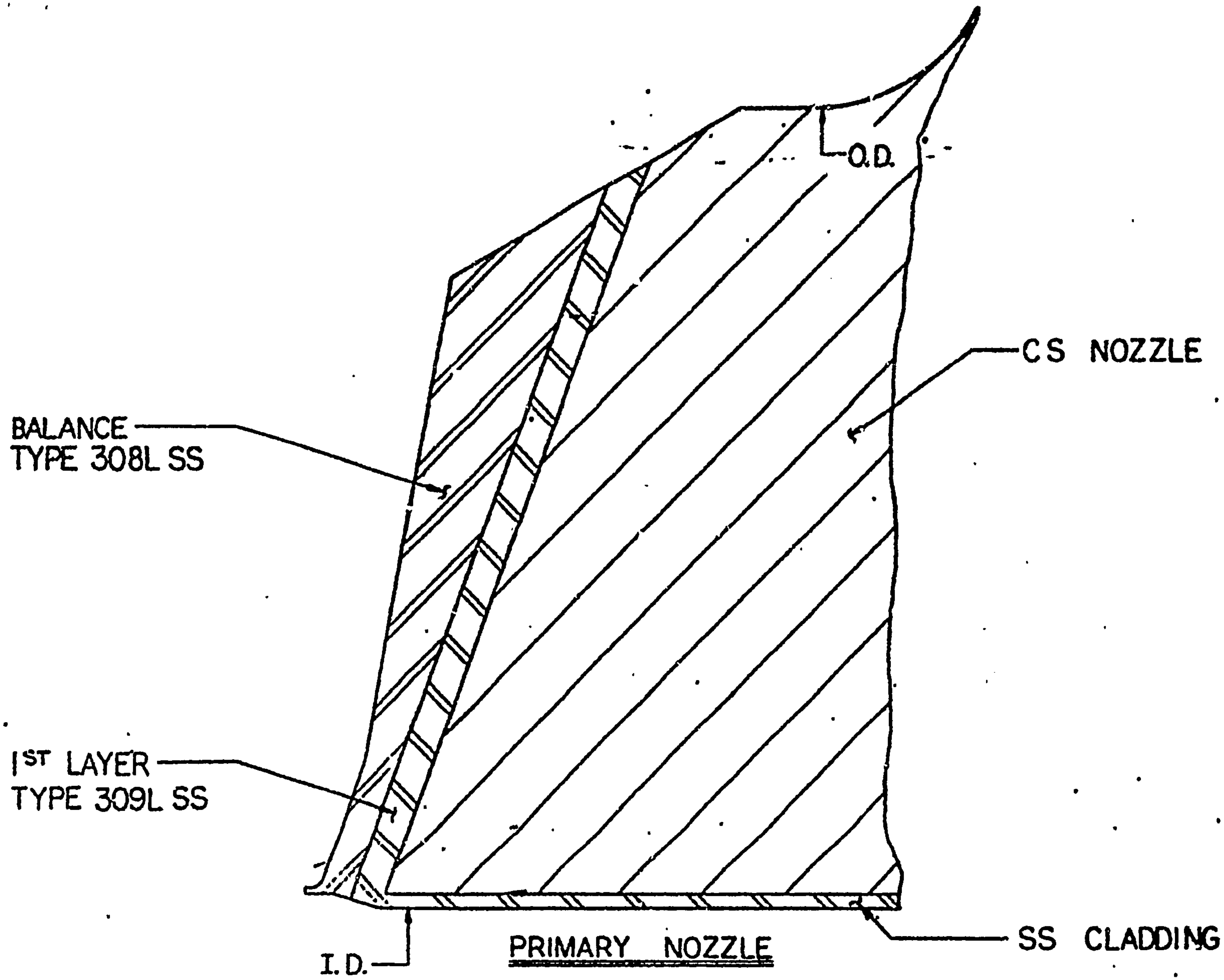

PRIMARY NOZZLE

TAMPA STEAM GENERATORS

Figure 4D-2

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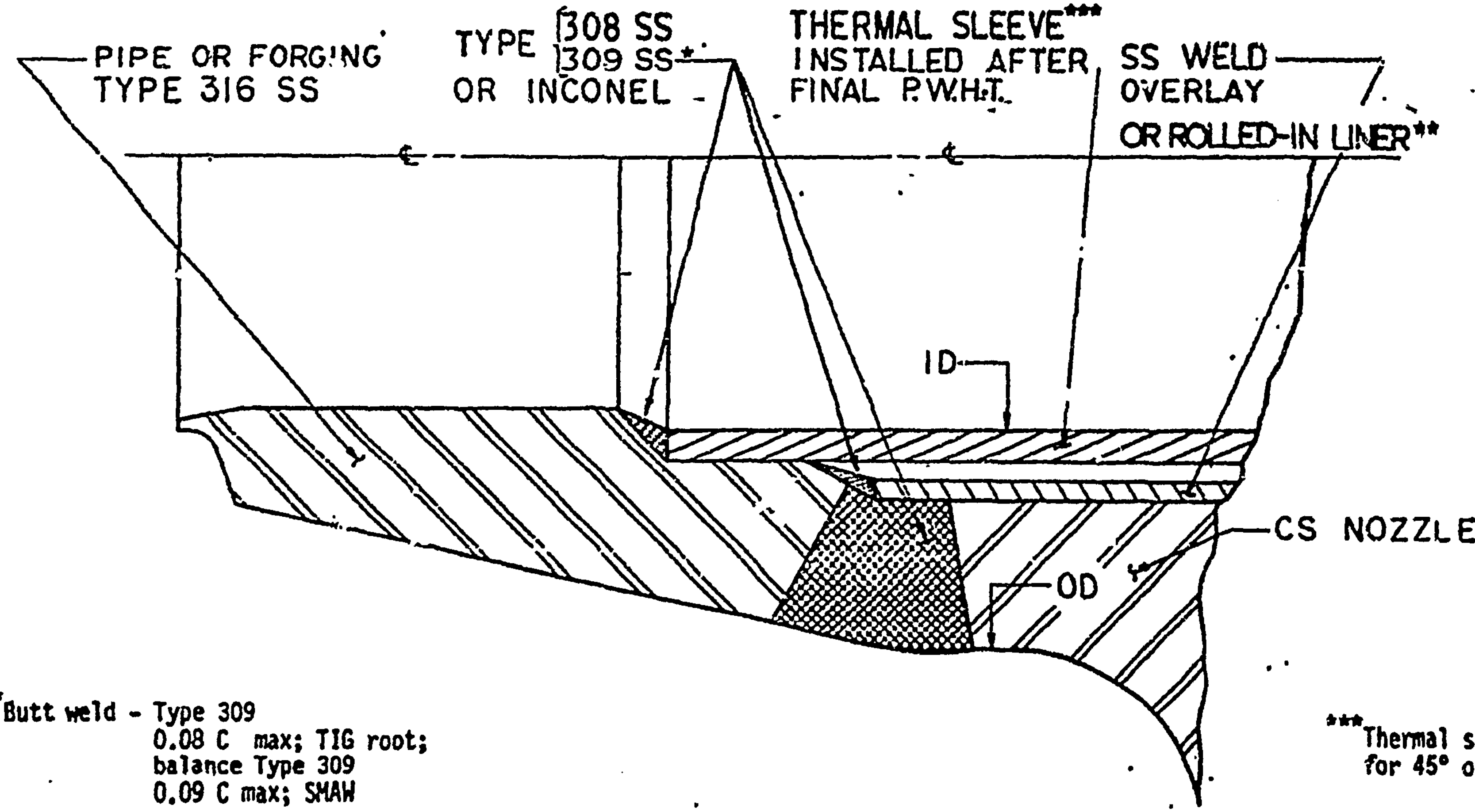

*Butt weld - Type 309
0.08 C max; TIG root;
balance Type 309
0.09 C max; SMAW

Attachment weld of thermal sleeve and rolled-in liner - Type 308 L 0.04 C max; TIG (made after final PWHT)

**Rolled-in liner welded top and bottom for spray, safety, and relief nozzles - Type 309 followed by Type 308 L weld overlay for surge nozzle

***Thermal sleeve welded for 45° of 360°

SPRAY OR SURGE NOZZLE

TAMPA PRESSURIZER

Figure 4D-3

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CHAPTER 5 - CONTAINMENT SYSTEM

5.1 CONTAINMENT SYSTEM STRUCTURES

5.1.1 DESIGN BASIS

The reactor containment completely encloses the entire reactor and reactor coolant system and ensures that essentially no leakage of radioactive materials to the environment would result even if gross failure of the reactor coolant system were to occur. The liner and penetrations are designed to prevent any leakage through the containment. The structure will provide biological shielding for both normal and accident situations.

The reactor containment is designed to safely withstand several conditions of loading and their credible combinations. The major loading conditions are:

a) Occurrence of a gross failure of the reactor coolant system which creates a high pressure and temperature condition within the containment.

b) Coincident failure of the reactor coolant system with an earthquake or wind.

5.1.1.1 Principal Design Criteria

Quality Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required. (GDC 1)

The containment system structure is of primary importance with respect to its safety function in protecting the health and safety of the public.

Quality standards of material selection, design, fabrication, and inspection governing the above features conforms to the applicable provisions of recognized codes and good nuclear practice. The concrete structure f the reactor containment conforms to the applicable portions of ACI-318-63. Further elaboration on quality standards of the reactor containment is g in Section 5.1.1.5.

Performance Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design. (GDC 2)

All components and supporting structures of the reactor containment are designed so that there is no loss of function of such equipment in the event of maximum potential ground acceleration acting in the horizontal and vertical directions simultaneiously. The dynamic response of the structure to ground acceleration, based on the side characteristics and on the structural damping, is included in the design analysis.

The reactor containment is defined as a Class I structure for purposes of seismic design (Appendix A). Its structural members have sufficient capacity to accept without exceeding specified stress limits a combination of normal operating loads, functional loads due to a loss of coolant accident, and the loadings imposed by the maximum potential earthquake.

Fire Protection

Criterion: A reactor facility shall be designed to ensure that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Non-combustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features. (GDC 3)

Fire protection in all areas of the nuclear electric plant is provided by structure and component design which optimizes the containment of combustible materials and maintains exposed combustible material below the ignition temperature. The station is designed on the basis of limiting the use of combustible materials in construction by using fire resistant materials to the greatest extent practical. The reactor containment system is designed to maintain its capability in case of fire to safely shut down and isolate the reactor. Since containment recirculation ventilation charcoal filters are required, special manually-actuated sprays are installed operable from the control room. Containment liner thermal insulation does not support combustion. The bearing oil systems for the reactor coolant pumps are self contained.

Records Requirement

Criterion: The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public.

Records of the design, fabrication, construction and testing of the reactor containment are maintained throughout the life of the reactor.

Reactor Containment

Criterion: The containment structure shall be designed (a) to sustain without undue risk to the health and safety of the public the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without loss of required integrity, and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires, the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public (GDC 10)

The design pressure and temperature of the containment exceeds the peak pressure and temperature occurring as the result of the complete blowdown of the reactor coolant through any rupture of the reactor coolant system up to and including the hypothetical double-ended severance of a reactor coolant pipe. Energy contribution from the steam system is included in the calculation of the containment pressure transient due to reverse heat transfer through the steam generator tubes. The supports for the reactor coolant system are designed to withstand the blowdown forces associated with the sudden severance of the reactor coolant piping so that the coincidental rupture of the steam system is not considered credible.

The containment structure and all penetrations are designed to withstand within design limits the combined loadings of the design basis accident and design and maximum potential seismic conditions.

All piping systems which penetrate the vapor barrier are anchored at the liner. The penetrations for the main steam, feedwater, blow down and sample lines are designed so that the penetration is stronger than the piping system and that the vapor barrier is not breached due to a hypothesized pipe rupture combined, for the case of the steam line, with the coincident internal pressure. The pipe capacity in flexure is assumed to be limited to the plastic moment capacity based upon the ultimate strength of the pipe material. All lines connected to the primary coolant system that penetrate the vapor barrier are also anchored in the secondary shield walls (i.e. walls surrounding the steam generators and reactor coolant pumps) and are each provided with at least one valve between the anchor and the reactor coolant system. These anchors are designed to withstand the thrust, moment and torque resulting from a hypothesized rupture of the attached pipe.

All isolation valves are suppported to withstand, without impairment of valve operability, the combined loadings of the design basis accident and design and maximum potential seismic conditions.

Section 5.1.5 includes a discussion of the details of the design of primary system supports. In addition, the design pressure will not be exceeded during any subsequent long-term pressure transient determined by the combined effects of heat sources, such as residual heat and limited metal-water reactions, structural heat sinks and the operation of the engineered safeguards; the latter utilizing only the emergency electric power supply.

<u>Reactor Containment Design Basis</u>

Criterion: The reactor containment structure, including openings and penetration and any necessary containment heat removal systems, shall be designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a loss-of-coolant accident, including the calculated energy from metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in the emergency core cooling system will not result in undue risk to the health and safety of the public (GDC 49)

The following general criteria are followed to assure conservatism in computing the required structural load capacity:

a) In calculating the containment pressure, rupture sizes up to and including a double-ended severance of reactor coolant pipe are considered.

b) In considering post-accident pressure effects, various malfunctions of the emergency systems are evaluated. Contingent mechanical or electrical failures are assumed to disable one of the diesel generators, two of the five fan-cooler units and one of the two containment spray units. Equipment which can be run from diesel power is described in Section 8.

c) The pressure and temperature loadings obtained by analyzing various loss-of-coolant accidents, when combined with operating loads and maximum wind or seismic forces, do not exceed the load-carrying capacity of the structure, its access opening or penetrations.

The most stringent case of these analyses is summarized below:

Discharge of reactor coolant through a double-ended rupture of the main loop piping, followed by operation of only these engineered safety features which can run simultaneously with power from two of the three on-site diesel generators (two high head safety injection pumps, one recirculation pump, three fan cooler units, one spray pump), results in a sufficiently low radioactive materials leakage from the containment structure that there is not undue risk to the health and safety of the public.

NDT Requirement for Containment Material

Criterion: The selection and use of containment materials shall be in accordance with applicable engineering codes. (GDC 50)

The selection and use of containment materials comply with the applicable codes and standards tabulated in Section 5.1.1.5.

The concrete containment is not susceptible to a low temperature brittle fracture.

The containment liner is enclosed within the containment and thus is not exposed to the temperature extremes of the environs. The containment ambient temperature during operation is between 50 and 120°F. This includes both hot operating and cold shutdown conditions. The minimum service metal temperature of the containment liner is well above the NDT temperature + 30°F for the liner material. Containment penetrations which can be exposed to the environment are also designed to the NDT + 30°F Criterion.

5.1.1.2 Supplementary Accident Criteria

Systems relied upon to operate under post-accident conditions, which are located external to the containment and communicate directly with the containment, are considered to be extensions of the leakage boundary.

The pressure retaining components of the containment structure are designed for the maximum potential earthquake ground motion of the site combined with the simultaneous loads of the design basis accident as follows:

(1) The liner is designed to ensure that no average strains greater than the strain at the guaranteed yield point occur at the factored loads.

(2) The mild steel reinforcement is designed to ensure that no strains greater than the strain at the guaranteed yield point occur at a cross section under the factored loads.

The pressure retaining components of containment subject to deterioration or corrosion in service are provided with appropriate protective means or devices (e.g. protective coatings).

5.1.1.3 Energy and Material Release

The design pressure is not exceeded during any subsequent long term pressure transient determined by the combined effects of heat sources such as residual heat and metal-water reactions, structural heat sinks and the operation of other engineered safety features utilizing only the emergency on-site electric power supply.

The design pressure and temperature on the containment structure are those created by the hypothetical loss-of-coolant accident. The reactor coolant system contains approximately 512,000 lb of coolant at a weighted average enthalpy of 595 Btu/lb for a total energy of 304,000,000 Btu. In a hypothetical accident, this water is released through a double-ended break in the largest reactor coolant pipe, causing a rapid pressure rise in the containment. The reactor coolant pipe used in the accident is the 29-in ID section because rupture of the 31-in ID section requires that the blowdown go through both the 29-in and the 27-1/2-in ID pipes and would, therefore, result in a less severe transient.

Additional energy release is considered from the following sources:

a) Stored heat in the reactor core.
b) Stored heat in the reactor vessel piping and other reactor coolant system components.

c) Residual heat production
d) Limited metal-water reaction energy and resulting hydrogen-oxygen reaction energy.

The following loadings are considered in the design of the containment in addition to the pressure and temperature conditions described above:

a) Structure dead load.
b) Live loads.
c) Equipment loads.
d) Internal test pressure.
e) Earthquake
f) Wind

The capability of the containment to withstand additional energy releases is discussed in Section 14.

5.1.1.4 Engineered Safety Features System Contributions

Five types of engineered safety features are included in the design of this facility to assure containment integrity. These systems are discussed in Section 6 and their effectiveness is analyzed in Section 14

5.1.1.5 Codes and Classifications

The design, materials, fabrication, inspection, and proof testing of the containment vessel complies with the applicable parts of the following:

I. STRUCTURAL

	Code	Title
1.	ASTM A-333, Gr. 1	Specification for Seamless and Welded Steel Pipe for Low Temperature Service
2.	ASTM A-181	Forged or Rolled Steel Pipe Flanges, Forged Fittings, and Valves and Parts for General Service
3.	ASTM A-300, Cl. 1	Specification for Notch Toughness Requirements for Normalized Steel Plates for Pressure Vessels
	Firebox A-201, Gr. B	Specification for Carbon Silicon Steel Plates of Intermediate Tensile Ranges for Fusion Welded Boilers and other Pressure Vessels
4.	ASTM A-36, Gr. C	Specification for Structural Steel
5.	ASTM A-131, Gr. C	Specification for Structural Steel for Ships
6.	ASTM A-240	Specification for Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion-Welded Unfired Pressure Vessels
7.	ASTM A-312	Specification for Seamless and Welded Austenitic Stainless Steel Pipe
8.	ASTM 442, Grade 60	Standard Specification for Carbon Steel Plates with Improved Transition Properties
9.	ASME Boiler & Pressure Vessel Code-Section III	Nuclear Vessels
10.	ASME Boiler & Pressure Vessel Code-Section VIII	Unfired Pressure Vessels
11.	ASME Boiler & Pressure Vessel Code-Section IX	Welding Qualifications
12.	ASTM C-33	Standard Specifications for Concrete Aggregates
13.	ASTM C-150	Standard Specifications for Portland Cement

	Code	Title
14.	ASTM C-172	Method of Sampling Fresh Concrete
15.	ASTM C-31	Method of Making and Curing Concrete Compression and Flexure Test Specimen in Field
16.	ASTM C-39	method of Test for Compressive Strength of Molded Concrete Cylinders
17.	ASTM C-350	Specification for Fly Ash for Use as an Admixture in Partland Cement Concrete
18.	ASTM C-94	Recommended Practice for Winter Concreting
19.	ASTM C-42	Methods of Securing, Preparing, and Testing Specimens from Hardened Concrete for Compressive and Flexural Strengths
20.	ASTM C-494	Specifications for Chemical Admixtures for Concrete
21.	ASTM A-305	Specifications for Minimum Requirements for Deformation of Deformed Bars for Concrete Reinforcement
22.	ASTM A-408	Specifications for Special Large Size Deformed Billet-Steel Bars for Concrete Reinforcement
23.	ASTM A-432	Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 PSI Minimum Yield Strength
24.	Research Council of Riveted & Bolted Structural Joints of the Engineering Foundation	Specification for Structural Joints using ASTM A-325 Bolts
26.	ACI-613	Recommended Practice for Selecting Proportions for Concrete
27.	ACI-306	Recommended Practice for Winter Concreting

Code	Title
28. ACI-318, Part IV-B	Structural Analysis and Proportioning of Members Ultimate Strength Design
29. ACI-318	Building Code Requirements for Reinforced Concrete
30. ACI-Code 505	Reinforced Concrete Chimney Design
31. ACI-315	Manual of Standard Practice for Detailing Reinforced Concrete Structures
32. ASME Nuclear Vessels Code	---
33. ASA N6.2	Safety Standards for the Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors
34. ASA A58.1	American Standard Code Requirements for Minimum Design Loads in Buildings and Other Structures
35. ---	State Building and Construction Code for the State of New York
36. SSPC-SP-6	Commercial Blast Cleaning

5.1.2 CONTAINMENT SYSTEM STRUCTURE DESIGN

5.1.2.1 General Description

The reactor containment structure is a reinforced concrete vertical right cylinder with a flat base and hemispherical dome. A welded steel liner with a minimum thickness of 1/4-inch is attached to the inside face of the concrete shell to insure a high degree of leak-tightness. The design objective of the containment structure is to contain all radioactive material which might be released from the core following a loss-of-coolant accident. The structure serves as both a biological shield and a pressure container.

The structure, as shown on Figures 5.1-1 through 5.1-7 consists of a side walls measuring 148-feet from the liner on the base to the springline of the dome, and has an inside diameter of 135-feet. The side walls fo the cylinder and the dome are 4-ft 6-in. and 3-ft. 6-in. thick respectively. The inside radius of the dome is equal to the inside radius of the cylinder so that the discontinuity at the springline due to the change in thickness is on the outer surface. The cylinderical part of the liner is substantially round. The difference between the minimum and maximum inside diameters at any selected cross section does not generally exceed 0.25% of the nominal diameter at that elevation. Between elevations +43 and +95, the maximum diameter of any cross section is 135'-2", and the minimum diameter is 134'-10" except at the liner closing the temporary opening in the northwest quadrant where a minimum diameter of 134'-8 5/8" was measured. This portion of the liner was erected after all exterior concrete work was completed and is within the local buckle allowance of the liner plates. Above elevation +95 the tolerance on inside diameter does not exceed 0.50% of the nominal diameter of the selected cross section. The liner is erected true and plumb so that the deviation does not exceed 1/500 of the height at the selected cross section (allowing for 2" local buckling of the liner plates).

Particular care is taken in matching edges of cylindrical and hemispherical sections to insure that all joints are properly aligned. Maximum permissible offset of completed joints is 25% of nominal plate thickness. Plates buckled beyond acceptable limits are cut out and replaced with new plates.

The flat concrete base mat is 9-ft. thick with the bottom liner plate located on top of this mat. The bottom liner plate is covered with 3-ft. of concrete, the top to which forms the floor of the containment.

Where uplift from pressure occurs at the outer areas of the mat, the 9 ft. thick mat has sufficient flexural capacity to resist the uplift until it is dissipated.

No hydraulic uplift exists since the bottom elevation of the mat is considerably higher than that of the high water level.

The large mass of the containmen including interior concrete and equipment makes the structure inherently stable from overturning due to seismic motion.

In addition, keying action from the reactor pit and sumps, plus friction between the concrete and rock, prevents sliding of the structure from horizontal ground motion.

The basic structural elements considered in the design of the containment structure are the base slab, side walls and dome acting as one structure under all possible loading conditions. The liner is anchored to the concrete shell by means of stud anchors. The reinforcing in the structure will have a total elastic response to all loads. The lower portions of the cylindrical liner is insulated to avoid thermal deformation of the liner under accident conditions.

The containment structure is inherently safe with regard to common hazards such as fire, flood and electrical storm. The thick concrete walls are invulnerable to fire and only an insignificant amount of combustible material, such as lubricating oil in pump and motor bearings, is present in the containment.

Internal structures consist of equipment supports, shielding, reactor cavity and canal for fuel transfer, and miscellaneous concrete and steel for floors and stairs All internal structures are supported on the mat with the exception of equipment supports secured to the intermediate floors.

A 3-foot thick concrete ring wall serving as a missile and partial radiation shield surrounds the reactor coolant system components and supports the polar-type reactor containment crane. A 2-foot thick reinforced concrete floor covers the reactor coolant system with removable gratings in the floor provided for crane access to the reactor coolant pumps. The four steam generators, pressurizer and various piping penetrate the floor. Spiral stairs provide access to the areas below the floor. A reinforced missile shield surrounds completely the portion of the pressurizer which protrudes above the operating floor thereby protecting the containment liner from postulated valve piece or instrument missiles connected to the pressurizer.

The refueling canal connects the reactor cavity with the fuel transport tube to the spent fuel pool. The floor and walls of the canal are concrete, with wall and shielding water providing the equivalent of 6-feet of concrete.

The floor is 4-foot thick. The concrete walls and floor are lined with 1/4-inch thick stainless steel plate. The linings provide a leakproof membrane that is resistant to abrasion and damage during fuel handling operation.

Waterproofing is provided in the areas of the containment in contact with backfill to prevent ground water seepage. This consists of a coat of bitumastic No. 50, a 5/8" thick layer of hardboard insulation, and a second coat of bitumastic No. 50. Fill for innermost five feet from containment walls is crushed rock of maximum size of 6" and minimum amount of fines. All fill is free of vegetable matter.

5.1.2.2 <u>Design Load Criteria</u>

The following loads are considered to act upon the containment structure creating stresses within the component parts.

a) Dead load consists of the weight of the concrete wall, dome, line, insulation, base slab and the internal concrete. Weights used for dead load calculations are as follows:

1.	Concrete	: 150 lb/ft^3
2.	Reinforcing Steel	: 490 lb/ft^3 using nominal cross-sectional areas of reinforcing as defined in ASTM for bar sizes.
3.	Steel Lining	: 490 lb/ft^3 using nominal cross-sectional area.
4.	Insulation	: 6-lb/ft^3 including stainless steel jacket.

b) Live load consist is of snow and construction loads on the dome and major components of equipment in the containment. Snow and ice loads are assumed to be applied uniformly to the top surface of the dome at an estimated value of 20 pounds per square foot of horizontal projection of the dome. This loading represents approximately 2-foot of snow, which is considered to be a conservative amount since the slope of the dome will tend to cause much of the snow which falls on it to slide off. A construction live load of 50 pounds per square foot has been used on the dome, but will not be considered to act concurrently with the snow load. Equipment loads are considered as specified on the drawings supplied by the manufacturets of the various pieces of equipment.

Design live loads inside the containment building are as follows:

@ El. 68'-0"	-	10-ft strip adjacent to crane wall	= 600 psf
		Remaining strip	= 100 psf
@ El. 95'-0"	-	Concrete slab	= 500 psf
		Grating areas	= 100 psf

c) The internal pressure transient used for the containment design and its variation with time is shown on the pressure-temperature transient curve, Figure 5.1-8. For the free volume of 2,610,000 cubic feet within the containment, the design pressure is 47 psig. This pressure transient is more severe than those calculated for various loss-of-coolant accidents which are presented in Section 14.

d) Thermal expansion stresses due to an internal temperature increase caused by a loss-of-coolant accident is considered. This temperature and its variation with time is shown on the pressure-temperature transient curve, Figure 5.1-8. The maximum temperature at the uninsulated section of the liner under accident conditions is 247°F. For the 1.25 times and 1.50 times design pressure loading conditions given in Section 5.1.2.4, the corresponding liner temperatures will be 235°F and 306°F respectively. The pressure-temperature transient curves for these loading conditons are shown in Figures 5-9 and 5-10 respectively. The maximum operating temperature is 120°F. The design 24-hour mean-low ambient temperature is -5°F.

e) The ground acceleration for the design earthquake has been determined to be 0.1g applied horizontally and 0.05g applied vertically. These values have been resolved as conservative numbers based upon recommendations from Dr. Lynch, Director of Seismic Observatory, Fordham University.

A dynamic analysis is used to arrive at equivalent design loads. Additionally, a hypothetical ground acceleration of 0.15 horizontal and 0.10 vertical is used to analyze for the no-loss of function. This is discussed in Section 5.1.3.8, Seismic Design.

f) The American Standards Association "American Standard Code Requirements for Minimum Design Loads in Buildings and Other Structures" (A58.1-1955) designates the site as being in a 25 psf zone. In this code, for height zones between 100 and 499 feet, the recommended wind pressure on a flat surface is 40 psf. Correcting for the shape of the containment by using a shape factor of 0.60, the recommended pressure becomes 24 psf. The State Building and Construction Code for the State of New York stipulates a wind pressure up to 30 psf on a flat surface for heights up to 600 feet. For design, a 30 psf basic wind load has been used from ground level up.

g) Internal pressure will be applied to test the structural integrity of the vessel up to 115 per cent of the design pressure. For this structure, the test pressure is 54 psig.

5.1.2.3 Material Specifications

Basically four materials are used for the construction of the containment vessel. These are:

a) Concrete
b) Reinforcing Steel
c) Plate Steel Liner
d) Insulation

Basic specifications for these materials are as follows:

a) Concrete is a dense, durable mixture of sound coarse aggregate, fine aggregate, cement and water. Aggregates conform to American Society for testing and Materials Specification C-150-65 "Standard Specification for Portland Cement," Type I (Normal), or Type II (moderate heat of hydration requirements). Whenever high early strength is required, Type III Cement is used. Water is free from any injurious amounts of acid, alkali, salts, oil, sediment or organic matter. The concrete has a minimum density of 150 lb/ft^3. The 28-day standard compressive strength of the concrete is 3,000 psi. Adequate means of control are used in the manufacture of the concrete. To assure this value is attained as a minimum, concrete samples are tested in accordance with the following ASTM Standards:

ASTM C-172 - Method of Sampling Fresh Concrete

ASTM C-31 - Method of Making and Curing Concrete Compression and Flexure Test Specimen in Field

ASTM C-39 - Method of Test for Compressive Strength of Molded Concrete Cyliners

b) All making and testing of concrete samples have been performed by Vacca Testing Laboratory and Research Co., Inc.

Reinforcing steel for the dome, cylindrical walls and base mat is high-strength deformed billet steel bars conforming to ASTM Designation A432-65 "Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 psi Minimum Yield Strength." This steel has a minimum yield strength of 60,000 psi, a minimum tensile strength of 90,000 psi, and a minimum elongation of 7 per cent in an 8-inch specimen. Reinforcing bars No. 11 and smaller in diameter are lapped spliced in the mat for flexural loadings and spliced by the Cadweld process in the walls and dome for tension loading. Bars No. 14S and 18S are spliced by the Caldweld process only. A certification of physical properties and chemical content of each heat of reinforcing steel delivered to the job site has been issued from the steel supplier. The splices used to join reinforcing bars have been tested to assure that they will develop at least 125% of the minimum yield point stress of the bar. The test program required cutting out, at random, approximately 3%, completed splices and testing to determine their breaking strength.

c) The plate steel liner is carbon steel conforming to ASTM Designation A442-65 "Standard Specification for Carbon Steel Plates with Improved Transition Properties," Grade 60. This steel has a minimum yield strength of 32,000 psi and a minimum tensile strength of 60,000 psi with an elongation of 22 per cent in an 8-inch gauge length at failure.

The liner is 1/4-inch thick at the bottom, 1/2-inch thick in the first three courses except 3/4-inch thick at penetrations and 3/8-inch thick for remaining portion of the cylindrical walls and 1/2-inch thick in the dome. The liner material has been tested to assure an NDT temperature more than 30°F lower than the minimum operating temperature of the liner material.

Impact testing has been done in accordance with Section N331 of Section III of the ASME Boiler and Pressure Vessel Code. A 100 per cent visual inspection of liner anchors was made prior to pouring concrete.

d) The material for insulating the liner plate is polyvinylchloride with stainless steel jacket. This insulation has been selected to withstand the calculated temperature and pressure conditions associated with Figures 5.1-8, 5.1-9 and 5.1-10.

e) Quality of both materials and construction of the containment vessel has been assured by a continuous program of quality control and inspection by Consolidated Edison Company of New York, Incorporated, and/or its field representatives, and Westinghouse Atomic Power Division, and United Engineers & Constructors Inc., as described in Section 5.1.2.6.

5.1.2.4 Design Stress Criteria

The design is based upon limiting load factors which are used as the ratio by which loads will be multiplied for design purposes to assure that the loading deformation behavior of the structure is one of elastic, tolerable strain behavior. The load factor approach is being used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach permits the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. In the case of the containment structure, therefore, this approach places minimum emphasis on the fixed gravity loads and maximum emphasis on accident and earthquake or wind loads. The loads utilized to determine the required limiting capacity of any structural element on the containment structure are computed as follows:

a) $C = 1.0D \pm 0.05D + 1.5\ P + 1.0\ (T + TL)$

b) $C = 1.0D \pm 0.05D + 1.25\ P + 1.0\ (T' + TL') + 1.25E$

c) $C = 1.0D \pm 0.05D + 1.0P + 1.0\ (T'' + TL'') + 1.0E'$

Symbols used in these formulae are defined as follows:

- C: = Required load capacity of section.
- D: = Dead Load of structure and equipment loads.
- P: = Accident pressure load as shown on pressure-temperature transient curves.
- T: = Load due to maximum temperature gradient through the concrete shell and mat based upon temperatures associated with 1.5 times accident pressure.
- TL: = Load exerted by the liner based upon temperatures associated with 1.5 times accident pressure.
- T': = Load due to maximum temperature gradient through the concrete shell and mat based upon temperatures associated with 1.25 times accident pressure.
- TL': = Load exerted by the liner based upon temperatures associated with 1.25 times accident pressure.

E: = Load resulting from either design earthquake or wind, whichever is greater.

T": = Load due to maximum temperature gradient through the concrete shell, mat based upon temperatures associated with the accident pressure.

TL": = Load exerted by the liner based upon temperatures associated with the accident pressure.

E': = Load resulting from assumed hypothetical earthquake.

A chart for allowable versus actual stresses will be included in the containment report.

Load condition (a) indicates that the containment will have the capacity to withstand loadings at least 50 per cent greater than those calculated for the postulated loss-of-coolant accident alone. Results of analysis using load condition (a) are shown in Figure 5.1-11.

Load condition (b) indicates that the containment will have the capacity to withstand loadings at least 25 per cent greater than those calculated for the postulated loss-of-coolant accident with a coincident design earthquake. Results of analysis using load condition (b) are shown in Figure 5.1-12.

Mathematical solutions of load conditions (c) are shown in Figure 5.1-13. They indicate that the containment will satisfy this relation for seismic loads of at least equal to those corresponding to the response of 0.15 g horizontal and 0.10 vertical ground accelerations occurring simultaneously.

The mat has been analyzed utilizing load conditions (a), (b) and (c) and also for loads occurring only at operating and test pressure conditions as shown in Figure 5-14. For loads, see Table 5.1-1, Flooded Weights - Containment Bldg.

The loads resulting from wind on any portion of the structure do not exceed those resulting from earthquake.

All structural components have been designed to have a capacity by the most severe loading combination. The loads resulting from the use of these equations will hereafter be termed "factored loads."

The load factors utilized in these equations are based upon the load factor concept employed in Part IV-B, "Structural Analysis and Proportioning of Members Ultimate Strength Design" of ACI 318-63. Because of the refinement of the analysis and the restrictions on construction procedures, the load factors in the design primarily provide for a safety margin on the load assumptions.

The design includes the consideration of both primary and secondary stresses. The design limit for tension member (i.e., the capacity required for the design load) is based upon the yield stress of the reinforcing steel.

No steel reinforced crossection experiences average strains beyond the yield point at the factored load. The load capacity so determined is reduced by a capacity reduction factor "Ø" which provides for the possibility that small adverse variations in material strengths, workmanship, dimensions, and control, while individually within required tolerances and the limits of good practice, occaisonally may combine to result in under capacity. For tension members, the factor "Ø" has been established as 0.95. The factor "Ø" is 0.90 for flexure and 0.85 for diagonal tension, bond and anchorage.

For the liner steel the factor "Ø" is 0.95 for tension. For compression and shear, the liner stress is maintained below 0.95 yield and elastic stability has been assured.

The liner is designed to assure that no strains greater than the strain at the guaranteed yield point will occur at the factored loads. Sufficient anchorage is provided to assure elastic stability of the liner. The basic design concept utilizing stud anchorage of the liner plate to the concrete structure assures stud failure or tear of the liner plate. See references 3 and 4. The studs in the 1/2 inch plate are installed on 24" horizontal and

28" vertical grid and in the 3/8 inch plate on a 24" horizontal and 14" vertical grid. The design considers the possibility of daily stress reversals due to ambient temperature changes for the life of the plant, and fatigue limit of the studs exceeds the design requirements. However, to accomodate possible fatigue failure in the plate-to-stud weldment, the depth of penetration to the liner plate is controlled to avoid impairment of liner integrity.

5.1.2.5 Missile Protection

High pressure reactor coolant system equipment is surrounded by the 3'-0" concrete shield wall enclosing the reactor coolant loop and pressurizer and by the 2'-0" concrete operating floor.

A structure is provided over the control rod drive mechanism to block any missiles generated from fracture of the mechanisms.

Systems containing hot pressurized fluids and which might affect the engineered safeguards components have been carefully checked against the possibility of being sources of missiles. The general criterion adopted has been to take provision, when necessary, against the generation of missiles rather than allow missile formation and try to contain their effects.

Once the design requirement that the above systems are not to be sources of missiles has been set forth, identification of potential deficiencies and generation of adequate fixes took place through the quality assurance program.

The following examples illustrate how this approach has been implemented.

Valves

All the valves installed in the Nuclear Steam Supply System have stems with back seat. This rules out the probability of ejecting valve stems as even if it were assumed that the stem threads fail, analysis shows that the back seat or the upset end cannot penetrate the bonnet and thereby become a missile. Additional interference is encountered with air and motor operated valves.

Valves with nominal diameter larger than 2" have been designed against bonnet-body connection failure and subsequent bonnet ejection by means of (a) using the design practice of ASME Section VIII which limits the allowable stress of bolting material to less than 20% of its yield strength; (b) using the design practice of ASME Section VIII for flange design; and (c) by controlling the load during the bonnet body connection stud tightening process.

The pressure containing parts except the flange and studs are designed per criteria established by the USAS B16.5. Flanges and studs are designed in accordance with ASME Section VIII. Materials of construction for these parts are procured per ASTM A182, F316, or A351. GR CF8M.

Stud and nut material is ASTM A193-B7 and A194-2H. The proper stud torquing procedures and the use of torque wrench, with indication of the applied torque, limit the stress of the studs to the allowable limits established in the ASME Code, i.e., 20,000 psi. This stress level is far below the material yield, i.e., about 105,000 psi. The complete valves are hydrotested per USAS B16.5 (1500# USAS valves are hydro to 5400 psi). The cast stainless steel bodies and bonnets are radiographed and dye penetrant tested to verify soundness.

Valves with nominal diameter of 2" or smaller are forged and have screwed bonnet with canopy seal. The canopy seal is the pressure boundary while the bonnet threads are designed to withstand the hydrostatic end force. The pressure containing parts are designed per criteria established by the USAS B16.5 specification.

Reactor Coolant Pump Flywheel

The reactor coolant pump flywheel is not considered to be a credible source of missiles because of conservative design and care in manufacture and inspection. The flywheel material is ASTM A-533 having an NDTT less than 10°F. The design results in a primary stress less than 50% of the material yield strength at operating speed. The flywheel is subjected to 100% volumetric ultrasonic inspection which will be repeated at intervals during plant life. The finished machined bore is subjected to either magnetic particle or liquid penetrant examination. The design overspeed of the pump is 125%. The maximum pump overspeed on loss of external load is 112%.

5.1.2.6 Quality Control

To insure a high degree of confidence in plant design, construction, workmanship, materials and performance, a quality control program has been in effect for this project in which the following principal organizations have their respective responsibilities:

1. Consolidated Edison Company of New York, Inc. as owner and operator of the plant.

2. Westinghouse Electric Corporation as the turnkey plant contractor and supplier of major equipment.

3. United Engineers & Constructors, Inc. as architect engineer, construction managers and constructors.

The function and responsibility in the quality control program of each of the above organizations is as follows:

Consolidated Edison Company of New York, Inc.

A qualified field representative is assigned to the field during the construction period. His responsibilities include continuous inspection of the construction of the containment building to insure that all materials used and work performed is strictly in accordance with the plans and specifications. The Consolidated Edison representative through instructions received from the home office has the power to stop the construction until any discrepancies are corrected and the work once more is in compliance with the specifications and plans.

The Consolidated Edison representative is in constant communication and consultation with the construction superintendent in matters regarding quality control. In addition, personnel from U.S. Testing Laboratories are assigned to this project to monitor the inspection of the construction and obtain samples of the materials for testing.

Westinghouse Electric Corporation

For the assurance of plant integrity and quality, Westinghouse performs the following functions regarding the containment building:

a) Review and approve the containment design criteria, material specifications and detail design concepts before they are released for construction. This work has been done by qualified structural engineers of the company's home office.

b) Review the construction and inspection methods employed by United Engineers & Constructors Inc.

Westinghouse, Pressurized Water Reactor Division, Nuclear Power Services Group has a field Quality Assurnace representative in residence during the construction period. His function is the same as the Consolidated Edison representative mentioned above. He reports discrepancies to the Westinghouse Construction and Services resident engineer '- has the authority to stop the work until the discrepancy is resolved.

In addition to this, he audits the construction files, verifies that records are complete, accurate and adequate for Quality Assurance.

Nuclear Power Service Headquarters Quality Assurance Engineers also make trips to the site to audit, monitor and review the project with regards to site Quality Assurance Construction practices are observed for conformance to codes, specifications and approved procedures.

United Engineers & Constructors Inc.

The responsibilities of United Engineers & Constructors Inc. in the quality control of the containment building have been as follows:

a) They inspect all materials delivered to the job site, and examine the suppliers' certified test reports of physical and chemical properties for those components furnished by them.

b) They inspect, in the shop, fabrication of major components of the containment structure. Trip reports are available at the job site.

c) They maintain an adequate force of qualified supervisory personnel at all times.

d) They supervise and are fully responsible for the quality of work performed by their subcontractors and for the craft labor employed and supervised by them.

e) They maintain as part of their field engineering force, qualified personnel who perform a thorough inspection of each construction operation.

[illegible] ges in design or specifications are allowed without the approval of the engineer in charge of design.

5.1.3 STRESS ANALYSIS

5.1.3.1 General

The structural design of the containment meets the requirements established by 1961 edition of "The State Building and Construction Code for the State of New York" so far as these provisions are applicable. All concrete structures have been designed, detailed and constructed in accordance with the provisions of "Building Code Requirements for Reinforced Concrete" (ACI 318-63) so far as these provisions are applicable.

5.1.3.2 Method of Analysis

Basically three separate structural components have been analyzed, each in equilibrium with loads applied to it and with constraints occurring at the junct[illegible]uctures. The three components are:

a) The 135-[illegible] hemispherical dome.
b) The 135-ft ID cylinder
c) The base slab.

Mathematically, the dome and cylinder have been treated as thin-walled shell structures, which results in a membrane analysis. Since the thickness of the dome and cylinder is small in comparison with the radius of curvature (1/15) and there are no discontinuities such as sharp bends in the meridonal curves, the stresses due to pressure and wind or earthquake is calculated by assuming that they are uniformly distributed across the thickness.

Since the concrete is not assumed to resist any tensile or shear forces, radial shear reinforcing has been introduced in the lower portion of the wall in the form of hooked diagonal stirrups and diagonally bent bars as shown in Figure 5.1-1. Likewise, diagonal shear reinforcing in the circumfrerential direction has been included to resist earthquake shears for the full height of the wall and a distance above the spring line into the dome until a point is reached where the dome liner can resist the total shear.

The base slab has been treated as a flat circular plate supported on a rigid non-yielding foundation.

Analyses of the liner is presented in Appendix C "Containment Liner Stress Analysis Report."

5.1.3.3 Dome Analysis

The analysis of the hemispherical dome has been performed by the superposition of membrane forces resulting from gravity, accident pressure and accident thermal loads. In addition, earthquake or wind loading create both direct and shear stresses in the dome and the operating

a) For membrane stress analysis, the dome and cylinder are treated as thin-walled shell structures. (The thickness to radius ratio for the dome is 1/20 and the cylinder 1/15. These ratios are smaller than the 1/10 criterion for thin-walled shell analysis.[5] Membrane forces are resisted by steel reinforcing and the building liner.

b. Discontinuity stresses occur at the juncture of the cylinder and the mat and the juncture of the cylinder and dome. Discontinuity effects are determined as follows:

1. The radial growth of the shell is computed based on membrane stress in the reinforcing and liner.

2. The flexural rigidity of the meridional wall section is determined based on a cracked section analysis in accordance with conventional reinforced concrete design techniques.

3. Moments and shears are calculated based on having consistent deformation for the two elements at the point of discontinuity.

Discontinuity effects at the spring line are very slight due to the small difference in radial growth between the dome and cylinder. Since the circumferential reinforcing in the dome and cylinder vary, stresses and, therefore, deformations are essentially equal.

temperature of the liner creates tension and compression. All of the combined direct stresses are developed in the reinforcing steel encased in the concrete. The liner of the dome above a certain point is used to resist shear load and the anchorages have been designed to assure composite action. The dome reinforcing is spliced to the vertical steel in the cylindrical concrete wall, so that a continuity between the dome and the cylinder is realized.

5.1.3.4 Cylinder Analysis

The analysis of the cylinder is by superposition of membrane forces resulting from gravity, pressure and thermal loads, over-turning due to earthquake or wind and shears due to earthquake or wind. The concrete has been reinforced circumferentially using steel hoops and vertically by straight bars. Diagonal bars have been placed to resist the horizontal and vertical shears due to earthquake or wind. The required capacity of the diagonal bars has been designed so that the horizontal component per foot of the diagonals is equal to the maximum value of shear flow. A check was made to insure that no net compressive force results in the diagonal bars because of the combination of seismic shear load and internal pressure load. Although, in the cylinder, the liner has some capacity available to resist the seismic shears, no credit is taken for this capacity.

Only in the upper area of the dome (beyond about 30° above the spring line) where the seismic shears are small is the liner counted on to resist shear. For all of the cylinder and the lower areas of the dome, the diagonal reinforcing has been designed to accommodate all seismic shears. No credit has been taken for the dowel action of the vertical and horizontal bars in resisting seismic shear.

5.1.3.5 Pressure Stresses

Pressure effects on the containment structure may be divided into two types:

a) Membrane stresses

b) Discontinuity stresses

The mat is considered as offering complete fixity, no credit is taken for the liner at the base in resisting moments since at the point of maximum shear the bond between the liner and concrete is insufficient to transmit complementary beam shear. A slip surface between the concrete and liner is formed and the liner is subjected to membrane forces only.

The 9 ft thick mat is subjected to the following due to pressure inside the containment building:

1. Uplift at the junctura with the wall.
2. Moment and shear due to discontinuity effects with the wall.
3. Downward pressure loading due to internal pressure.

The 9 ft mat is designed to accommodate these loads until there is no further flexural consideration. At the crane wall, the mat is founded on the unyielding rock and further pressure loads are transmitted through bearing directly into the rock.

Resistance to these loads is based on a cracked concrete section. No credit is taken for the liner for the same reasons given for the wall.

Discontinuity shears in both the cylinder and mat are resisted by either bent bars or stirrups.

5.1.3.6 Thermal Stresses

Temperature effects on the containment structure may be divided into two separate considerations; one effect is due to a thermal gradient through the wall, the other is caused by the rapid temperature rise of the liner under the accident conditions. The reinforced concrete wall restrains the liner from growing, resulting in compression in the liner and additional tension in the reinforcing.

a) Calculation of gradient stresses is based on method of analysis outlined in ACI Code 505, "Reinforced Concrete Chimney Design."[6] The gradient used is linear with 120°F on the inside and 0°F exterior concrete temperature (-5°F ambient).

The ACI method assumes a cracked section in which the concrete carries no tension. The neutral surface (surface at which no thermal stress exists) is determined. Stresses in the liner and reinforcing are calculated based on the assumption that there is no distortion of the wall; i.e., strains through the wall are constant.

b) To determine the effects due to rapid rise in liner temperature, there are two basic assumptions made. The first is that the effects are internal in nature; i.e., the compressive force in the liner is balanced by a tensil force in the reinforcing. The second is that there is no distortion of the wall.

Because temperature effects are internal in nature and do not effect the overall load carrying capability of the structure, local yielding of reinforcing under accident conditions is acceptable.

The temperature gradient through the wall is essentially linear on both the insulated and uninsulated portions and is a function of the operating temperature internally and the average ambient temperature externally. Accident temperatures mainly affect the liner, rather than the concrete and reinforcing bars, due to the insulating properties of the concrete. By the time the temperature of the concrete adjacent to the liner begins to rise significantly, the internal pressure and temperature in the containment shell due to maximum thermal gradient will not influence the capacity of the structure to resist the other forces. Temperature effects induce stresses in the structure which are internal in nature; tension outside the compression in the inside of the shell such that the resultant force is zero. Loading combinations concurrent with these temperature effects may cause local stresses in the outside horizontal and vertical bars to reach yield; however, as local yielding is reached, any further load is transferred to the unyielded elements. At the full yield conditio the magnitude of final load resisted across a horizontal and vertical section remains identical to that which would be carried if the temperatur affects were not considered. Thus, the overall carrying capacity of the structure and the factor of safety of the structural elements are not affected.

5.1.3.7 Large Openings

In the cylindrical section of the containment, where there are large openings for access hatchways and penetrations, the reinforcing bars (hoop, vertical and diagonal) are continued without interruption around the openings.

No bar terminates at any opening as illustrated around the penetration in Figure 5.1-1. Also additional bars have been furnished locally to take the stresses developed around these openings. Concrete is locally thickened at the equipment access hatchway area to accommodate all the reinforcing bars required in this area.

The liner plate is locally thickened at the penetrations to take care of additional stresses.

A finite element analysis is performed of the large openings. Representation of the structure is by rectangular elements; each element consists of ten layers of orthotropic, elastic material to represent the reinforcement, concrete and the liner. About 1000 degrees of freedom are considered in the model. This analysis is used as a check on the adequacy of the large openings. Results appear in the "Indian Point No. 2 Containment Design Report."[7]

5.1.3.8 Seismic Design

The design of the containment which is a Class I structure (see Appendix A) is based on a "response spectrum" approach in the analysis of the dynamic loads imparted by earthquake. The seismic design takes into account the acceleration response spectrum curves developed by G. Housner. Seismic accelerations have been computed as outlined in the AEC TID-7024[1] and Portland Cement Publication.[2]

The following damping factors have been used:

DAMPING FACTORS

	Component	Per Cent Critical Damping
1.	Containment Structure	2.0
2.	Concrete Support Structure of Reactor Vessel	2.0
3.	Steel Assemblies:	
	(a) Bolted or Riveted	2.5
	(b) Welded	1.0
4.	Vital Piping Systems	0.5
5.	Concrete Structures above Ground:	
	(a) Shear Wall	5.0
	(b) Rigid Frame	5.0

As indicated in Sections 5.1.2.2, ground accelerations used for design purposes are 0.1g applied horizontally and 0.05g applied vertically. The natural period of vibration is computed by the Rayleigh method; in this method, the containment structure is analyzed as a simple cantilever intimately associated with the rock base and with broad base sections of adequate strength to assure full and continued elastic response during seismic motions. Further, both bending and shear deformations are considered.

The structure is divided into sections of equal length and loaded laterally by dead weight of the section and any equipment and live load occurring at the section. Deflections caused by shear and moments are then determined, and the end deflection is given the value $\phi' = 1.0$ with corresponding values determined for other sections. The natural period of vibration for the structure is then determined by setting potential energy equal to kinetic energy and solving for the period.

Y_o = Maximum Actual Deflection

$\phi' = \dfrac{\text{Deflection of Section Under Consideration}}{\text{Maximum Actual Deflection}}$

g = Acceleration Due to Gravity

dm = Weight of Section Under Consideration

Based on an uncracked concrete section, the period is determined to be 0.241 Sec. A more realistic ca[illegible]lation for a cracked section, using reinforcing steel and liner as the [illegible]sisting elements, yields a period 0.0936 Se[illegible]

Using the derived period and entering the acceleration spectral curves, Figures A-1 and A-2 of Appendix A, and applying a 2% critical damping, a spectral acceleration for the containment was selected. This value was derived to determine the base shear. The distribution of base shear is a triangular loading assumption.

This assumption yields a load distribution pattern with zero loading at the base to a maximum loading at the spring line of the dome. Above this line, the loading decreases due to a change in section and consequently change in weight. This load distribution allows the determination of shears and moments at any critical section through the containment from which the appropriate unit stresses are obtained.

Seismic shears are resisted by diagonal reinforcing except in the upper areas of the dome. No credit is taken for the reinforcing in compression.

From 30° above the springline, the shear is resisted by the liner. The shear is transmitted to the liner by means of tees welded to the liner.

5.1.4 PENETRATIONS

5.1.4.1 General

In general, a penetration consists of a sleeve embedded in the concrete wall and welded to the containment liner. The weld to the liner is shrouded by a continuously pressurized channel which is used to demonstrate the integrity of the penetration-to-liner weld joint. The pipe, electrical conductor cartridge, duct or equipment access hatch passes through the embedded sleeve and the ends of the resulting annulus are closed off, either by welded end plates, bolted flanges or a combination of these.

Differential expansion between a sleeve and one or more hot pipes passing through it is accomodated by using a bellows type expansion joint between the outer end of the sleeve and the outer end plate, as shown on Figure 5-10.

Pressurizing connections are provided to continuously demonstrate the integrity of the penetration assemblies.

5.1.4.2 Types

a) Electrical Penetrations

"Cartridge" type penetrations are used for all electrical conductors passing through the containment. The penetrations are provided with a pressure connection to allow continuous pressurization. Insulating bushings or fused glass seals are used to provide a pressure barrier for the conductor.

Figure 5.1-17 shows a design of typical electrical penetrations. There are approximately 60 electrical penetrations.

b) Piping Penetrations

Double barrier piping penetrations are provided for all piping passing through the containment. The pipe is centered in the embedded sleeve which is welded to the liner. End plates are welded to the pipe at both ends of the sleeve. Several pipes may pass through the same embedded sleeve to minimize the number of penetrations required. In this case, each pipe is welded to both end plates. A connection to the penetration sleeve is provided to allow continuous pressurization of the compartment formed between the piping and the embedded sleeve. In the case of piping carrying hot fluid, the pipe is insulated and cooling is provided to maintain the concrete temperature adjoining the embedded sleeve at or below 150°F.

Cooling is provided for hot penetrations through the use of air-to-air heat exchangers. These are made in accordance with the ASME UPV Code, Section VIII, by welding together one flat sheet and one embossed sheet of 10 gage carbon steel material, the embossment forming coolant passages. The unit is rolled into the form of a cylinder with an outside diameter slightly smaller than the respective inside diameter of the penetration sleeve. The exchanger is placed inside the sleeve and outside the pipe insulation, with the inlet and outlet coolant connections penetrating the sleeve between the outside concrete wall surface and the bellows expansion joint. The coolant to be used is ambient air fed by a centrifugal blower which is backed up with a full sized spare. The isolation features and criteria for piping penetrations are given in Chapter 6. Figure 5.1-18 shows typical and hot and cold pipe penetrations.

A total of approximately 80 pipes pass through approximately 50 penetration sleeves, 23 of which are considered thermally hot. In addition, several spare sleeves (capped and pressurized) are provided for the possible future addition of piping.

c) Equipment and Personnel Access Hatches

An equipment hatch has been provided. It is fabricated from welded steel and furnished with a double-gasketed flange and bolted dished door. The hatch barrel is embedded in the containment wall and welded to the liner. Provision is made to continuously pressurize the space between the double gaskets of the door flanges and the weld seam channels at the liner joint, hatch flanges and dished door. Pressure is relieved from the double gasket spaces prior to opening the joints. The personnel hatch is a double door, mechanically-latched, welded steel assembly. A quick-acting type, equalizing valve connects the personnel hatch with the interior of the containment vessel for the purposes of equalizing pressure in the two systems when entering or leaving the containment. The personnel hatch doors are interlocked to prevent both being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened.

Remote indicating lights and annunciators situated in the control room indicate the door operational status. An emergency lighting and communication system operating from an external emergency supply is provided in the lock interior. Emergency access to either the inner door, from the containment interior; or to the outer door, from outside, is possible by the use of special door unlatching tools. The design is in accordance with Section VIII of the ASME Code. For calculations, see the Containment Design Report.[7]

d) Special Penetrations

1. Fuel Transfer Penetration

A fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the reactor containment and the spent fuel pit. The penetration consists of a 20-inch stainless steel pipe installed inside a 24-inch pipe. The inner pipe acts as the transfer tube and is fitted with a pressurized double-gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pit. This arrangement prevents leakage through the transfer tube in the event of an accident. The outer pipe is welded to the containment liner and provision is made by use of a special seal ring for pressurizing all welds essential to the integrity of the penetration during plant operation. Bellows expansion joints are provided on the pipes to compensate for any differential movement between the two pipes or other structures. Figure 5.1-19 shows a sketch of the fuel transfer tube.

2. Containment Supply and Exhaust Purge Ducts

The ventilation system purge ducts are each equipped with two quick-acting tight-sealing valves (one inside and one outside of the containment) to be used for isolation purposes. The valves are normally opened for containment purging, but are automatically closed upon a signal of high containment pressure or high containment radiation level. The space between the valves is pressurized above design pressure, while the valves are normally closed during plant operation. See Section 5.3, Containment Ventilation System, and Section 6.4, Containment Air Recirculation Cooling and Filtration System.

Two solenoid controlled, pneumatically operated butterfly valves are provided for each purge penetration, one on each side of the containment building wall. Two penetrations, one supply and one exhaust, are required. Valves are spring-loaded to fail closed.

The space between the valves is pressurized from the pressurization system through an electrically operated three-way solenoid valve. This pressure is maintained only when valves are closed and must be relieved before butterfly valves can be opened. Failure to release this pressure will prevent valves from opening.

Failure of any of the valves to open will prevent the fans from running. Tripping of either of the purge fans will automatically close the butterfly valves, and pressurize the space between the valves. Failure of any of the valves to close will prevent the adjacent space from being pressurized, and sound the loss of pressurization alarm. Loss of pressure for either zone will be displayed by individual indicating lights at the Main Control Board.

The valve control solenoids and pressurization solenoids are controlled from a single control switch on the fan room control panel. The cycle is initiated by setting the control switch to "open" position. This will energize the pressurization alarm.

When the pressure between the valves has been relieved, the valve control solenoids are energized and the valves opened. If for any reason, any of the four valves fail to open within a given time (say 35 seconds) after the cycle is initiated, all four valves will close and pressure will be restored. The circuit is interlocked to prevent inadvertent opening of the valves during S.I. condition.

Once all four valves have been opened, the operator has a predetermined time (say 1 minute) to start the purge supply fan. Failure to do so will cause all four valves to close.

Position indicating lights for each of the four valves are provided on the Fan Room Control Panel and the Main Control Board.

3. Sump Penetrations

The piping penetration in the containment sump area is not of the typical sleeve to liner design. In this case, the pipe is welded directly to the base liner. The weld to the liner is shrouded by a test channel which is used to demonstrate the integrity of the liner.

5.1.4.3 Design of Penetrations

a) Criteria

The liner is basically not a load-carrying member because it is subjected to strains imposed by the reinforced concrete; nevertheless, the liner has been reinforced at each penetration in accordance with the ASME Code Section VIII. The weldments of liner to penetration sleeve are of sufficient strength to accommodate stress concentrations and adhere strictly to ASME Code Section VIII requirements for both type and strength. The penetration sleeves and plates are designed to accommodate all loads imposed on them under operating conditions (thermal effects and internal penetrations and test pressures) and accident conditions (loads resulting from all strains, internal pressures, and seismic movements).

b) Materials

The materials for penetrations including the personnel and equipment access hatches, together with the mechanical and electrical penetrations will be carbon steel, conform with the requirements of the ASME Nuclear Vessels Code and exhibit ductility and welding characteristics compatible with the main liner material. As required by the Nuclear Vessels Code, the penetration materials meet the necessary Charpy V-notch impact values at a temperature 30°F below lowest service metal temperature which is 50°F within containment and -5°F outside the containment.

The stainless steel bellows of the hot penetration expansion joints will be protected from damage in transit and during construction by sheet metal covers fastened in place at the fabricator's shop. These can be left in place permanently if there is no interference with nearby piping or equipment.

1. <u>Piping Penetrations: Materials</u>

Piping Penetration Material	Specification
Penetration Sleeve - 12" ∅ and under	ASTM - A333, Gr. 1
-Over 12" ∅	ASTM - A201, Gr. B to A300
Rolled Shapes	ASTM - A36, A131, Gr. C

2. <u>Electrical Penetrations: Materials</u>

The penetration sleeves to accommodate the electrical penetration assembly cartridges are Schedule 80 carbon steel in accordance with

ASTM-A333, Gr. 1, except where otherwise noted. The electrical cartridges have been secured to the penetration sleeve so that all possible leak paths between the cartridge and sleeve will be blocked by a pressurized zone.

3. Access Penetrations: Materials

The equipment and personnel access hatch material are as follows:

Item	Material Specification
Equipment Hatch Insert:	ASTM A300, Cl. 1, Firebox A-516, Gr. 60
Equipment Hatch Flanges:	ASTM A300, Cl. 1, Firebox A-516, Gr. 60
Equipment Hatch Head:	ASTM A300, Cl. 1, Firebox A-516, Gr. 60
Personnel Hatch:	ASTM A300, Cl. 1, Firebox A-516, Gr. 60

5.1.4.4 Leak Testing of Penetration Assemblies

A proof test is applied to each penetration by pressurizing the necessary areas to 54 psig. This pressure is maintained for a sufficient time to allow soap buddle and Freon sniff tests of all welds and mating surfaces. Any leaks found are to be repaired and retested; this procedure is repeated until no leak exists.

5.1.4.5 Construction

The qualification of welding procedures and welders has been in accordance with Section IX, "Welding Qualifications" of the ASME Boiler and Pressure Vessel Code. The repair of defective welds has been in accordance with Para. UW-38 of Section VIII "Unfired Pressure Vessels."

5.1.4.6 Testability of Penetrations and Weld Seams

All penetrations, the personnel air lock and the equipment hatches are designed with double seals which will be normally pressurized at 50 psig. Individual testing at 115% of containment design pressure is also possible.

The containment ventilation purge ducts are equipped with double isolation valves and the space between the valves is permanently piped into the penetration pressurization system. The space can be pressurized to 115% of
14 design pressure when the isolation valves are closed. The purge valves fail in the closed position upon loss of power (electric or air).

All welded joints in the liner have steel channels welded over them on the inside of the vessel. During construction, the channel welds were tested by means of pressurizing sections with Freon gas and checking for leaks by means of a Freon sniffer. These welds were also then continuously pressurized at 50 psig.

5.1.4.7 Accessibility Criteria

The containment will be completely closed whenever the core is critical or whenever the primary system temperature is above 200°F and the pressure above 300 psig with nuclear fuel in place except as required for brief periods necessary to relieve the containment to keep the pressure below a reasonable level (1-2 psig).

Limited access to the containment through personnel air locks will be possible with the reactor at power or with the primary system at design pressure and temperature at hot shutdown. This type of access would be restricted to the areas external to the reactor equipment compartment, primarily for inspection and maintenance of the air recirculation equipment and the incore ion chamber drives.

After shutdown, the containment vessel will be purged to r uce the concentration of radioactive gases and airborne particulates. This purge system has been designed to reduce the radioactivity level to doses defined by 10 CFR 20 for a 40-hour occupational work week, within 2-6 hours after plant shutdown. Since negligible fuel defects are expected for this reactor, much less than the 1% fuel rod defects used for design, purging of the containment will normally be accomplished in less than 2 hours. To assure removal of particulate matter, the purge air will be passed through a high efficiency filter before being released to the atmosphere through the purge vent.

The primary reactor shield has been designed so that access to the primary equipment is limited by the activity of the primary system equipment and not the reactor.

5.1.5 PRIMARY SYSTEM SUPPORTS

The primary system supports, steam generator, reactor coolant pump, pressurizer and reactor vessel are designed to withstand pipe break or seismic acceleration based on the following:

1. The break is either a circumferential or longitudinal pipe rupture of area equivalent to the pipe cross section occuring anywhere in the system piping. The longitudinal rupture occurs at any point 360° around the pipe. The support system is designed to withstand the steady thrust equivalent to the product of system operating pressure and pipe rupture area without exceeding yield stress in the support members. The stress limits on the vessels and piping are tabulated in Appendix A. The component supports prevent rupture of reactor coolant piping in the remaining intact loops as a result of an assumed rupture in any one loop, thereby assuring the path for safety injection flow to the core is available. Additionally, the supports are designed to prevent secondary piping rupture as a result of rupture in the primary loop and vice versa.

2. The nuclear steam supply system and its support system are designed such that the nuclear steam supply system is capable of continued safe operation for the combination of normal loads and the design earthquake loading. The equipment and supports operate within normal design limits for the design earthquake. The system and its supports are also designed to withstand the maximum potential earthquake without loss of function. The seismic response curves for both the design and maximum potential earthquake and the stress limits are presented in Appendix A. Component loads are obtained from the curves using the appropriate period and damping.

Steam Generators

The steam generators are supported within a caged structural system, consisting of four connected trusses, all welded together, fabricated of carbon steel members, with provisions for limited movement of the structure in a horizontal direction to accommodate piping expansion with a system of "Lubrite" plates, hydraulic snubbers, guides and stops, The "Lubrite" plates, hydraulic snubbers, guides and stops are designed as a rigid support to resist the action of seismic and pipe break loads.

The following are loading conditions that the sturcture has been designed to resist:

a) Vertical dead weight of pipe and vessel flooded = 1,000 kips

b) Seismic Loads:

 1) Horizontal load of 474 kips acting at the centroid of the steam generator vessel, located near top of support structure, which is directly transferred to the hydraulic snubbers, guides and stops at elevation 95'-0" and in turn to the concrete slab.

 2) Vertical load of 320 kips transferred as axial load to the base plates and anchor bolts at elevation 46'-0".

c) Primary System - Longitudinal Pipe Rupture:

 1) Reaction at the nozzle of the steam generator from the pipe between the reactor and the steam generator elbow, produces a force of 1090 kips in any direction and an overturning moment or torsional moment of (1090 kips x 4.25 ft) 4632 ft-kips. Overturning and torsional moments are resisted by the support system at elevation 46'-0" and horizontal forces are distributed, through the truss action, to elevations 46'-0" and 93'-0".

2) Reactions at the nozzle of the steam generator from the pipe between the steam generator elbow and reactor coolant pump elbow, produces a force of 785 kips in any direction and a torsional moment or overturning moment of (785 kips x 5.0 ft) 3925 ft-kips. Overturning and torsional moments are resisted by the support system at elevation 46'-0", and horizontal forces are distributed, through the truss action, to elevations 46"-0" and 93'-0".

d) Primary System - Circumferential Break:

1) Reactions at the nozzle of the steam generator from the pipe between the reactor and steam generator produces a horizontal force of 1,490 kips. This force is transferred through the vessel support to the two vertical trusses of the structural system which, in turn, transmits it as horizontal reactions at the slabs at elevations 46'-0" and 93'-0". The moment produced by this force is (1,490 kips x 2 ft) 2,980 ft-kips and is less than the dead load resisting moment (500 kips x 10 ft) 5,000 ft-kips, and the vertical forces at elevation 46"-0" are all compressive, no uplift.

2) Reactions at the nozzle of the steam generator from a pipe between the steam generator and the reactor coolant pump produces a horizontal force of 1700 kips plus an overturning moment of (1700 kips x 4.25 ft) 7225 ft-kips, or a vertical force of 1700 kips and an overturning moment of (1700 kips x 5.33 ft) 9061 ft-kips. The horizontal force and moments are transferred to the structural system and the reactions are resisted at the slabs at elevations 46'-0" and 93'-0", or the vertical force and moment are resisted at elevation 46'-0". to my knowledge.

e) Secondary System - Longitudinal Rupture in Steam Pipe:

1) Reactions at the nozzle of the steam generator from the steam pipe longitudinal rupture at the top of thn vessel produces:

a) Horizontal force of 600 kips and a torsional moment of 2400 ft-kips. Horizontal force is transferred through the vessel to the structural support system, which in turn transmits it as horizontal reactions to the slabs at elevations 46'-0" and 95'-0". The torsional moment is transferred through the vassel to the structural system which, in turn, transmits it to the base at elevation 46'-0".

b) Vertical upward or downward of 600 ft-kips and an overturning moment of 2400 ft-kips. Upwurd forces are overcome by the operating weight of the steam generator. Downward force is added to the operating weight and transferred to the base at elevation 46'-0". Overturning moment is transferred through the vessel supports to the structural system which, in turn, transmits it as vertical reactions at the base, elevation 46'-0".

f) Secondary System - Circumferential Break:

Reaction at the nozzle of the steam generator from the steam pipe guillotine break at the top of the vessel produces a horizontal force of 600 kips. This force is transferred through the vessel to the structural system which, in turn, transmits it as horizontal reaction of 1085 kips at elevation 93'-0" and 485 kips at elevation 46'-0".

g) Secondary System – Feedwater Pipe Breaks:

The reactions from circumferential and longitudinal pipe breaks in the feedwater system are resisted in a manner similar to steam pipe breaks listed under preceeding sections (e) and (f), but are much smaller in magnitude. Maximum longitudinal 1600 ft-kips, maximum circumferential 400 kips.

Reactor Coolant Pump

The reactor coolant pump is supported on a three-legged structrual system consisting of three connected trusses fabricated of carbon steel members, structural sections and pipe, supported from elevation 48'-6". Provisions for limited movement of the structure in any horizontal direction to accommodate piping expansion is accomplished with a sliding "Lubrite" base plate arrangement and a system of tie rods and anchor bolts which restrain the structure from movement beyond the calculated limits.

The following are loading conditions that the structure has been designed to resist:

a) Vertical Dead Weight of pipe and pump flooded = 206 kips.

b) Seismic:

1. Horizontal load of 100 kips acting at the centroid of the pump assembly which is transferred by the structural system and piping to the tie rods and base of supporting structure at elevation 48'-6".

2. Vertical seismic load of 66 kips transferred directly as axial load to the base plates and anchor bolts.

c) Primary System - Longitudinal Rupture:

1. Reaction at the nozzle of the pump from a pipe break in the pipe between the steam generator elbow and pump elbow produces a torsional moment of 3815 ft-kips, together with a horizontal force of 850 kips or an overturning moment of 3815 ft-kips, together with a vertical up or down force of 850 kips. Torsional forces are resisted by the structural stability of the primary piping connected to the pump.

   Reactions from horizontal forces are resisted by the tie rods connected to the steam generator support structure. Forces caused by an overturning moment are resolved into horizontal and vertical components which are resisted by tension in the anchor bolts, axial load on the foundations and tension in the tie rods.

2. Reaction at the nozzle of the pump from a pipe break in the pipe between the pump and the reactor, produces a torsional moment of 6880 ft-kips, together with a horizontal force of 1165 kips, or an overturning moment of 6880 ft-kips, together with a vertical up or down force of 1165 kips.

   Torsional forces are resisted by the structural stability of the primary piping connected to the pump. Reactions from the horizontal forces are resisted by the tie rods connected to the walls.

Forces caused by an overturning moment are resolved into horizontal and vertical components which are resisted by:

Tension in the anchor bolts,
Axial load on the foundations, and
Tension in the tie rods.

d) Primary System - Circumferential Break:

1. Reactions at the nozzle of the pump from a pipe break in the pipe between the steam generator and pump, produces a horizontal force on the stucture of 1700 kips. This force is resisted directly by the bumper located against the elbow of the pipe. Components of the force are then transferred to the base of the structure and the tie rods connecting the pump support to the steam generator support system.

2. Reactions at the nozzle of the pump from a pipe break in the pipe between the pump and the reactor produces a torsional moment of 3240 ft-kips and a horizontal force of 1340 kips on the structure.

   Torsional forces are resisted by the structural stability of the remaining primary piping connected to the pump.

   Reactions from the horizontal forces are resisted by tie rods connected to the walls.

<u>Pressurizer</u>

Pressurizer is supported on a free-standing structural system, consisting of six connected trusses fabricated of carbon steel members, all welded together and secured at the base by anchor bolts at elevation 46'-0".

The following are loading conditions that the structure has been designed to resist:

a) Vertical dead weight of pipe and vessel flooded is 360 kips.

b) Seismic:

1. Horizontal seismic load of 173 kips acting at the centroid of the pressurizer vessel which coincides in elevation with the slab at elevation 95'-0" is directly transferred through the concrete embedded guides to the slab.

2. Vertical seismic load of 115 kips transferred through the structural system as axial forces to the base plates and anchor bolts at elevation 46'-0".

c) Longitudinal Pipe Rupture:

1. Reaction at the surge pipe nozzle of the pressurizer produces either a torsional moment of 734 ft-kips and a horizontal force of 234 kips or an overturning moment of 734 ft-kips and a horizontal or vertical force of 234 kips.

These moments and forces are resisted by the structural system and transferred to the base at elevation 46'-0".

d) Circumferential Pipe Break:

1. Reaction at the surge pipe nozzle of the pressurizer produces a horizontal force of 234 kips and an overturning moment of 734 ft-kips.

These moments and forces are resisted by the structural system and transferred to the base at elevation 46'-0".

Reactor Vessel Support Girder

The reactor vessel is supported on four cooling plates which are fastened to the top flange of a circular box section ring girder, fabricated of carbon steel plates. The bottom flange of the girder is in continuous contact with a non-yielding concrete foundation.

In addition to the reactor vessel weight and piping reactions of 3736 kips, the girder has been designed to support the condtiions of loading for pipe break and seismic forces as outlines or Figure 5.1-20.

5.1.6 SYSTEM DESIGN EVALUATION

RELIANCE ON INTERCONNECTED SYSTEMS

The containment leakage limiting boundary is provided in the form of a single, carbon steel liner on the vessel having double barrier weld channels and penetrations. Each system whose piping penetrates this boundary is designed to maintain isolation of the containment from the outside environment. Provision is made to continuously pressurize penetrations and weld channels and to monitor leakage from this pressurization.

SYSTEM INTEGRITY AND SAFETY FACTORS

Pipe Rupture - Penetration Integrity

The penetrations for the main steam, feedwater, blowdown and sample lines are designed so that the penetration is stronger than the piping system and that the vapor barrier will not be breeched due to a hypothesized pipe rupture.

Major Component Support Structures

The support structures for the major components are designed to resist all thrust forces, moments and torques associated with either a reactor coolant system or main steam pipe break. All primary structural steel elements are designed for stresses not exceeding yield stress due to these forces.

Containment Structure Components Analyses

The details of radial, longitudinal and horizontal shear analyses for the containment reinforced concrete are given in Section 5.1.3.

PERFORMANCE CAPABILITY MARGIN

The containment structure is designed based upon limiting load factors which are used as the ratio by which accident and earthquake loads are multiplied for design purposes to ensure that the load/deformation behavior of the structure is one of elastic, low strain behavior. This approach places minimum emphasis on fixed gravity loads and maximum emphasis on accident and earthquake loads. Because of the refinement of the analysis and the restrictions on construction procedures, the load factors primarily provide for a safety margin on the load assumptions. Load combinations and load factors utilized in the design which provide an estimate of the margin with respect to all loads are tabulated in Section 5.1.2.

5.1.7 MINIMUM OPERATING CONDITIONS

CONTAINMENT INTEGRITY

1. Containment integrity shall be maintained whenever the reactor coolant system is above 300 psig and 200°F. The shutdown margin shall be greater than 3% Δk with all rods inserted when the containment is open.

   The reactor coolant system conditions of 300 psig and 200°F assure that no steam will be formed and hence there would be no pressure buildup in the containment if a reactor coolant system rupture were to occur.

2. Containment integrity shall not be violated when the reactor vessel head is removed unless a shutdown margin greater than 10% Δk is constantly maintained.

   The shutdown margins are selected based on the type of activities that are being carried out. The shutdown margin during refueling precludes criticality under any circumstances, even though fuel is being moved. When the reactor head is not to be removed, the specified shutdown margin of 3% Δk is constantly maintained.

3. Positive reactivity changes shall not be made by rod drive motion or boron dilution whenever the containment integrity is not intact.

INTERNAL PRESSURE

The reactor shall not be critical if the containment internal pressure exceeds 2.0 psig, or the internal vacuum exceeds 2.0 psig.

Regarding internal pressure limitations, the containment design pressure of 47 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 6 psig. The containment is designed to withstand an internal vacuum of 2.5 psig. The 2.0 psig vacuum is specified as an operating limit to avoid any difficulties with motor cooling.

LEAKAGE

The reactor shall not be critical if the containment leakage exceeds 0.1 weight per cent of the contained air per 24 hours at an internal pressure of 47 psig. This will be demonstrated by the initial pre-operational integrated leakage rate test (which will verify the 0.1% per day leak rate limit with the double penetrations and weld channels vented to the containment atmosphere) and subsequent monitoring of double penetrations and weld channel zone leakage less than 0.2 per cent of the containment free volume per day.

A containment leakage rate of 0.1 weight per cent of the contained air per 24 hours at an internal pressure of 47 psig, 271°F, under hypothetical accident conditions with 3 of 5 air recirculation units operating will maintain public exposure well below 10 CFR 100 values.

5.1.8 CONTAINMENT SYSTEM STRUCTURE-INSPECTION AND TESTING

An appropriate containment inspection plan is presently in the development stage. It is our intention however that, such an inspection will be conducted annually.

Initial Containment Leakage Rate Testing

Criterion: Containment shall be designed so that integrated leakage rate testing can be conducted at the peak pressure calcu-lated to result from the design basis accident after completion and installation of all penetrations and the leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance.

After completion of the containment structure and installation of all penetrations and weld channels, an initial integrated leakage rate test will be conducted at the containment design pressure (47 psig), maintained for a minimum of 24 hours, to verify that the leakage rate is no greater than 0.1 per cent by weight of the containment volume per day at design basis accident conditions. This leakage rate test will be performed using the absolute method. In addition, a reduced pressure integrated leakage rate test will be conducted at a pressure not less than 50% of the containment design pressure, maintained for a minimum of 24 hours.

Periodic Containment Leakage Rate Testing

Criterion: The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime.

The full pressure (47 psig) integrated leakage rate test will not be repeated unless major maintenance or modifications are made. Integrated leakage rate tests at the reduced test pressure will be conducted at periodic intervals during the life of the plant.

A leak rate test at the containment design pressure using the same method as the initial leak rate test can be performed at any time during the operational life of the plant, provided the plant is not in operation and precautions are taken to protect instruments and equipment from damage.

Provisions for Testing of Penetrations

Criterion: Provisions shall be made to the extent practical for periodically testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at the peak pressure calculated to result from occurrence of the design basis accident.

Penetrations are designed with double seals which are continuously pressurized above accident pressure. The large access openings such as the equipment hatch and personnel air locks are equipped with double gas doors and flanges with the space between the gaskets connected to the pressurization system. The system utilizes a supply of clean, dry, compressed air which will place the penetrations under an internal pressure above the peak calculated accident pressure.

A permanently piped monitoring system is provided to continuously measure leakage from all penetrations.

Leakage from the monitoring system is checked by continuous measurement of the integrated makeup air flow. In the event excessive leakage is discovered, each penetration can then be checked separately at any time.

Provisions for Testing of Isolation Valves

Criterion: Capability shall be provided to the extent practical for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus during periods of reactor shutdown.

Initiation of containment isolation employs coincidence circuits which allow checking of the operability and calibration of one channel at a time. Removal or bypass of one signal channel places that circuit in the half-tripped mode.

Hydrostatic tests of isolation valves in series are performed by first testing the upstream valve with the second valve open, then opening the upstream valve and closing the second valve, so that each valve will have an independent test.

The main steam and feedwater barriers and isolation valves in systems which connect to the Reactor Coolant System are hydrostatically tested to measure leakage.

Valves in the Residual Heat Removal System are not considered to be isolating valves in the unusual sense inasmuch as the system would be in operation under accident conditions.

Field and operational inspection and testing have been divided into three phases:

a) That taking place during erection of the containment building liner; – construction tests.

b) That taking place after the containment structure is erected and all penetrations are complete and installed; pre-operational tests.

c) Monitoring during reactor operation; post-operational tests.

5.1.8.1 Construction Tests

During erection of the liner, the following inspection and tests will be or have been performed:

Bottom Liner Plates

All liner plate welds are tested for leak tightness by vacuum box. The box is evacuated to at least a 5 psi pressure differential with the atmospheric pressure.

After completion of a successful leak test, the welds are covered by channels. A strength test is performed by applying 54 psig air pressure to the channels in the zone for a period of 15 minutes.

The zone of channel-covered welds is pressurized to 47 psig with a 20% by weight of Freon-air mixture. The entire run of the channel to plate welds is then traversed with a halogen leak detector.

The sensitivity of the leak detector is 1×10^{-9} standard CC per second. The sniffer is held approximately 1/2-inch from the weld and traversed at a rate of about 1/2-inch/second. The detection of any amount of halogen indicates a leak requiring weld repairs and retesting.

After the halogen test is complete all liner welds not accessible for radiography are pressurized with air to 47 psig and soap-tested. Any leaks indicated by bubbles are repaired and retested. Where leaks occur, welds are removed by arc gouging, grinding, chipping and/or machining, before rewelding. In addition, the zone of channels are held at the 47 psig air pressure for a period of at least two hours. The drop in pressure is not to exceed the equivalent of a leakage of 0.05% of the containment building volume per day. Compensation for change in ambient air temperature is made if necessary.

Vertical Cylindrical Walls and Dome

For a liner a complete radiograph is made of the first 10 feet of full penetraton weld made by each welder or welding operation. A minimum of a 12" film "spot" radiograph is made every 50 feet of weld thereafter on the side walls and dome, except where back-up plates are used. The radiograph films are given to United Engineers and Constructors for their review.

When a spot radiograph shows defects that require repair, two adjacent spots shall be radiographed. If defects requiring repair are shown in either of these, all of the welding performed by the responsible operator or welder shall be 100% radiographed to determine the end of defect.

The performance and acceptance standards for all radiography shall be ASME Section VIII, Paragraph UW51.

The liner plate to plate welds are tested for leak tightness by vacuum box techniques. After successful completion of the spot radiography and vacuum box tests and subsequent repair of all defects, the channels are welded in place over all seam welds in a pre-determined zone. A strength test is performed on the liner plate weld and the channel weld by pressurizing the channel with air at 54 psig for 15 minutes. In addition, each zone of channel covered weld is leak tested using the Freon-air mixture at 47 psig.

In locations where radiography is not possible, such as the lower courses of shell plates where back-up plates are used, and where liner bottom welds and floor plate welds are made to angles and tees, the liner fabricator welds on a 2" long overrun coupon. The overrun coupon is chipped off, marked for location and given to United Engineers and Constructors for testing. These welds are also vacuum box tested.

Welded studs are visually inspected, and at least one at the beginning of each day's work and another at approximately mid-day are bend-tested to 45° for each welder. Studs failing visual or bend-testing are removed.

While the liner is not a pressure vessel, industry experience has shown that leaks in pressure vessels normally occur at joints. For this reason, and following current liner fabrication practice, there is no radiographic or other non-destructive examination of liner plate.

Penetrations

Strength and leak tests of individual penetration internals and closures and sleeve weld channels are performed in a similar manner to the above and all leaks repaired and the penetration or weld channel retested until no further leaks are found.

5.1.8.2 PRE-OPERATIONAL TESTS

All penetrations and the welds joining these penetrations to the containment liner and the liner seam welds have been designed to provide a double barrier which can be continuously pressurized at a pressure higher than the design pressure of the containment. This blocks all of these potential sources of leakage with a pressurized zone and at the same time provides a means of monitoring the leakage status of the containment which is more sensitive to changes in the leakage characteristics of these potential leakage sources.

After the containment building is complete with liner, concrete structures, and all electrical and piping penetrations, equipment hatch and personnel locks in place, the following tests will be performed:

a) Strength Test:

A pressure test will be made on the completed building using air at 54 psig. This pressure will be maintained on the building for a period of at least one hour. During this test, measurements and observations will be made to verify the adequacy of the structural design. For a description of observations, cracks, strain gauges, etc., refer to the Containment Report.(7)

b) **Integrated Leakage Rate Tests:**

The integrated leakage rate tests which will be performed on the completed building at 47 psig using the absolute method. This leakage test will be performed with the double penetration and weld channel zones open to the containment atmosphere. The leakage rate to be demonstated by this test will be equal to or less than 0.1% of the containment free volume per day at design basis accident conditions. After it has been assured that there are no defects remaining from construction, a sensitive leak rate test will be conducted.

c) Sensitive Leak Rate Test:

The sensitive leak rate test will include only the volume of the weld channels and double penetrations. This test is considered more sensitive than the integrated leakage rate test, as the instrumentation used permits a direct measurement of leakage from the pressurized zones. The sensitive leak rate test will be conducted with the penetrations and weld channels at 50 psig and with the containment building at atmospheric pressure. The leak rate for the double penetrations and weld channel zones will be equal to or less than 0.2% of the containment free volume per day.

5.1.8.3 POST-OPERATIONAL TESTS

The double penetrations and the weld seam channels which are installed on the inside of the liner in the containment will be continuously pressurized to provide a continuous, sensitive and accurate means of monitoring their status with respect to leakage.

There is no need to repeat the full pressure integrated leakage rate test of the containment building, unless major maintenance or modifications of the containment are made. To allow for this possibility, it is permissible to pressurize the containment building at 54 psig, after the major modifications have been completed. Periodic reduced pressure containment integrated leakage rate tests will be performed.

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1. United States Atomic Energy Commission - 1963 - Nuclear Reactors and Earthquakes, TID-7024.

2. J. Blume, N. Newmark, L. Corning - Design of Multistory Reinforced Concrete Building for Earthquake Motions - Portland Cement Association

3. J. B. Stallmeyer, W. H. Munse and E. A. Selby, Fatigue Tests of Plates and Beams with Stud Shear Connections, HIGHWAY RESEARCH RECORD, No. 76.

4. Robert C. Singleton, The Growth of Stud Welding, WELDING ENGINEER, July 1963.

5. S. Timoshenko and S. Woinowsky-Kreiger, Theory of Plates and Shells, Second Edition, McGraw-Hill, 1954.

6. American Concrete Institute, Code for Reinforced Concrete Chimney Design, ACI-505.

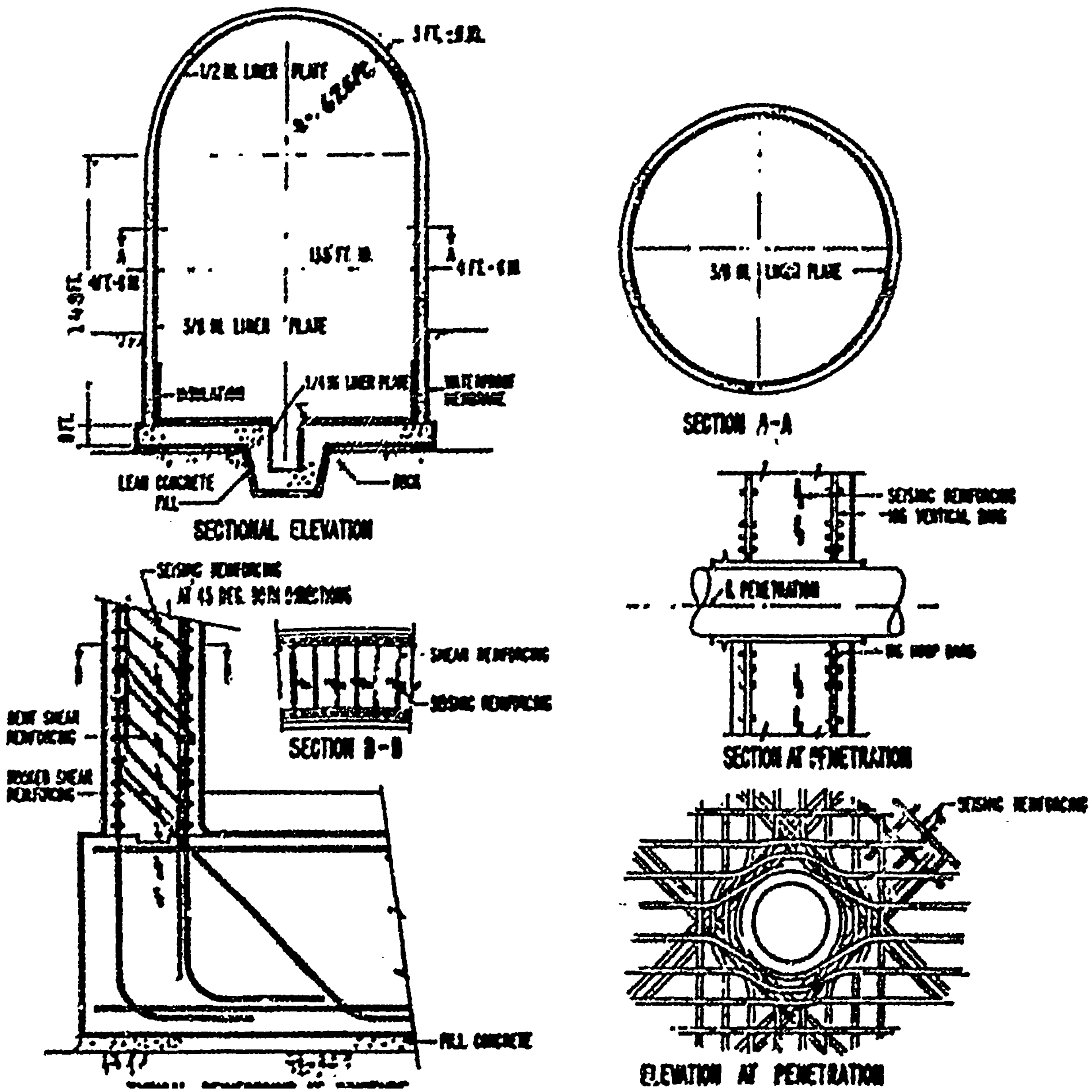
7. "Indian Point No. 2 Containment Design Report", to be issued.

TABLE 5.1-1

FLOODED WEIGHTS - CONTAINMENT BUILDING

Item	Flooded Operating Weight, lb
Pressurizer - 1	346,000
Steam Generators - 4	3,746,000
Reactor - 1	
(a) Vessel	868,000
(b) Internals	420,000
(c) Piping	1,000,000
Reactor Pumps - 4	824,000
Accumulator Tanks - 4	529,000
175 Ton Polar Crane - 1	650,000
Ventilation Fans - 4	656,000
Reactor Coolant Drain Tank - 1	20,000
Pressure Relief Tank - 1	100,000
Other Miscellaneous Equipment	100,000
TOTAL	9,259,000

CONTAINMENT STRUCTURE



Indian Point
Final Facility Description and Safety Analysis Report
Dated: October 15, 1968

Figure 5.1-2, Titled "Containment Building General Arrangement Plans" Ex 4

The above was **redacted** by NRC staff as **sensitive information** to allow release of this document to the public.

Indian Point
Final Facility Description and Safety Analysis Report
Dated: October 15, 1968

Ex 4

Figure 5.1-3, Titled "Containment Building General Arrangement Plans"

The above was **redacted** by NRC staff as **sensitive information** to allow release of this document to the public.

Indian Point
Final Facility Description and Safety Analysis Report
Dated: October 15, 1968

Figure 5.1-4, Titled "Containment Building General Arrangement Plans"

Ex 4

The above was **redacted** by NRC staff as **sensitive information** to allow release of this document to the public.

Indian Point
Final Facility Description and Safety Analysis Report
Dated: October 15, 1968

Figure 5.1-5, Titled "Containment Building General Arrangement Elevations" Ex 4

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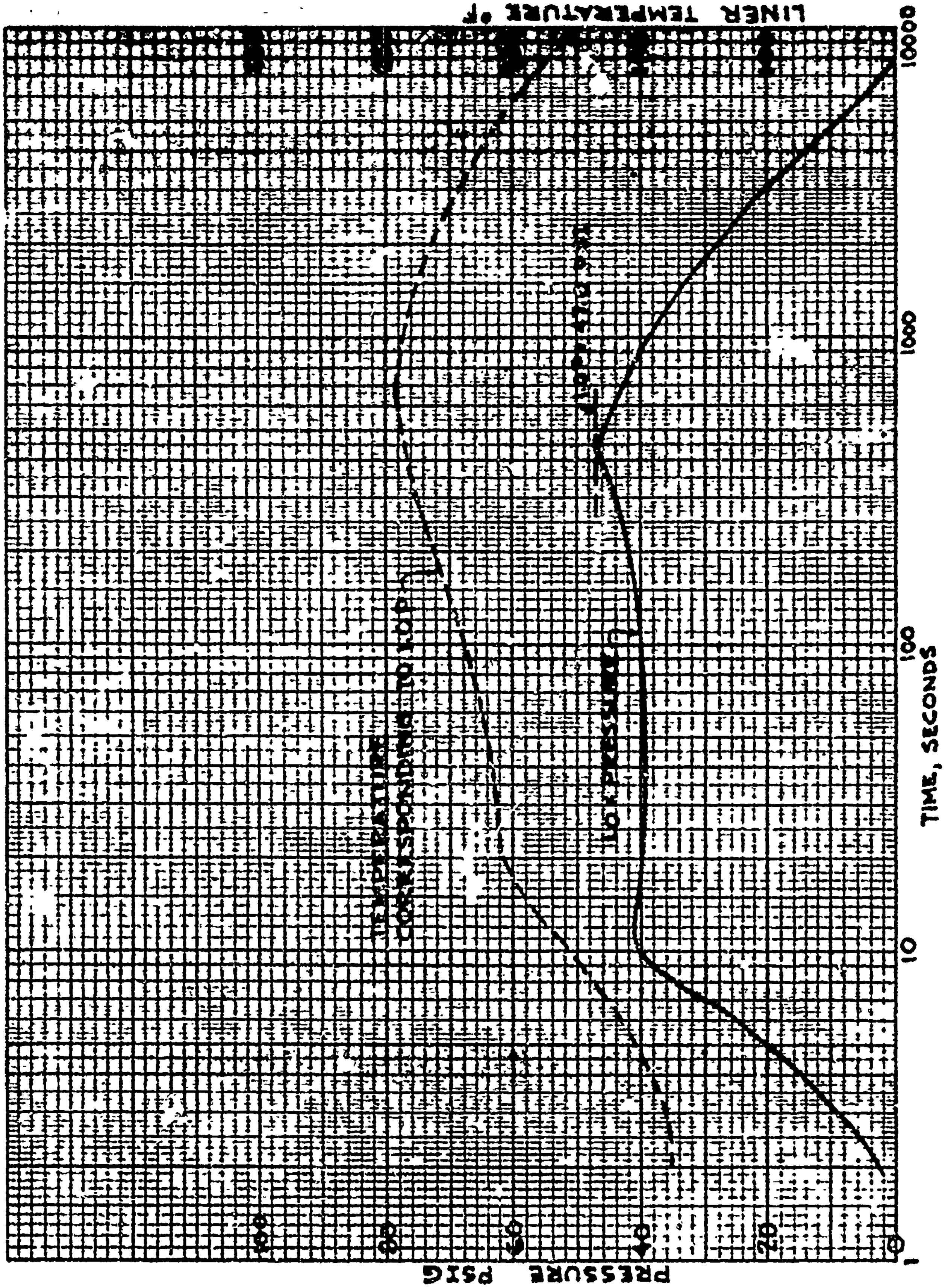
Indian Point
Final Facility Description and Safety Analysis Report
Dated: October 15, 1968

Ex 4

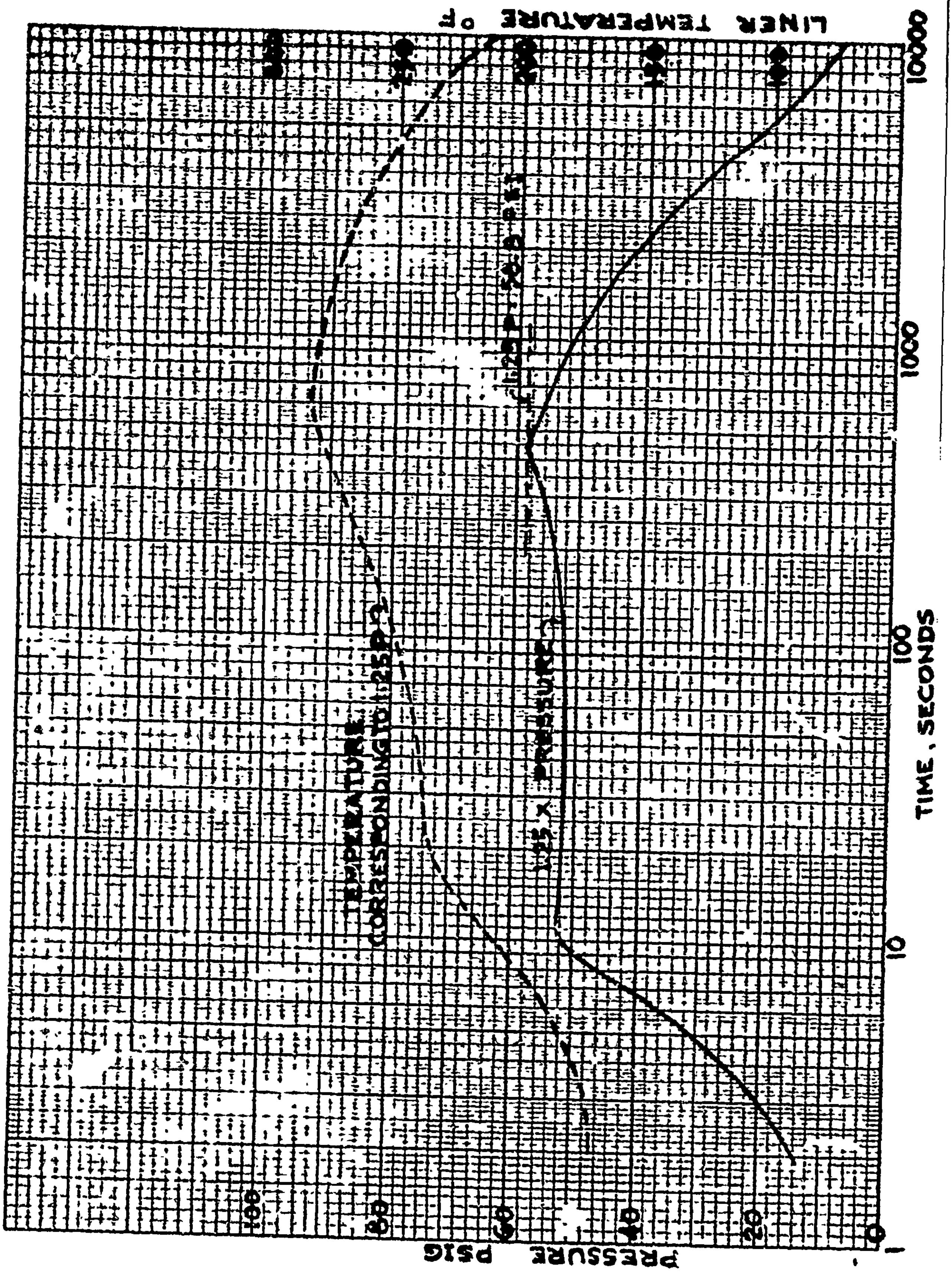
Figure 5.1-6, Titled "Containment Building General Arrangement Elevations"

The above was **redacted** by NRC staff as **sensitive information** to allow release of this document to the public.



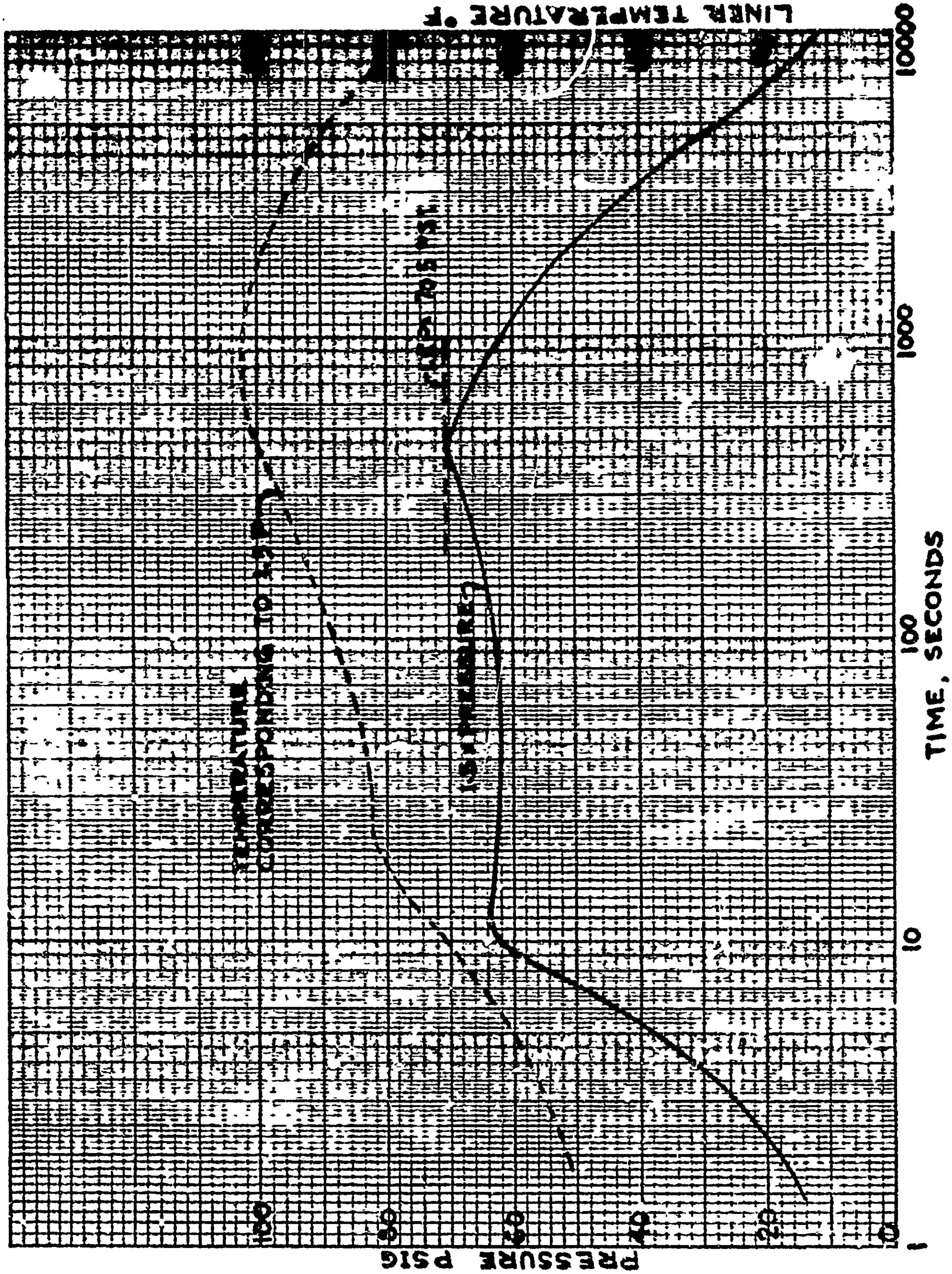
DESIGN PRESSURE-TEMPERATURE TRANSIENT

Figure 5.1-8



1.25 TIMES DESIGN PRESSURE-TEMPERATURE TRANSIENT

Figure 5.1-9



1.50 TIMES DESIGN PRESSURE-TEMPERATURE TRANSIENT

Figure 5.1-10

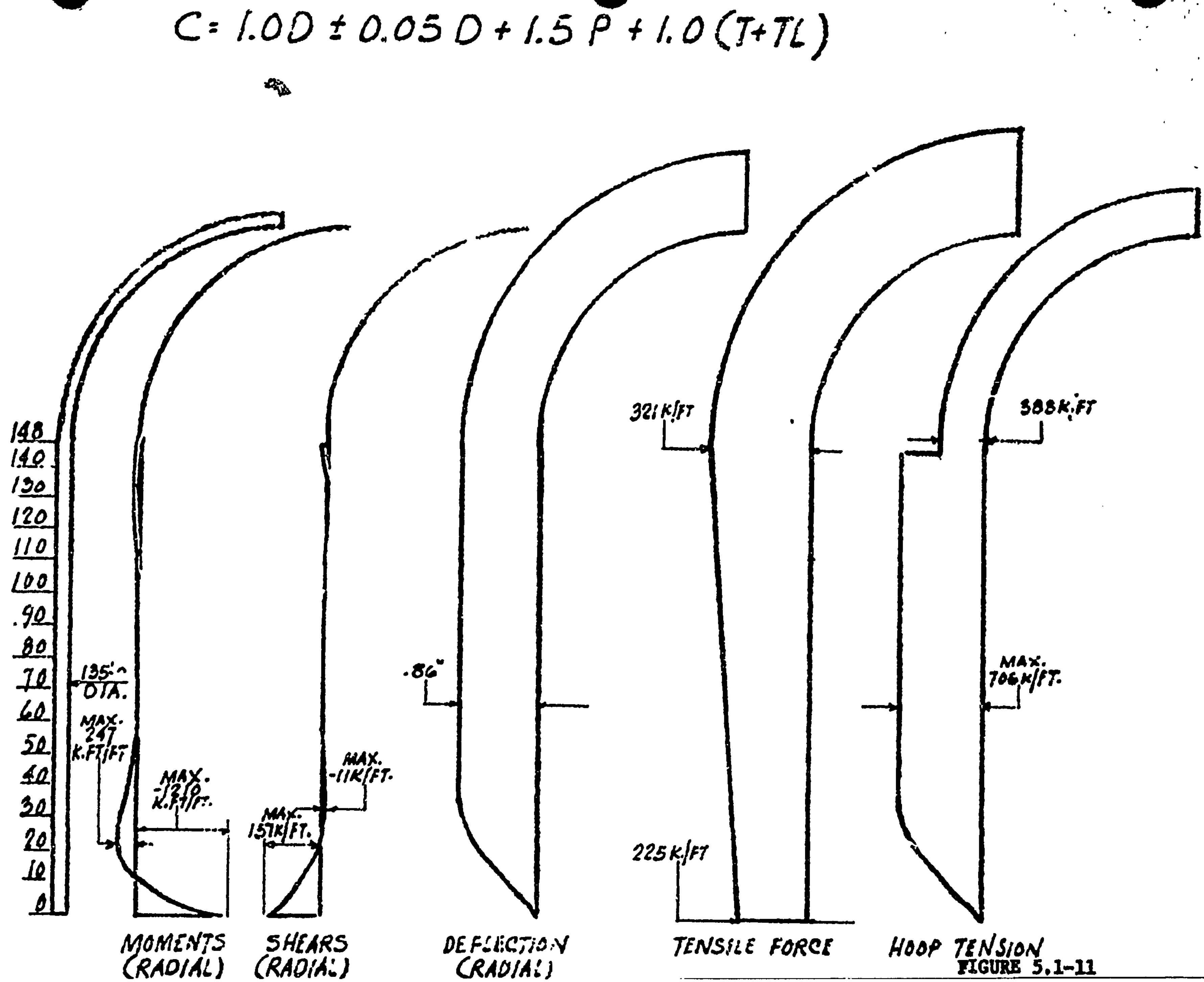
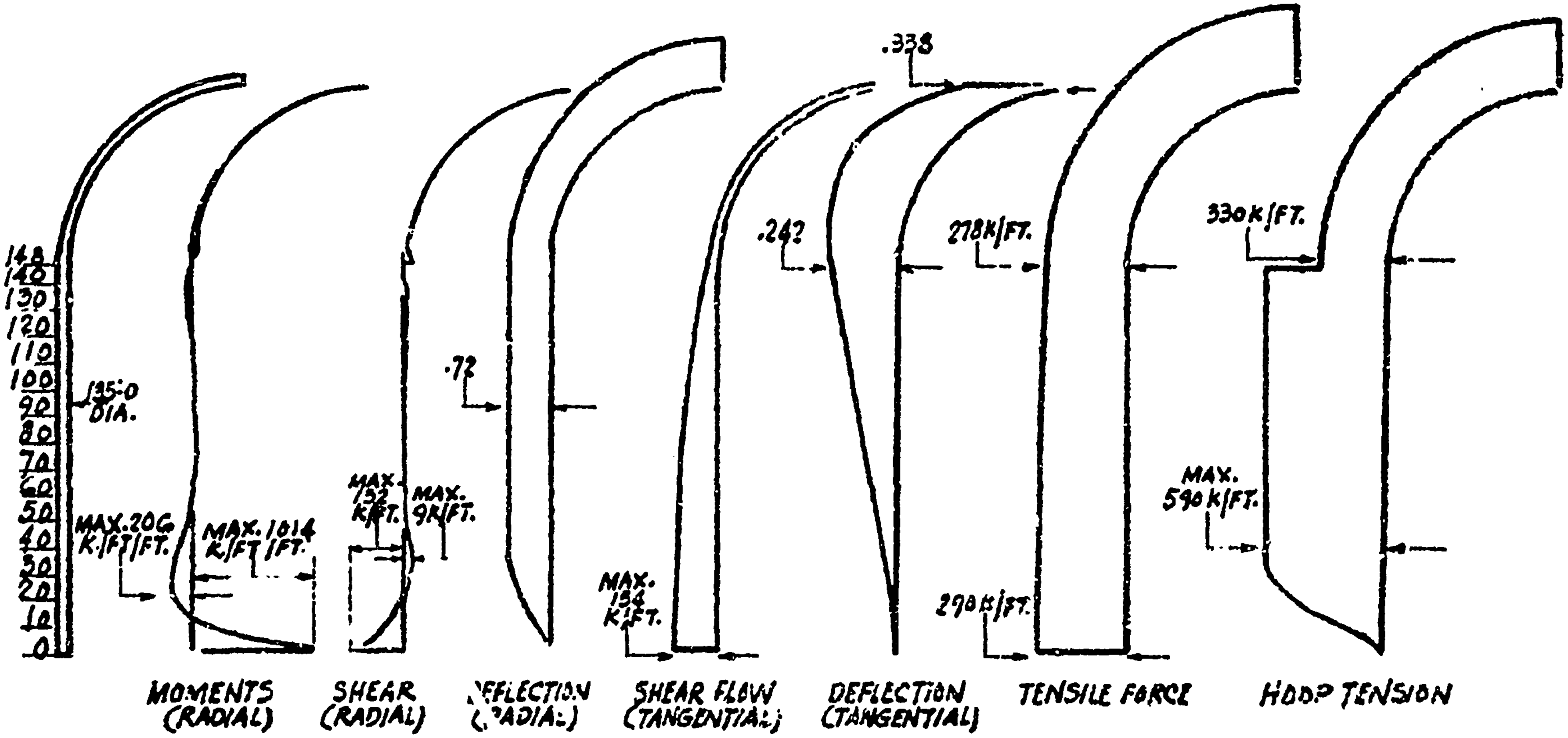

C= 1.0D ± 0.05D + 1.5P + 1.0(T+TL)
148
140
130
120
110
100
90
80
70
60
50
40
30
20
10
0
135' DIA.
MAX. 247 K.FT/FT
MAX. -120 K.FT/FT.
MAX. 157K/FT.
MAX. -11K/FT.
.86"
321 K/FT
225 K/FT
388 K/FT
MAX. 706 K/FT.
MOMENTS (RADIAL)
SHEARS (RADIAL)
DEFLECTION (RADIAL)
TENSILE FORCE
HOOP TENSION

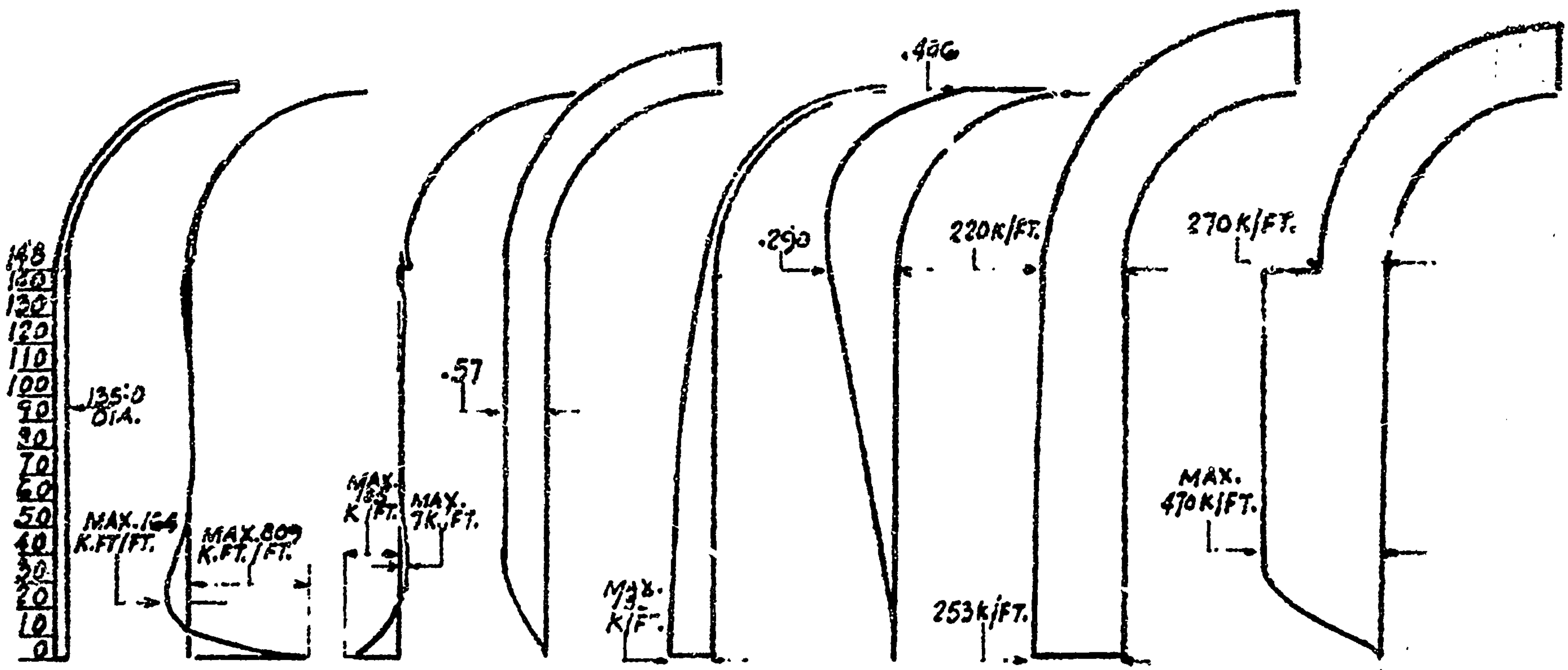
FIGURE 5.1-11

$$C = 1.0D \pm 0.05D + 1.25P + 1.0(T' + TL') + 1.25E$$



(EARTHQUAKE)

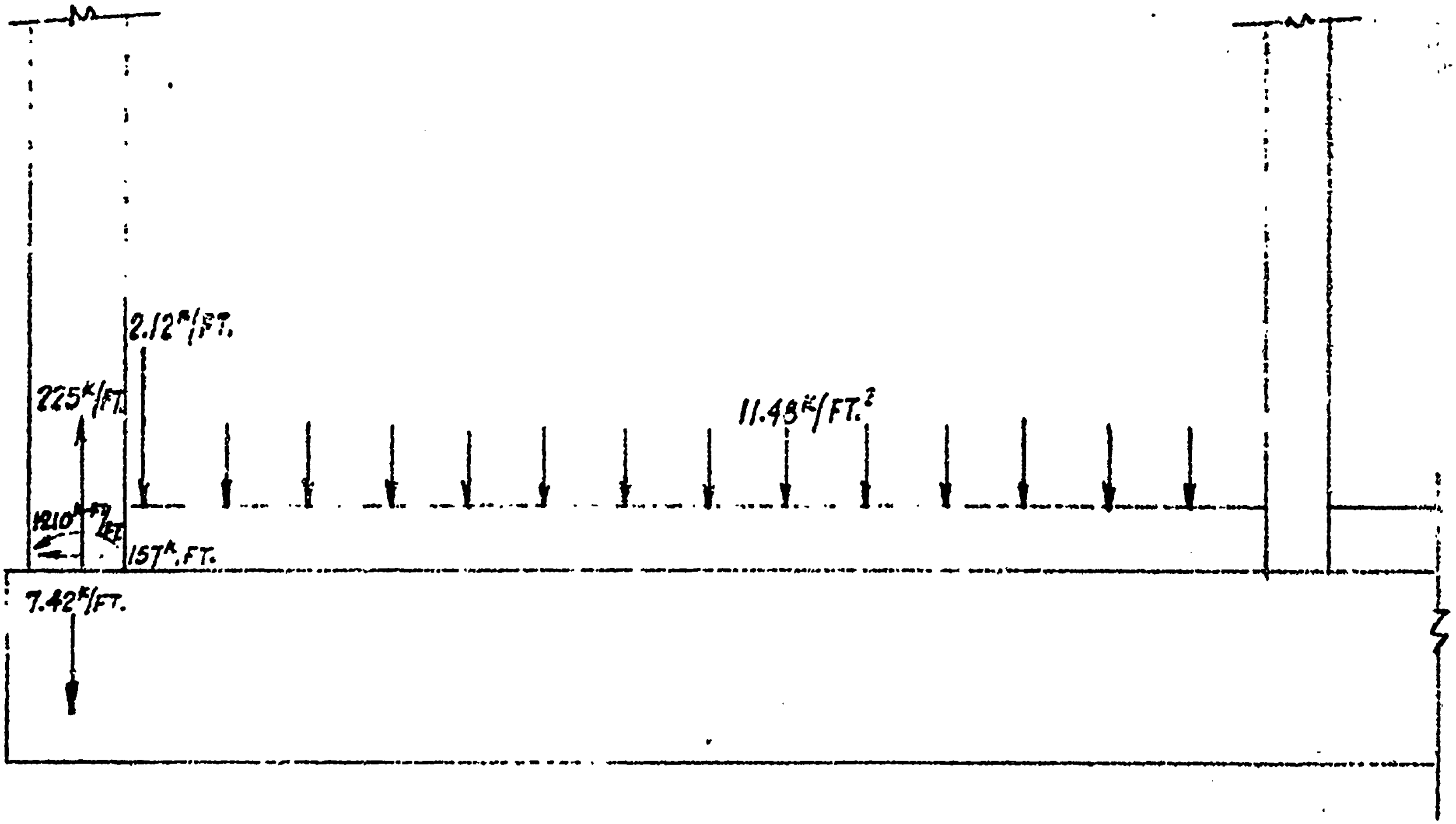
$$C = 1.0D \pm 0.05D + 1.0P + 1.0(T'' + TL'') + 1.0E'$$



MOMENTS (RADIAL) | SHEAR (RADIAL) | DEFLECTION (RADIAL) | SHEAR FLOW (TANGENTIAL) | DEFLECTION (TANGENTIAL) | TENSILE FORCE | HOOP TENSION

(EARTHQUAKE)

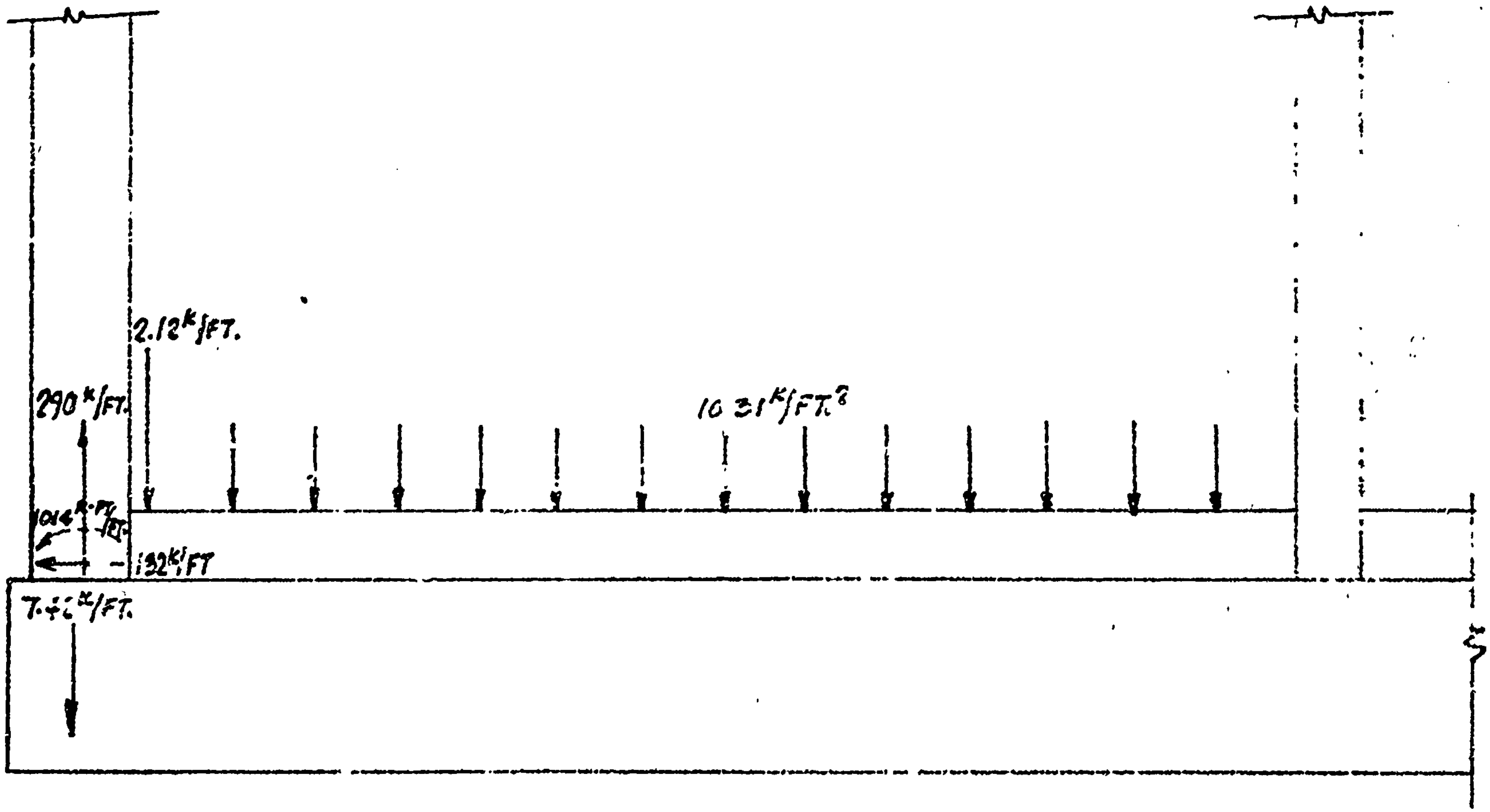
$$C = 1.0D \pm 0.05D + 1.5P + 1.0(T + T_L)$$



LOADING DIAGRAM IN MAT

FIGURE 5.1-16

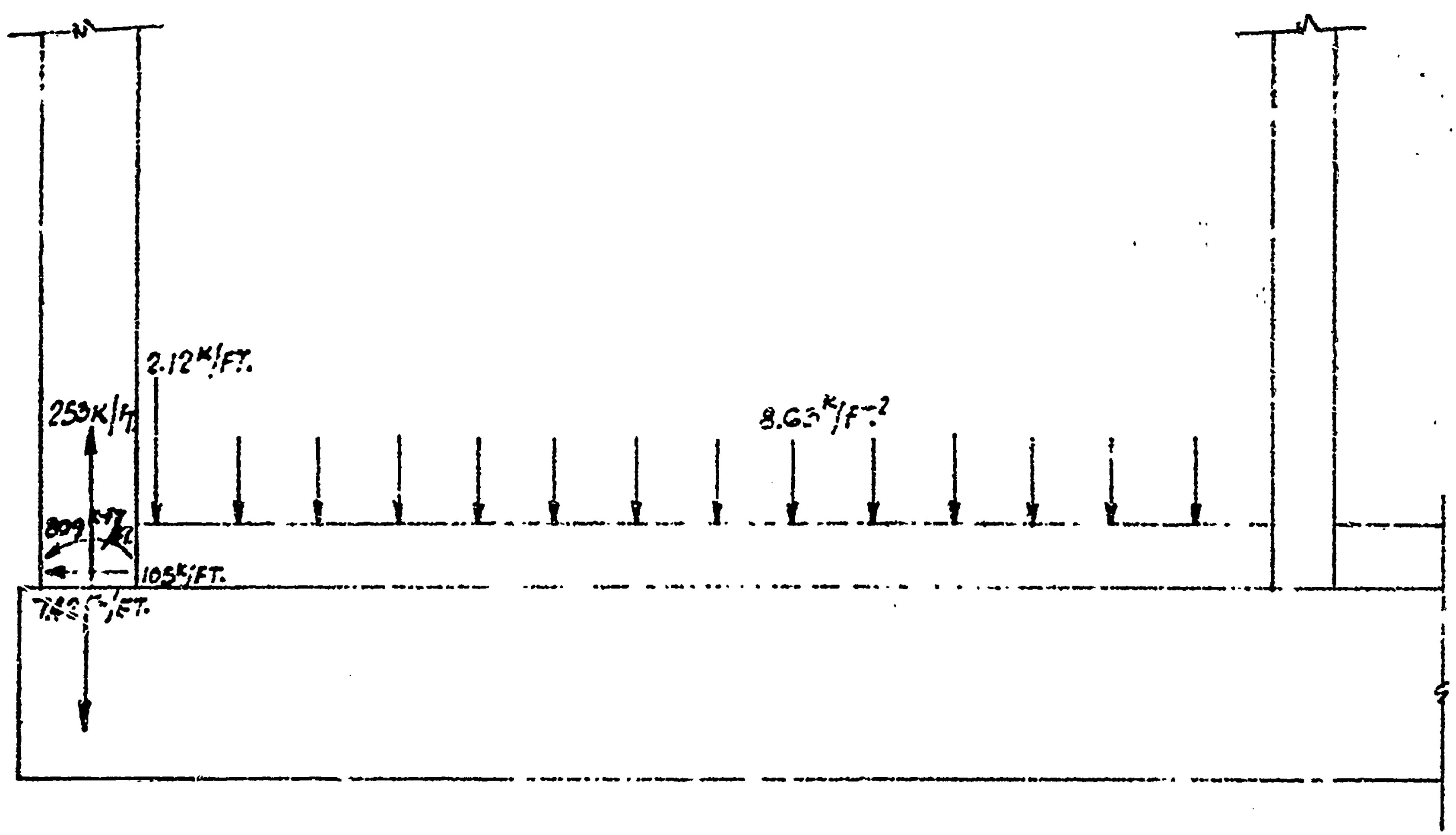
$$C = 1.0D \pm 0.05D + 1.25P + 1.0(T + TL') + 1.25E$$



LOADING DIAGRAM IN MAT

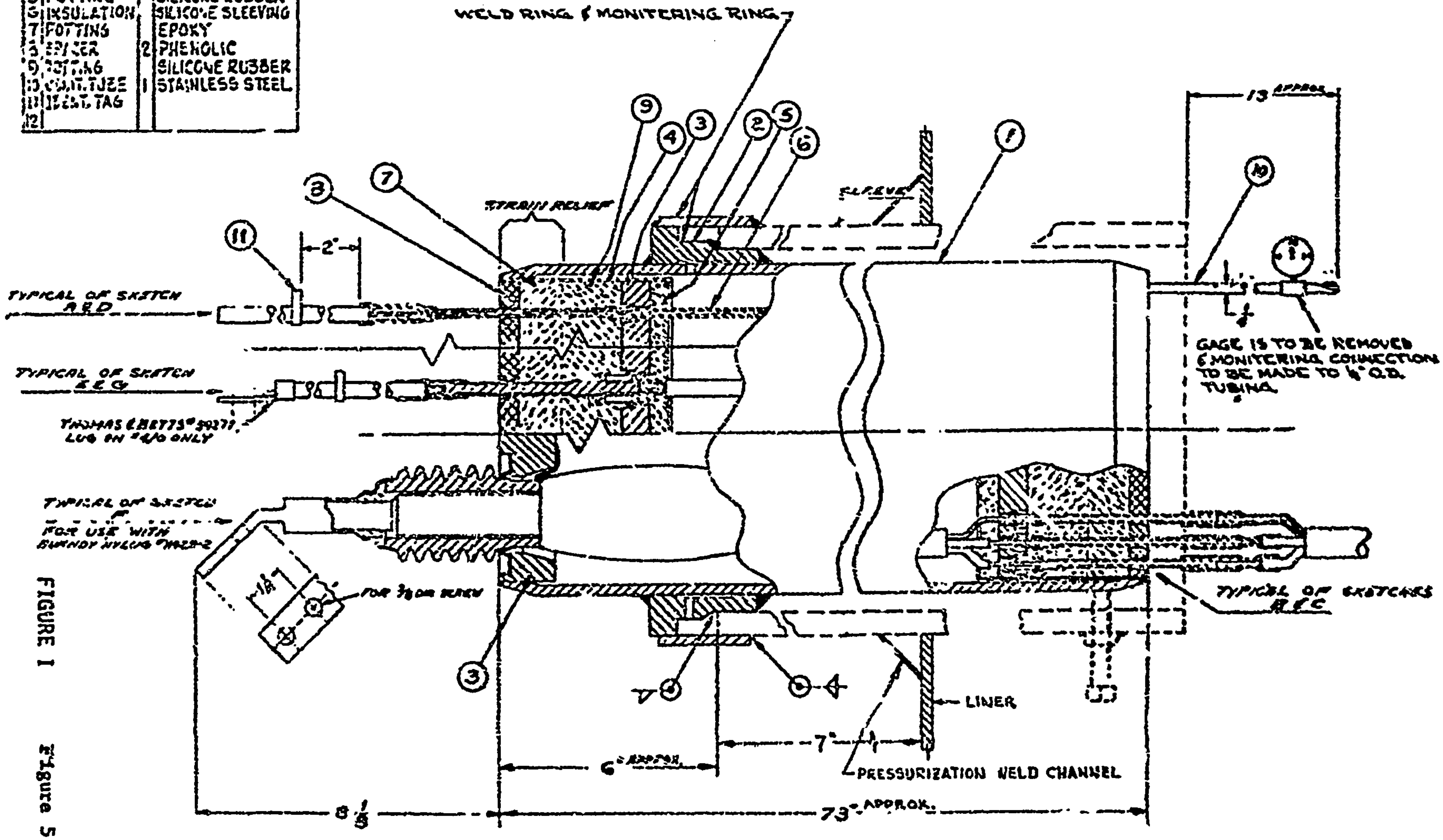
FIGURE 5.1-15

$$C = 1.0D \pm 0.05D + 1.0P + 1.0(T' + TL') + 1.0E'$$



LOADING DIAGRAM IN MAT

NO	PART NAME	QTY	DESCRIPTION
1	CONTAINER	1	STAINLESS STEEL
2	FLANGE	1	CARBON STEEL
3	HEADER	2	STAINLESS STEEL
4	TUBE	2	STAINLESS STEEL
5	POTTING		SILICONE RUBBER
6	INSULATION		SILICONE SLEEVING
7	POTTING		EPOXY
8	SPACER	2	PHENOLIC
9	POTTING		SILICONE RUBBER
10	CONT. TUBE	1	STAINLESS STEEL
11	IDENT. TAG		
12			



TYPICAL ELECTRICAL PENETRATIONS

FIGURE 1

Figure 5.1-17

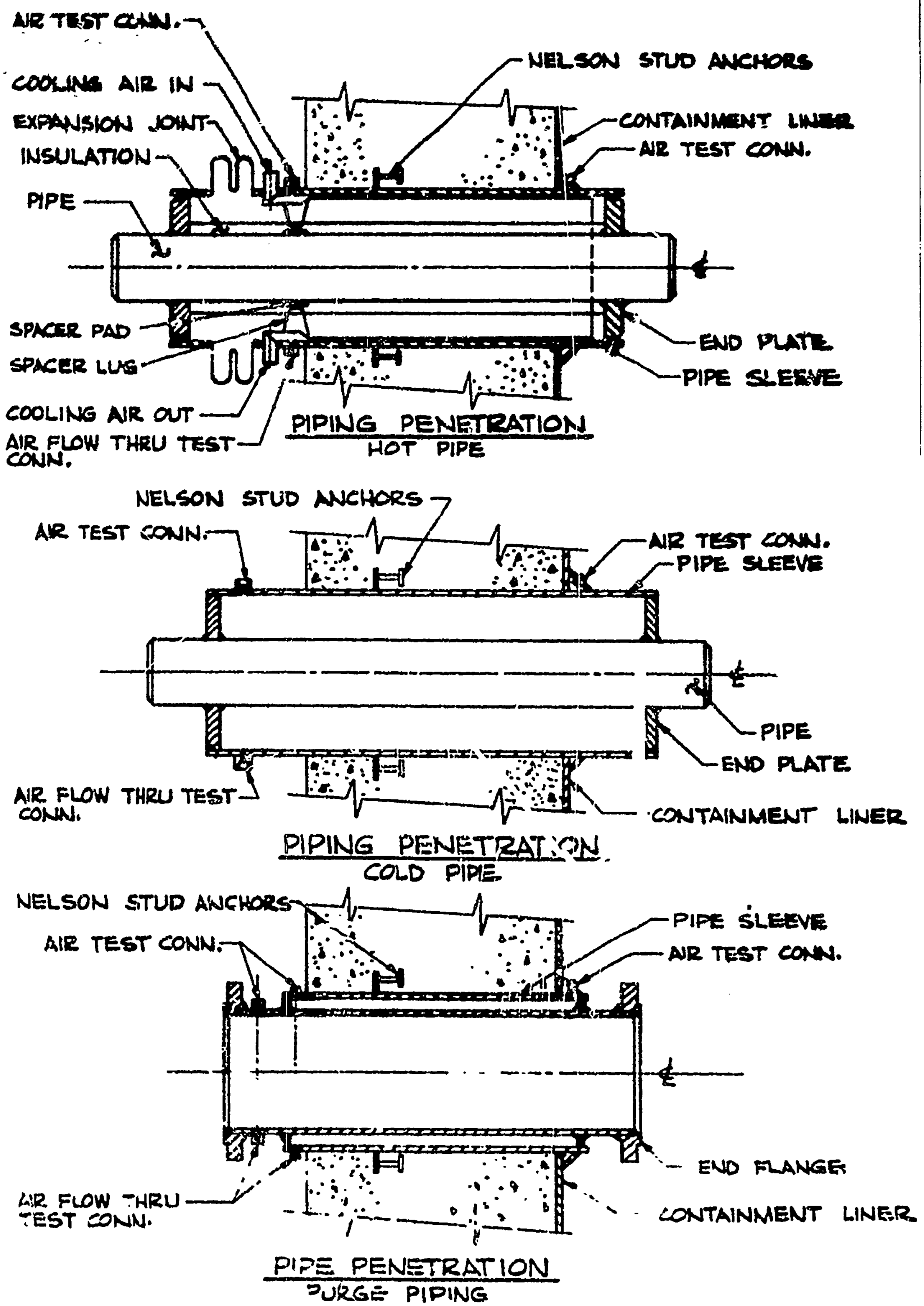
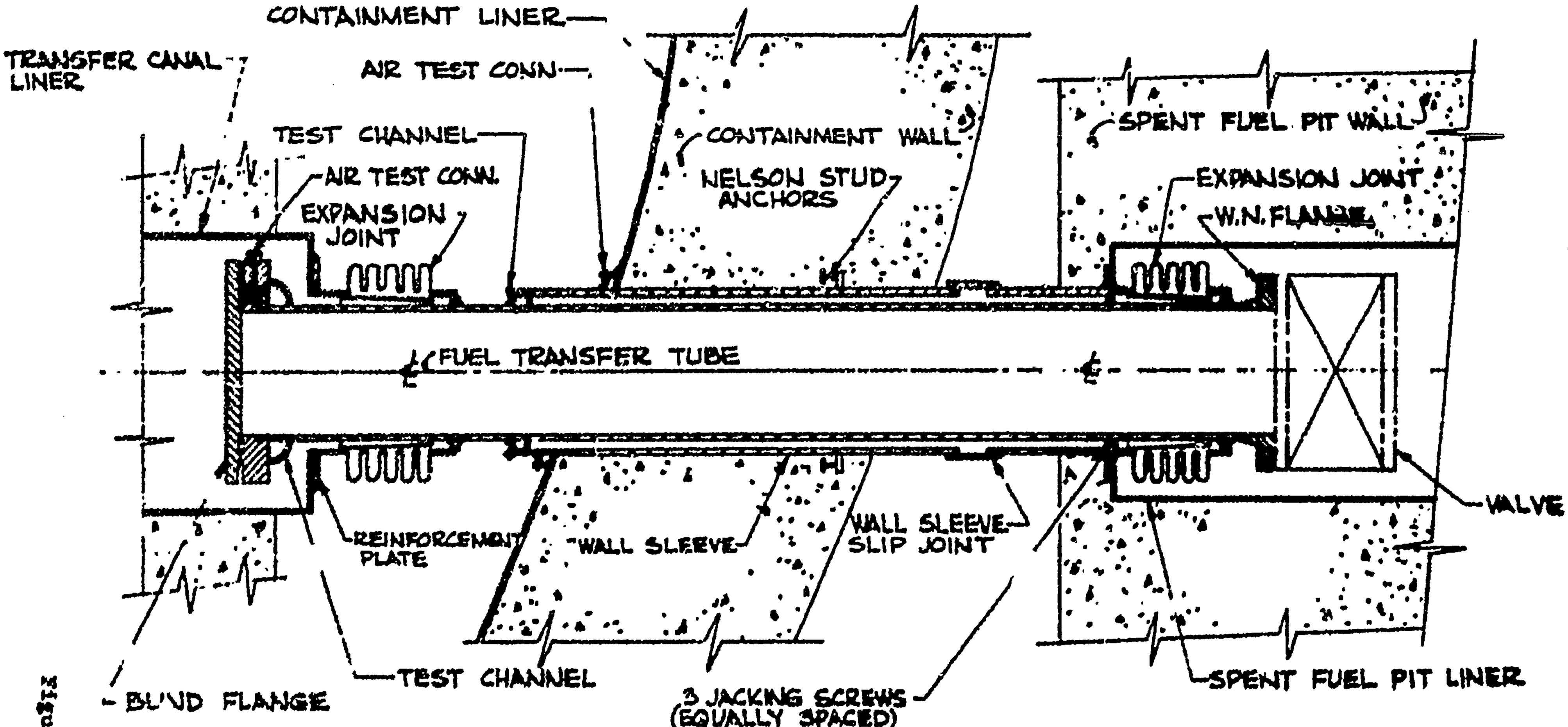

AIR TEST CONN.
COOLING AIR IN
EXPANSION JOINT
INSULATION
PIPE
NELSON STUD ANCHORS
CONTAINMENT LINER
AIR TEST CONN.
SPACER PAD
SPACER LUG
END PLATE
PIPE SLEEVE
COOLING AIR OUT
AIR FLOW THRU TEST CONN.
PIPING PENETRATION
HOT PIPE
NELSON STUD ANCHORS
AIR TEST CONN.
AIR TEST CONN.
PIPE SLEEVE
PIPE
END PLATE
AIR FLOW THRU TEST CONN.
CONTAINMENT LINER
PIPING PENETRATION
COLD PIPE
NELSON STUD ANCHORS
AIR TEST CONN.
PIPE SLEEVE
AIR TEST CONN.
END FLANGE
AIR FLOW THRU TEST CONN.
CONTAINMENT LINER
PIPE PENETRATION
PURGE PIPING

Figure 5.1-18

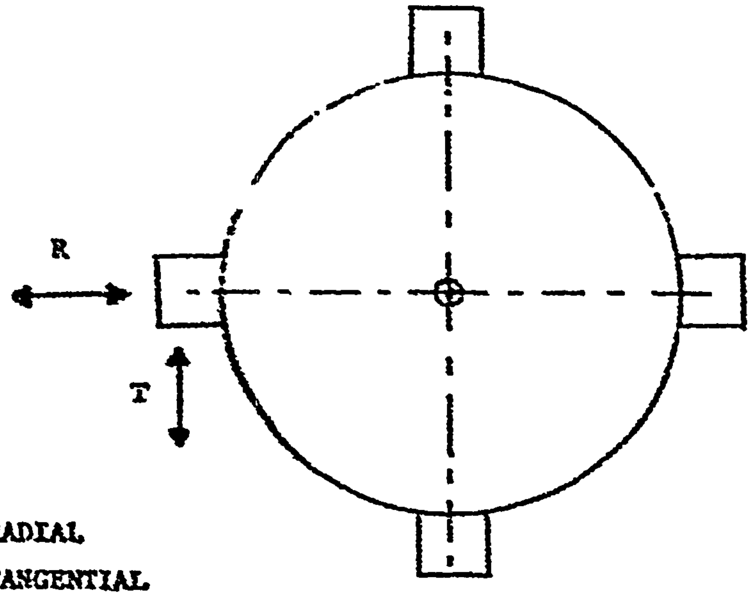


FUEL TRANSFER TUBE PENETRATION
(CONCEPTUAL DRAWING)

Figure 5.1-

MAXIMUM FORCES
ACTING ON A REACTOR
VESSEL SUPPORT

	A	B	C	D	Σ		
	REACTOR VESSEL WEIGHT & PIPING REACTION	PIPE BREAK CASE II	PIPE BREAK CASE III	EARTH-QUAKE Z & VERTICAL DIRECTION	A+B	A+C	A+D
P (lb)	934,000	-	525,000	395,000	934,000	1,459,000	1,329,00
R (lb)	322,000	-	-	-	322,000	322,000	322,00
T (lb)	140,000	1,187,000	710,000	969,000	1,327,000	850,000	1,109,00



PIPE BREAK CASE II
IS BETWEEN PUMP AND
ELL ON REACTOR NOZZL

PIPE BREAK CASE III
IS AT STEAM GENERATO
INLET

EARTHQUAKE IN Z
DIRECTION ACTS NORMA
TO REACTOR COOLANT
OUTLET LINE

R-RADIAL
T-TANGENTIAL
P-VERTICAL

PLAN

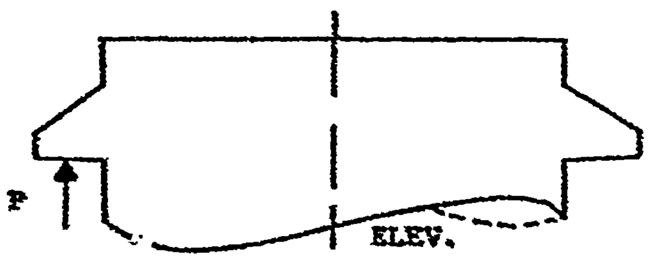


FIGURE 5.1-2

5.2 CONTAINMENT ISOLATION SYSTEM

5.2.1 DESIGN BASIS

Each system whose piping penetrates the containment leakage limiting boundary is designed to maintain or establish isolation of the containment from the outside environment under the following postulated conditions:

a. Any accident for which isolation is required (severely faulted conditions) with

b. A coincident independent single failure or malfunction (expected fault condition) occurring in any active system component within the isolated bounds.

Piping penetrating the containment is designed for pressures at least equal to the containment design pressure. Containment isolation valves are provided as necessary in lines penetrating the containment to assure that no unrestricted release of radioactivity can occur. Such releases might be due to rupture of a line within the containment concurrent with a loss-of-coolant accident, or due to rupture of a line outside the containment which connects to a source of radioactive fluid within the containment.

In general, isolation of a line outside the containment protects against rupture of the line inside concurrent with a loss-of-coolant accident, or closes off a line which communicates with the containment atmosphere in the event of a loss-of-coolant accident.

Isolation of a line inside the containment prevents flow from the reactor coolant system or any other large source of radioactive fluid in the event that a piping rupture outside the containment occurs. A piping rupture outside the containment at the same time as a loss-of-coolant accident is not considered credible, as the penetrating lines are seismic Class I design up to and includi the second isolation barrier and are assumed to be an extension of containment.

The isolation valve arrangement provides two barriers between the Reactor Coolant System or containment atmosphere, and the environment.

System design is such that failure of one valve to close will not prevent isolation, and no manual operation is required for immediate isolation. Automatic isolation is initiated by a containment isolation signal Section 7, derived either from any automatic safety injection signal, ("T" signal), or manually.

The containment isolation valves have been examined to assure that they are capable of withstanding the maximum potential seismic loads.

To assure their adequacy in this respect:

a) Valves are located in a manner to reduce the accelerations on the valves. Valves suspended on piping spans are reviewed for adequacy for the loads to which the span would be subjected. Valves are mounted in the position recommended by the manufacturer.

b) Valve yokes are reviewed for adequacy and strengthened as required for the response of the valve operator to seismic loads.

c) Where valves are required to operate during seismic loading, the operator forces are reviewed to assure that system function is preserved. Seismic forces on the operating parts of the valve are small compared to the other forces present.

d) Control wires and piping to the valve operators are designed and installed to assure that the flexure of the line does not endanger the control system. Appendages to the valve, such as position indicators and operators, are checked for structural adequacy.

Containment Isolation Valves Criterion

Isolation valves are provided as necessary for all fluid system lines penetrating the containment to assure at least two barriers for redundance against leakage of radioactive fluids to the environment in the event of a loss-of-coolant accident. These barriers, in the form of isolation valves or closed systems, are defined on an individual line basis. In addition to satisfying containment isolation criteria, the valving is designed to facilitate normal operation and maintenance of the systems and to ensure reliable operation of other engineered safeguards systems.

With respect to numbers and locations of isolation valves, the criteria applied are generally those outlined by the six classes described in Section 5.2.2 below.

5.2.2 SYSTEM DESIGN

The six classes listed below are general categories into which lines penetrating containment may be classified. The seal water referred to in the listing of categories is provided by the Isolation Valve Seal Water System described in Section 6.5. The following notes apply to these classification

1. The "not missile protected" designation refers to lines that are not protected throughout their length inside containment against missiles generated as the result of a loss of coolant accident. These lines, therefore, are not assumed invulnerable to rupture as a result of a loss of coolant.

2. In order to qualify for containment isolation, valves inside the containment must be located behind the missile barrier for protection against loss of function following an accident.

3. Manual isolation valves that are locked closed or otherwise closed and under administrative control during power operation qualify as automatic trip valves.

4. A check valve qualifies as an automatic trip valve in certain incoming lines not requiring seal water injection.

5. The double disk type of gate valve is used to isolate certain lines. When sealed by water injection, this valve provides two barriers against leakage of radioactive liquids or containment atmosphere.

6. In lines isolated by globe valves and provided with seal water injection, the valves are installed so that the seal water wets the stem packing.

7. Excessive loss of seal water through an isolation valve that fails to close on signal, is prevented by the high resistance of the seal water injection line. A water seal at the failed valve is assured by proper slope of the protected line, or a loop seal, or by additional valves on the side of the isolation valves away from the containment.

8. Isolated lines between the containment and the second outside isolation valve are designed to the same seismic criteria as the containment vessel, and are assumed to be an extension of containment.

Class 1 (Outgoing Lines, Reactor Coolant System)

Normally operating outgoing lines connected to the Reactor Coolant System are provided with at least two automatic trip valves in series located outside the containment. Automatic seal water injection is provided for lines in this classification.

An exception to the general classification is the residual heat removal loop outlet line, which has two barriers established by normally closed valves.

Class 2 (Outgoing Lines)

Normally operating outgoing lines not connected to the Reactor Coolant System, and not missile protected or which can otherwise communicate with the containment atmosphere following an accident, are provided at a minimum with two automatic trip valves in series outside containment. Automatic seal water injection is provided for lines in this classification. Most of these lines are not vital to plant operation following an accident.

An exception is the residual heat exchanger cooling water return line, which is valved in accordance with safeguards operation.

Class 3 (Incoming Lines)

Incoming lines connected to open systems outside containment, and not missile protected or which can otherwise communicate with the containment atmosphere following an accident are provided with one of the following arrangements outside containment:

1. Two automatic trip valves in series, with automatic seal water injection. This arrangement is provided for lines which are not necessary to plant operation after an accident.

2. Two manual isolation valves in series, with manual seal water injection. This arrangement is provided for lines which remain in service for a time, or are used periodically, subsequent to an accident.

Incoming lines connected to closed systems outside containment, and not missile protected or which can otherwise communicate with the containment atmosphere are provided, at a minimum, with one check valve or normally closed isolation valve located either inside or outside containment. The closed piping system outside containment provides the necessary isolation redundancy. Most lines in this category are provided with additional isolation valves which satisfy particular systems or safeguards requirements. Seal water injection is provided for certain lines in this category.

Exceptions are the containment spray headers and residual heat exchangers cooling water supply line, for which valving is based on safeguards requirements.

Class 4 (Missile Protected)

Normally operated incoming and outgoing lines which penetrate the containment and are connected to closed systems inside the containment and protected from missiles throughout their length and are provided with at least one manual isolation valve located outside the containment. Seal water injection is not required for this class of penetration.

Class 5 (Normally Closed Lines Penetrating the Containment)

Lines which penetrate the containment and which can be opened to the containment atmosphere but which are normally closed during reactor operation are provided with two isolation valves in series or one isolation valve and one blind flange. One valve or flange is located inside and the second valve or flange located outside the containment.

Class 6 (Special Service)

There are a number of special groups of penetrating lines and containment access openings. These are discussed below.

Each ventilation purge duct penetration is provided with two tight-closing butterfly valves, which are closed during reactor power operation and are actuated to the closed position automatically upon a containment isolation or a containment high radiation signal. One valve is located inside and one valve is located outside the containment at each penetration. The space between valves is pressurized by air from the Penetration and Weld Channel Pressurization System whenever they are closed.

The containment pressure relief line is similarly protected. However, since the line can be opened during reactor power operation, three tight closing butterfly valves in series are provided, one inside and two outside the containment. These valves also are actuated to the closed position upon a containment isolation or containment high radiation signal. The two intra-valve spaces are pressurized by air from the Penetration and Weld Channel Pressurization System whenever they are closed.

The equipment access closure is a bolted, gasketed closure which is sealed during reactor operation. The personnel air locks consist of two doors in series with mechanical interlocks to assure that one door is closed at all times. Each air lock door and the equipment closure are provided with double gaskets to permit pressurization between the gaskets by the Penetration and Weld Channel Pressurization System, Section 6.6.

The fuel transfer tube penetration inside the containment is designed to present a missile protected and pressurized double barrier between the containment atmosphere and the atmosphere outside the containment. The penetration closure is treated in a manner similar to the equipment access hatch. A positive pressure is maintained between these gaskets to complete the double barrier between the containment atmosphere and the inside of the fuel transfer tube. The interior of the fuel transfer tube is not pressurized. Seal water injection is not required for this penetration.

The following lines would be subjected to pressure in xcess of the Isolation Valve Seal Water System design pressure (150 psig) in the event of an accident, due to operation of the recirculation pumps:

1. Residual heat removal loop inlet line
2. Residual heat removal loop outlet line
3. By pass line from residual heat exchanger outlet to safety injection pumps suction
4. Residual heat removal pumps miniflow line
5. Residual heat removal loop sample line
6. Recirculation pump discharge sample line

Lines 1, 2 and 3 are isolated by double disc gate valves, while 5, and 6, are each isolated by two globe valves in series. Line 4 is isolated by a globe and a gate valve in series. These valves can be sealed by nitrogen gas from the high pressure nitrogen supply of the Isolation Valve Seal Water System. A self contained pressure regulator, operates to maintain the nitrogen injection pressure slightly higher than the maximum expected line pressure. These valves are closed during power operation, and the nitrogen gas injection is manually initiated.

Lines which communicate with the containment atmosphere at all times (normally filled with air or vapor) include:

1. Steam jet air ejector return line to containment
2. Containment radiation monitor inlet and outlet lines

In an accident condition the space between the two containment isolation valves in each line are sealed by pressurizing with air from the Penetration and Weld Channel Pressurization System. The air is introduced into each space at approximately 2 psi above the containment design pressure through a separate line from the Penetration and Weld Channel Pressurization System. Parallel (redundant) fail open valves in each injection line open on the appropriate containment isolation signal to provide a reliable supply of pressurizing air. A flow limiting orifice in each injection line presents excessive air consumption if one of these valves spuriously fails to open, or if one of the containment isolation valves fails to respond to the "trip" signal.

5.2.2.1 Isolation Valves and Instrumentation Diagrams

15 Figures 5.2-1 through 5.2-24 show all valves in lines leading to the atmosphere or to closed systems on both sides of the containment barrier, valve actuation and preferential failure modes, the application of "trip" (containment isolation signals, relative location of the valves with respect to missile barriers, and
15 the boundaries of Seismic Class I designed lines. Figure 5.2-25 defines the nomenclature and symbols used.

5.2.2.2 Valve Parameters Tabulation

A summary of the fluid systems lines penetrating containment and the valves and closed systems employed for containment isolation is presented in Table 5.2-1. Each valve is described as to type, operator, position indication and open or closed status during normal operation, shutdown and accident conditions. Information is also presented on valve preferential failure mode, automatic trip by the containment isolation signal, and the fluid carried by the line.

Containment isolation valves are provided with actuation and control equipment appropriate to the valve type. For example, air operated globe and diaphragm (Saunders Patent) valves are generally equipped with air diaphragm operators, with fail-safe operation provided by the control devices in the instrument air supply to the valve. Motor operated gate valves are capable of being supplied from reliable on-site emergency power as well as their normal power source. Manual and check valves, of course, do not require actuation or control systems.

The automatically tripped isolation valves are actuated to the closed position by one of two separate containment isolation signals. The first of these signals is derived in conjunction with automatic safety injection actuation, and trips the majority of the automatic isolation valves. These are valves in the so-called "non-essential"* process lines penetrating the containment. This is defined as "Phase A" isolation and the trip valves are designated by the letter "T" in the isolation diagrams, Figures 5.2-1 through 5.2-23 This signal also initiates automatic seal water injection (See Section 6.5). The second, or "Phase B", containment isolation signal is derived upon actuation of the containment spray system, and trips the automatic isolation valves in the so-called "essential"* process lines penetrating the containment. These trip valves are designated by the letter "P" in the isolation diagrams.

A manual containment isolation signal can be generated from the control room. This signal performs the same functions as the automatically derived "T" signal, i.e. "Phase A" isolation and automatic seal water injection.

Non-automatic isolation valves, i.e., remote stop valves and manual valves, are used in lines which must remain in service, at least for a time, following an accident. These are closed manually if and when the lines are taken out of service.

---

*"Non-essential" process lines are defined as these which do not increase the potential for damage to in-containment equipment when isolated. "Essential" process lines are those providing cooling water and seal water flow through the reactor coolant pumps. These services should not be interrupted unless absolutely necessary while the reactor coolant pumps are operating.

Standard closing times available with commercial valve models are adequate for the sizes of containment isolation valves used. Valves equipped with air-diaphragm operators generally close in approximately two seconds. The typical closing time available for large motor operated gate valves is ten seconds.

The large butterfly valves used to isolate the containment ventilation
purge ducts are equipped with air-diaphragm operators, with spring returns 15
capable of closing the valves in two seconds. These valves fail to the
closed position on loss of control signal or instrument air.

5.2.2.3 Valve Operability

All containment isolation valves, actuators and controls are located so as to be protected against missiles which could be generated as the result of a loss of coolant accident. Only valves so protected are considered to qualify as containment isolation valves.

Only isolation valves located inside containment are subject to the high pressure, high temperature, steam laden atmosphere resulting from an ac ident. Operability of these valves in the accident environment is ensured by proper design, construction and installation, as reflected by the following considerations:

a. All components in the valve installation, including valve bodies, trim and moving parts, actuators, instrument air and control and power wiring, are constructed of materials sufficiently temperature resistant to be unaffected by the accident environment. Special attention is given to electrical insulation, air operator diaphragms and stem packing material.

b. In addition to normal pressures, the valves are designed to withstand maximum pressure differentials in the reverse direction imposed by the accident conditions. This criterion is particularly applicable to the butterfly type isolation valves used in the containment purge lines.

FAI – Fail As Is
FC – Fail Closed
FO – Fail Open
LC – Locked Closed
LO – Locked Open
BV – Butterfly Valve
DDV – Double Disc Gate Valve
Dia. – Diaphragm Valve
T – Containment Isolation Signal – Phase A
P – Containment Isolation Signal – Phase B
A – Automatic
M – Manual

RCS – Reactor Coolant System
ACS – Auxiliary Coolant System
WDS – Waste Disposal System
SIS – Safety Injection System
SS – Sampling System
CVCS – Chemical and Volume Control System
VENT – Ventilation System
SWS – Service Water System
FH – Fuel Handling
PPS – Penetration Pressurization System

TABLE 5.2-1

CONTAINMENT PIPING PENETRATIONS AND VALVING

(Page 1 of 10)

Penetration and System	Diagram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Sealing Method	Used After Accid.	Fluid Gas Or Water	Temp. Hot > 200°F Cold < 200°F	Notes
1. Pressurizer Relief Tank to Gas Analyzer – RCS	5.2-1	A	Globe	Air	Yes	Open	Open	Closed	FC	T	Water (A)	No	G	Cold	
		B	Globe	Air	Yes	Open	Open	Closed	FC	T	Water (A)				
2. Pressurizer Relief Tank N_2 Supply – RCS	5.2-1	A	Check	–	No	Closed	Closed	Closed	–	–	–	No	G	Cold	
		B	Dia.	Manual	No	Open	Open	Closed	As is	–					
		C	Pres. Reg.	Self-Contain.	No	Closed	Closed	Closed	FC	–					
3. Pressurizer Relief Tank Makeup – RCS	5.2-1	A	Check	–	No	Closed	Closed	Closed	–	–		No	W	Cold	
		B	Dia.	Air	Yes	Closed	Closed	Closed	FC	T	Water (A)				
		C	Dia.	Air	Yes	Closed	Closed	Closed	FC	T	Water (A)				
4. Residual Heat Removal Return – ACS/SIS	5.2-2	A	Check	–	No	Closed	Open	Open	–	–		Yes	W	Cold	
		B	DDV	Motor	Yes	Open	Open	Open	FAI	–	Nitrogen (N)				
5. Resid. Heat Removal Loop – To S.I. Pumps – ACS/SIS	5.2-2	A_1	DDV	Motor	Yes	Closed	Closed	Open	FAI	–	Nitrogen (N)	Yes	W	Cold	
		A_2	DDV	Motor	Yes	Closed	Closed	Open	FAI	–					
		C.S.	–	–	–	–	–	–	–	–	–	–	–	–	Safety Injection System
–To Sampling System– ACS/SS		A_3	Globe	Manual	No	Closed	Closed	Closed	As is	–	–	Yes	W	Cold	May be used for sampling during shutdown and after accident
		B_3	Globe	Air	Yes	Closed	Closed	Closed	FC	–					

TABLE 5.2-1 (Cont'd)

CONTAINMENT PIPING PENETRATIONS AND VALVING
(Page 2 of 10)

		Penetration and System	Diagram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Sealing Method	Used After Accid.	Fluid Gas Or Water	Temp. Hot >200°F Cold <200°F	Notes
	6.	Residual Heat Removal Loop Out - ACS	5.2-2	A	Gate	Motor	Yes	Closed	Open	Closed	FAI	-		No	W	Hot	
				B_1	DBV	Manual	No	L.C.	Open	Closed	As is	-	Water (A)				
				B_2	Globe	Motor	Yes	Closed	Open	Closed	FAI	-	Water (A)	No	W	Hot	
				C_2	Gate	Motor	Yes	Closed	Open	Closed	FAI	-	Water (A)				
	7.	Containment Sump Recirculation Line - ACS/SIS	5.2-2	A	Gate	Motor	Yes	Closed	Closed	Open	FAI	-	-	Yes	W	Cold	
15				B	Gate	Motor	Yes	Closed	Closed	Open	FAI	-					
	8.	Letdown Line - CVCS	5.2-3	A	Globe	Air	Yes	Open	Closed	Closed	FC	T	Water (A)	No	W	Hot	
				B	Globe	Air	Yes	Open	Closed	Closed	FC	T	Water (A)				
	9.	Charging Line - CVCS	5.2-3	A	Check	-	No	Open	Closed	Closed	-	-		No	W	Cold	
				B	Gate	Manual	No	Open	Closed	Closed	As is	-	Water (X)				
				C_1	Globe	Manual	No	Open	Closed	Closed	As is	-	Water (X)				
				C_2	Globe	Manual	No	Closed	Closed	Closed	As is	-					
	10.	Reactor Coolant Pump Seal Water Supply Lines - CVCS	5.2-4	A	Check	-	No	Open	Closed	Closed	-	-		Yes*	W	Cold	*Manual isolation when RC pumps are stopped
				B	Globe	Manual	No	Open	Closed	Closed	As is	-	Water (X)				
13				C	Needle	Manual	No	Open	Open	Closed	As is	-	Water (X)				

TABLE 5.2-1 (Cont'd)

CONTAINMENT PIPING PENETRATIONS AND VALVING
(Page 3 of 19)

	Penetration and System	Diagram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Sealing Method	Used After Accid.	Fluid Gas Or Water	Temp. Hot >200°F Cold <200°F	Notes
11.	Reactor Coolant	5.2-4	A	DDV	Motor	Yes	Open	Open	Closed	FAI	P	Water (A)	No	W	Cold	
	Pump Seal Water		B_1	Dia.	Manual	No	Open	Open	Closed	As is	-					
	Return - CVCS		B_2	Dia.	Manual	No	Closed	Open	Closed	As is	-					
12.	Reactor Coolant	5.2-5	A	Globe	Air	Yes	Closed	Closed	Closed	FC	T	Water (A)	No	W	Hot	
	Sample Line - SS		B	Globe	Air	Yes	Closed	Closed	Closed	FC	T					
13.	Fuel Transfer Tube - FH	5.2-5	A	Blind Flange	-	No	Closed	-	-	As is	-	-	No	W	Cold	Flange is double gasketed type, located in refueling canal (Missile Protected)
			B	Gate	Manual	No	Closed	Open	Closed	As is	-					
14.	Containment Spray	5.2-6	A	DDV	Manual	No	Open	Closed	Open	As is	-	Water (N)	Yes	W	Cold	
	Headers (2) - SIS		B_1	Check	-	No	Closed	Closed	Open	-	-					
			B_2	Globe	Manual	No	LC	Closed	Closed	As is	-					
15.	Safety Injection	5.2-7	A	Gate	Motor	Yes	Closed	Closed	Open	FAI	-		Yes	W	Cold	
	Headers (2) - SIS		B_1	DDV	Manual	No	LO	Open	Open	As is	-	Water (N)				
			B_2	DDV	Motor	No	Open	Closed	Open	FAI	-	Water (N)				
16.	Safety Injection	5.2-7	A	Globe	Manual	No	LC	Closed	Closed	As is	-	Water (A)	No	W	Cold	
	Test Line - SIS		B	Globe	Manual	No	LC	Closed	Closed	As is	-	Water (A)				

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TABLE 5.2-1 (Cont'd)

CONTAINMENT PIPING PENETRATIONS AND VALVING
(Page 4 of 10)

	Penetration and System	Diagram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Sealing Method	Used After Accid.	Fluid Gas Or Water	Temp. Hot >200°F Cold <200°F	Notes
17.	Accumulator N_2 Supply - SIS	5.2-8	A	Globe	Air	Yes	Open	Open	Closed	FC	-	-	No	G	Cold	
			B_1	Press.-Reg.	Self-Contain.	No	Open	Closed	Closed	As is	-					
			B_2	Globe	Manual	No	Closed	Closed	Closed	As is	-	-	No	G	Cold	
18.	Accumulator Sample - SS	5.2-8	A	Globe	Air	Yes	Open	Closed	Closed	FC	T	Water (A)	No	W	Cold	Valves A & B opened intermittently to take sample
			B	Globe	Air	Yes	Open	Closed	Closed	FC	T	Water (A)				
19.	Primary System Vent Header And N_2 Supply Line - WDS	5.2-9	A_1	Dia	Air	Yes	Open	Closed	Closed	FC	T	Water (A)	No	G	Cold	
			B_1	Dia	Air	Yes	Open	Closed	Closed	FC	T	Water (A)				
			A_2	Dia	Manual	No	Open	Open	Open	As is	-	-	No	G	Cold	
			B_2	Check	-	No	Closed	Closed	Closed	-	-					
			C_2	Press.-Reg.	Self-Contain.	No	Closed	Closed	Closed	FC	-					
20.	Reactor Coolant Drain Tank To Gas Analyzer - WDS	5.2-9	A	Dia	Air	Yes	Open	Open	Closed	FC	T	Water (A)	No	G	Cold	Valve B is opened periodically by the gas analyzer
			B	Dia	Air	Yes	Open	Closed	Closed	FC	T					
21.	RCDT Pump Discharge - WDS	5.2-9	A	Dia	Air	Yes	Open	Open	Closed	FC	T	Water (A)	No	W	Cold	
			B	Dia	Air	Yes	Open	Open	Closed	FC	T					
22.	Reactor Coolant Pump Cooling Water In - ACS	5.2-10	A	Check	-	No	Open	Closed	Closed	-	-	Water (A)	-	W	Cold	
			B	DOV	Motor	Yes	Open	Closed	Closed	FAI	P					
			C	Gate	Motor	Yes	Open	Closed	Closed	FAI	P					

TABLE 5.2-1 (Cont'd)

CONTAINMENT PIPING PENETRATIONS AND VALVING
(Page 3 of 10)

	Penetration and System	Diagram	Valve No or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Sealing Method	Used After Accid.	Fluid Gas Or Water	Temp. Hot >250°F Cold <250°F	Notes
23.	Reactor Coolant Pump Water Out (6") - ACS	5.2-10	A B	MOV Gate	Motor Motor	Yes Yes	Open Open	Closed Closed	Closed Closed	FAI FAI	P P	Water (A)		W	Cold	
24.	Reactor Coolant Pump Cooling Water Out (3") - ACS	5.2-10	A B	MOV Gate	Motor Motor	Yes Yes	Open Open	Closed Closed	Closed Closed	FAI FAI	P P	Water (A)		W	Cold	
25.	Resid. Heat Exch. Cooling Water In - ACS	5.2-11	C.S. C.S.	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	Residual heat exchanger is a missile protected closed system Component Cooling System
26.	Resid. Heat Exch. Cooling Water Return - ACS	5.2-11	A C.S.	Gate -	Motor -	Yes -	Closed -	Open -	Open -	FAI -	- -	- -	Yes -	W -	Cold -	Component Cooling System
27.	Recirc. Pump Cooling Water Supply - ACS	5.2-12	A C.S.	Gate -	Manual -	No -	Open -	Open -	Closed -	As is -	- -	- -	No -	W -	Cold -	Component Cooling System
28.	Recirc. Pump Cooling Water Return - ACS	5.2-12	A B C.S.	Gate Gate -	Manual Manual -	No No -	Open Open -	Open Open -	Closed Closed -	As is As is -	- - -	- -	No -	W -	Cold -	Component Cooling System

TABLE 5.2-1 (Cont'd)

CONTAINMENT PIPING PENETRATIONS AND VALVING
(Page 8 of 10)

	No.	Penetration and System	Diagram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Sealing Method	Used After Accid.	Fluid Gas Or Water	Temp. Hot >200°F Cold <200°F	Notes
	29.	Excess Letdown Heat Exchanger Cooling Water In - ACS	5.2-13	A B	Dia. Dia.	Air Air	Yes Yes	Closed Closed	Open Open	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Cold	
	30.	Excess Letdown Heat Exchanger Cooling Water Out - ACS	5.2-13	A B	Globe Dia.	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Cold	
	31.	Containment Sump Pump Discharge - WDS	5.2-13	A B	Dia. Dia.	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Cold	
	32.	Containment Air Sample In - Rad. Mon.	5.2-14	A B	Dia. Dia.	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Air (A)	No*	G	Cold	*May be opened for air sampling following accident when the containment pressure is below 5 psig
	33.	Containment Air Sample Out - Rad. Mon.	5.2-14	A B	Dia. Dia.	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Air (A)	No*	G	Cold	*May be opened for air sampling following accident when the containment pressure is below 5 psig
15	34.	Air Ejector Discharge To Containment - Sec. Sys.	5.2-14	A B	Globe Globe	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	Air (A)	No	G	Cold	

TABLE 5.2-1 (Cont'd)

CONTAINMENT PIPING PENETRATIONS AND VALVING
(Page 7 of 10)

	Penetration and System	Diagram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Sealing Method	Used After Accid.	Fluid Gas Or Water	Temp. Hot >200°F Cold <200°F	Notes
35.	Main Steam Headers	5.2-15	A_1	Globe	Manual	No	Open	Open	Open	As is	-	-	Yes*	G	Hot	
			A_2	Relief	-	No	Closed	Closed	Closed	As is	-					*Automatic isolation for steam line break only
			A_3	Globe	Manual	No	Open	Open	Open	As is	-					
			A_4	Swing Disc	Air Piston	Yes	Open	Closed	Open*	FC	-					
			C.S.	-	-	-	-	-	-	-	-	-		-	-	Steam Generator
35a.	(To Aux. F.W. Pump Turbine)		A_5	Stop-Check	-	No	Open	Closed	Open	As is	-	-	Yes	G	Hot	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Steam Generator
36.	Main Feedwater Headers - Sec. Sys.	5.2-15	A	Gate	Manual	No	Open	Closed	Closed	As is	-	-	No	W	Hot	
			B	Check	-	No	Open	Closed	Closed	-						
36a.	Auxiliary Feedwater Turbine Driven - Sec Sys.	5.2-15	A	Gate	Manual	No	Open	Closed	Open	As is	-	-	Yes	W	Hot	
			B	Check	-	No	Open	Closed	Open	-						
36b.	Auxiliary Feedwater Motor Driven - Sec. Sys.	5.2-15	A	Gate	Manual	No	Open	Closed	Open	As is	-	-	Yes	W	Hot	
			B	Check	-	No	Open	Closed	Open	-						
37.	Steam Generator Blowdown - Sec. Sys.	5.2-15	A	Globe	Air	Yes	Closed	Closed	Closed	FC	T	-	No	W	Hot	
			B	Globe	Air	Yes	Closed	Closed	Closed	FC	T					

TABLE 5.2-1 (Cont'd)

CONTAINMENT PIPING PENETRATIONS AND VALVING
(Page 4 of 10)

	Penetration and System	Diagram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Sealing Method	Used After Accid.	Fluid Gas Or Water	Temp. Hot >200°F Cold <200°F	Notes
38.	S.G. Blowdown Sample – Sec. Sys.	5.2-15	A B	Globe Globe	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	–	No	W	Hot	
39.	Ventilation System Cooling Water In – SWS	5.2-16	A C.S.	Gate –	Manual –	No –	Open –	Open –	Open –	As is –	– –	– –	Yes –	W –	Cold –	Fan Cooler Units
40.	Ventilation System Cooling Water Out – SWS	5.2-16	A C.S.	Gate –	Manual –	No –	Open –	Open –	Open –	As is –	– –	– –	Yes –	W –	Cold –	Fan Cooler Units
41.	Service Air (Sec. Sys.)	5.2-17	A B	Globe Globe	Manual Manual	No No	Closed Closed	Closed Closed	Closed Closed	As is As is	– –	Water (A)	No	G	Cold	
42.	Instrument Air (Sec. Sys.)	5.2-17	A B	Check Press.-Reg.	– Air	No Yes	open Open	Open Open	Closed Closed	– Closed	– T	–	No	G	Cold	
43.	Weld Channel Pressurization Air Supply (PPS)	5.2-17	A C.S.	Dia. –	Manual –	No –	Open	Open	Open	As is –	– –	– –	Yes –	G –	Cold –	Penetration Pressurization System

TABLE 5.2-2 (Cont'd)

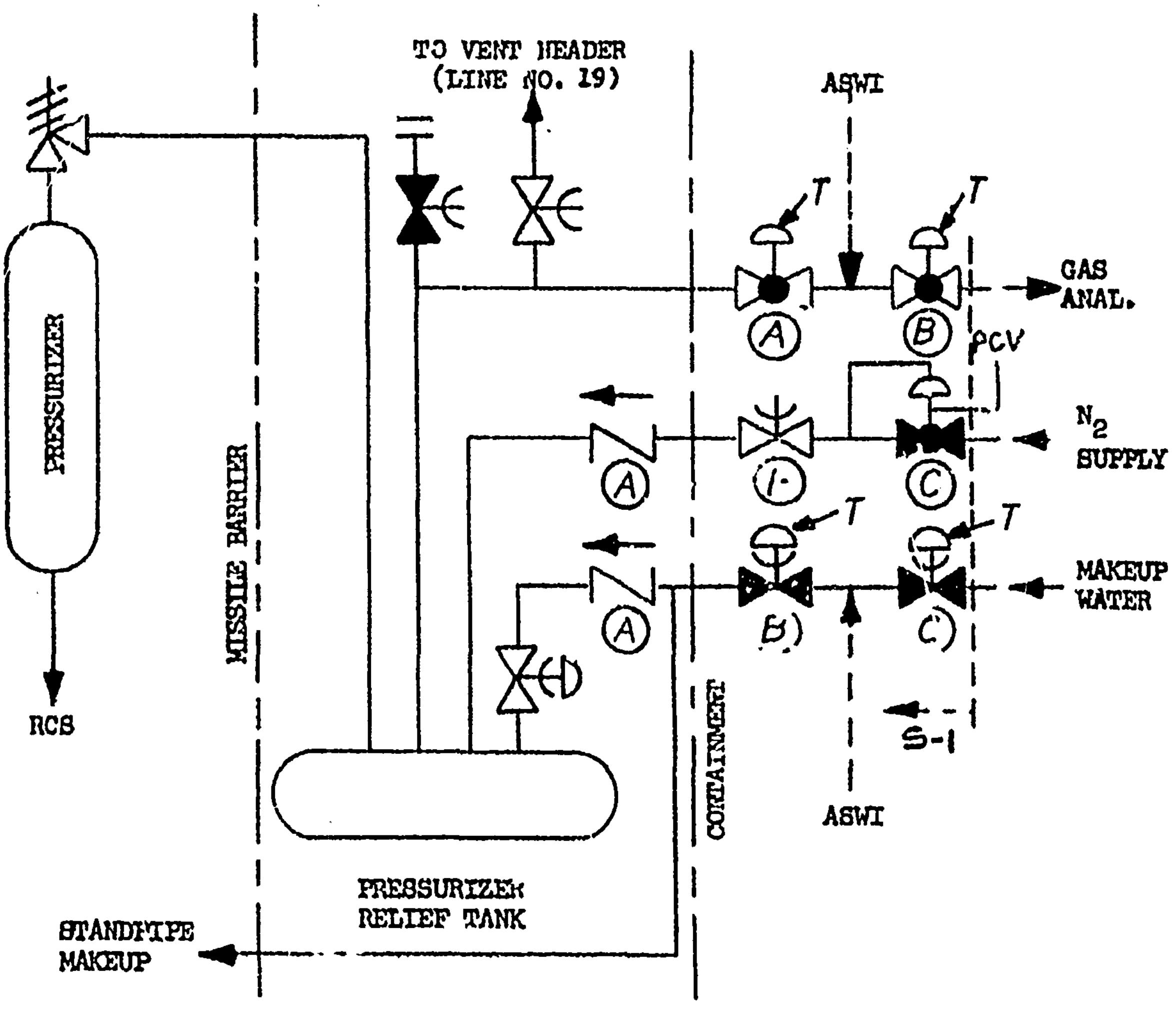
CONTAINMENT PIPING PENETRATIONS AND VALVING
(Page 9 of 10)

	Penetration and System	Diagram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Sealing Method	Used After Accid.	Fluid Gas Or Water	Temp. Hot >200°F Cold <200°F	Notes
44.	Dead Weight Tester (Misc.)	5.2-17	A B	Needle Needle	Manual Manual	No No	Closed Closed	Closed Closed	Closed Closed	As is As is	-	-	No	G	Cold	
45.	Auxiliary Steam Supply - Sec. Sys.	5.2-18	A C.S.	SDV -	Manual -	No -	LC -	Closed* -	Closed -	As is -	-	Water (A) -	No -	G -	Hot -	*May be opened during shutdown for cont. heating Auxil. Steam System
46.	Auxiliary Steam Condensate Return - Sec. Sys.	5.2-18	A C.S.	DDV -	Manual -	No -	LC -	Closed* -	Closed -	As is -	- -	Water (A) -	No -	W -	Cold -	*May be opened during shutdown for containment heatin Auxiliary Steam System
47.	City Water - Sec. Sys.	5.2-18	A B	Dia. Dia.	Manual Manual	No No	Closed Closed	Closed Closed	Closed Closed	As is As is	- -	Water (A)	No	W	Cold	
48.	Purge Supply Duct - Vent. Sys.	5.2-19	A B	BV BV	Air Air	Yes Yes	Closed Closed	Open Open	Closed Closed	FC FC	T* T*	Air (A)	No	G	Cold	*Also tripped closed by high radiation signal
49.	Purge Exhaust Duct - Vent. Sys.	5.2-19	A B	BV BV	Air Air	Yes Yes	Closed Closed	Open Open	Closed Closed	FC FC	T* T*	Air (A)	No	G	Cold	*Also tripped closed by high radiation signal
50.	Containment Pressure Relief - Vent. Sys.	5.2-19	A B C	BV BV BV	Air Air Air	Yes Yes Yes	Closed Closed Closed	Closed Closed Closed	Closed Closed Closed	FC FC FC	T* T* T*	Air (A)	No	G	Cold	Opened intermittently for pressure relief *Also tripped closed by high radiation signal

LINE NO. 1 PRESSURIZER RELIEF TANK TO GAS ANALYZER

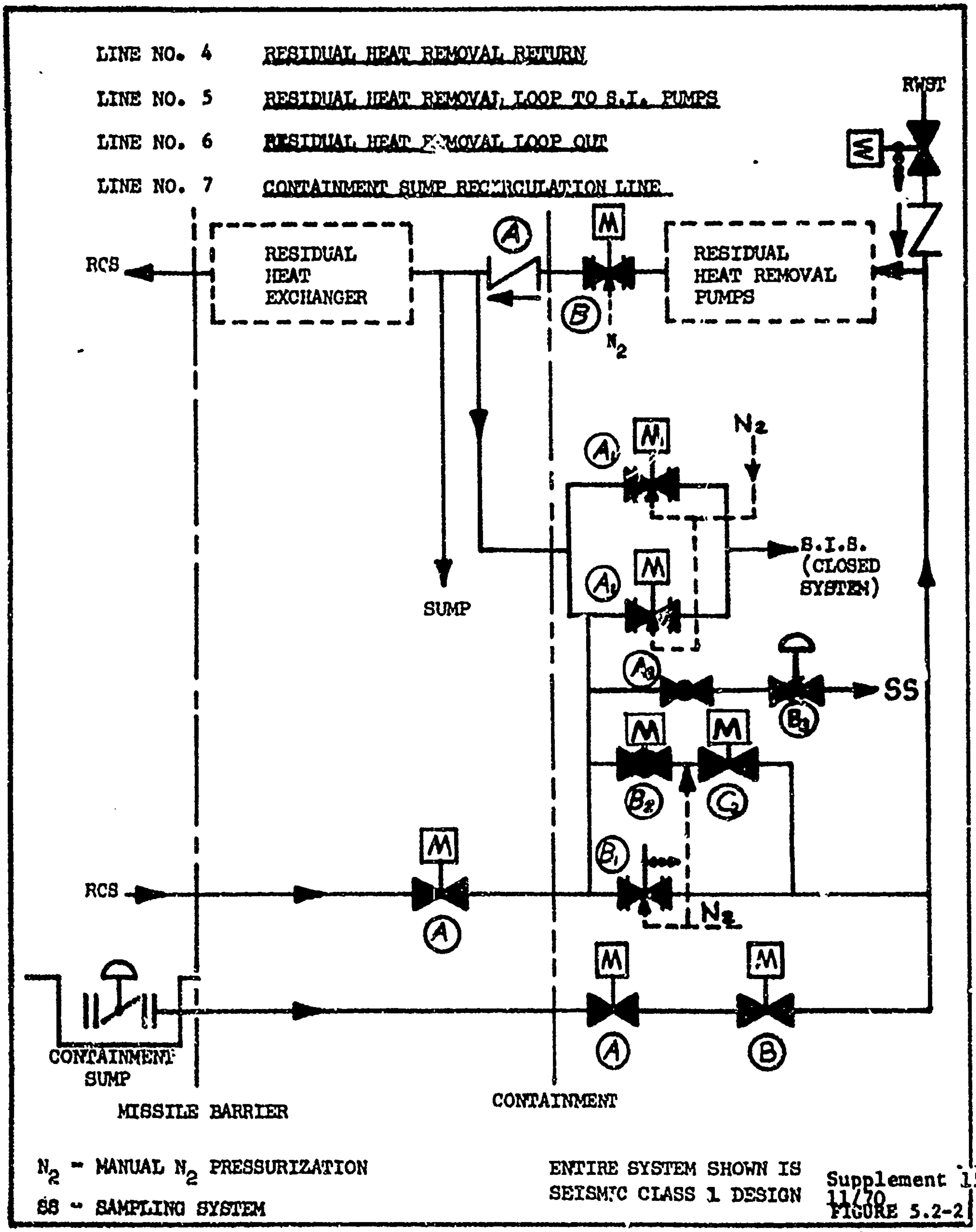
LINE NO. 2 PRESSURIZER RELIEF TANK N_2 SUPPLY

LINE NO. 3 PRESSURIZER RELIEF TANK MAKEUP



ALTHOUGH THE PRESSURIZER RELIEF TANK IS MISSILE PROTECTED, THESE PENETRATING LINES CAN BECOME EXPOSED TO CONTAINMENT ATMOSPHERE IF THE PRESSURIZER DISCHARGE HEADER IS BREACHED DURING THE ACCIDENT.

Supplement 13
8/70
Figure 5.2-1



Supplement 15
11/70
FIGURE 5.2-2

15

13

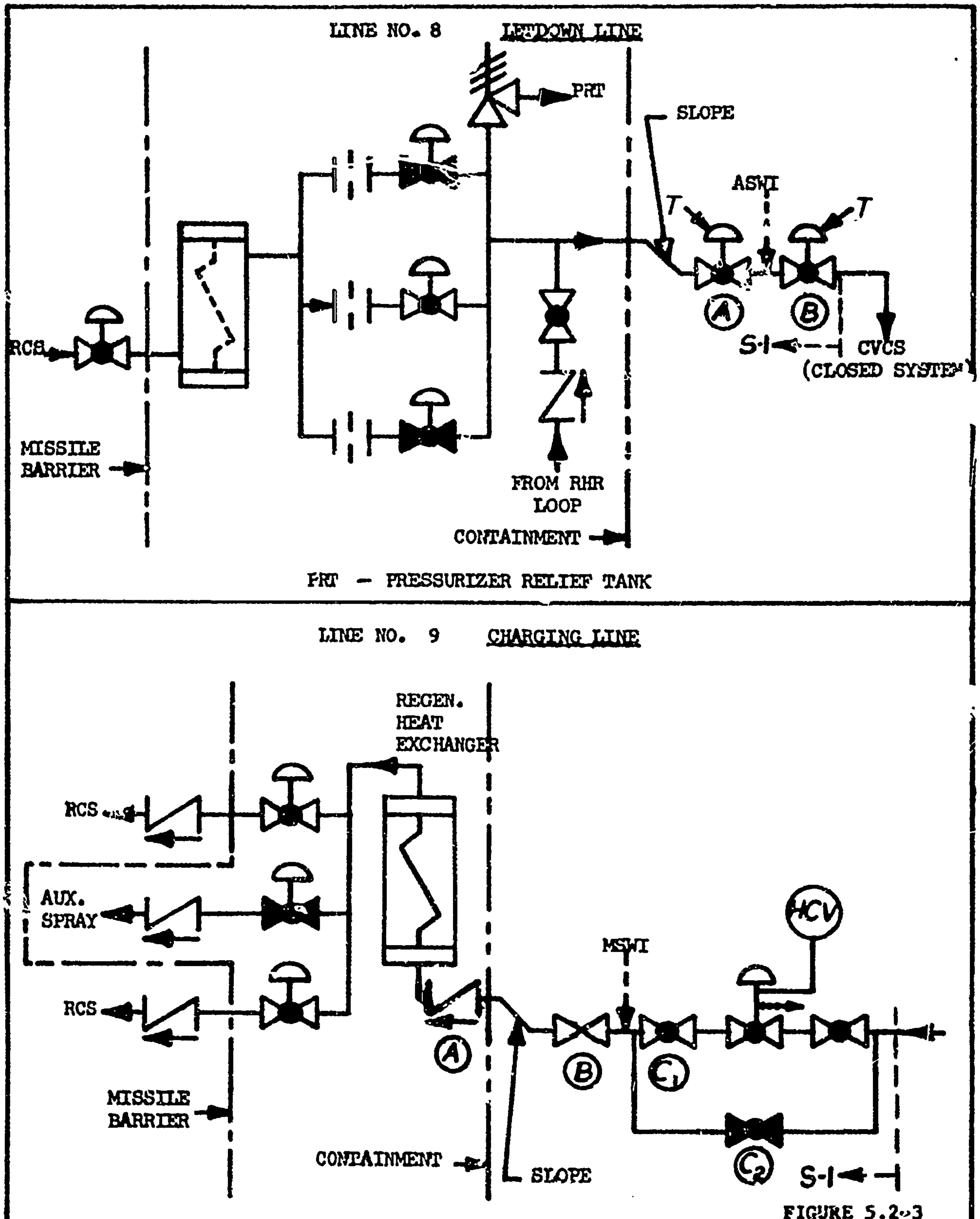


FIGURE 5.2-3

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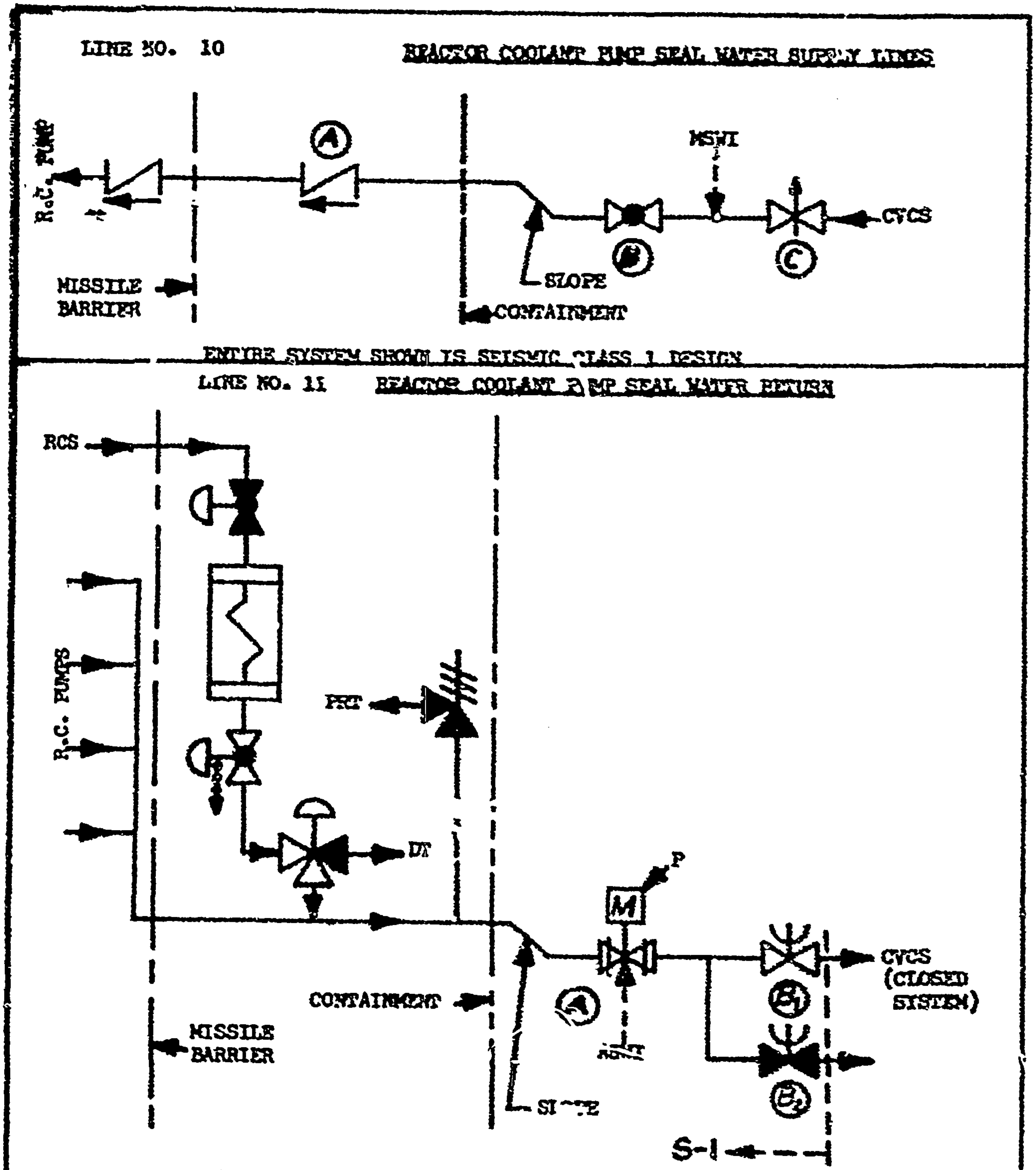

DT - REACTOR COOLANT DRAIN TANK

P - TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL PHASE B

PRT - PRESSURIZER RELIEF TANK

FIGURE 5.2-4
Supplement 13
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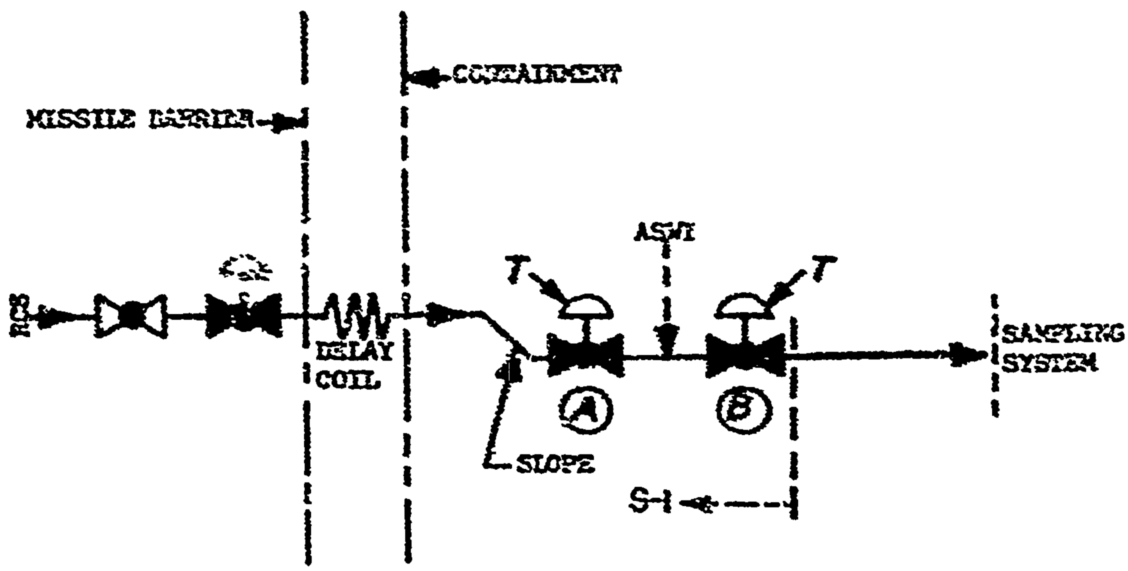
LINE NO. 12 REACTOR COOLANT SYSTEM SAMPLE LINES



LINE NO. 13 FUEL TRANSFER TUBE

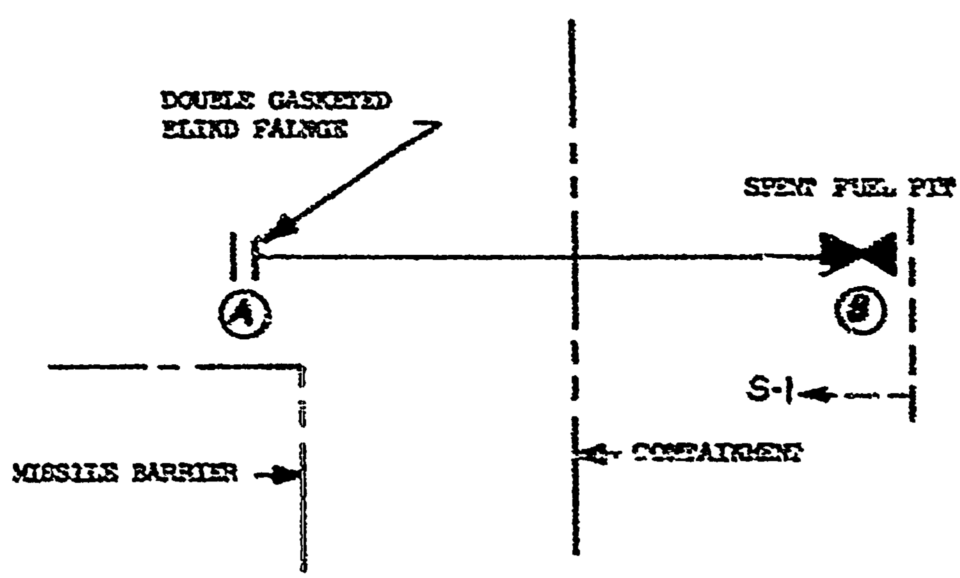


FIGURE 5.2-5
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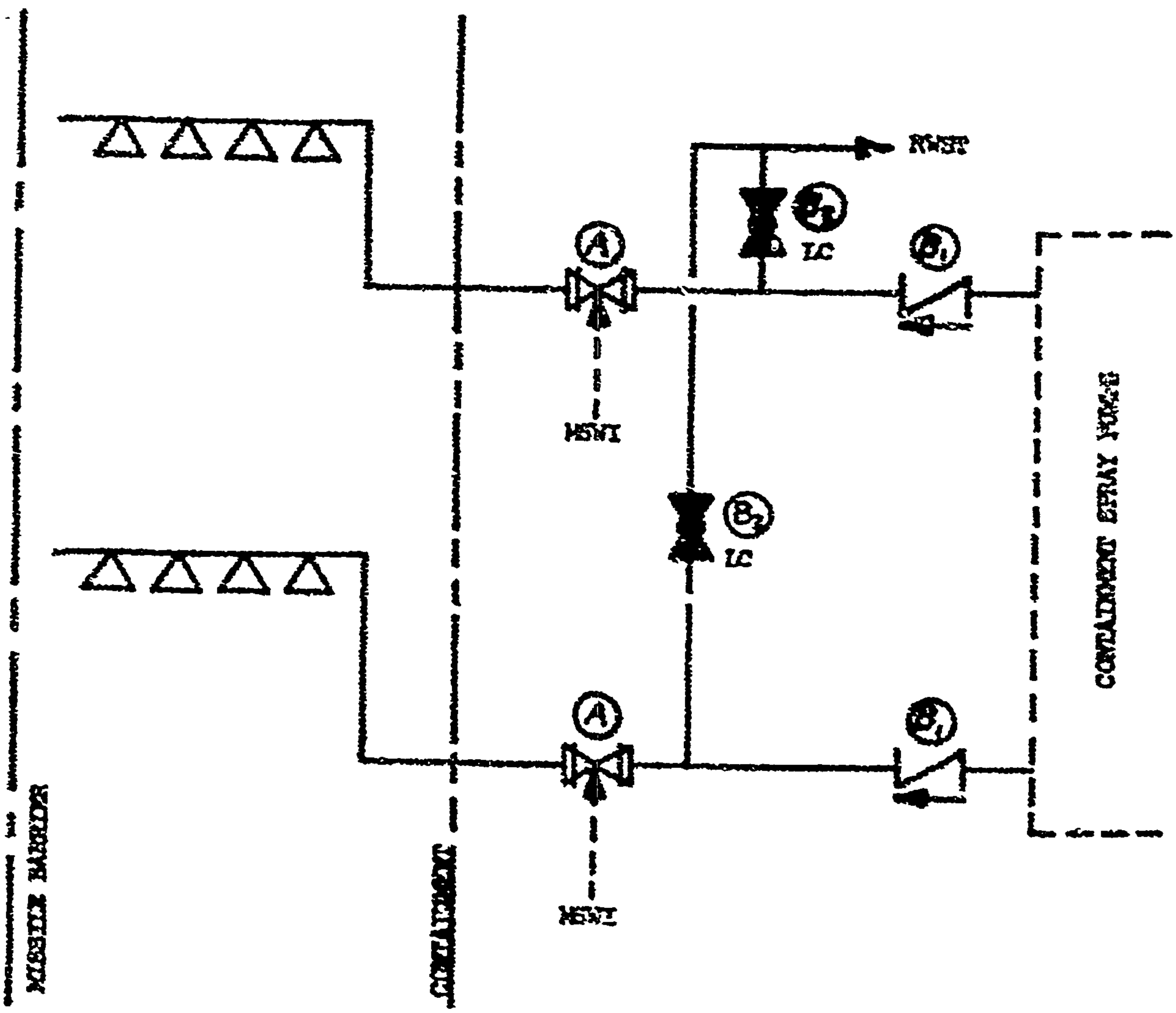
LINE NO. 14 CONTAINMENT SPRAY HEADERS



RWST - REFUELING WATER STORAGE TANK

ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

FIGURE 5.2-6
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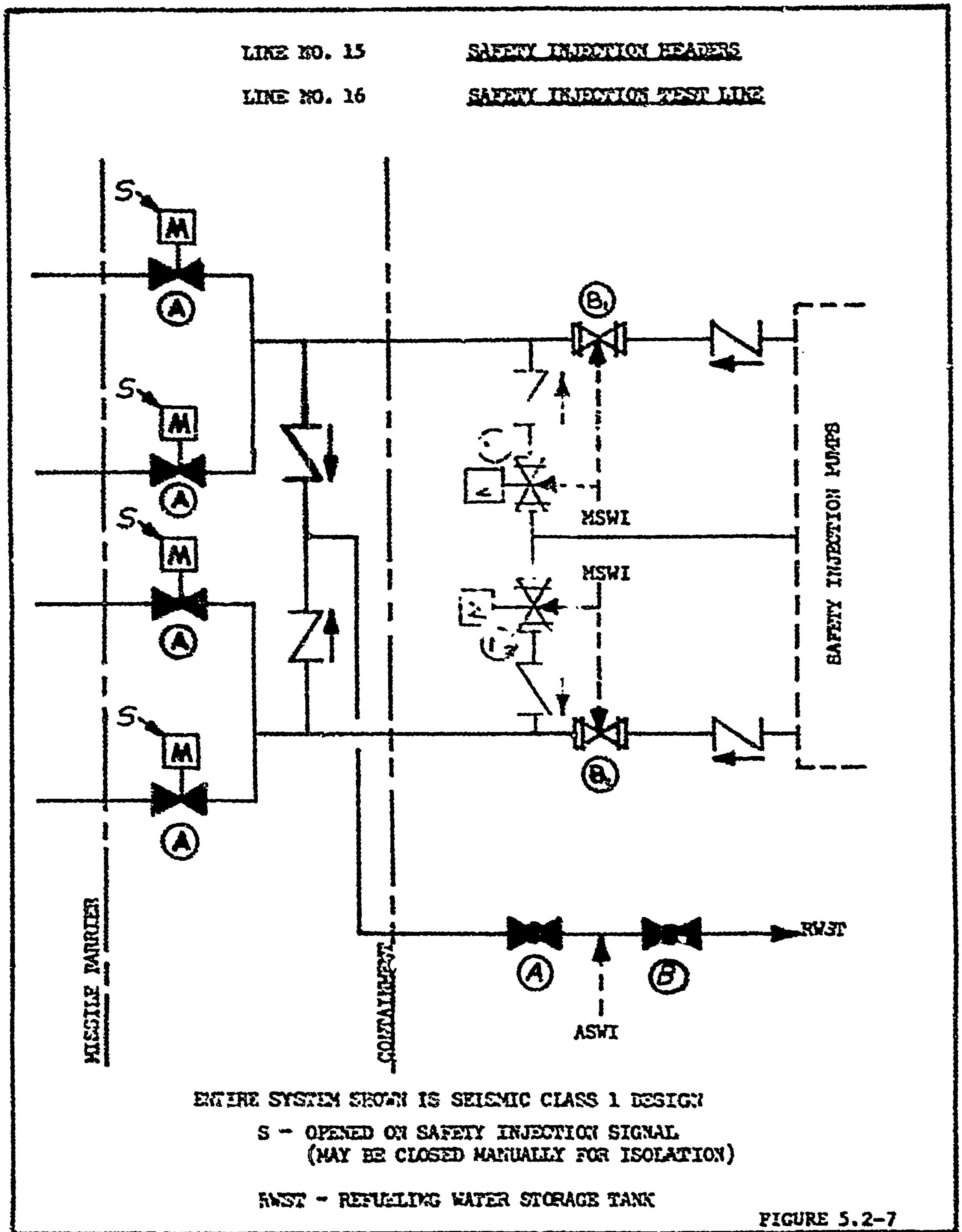

FIGURE 5.2-7
Supplement 13
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LINE 17 ACCUMULATOR N_2 SUPPLY

LINE 18 ACCUMULATOR SAMPLE

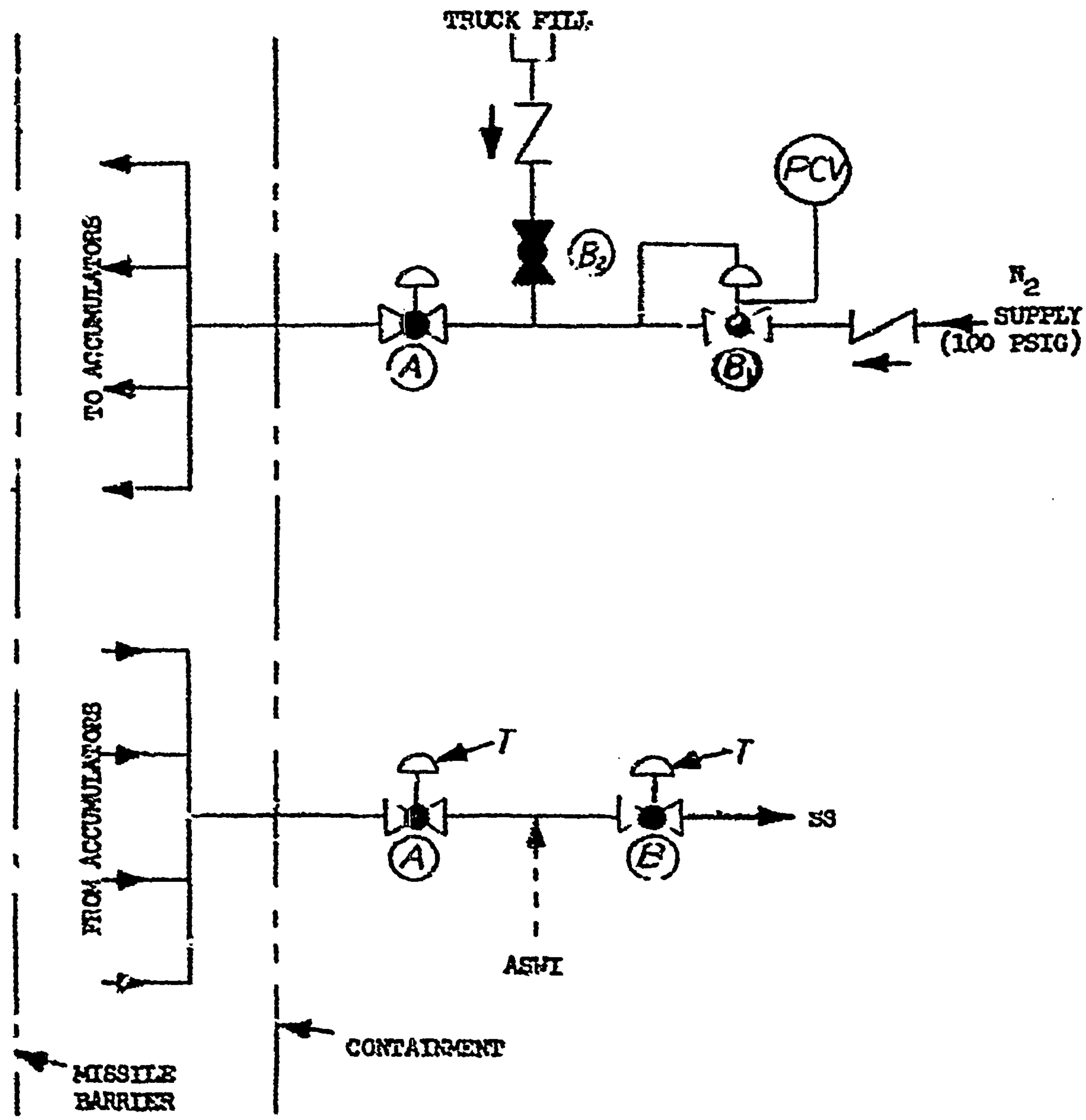

FIGURE 5.2-8

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LINE NO. 19 PRIMARY SYSTEM VENT HEADER AND N_2 SUPPLY LINE

LINE NO. 20 REACTOR COOLANT DRAIN TANK TO GAS ANALYZER

LINE NO. 21 RCDT PUMP DISCHARGE

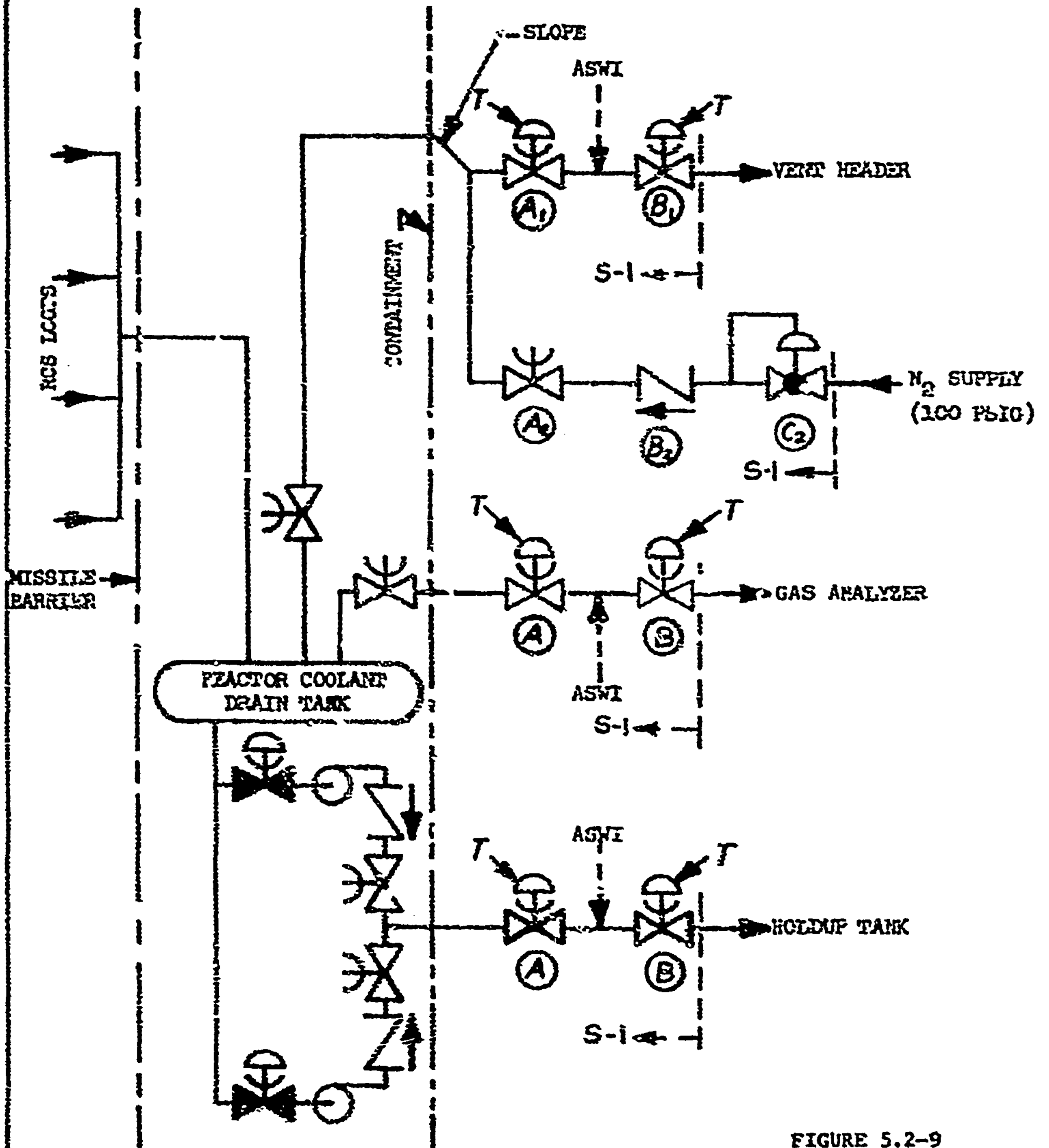

FIGURE 5.2-9

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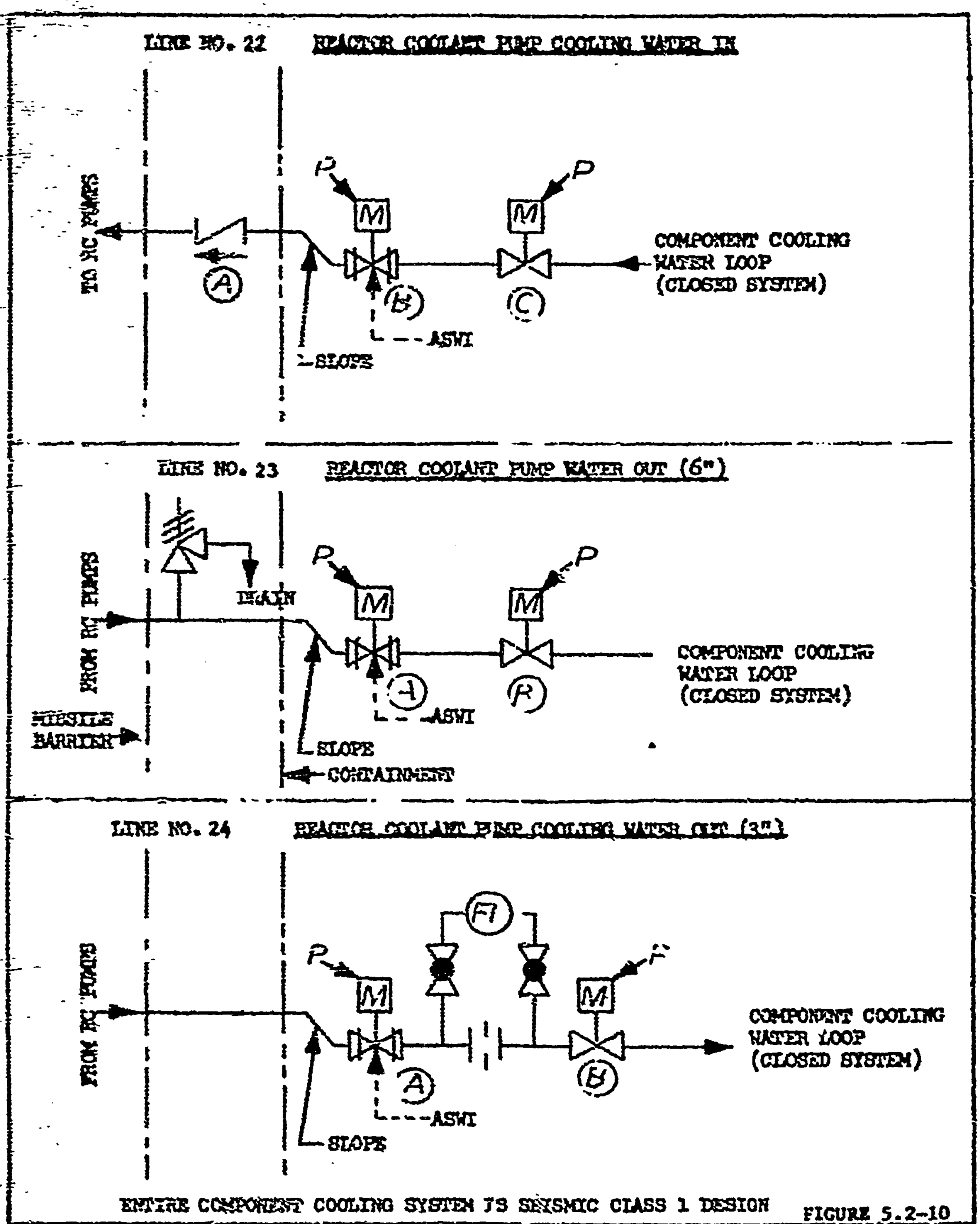

FIGURE 5.2-10

Supplement 13
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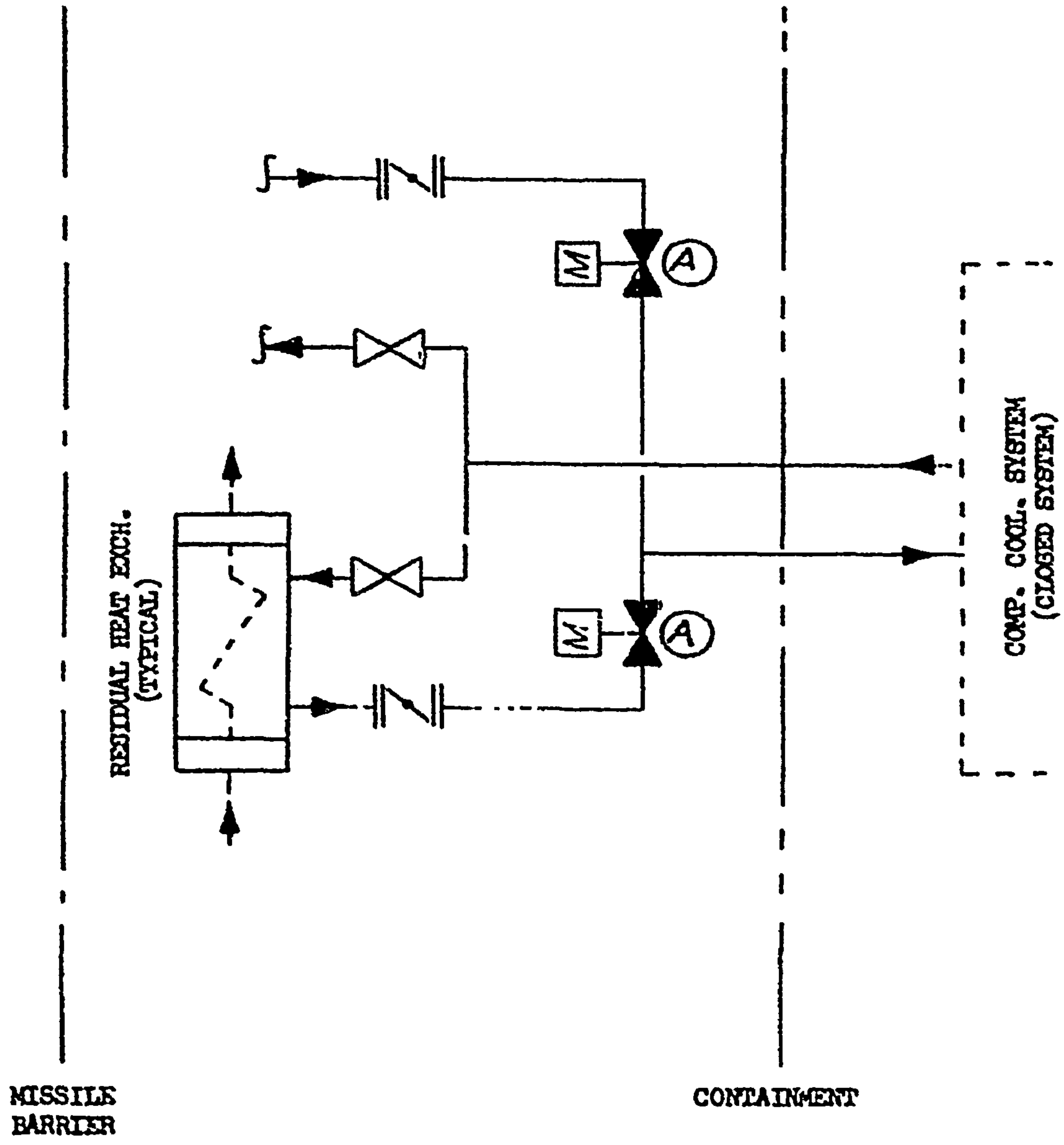
LINE NO. 25 RESIDUAL HEAT EXCHANGER COOLING WATER IN

LINE NO. 26 RESIDUAL HEAT EXCHANGER COOLING WATER RETURN



ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

FIGURE 5.2-11

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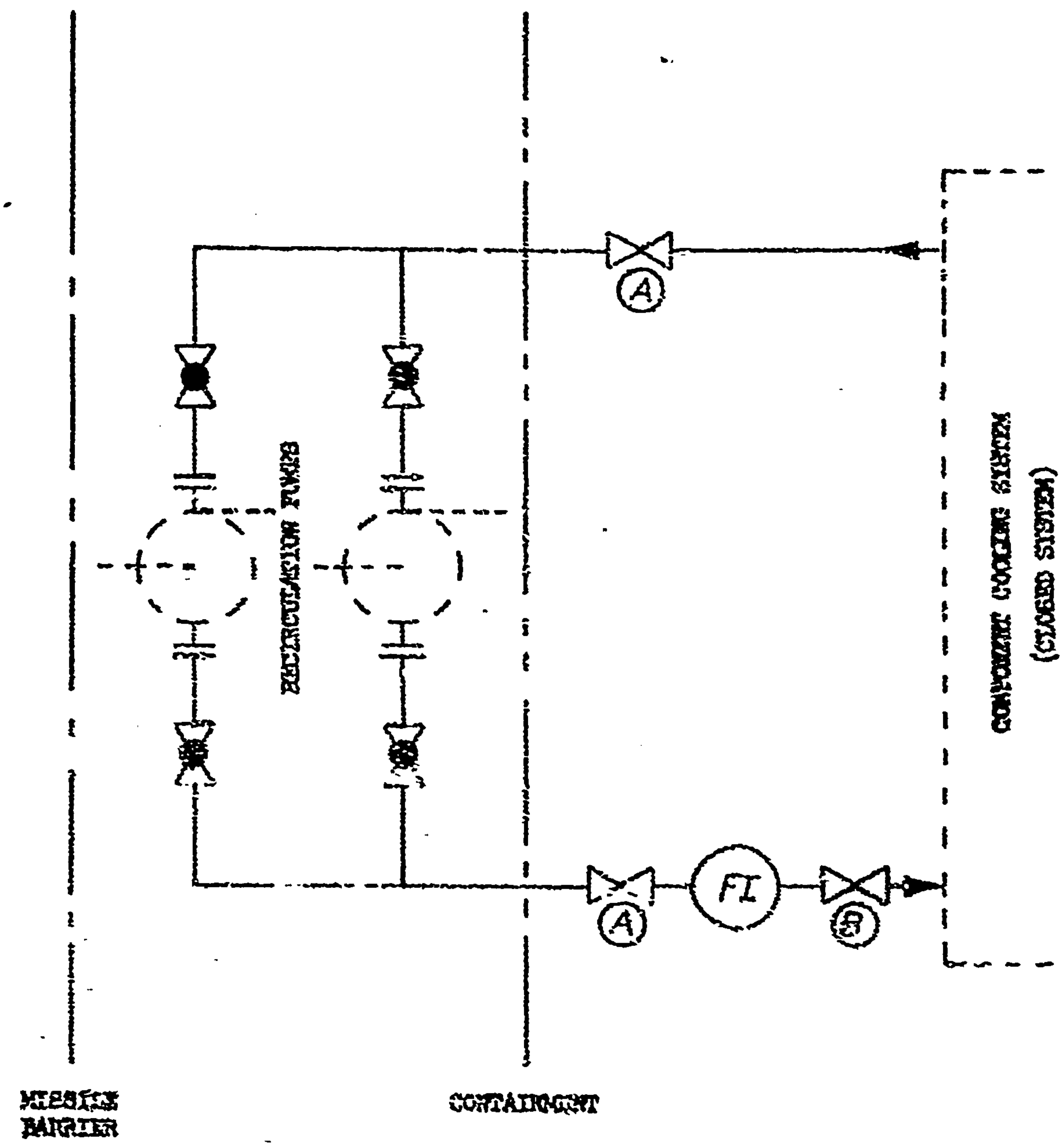
LINE NO. 27 RECIRCULATION PUMP COOLING WATER SUPPLY

LINE NO. 28 RECIRCULATION PUMP COOLING WATER RETURN



ENTIRE COMPONENT COOLING SYSTEM IS SEISMIC CLASS 1 DESIGN

FIGURE 5.2-12
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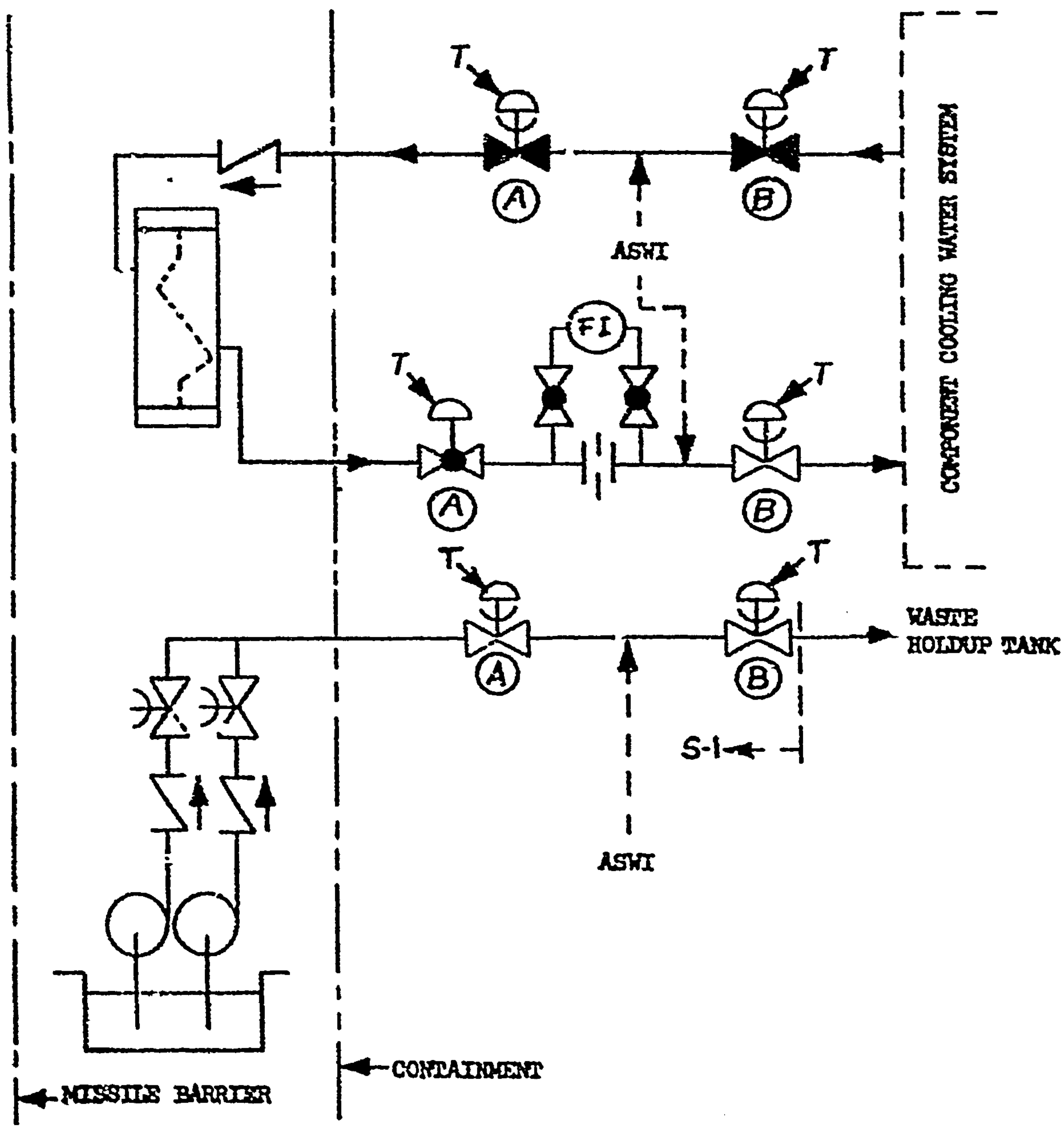
LINE NO. 29 EXCESS LETDOWN HEAT EXCHANGER COOLING WATER IN

LINE NO. 30 EXCESS LETDOWN HEAT EXCHANGER COOLING WATER OUT

LINE NO. 31 CONTAINMENT SUMP PUMP DISCHARGE



ENTIRE COMPONENT COOLING SYSTEM IS SEISMIC CLASS 1 DESIGN FIGURE 5.2-13

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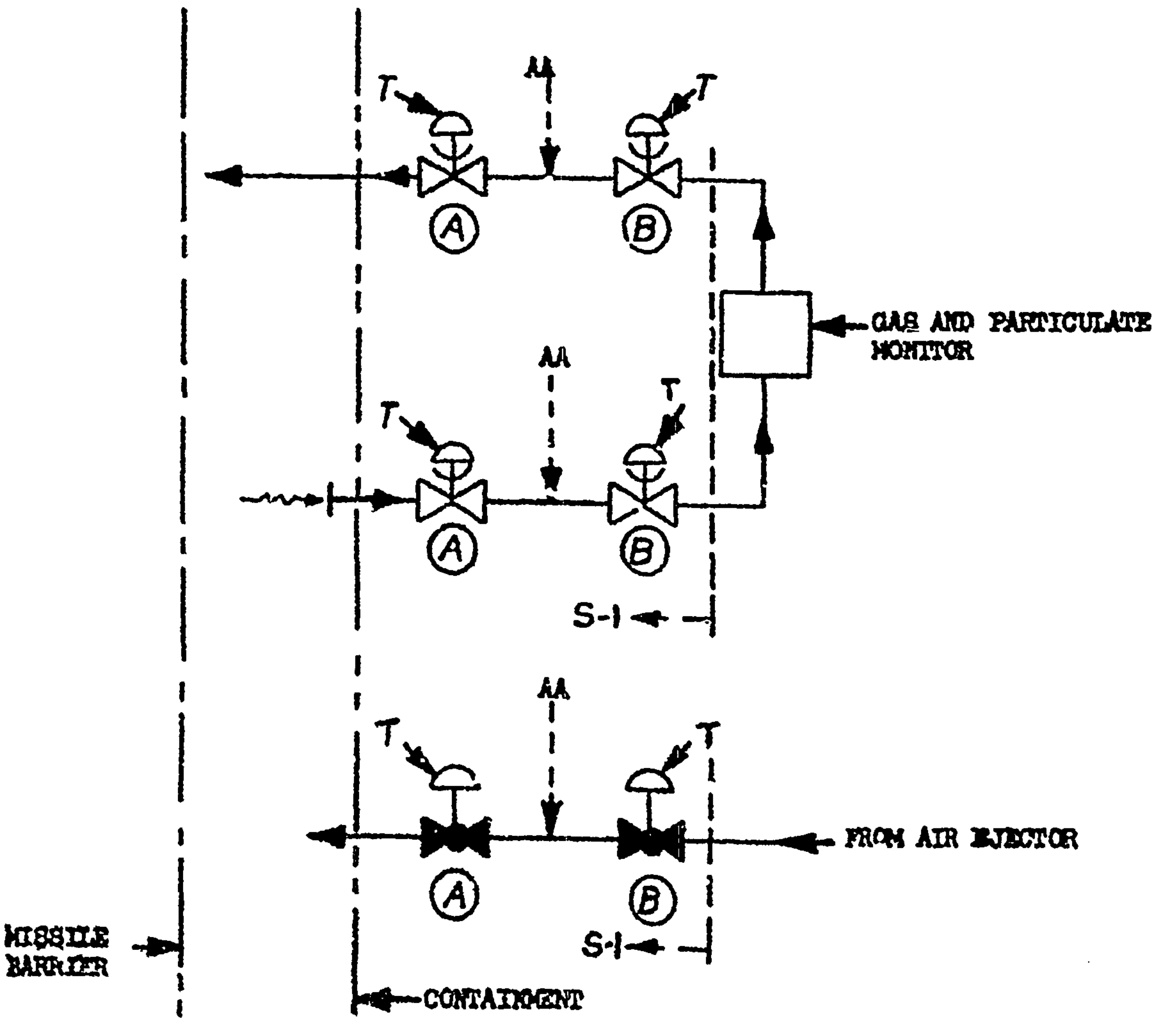
LINE NO. 32 CONTAINMENT AIR SAMPLE IN

LINE NO. 33 CONTAINMENT AIR SAMPLE OUT

LINE NO. 34 AIR EJECTOR DISCHARGE TO CONTAINMENT



AA - AUTOMATIC PRESSURIZATION WITH AIR FROM PENETRATION PRESSURIZATION SYSTEM

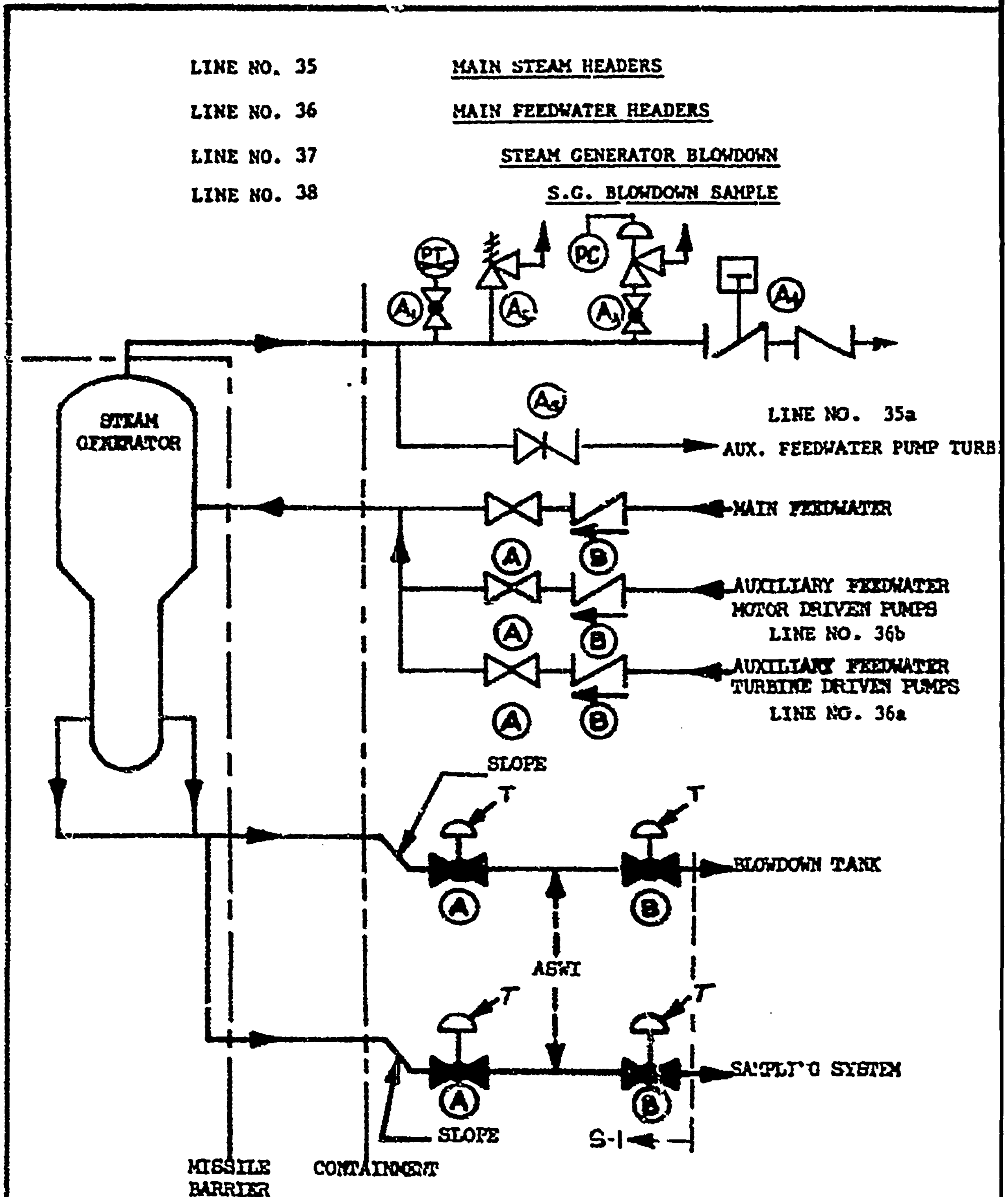


FIGURE 5.2-15

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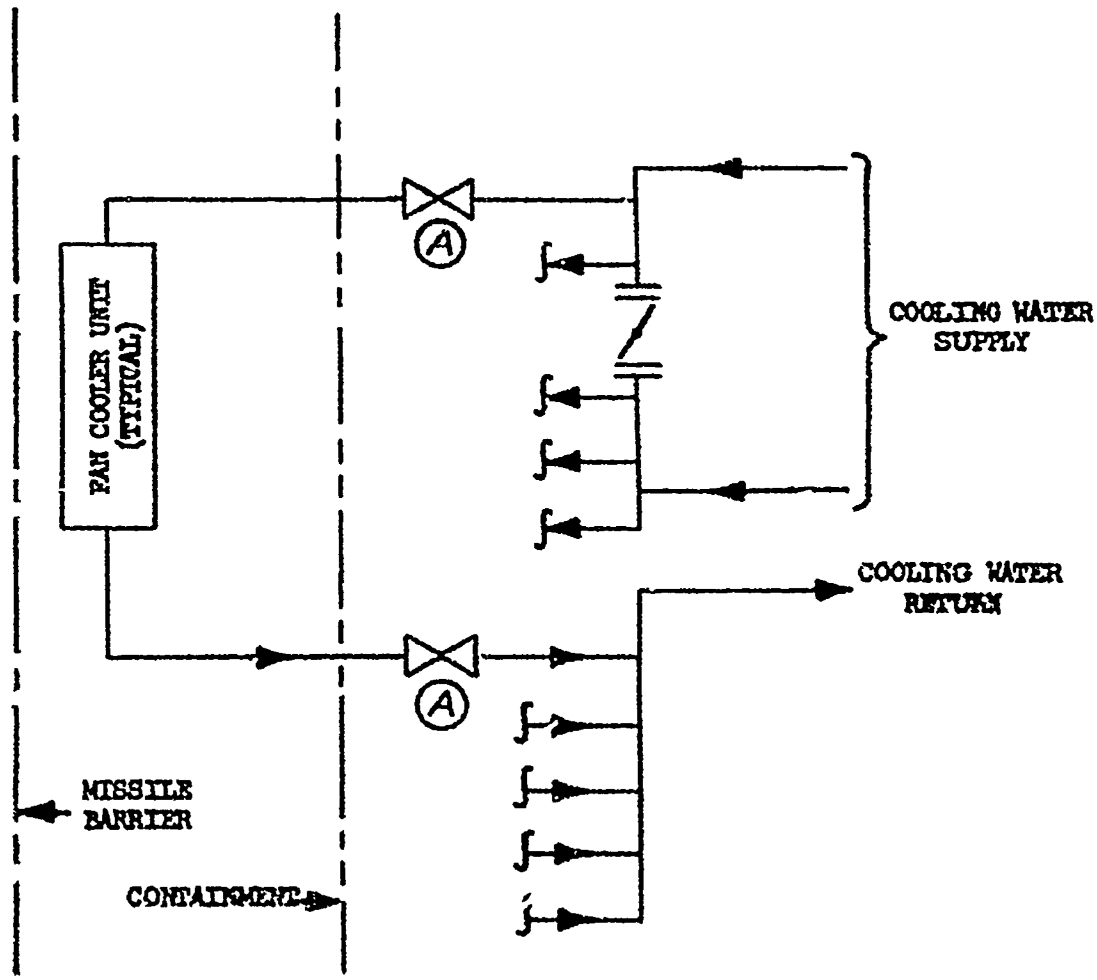
LINE NO. 39 VENTILATION SYSTEM COOLING WATER IN

LINE NO. 40 VENTILATION SYSTEM COOLING WATER OUT



ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

FIGURE 5.2-16

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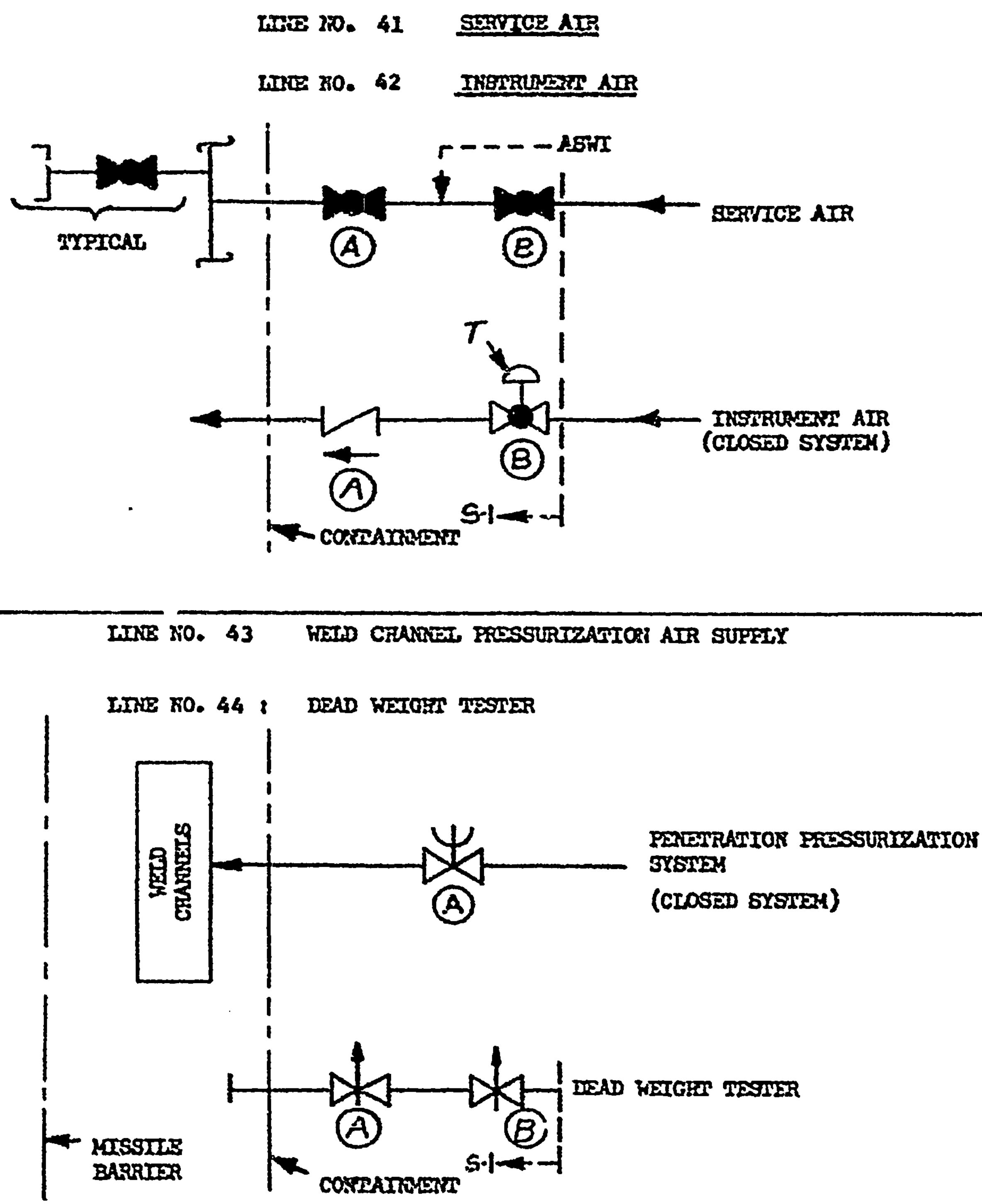
LINE NO. 41 SERVICE AIR
LINE NO. 42 INSTRUMENT AIR
ASWI
TYPICAL
SERVICE AIR
A
B
T
INSTRUMENT AIR
(CLOSED SYSTEM)
A
B
S
CONTAINMENT
LINE NO. 43 WELD CHANNEL PRESSURIZATION AIR SUPPLY
LINE NO. 44 : DEAD WEIGHT TESTER
WELD CHANNELS
PENETRATION PRESSURIZATION SYSTEM
(CLOSED SYSTEM)
A
DEAD WEIGHT TESTER
A
B
S
MISSILE BARRIER
CONTAINMENT

FIGURE 5.2-17

LINE NO. 45 AUXILIARY STEAM SUPPLY

LINE NO. 46 AUXILIARY STEAM CONDENSATE RETURN

LINE NO. 47 CITY WATER

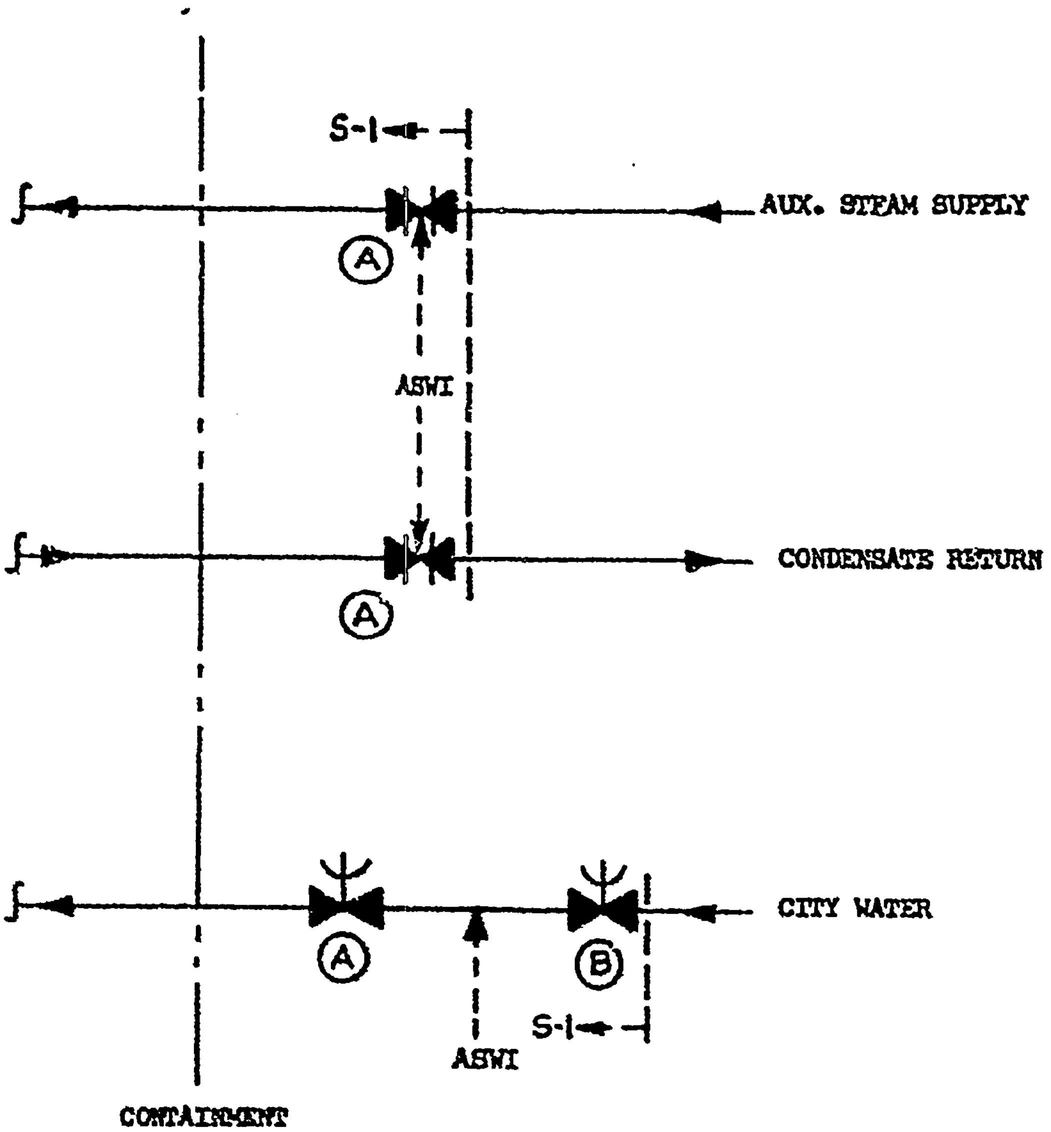

FIGURE 5.2-18

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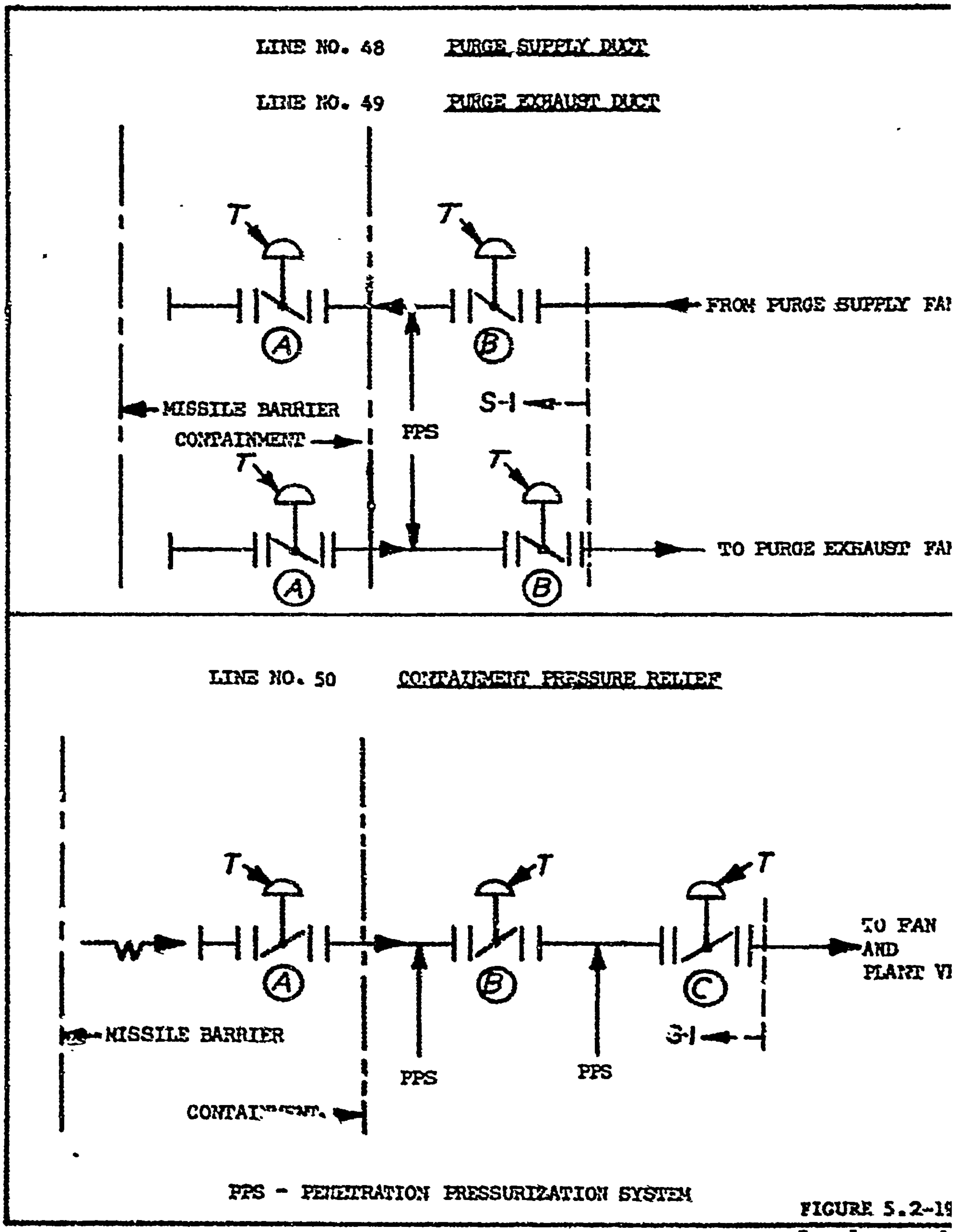
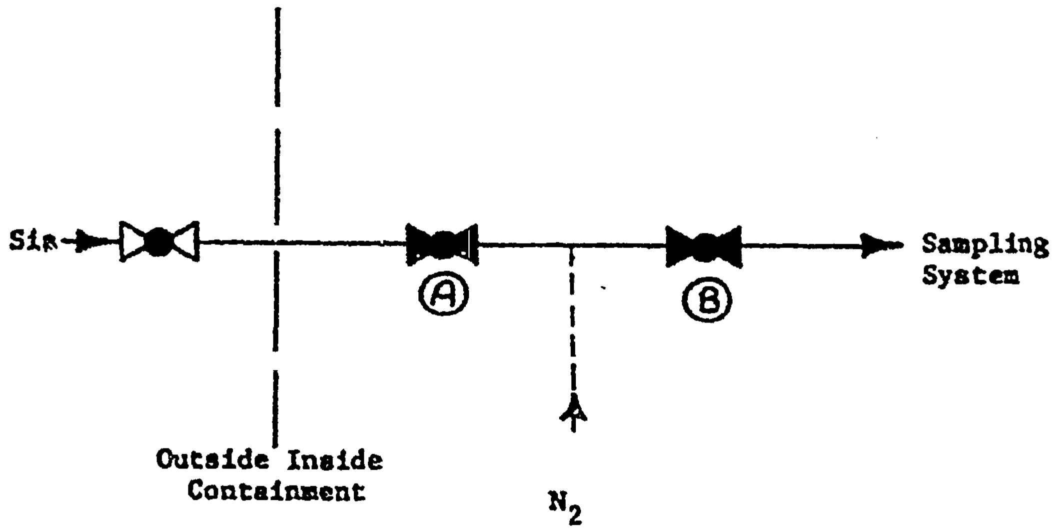

LINE NO. 48 PURGE SUPPLY DUCT
LINE NO. 49 PURGE EXHAUST DUCT
T
A
B
FROM PURGE SUPPLY FA
MISSILE BARRIER
CONTAINMENT
PPS
S-1
TO PURGE EXHAUST FA
LINE NO. 50 CONTAINMENT PRESSURE RELIEF
W
C
TO FAN AND PLANT V
PPS - PENETRATION PRESSURIZATION SYSTEM

FIGURE 5.2-1

Line No. 51 <u>RECIRCULATION PUMP DISCHARGE SAMPLE LINE</u>

15



Line No. 52 <u>PRESSURIZER STEAM SPACE SAMPLE</u>
Line No. 53 <u>PRESSURIZER LIQUID SPACE SAMPLE</u>

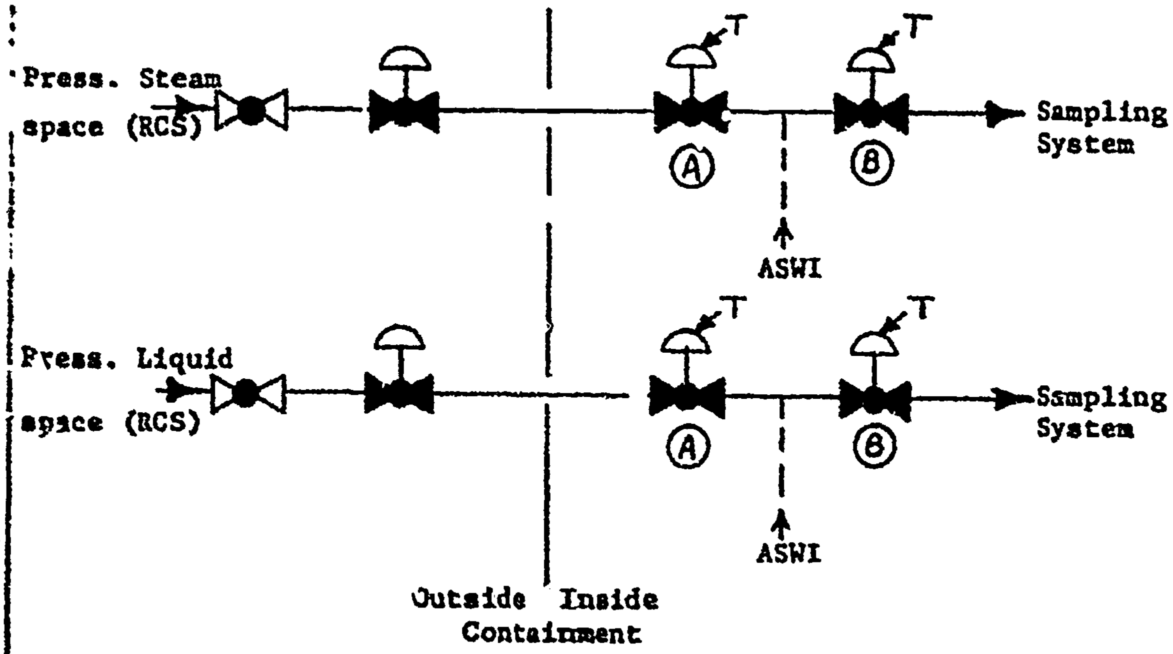


Figure 5.2-20

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Line Nos. 54, 55, and 56.

CONTAINMENT PRESSURE INSTRUMENTATION LINES

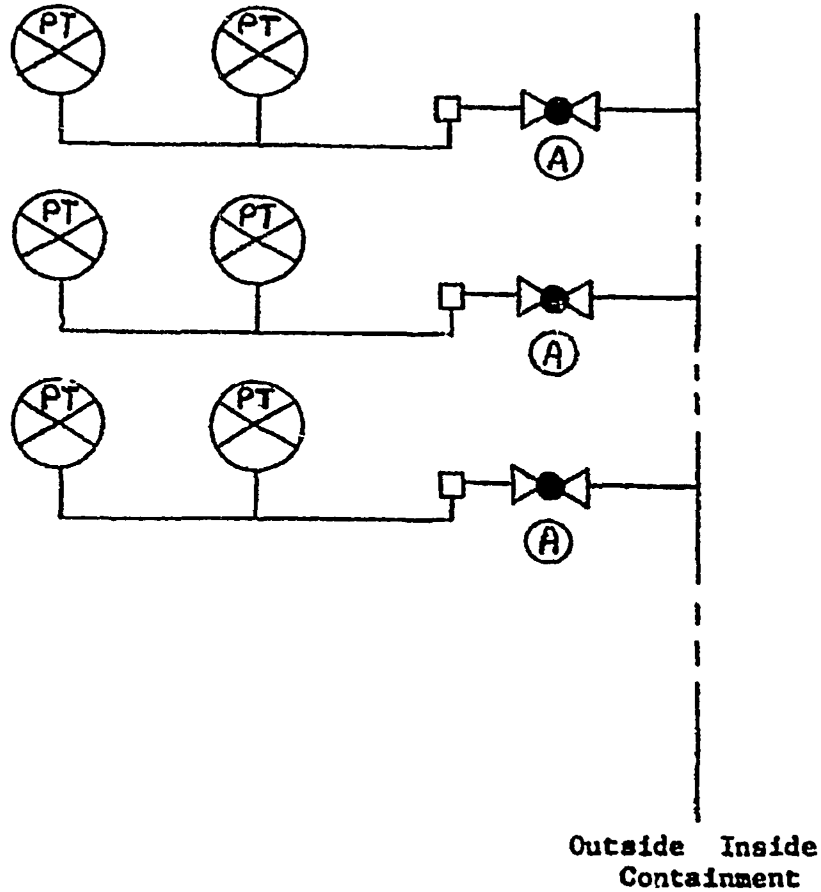

Figure 5.2-21
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Line No. 57 Post Accident Containment Sampling Lines (Supply and Return)

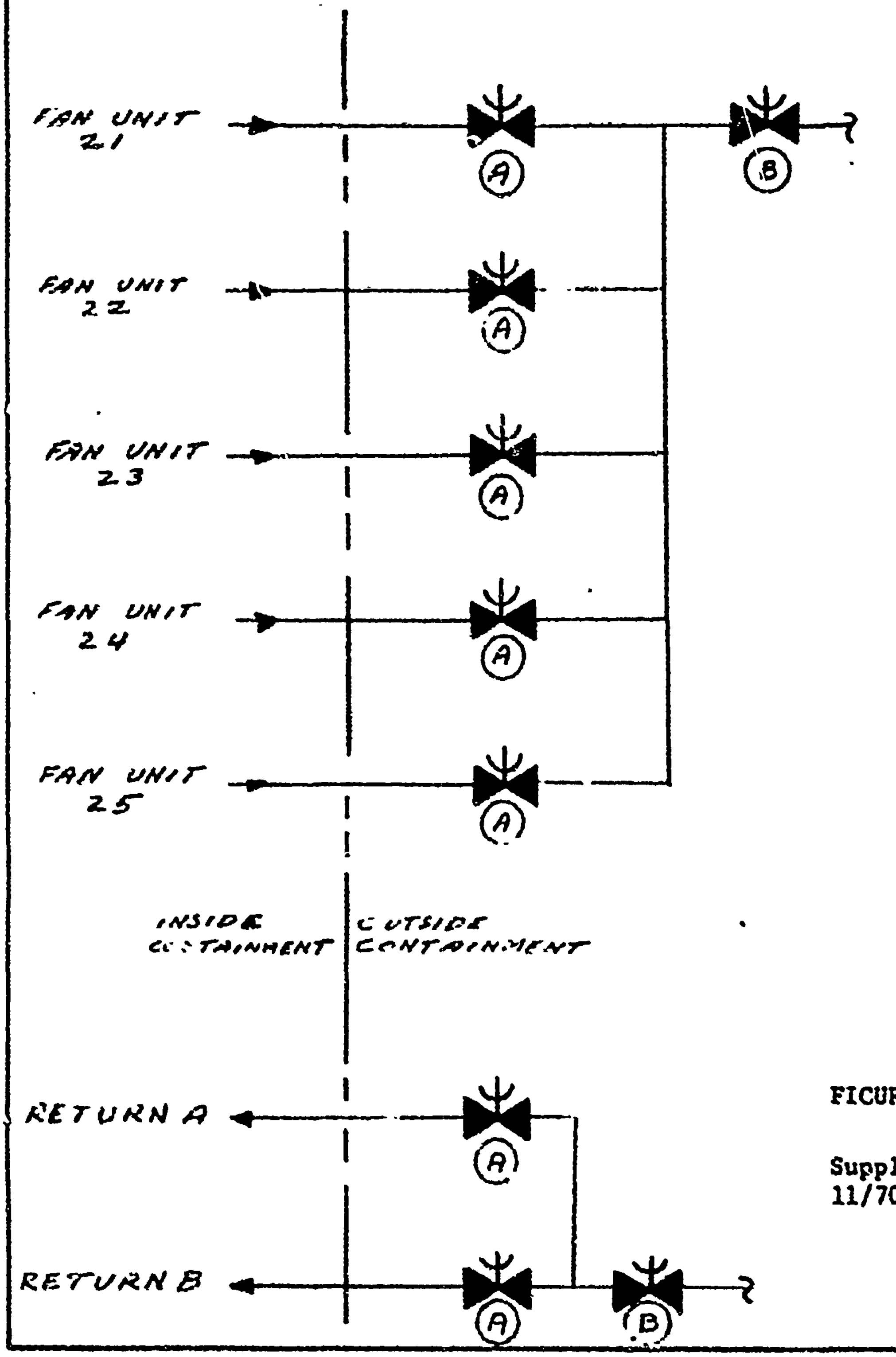

FIGURE 5.2-22

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Line No. 58 Oxygen Supply To Containment

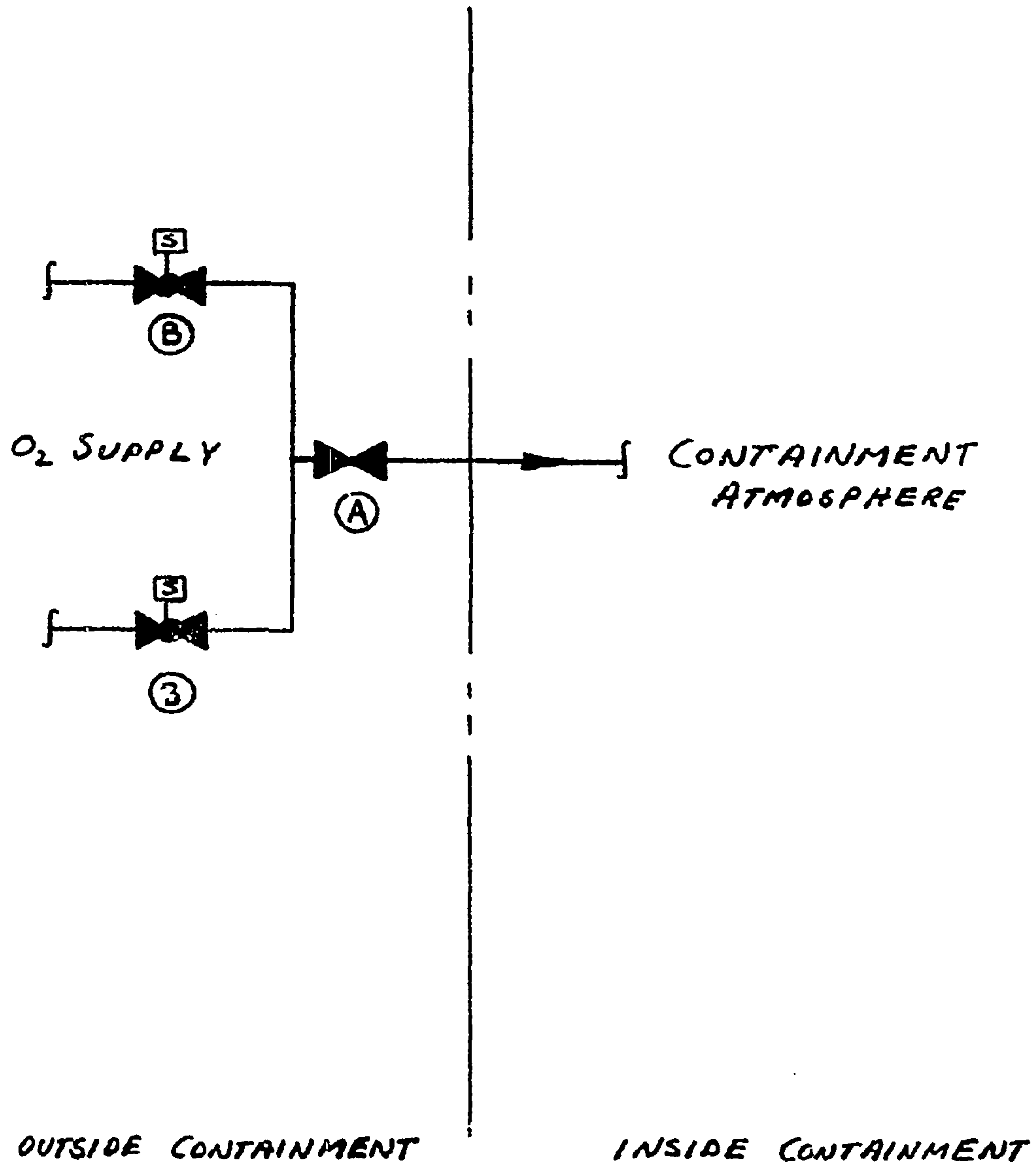

FIGURE 5.2-23

Line No. 59 H_2 Supply To H_2 Recombiner

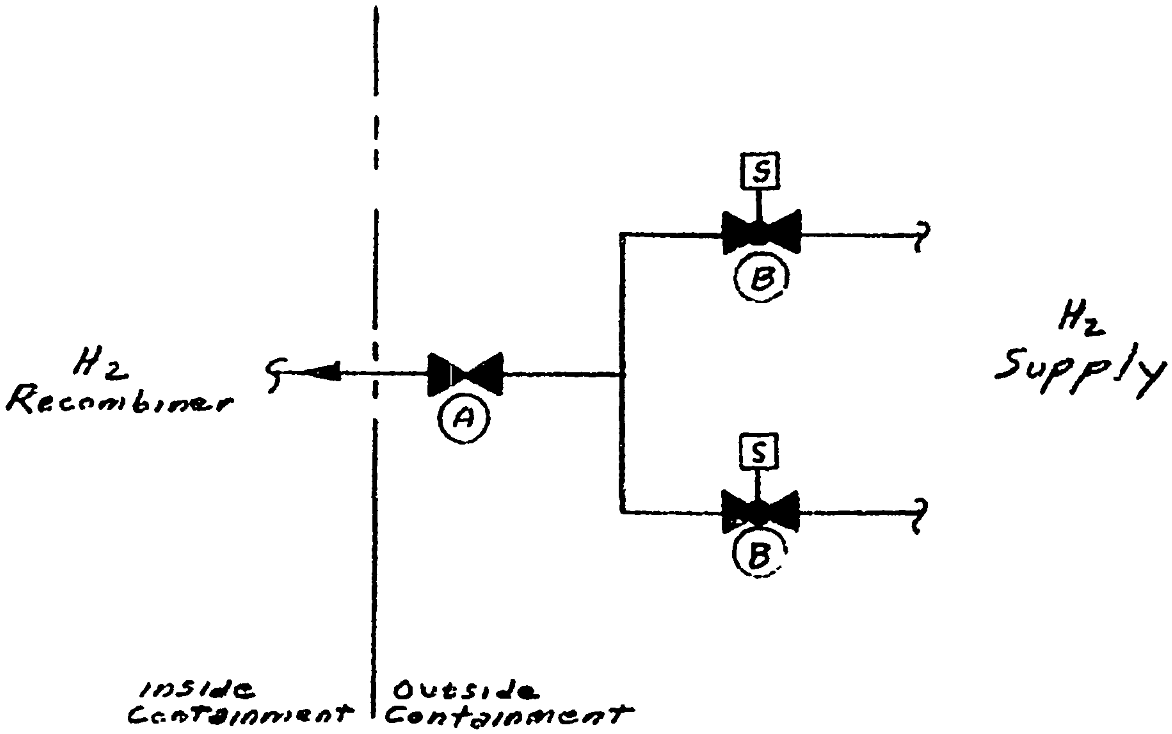

FIGURE 5.2-24

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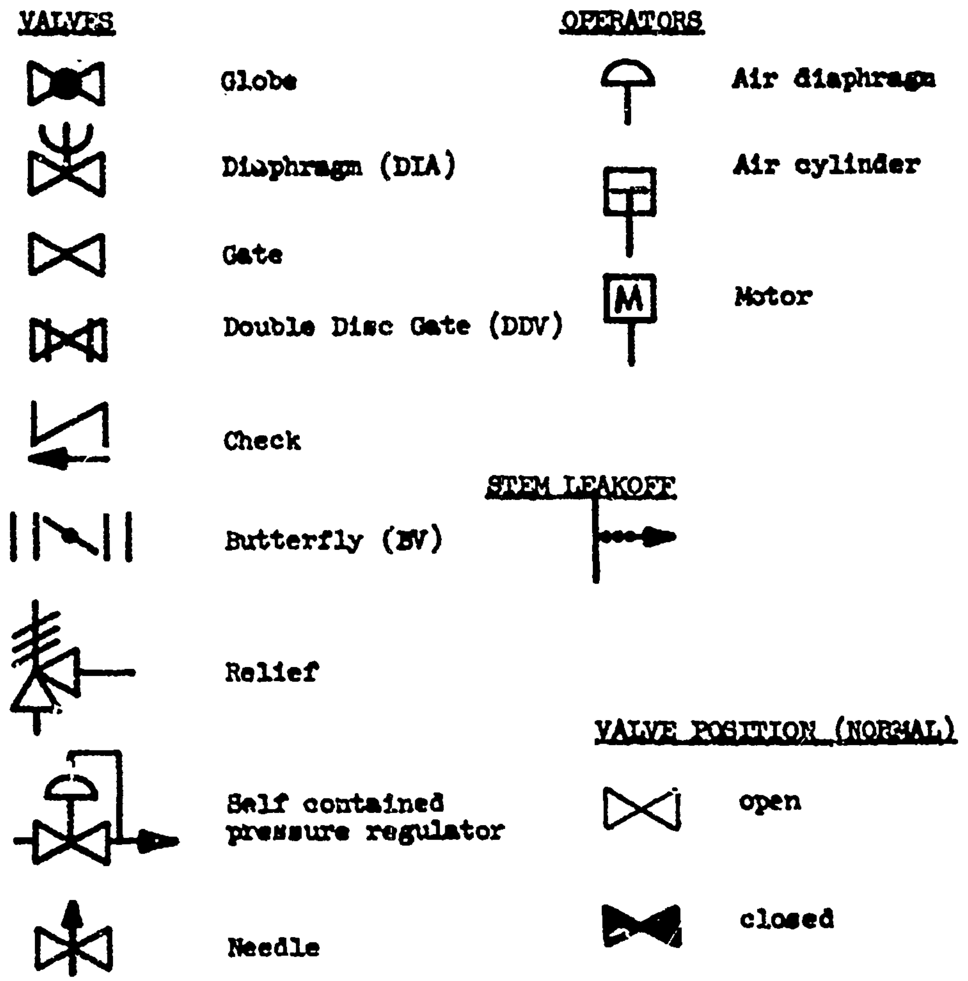
LEGEND



NOTATION

ASWI - AUTOMATIC SEAL WATER INJECTION
MSWI - MANUAL SEAL WATER INJECTION
AA - AUTOMATIC PRESSURIZATION WITH AIR
N_2 - MANUAL PRESSURIZATION WITH NITROGEN
LO - LOCKED OPEN
LC - LOCKED CLOSED
T - TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL, PHASE A
P - TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL, PHASE B

LEGEND FOR SYMBOLS, CONTAINMENT ISOLATION SYSTEM

FIGURE 5.2-25

5.3 CONTAINMENT VENTILATION SYSTEM

5.3.1 DESIGN BASIS

5.3.1.1 Performance Objectives

The containment ventilation system is designed to accomplish th llowing:

a) Remove the normal heat loss from all equipment and piping in the reactor containment during plant operation and to maintain a normal ambient temperature of 120°F or less.

b) Provide sufficient air circulation and filtering throughout all containment areas to permit safe and continuous access to the reactor containment within two hours after reactor shutdown assuming defects exist in 1% of the fuel rods.

c) Provide for positive circulation of air across the refueling water surface to assure personnel access and safety during shutdown.

d) Provide a minimum containment ambient temperature of 50°F during reactor shutdown.

e) Provide for purging of the containment vessel to the plant vent for dispersion to the environment. The rate of release does not permit off-site dose to exceed one-tenth of that permitted by 10 CFR 20.

f) Provide for dperessurization of the containment vessel following an accident. The post-accident design and operating criteria are detailed in Section 6.

In order to accomplish these objectives the following systems are provided:

a) Containment Recirculation Cooling and Filtration System
b) Control Rod Drive Mechanism Cooling System
c) Reactor Compartment Cooling System
d) Containment Purge System
e) Containment Auxiliary Charcoal Filter System
f) Containment Post-Accident Charcoal Filter System (Described in Section 6.4)
g) Steam Heating System

5.3.1.2 Design Characteristics - Sizing

The design characteristics of the equipment required in the containment for cooling, filtration and heating to handle the normal thermal and air cleaning loads during normal plant operation are presented in Table 5.3.1-1. In certain cases where engineered safeguards functions also are served by the equipment, component sizing is determined from the heavier duty specifications associated with the design basis accident (DBA), detailed further in Section 6.

5.3.2 SYSTEM DESIGN

5.3.2.1 Piping and Instrumentation Diagram

The containment ventilation, purging and recirculation cooling and filtration systems flow diagram is shown in Figure 5.3-1. The containment ventilation systems and main plant vent are designed as Class I structures.

5.3.2.2 Containment Recirculation Ventilation

Air recirculation cooling and filtering during normal operation is accomplished using all five air handling units discharged to a common headered ductwork distribution system to assure adequate flow of filtered and cooled air throughout the containment. The cooling coils in each air handling unit transfer up to 2.2×10^6 Btu/hr to the service water system during normal plant operation and 76.32×10^6 Btu/hr in the event of an accident when supplied with 2000 gpm cooling water at 85°F inlet temperature.

Each air handling unit consists of the following equipment arranged so that, during normal operation, air flows through the unit in the following sequence: cooling coils, moisture separators (demisters), HEPA filters, centrifugal fan with direct-drive motor, and distribution header. The fans and motors of these units are equipped with vibration sensors to detect abnormal operating conditions in the early stages of the disturbance. In the event of an accident, the air will continue this same flow path except that, after passing through the fan, it will be diverted automatically by air operated butterfly valves to a compartment containing charcoal filter before entering the distribution header. The normal air flow rate per air handling unit is approximately 70,000 cfm and the post-accident flow rate will be approximately 65,000 cfm, with 8000 cfm through the charcoal filter section. Section 6.4.2 provides additional information on the operation of this system.

The following additional systems supplement the main containment recirculation system:

a) Control rod drive cooling system consisting of fans and ductwork to circulate air through the control drive mechanism shroud and discharge it to the main containment volume. Four 1/3 capacity direct driven axial flow fans are used.

b) Four unit heaters supplied with 25 psig steam for containment heating, designed to maintain a minimum temperature of 50°F in the containment building during winter shutdown. Capacity of the four units is 1.6×10^6 Btu/hr.

5.3.2.3 Containment Purge System

The containment purge system is independent of the primary auxiliary building exhaust system and includes provisions for both supply and exhaust air. The supply system includes roughing filters, heating coils, fan, supply penetration with two butterfly valves for bubble tight shutoff, and a purge supply distribution header inside containment. The exhaust system includes exhaust penetration with two butterfly valves identical to those above, exhaust ductwork, filter bank with roughing and HEPA filters, fans and exhaust vent. The full purge flow rate is 40,000 cfm. The quick closing purge isolation valves are capable of closing within two seconds of receipt of the accident signal.

During power operation, containment integrity is maintained with no release from the containment ventilation system to the atmosphere. Prior to purging the containment air, particulate and gas monitor indications of the closed containment activity levels will be used to guide routine releases from the containment. During power operation, the containment air particulate and gas monitor indications will help determine the desirability of using either one or both of two auxiliary particulate and charcoal filter units installed in the containment primarily for pre-access cleanup.

When containment purging for access following reactor shutdown is in progress, releases from the plant vent are continuously monitored with a gas monitor.

5.3.2.4 Isolation Valves

The purge supply and exhaust ducts butterfly valves, both inside and outside the containment, are closed during power operation. The spaces between the closed valves are pressurized with air by the Penetration and Weld Channel Pressurization System. The valves are designed for rapid automatic closing by the containment isolation signal (derived from any automatic safety injection signal), or upon a signal of high activity level within the containment in the event of a radioactivity release when the purge line is open.

5.3.2.5 Containment Pressure Relief Line

The normal pressure changes in the containment during reactor power operation, and during plant cooldown if the containment purge system is not operating, will be handled by the containment pressure relief line. This line is equipped with three quick-closing butterfly type isolation valves, one inside and two outside the containment. The valves will be automatically actuated to the closed position by the containment isolation signal, or by a containment high radioactivity signal. The two intra-valve spaces are pressurized with air by the Penetration and Weld Channel Pressurization System when the valves are closed. The pressure relief line discharges to the plant vent.

TABLE 5.3.1-1

PRINCIPAL COMPONENT DATA SUMMARY

System	Units Installed	Unit Capacity	Units Required for Normal Operati
Containment Recirculation			
Demister	5	65,000* cfm	5
Cooling Coils - Normal	5	2.2×10^6 Btu/hr	5
Cooling Coils - DBA	5	76.32×10^6 Btu/hr	5
HEPA Filters	5	65,000* cfm	5
Fans	5	65,000* cfm	5
Fan Pressure - Normal	-	7.70 in. H_2O	
Fan Motors (440 V, 3 phase)	5	350 Hp	5
DBA Charcoal Filters	5	65,000 cfm	0
Control Rod Drive Mechanism Cooling			
Fans, Standard Conditions	4	15,000 cfm	3
Fan Pressure	-	5-1/2 in. H_2O	
Fan Motors	4	25 Hp	3
Reactor Compartment Cooling			
Part of CB Recirculation System	-	12,000 cfm	
Refueling Canal Air Sweep			
Part of CB Recirculation System	-	17,500 cfm	

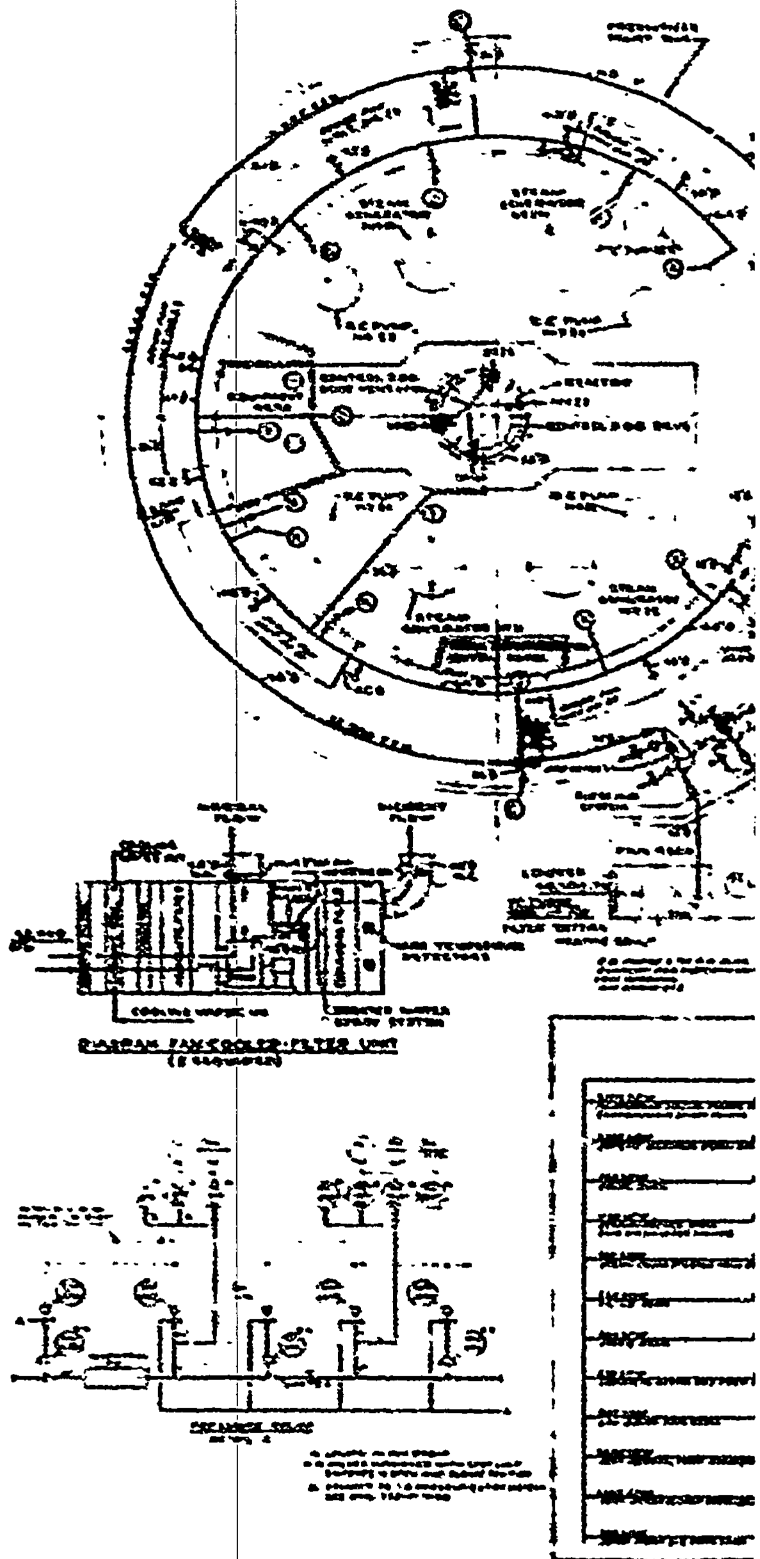
* 70,000 cfm during normal operation

TABLE 5.3.1-1 (CONT'D)

System	Units Installed	Unit Capacity	Units Required for Normal Operation
Purge Supply			
Fans, Standard Conditions	1	40,000 cfm	Optional
Fan Pressure	-	4 in. H_2O	
Fan Motors	1	40 HP	
Pre-heat Coils	1 Set		Optional
Air Filters, Roughing	-	40,000 cfm	1
* Purge Exhaust			
Fans, Standard Conditions	2	53,200 cfm	Optional
Fan Pressure	-	8.5 in. H_2O	"
Fan Motors	2	100 HP	"
Plenums	2	53,200 cfm	"
HEPA Filters	2	53,200 cfm	"
Containment Auxiliary Charcoal Filter			
Fans, Standard Conditions	2	8,000 cfm	"
Fan Pressure	-	4.75 in. H_2O	"
Fan Motors	2	10 HP	"
Filters; Roughing, HEPA and Charcoal Filters	2	8,000 cfm	"
Steam Heating			
Heaters, 25 psig steam	4	400,000 Btu/hr each	"

* Note: The two exhaust fan are used interchangeably or as backup for:

1. Ventilation of Primary Auxiliary Building.
2. Containment Building purge system.



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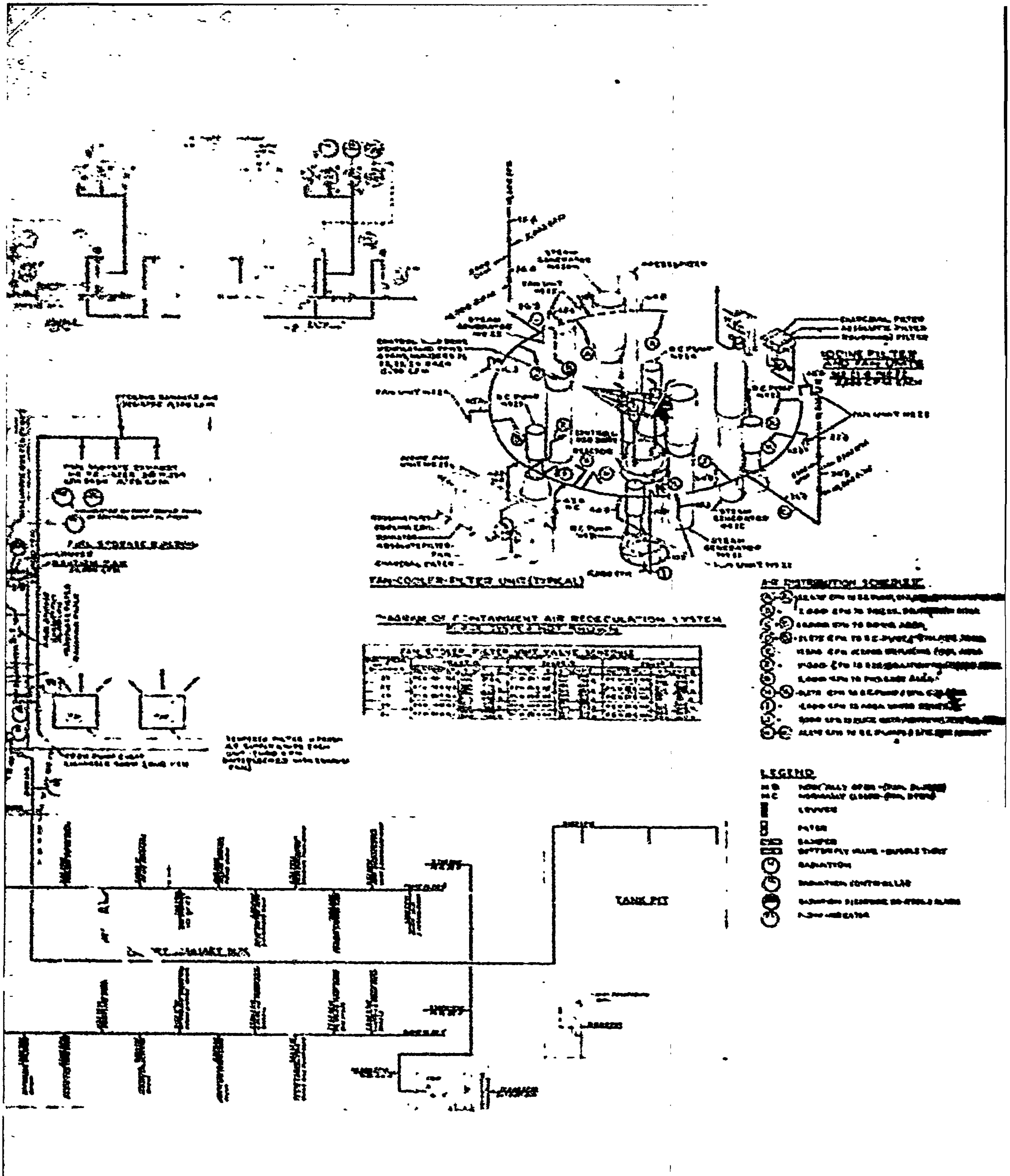

CONTAINMENT VENTILATION SYSTEM

FIGURE 5.3-1

Chap 6

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BY QUESTION RESPONSES IN
VOLUMES 5 AND 6

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	Design and tests of the valve and valve operator which are added to the sump suction lines to limit the consequences of a passive failure in the sump suction line.	14.3.2
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6 ENGINEERED SAFETY FEATURES

The central safety objective in reactor design and operation is control of reactor fission products. The methods used to assure this objective are:

a. Core design to preclude release of fission products from the fuel (Section 3).

b. Retention of fission products in the reactor coolant for whatever leakage occurs (Sections 4, and 6).

c. Retention of fission products by the containment for operational and accidental releases beyond the reactor coolant boundary (Sections 5, and 6).

d. Optimizing fission product dispersal to minimize population exposure. (Sections 2, and 11).

The Engineered Safety Features are the provisions in the plant which embody methods b and c above to prevent the occurrence or to ameliorate the effects of serious accidents.

The Engineered Safety Features systems in this plant are the Containment System, detailed in Section 5; the Safety Injection System, detailed in Section 6.2; the Containment Spray System, detailed in Section 6.3; the Containment Air Recirculation Cooling and Filtration System, detailed in Section 6.4; the Isolation Valve Seal Water System detailed in Section 6.5; and the Containment Penetration and Weld Channel Pressurization System, detailed in Section 6.6.

Evaluations of techniques and equipment used to accomplish the central objective including accident cases are detailed in Section 5, 6 and 14.

6.1 GENERAL DESIGN CRITERIA

Criteria applying in common to all engineered safety features are given in 6.1.1. Thereafter, criteria which are related to engineered safety features but which are more specific to other plant features or systems, are listed and cross referenced in Section 6.1.2.

6.1.1 ENGINEERED SAFETY FEATURES CRITERIA

Engineered Safety Features Basis for Design

Criterion: Engineered safety features shall be provided in the facility to back up the safety provided by the core design. the reactor coolant pressure boundary, and their protection systems. Such engineered safety feature. shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary, assuming unobstructed discharge from both ends. (GDC 37)

The design, fabrication, testing and inspection of the core, reactor coolant pressure boundary and their protection systems give assurance of safe and reliable operation under all anticipated normal, transient, and accident conditions. However, engineered safety features are provided in the facility to back up the safety provided by these components. These engineered safety features have been designed to cope with any size reactor coolant pipe break up to and including the circumferential rupture of any pipe assuming unobstructed discharge from both ends, and to cope with any steam or feedwater line break up to and including the main steam or feedwater headers.

Limiting the release of fission products from the reactor fuel is accomplished by the Safety Injection System which, by cooling the core, keeps the fuel in place and substantially intact and limits the metal water reaction to an insignificant amount.

The Safety Injection System consists of high and low head centrifugal pumps driven by electric motors, and passive ac mulator tanks which are self energized and which act independently of any actuation signal or power source.

The release of fission product from the containment is limited in three ways:

1. Blocking the potential leakage paths from the containment. This is accomplished by:

   a. A steel-lined, reinforced concrete reactor containment with testable, doubly sealed penetrations and liner weld channels, the spa of which are continuously pressurized above accident pressure and wh form a virtually leak-tight barrier to the escape of fission products should a loss of coolant accident occur.

   b. Isolation of process lines by the Containment Isolation System which imposes double barriers in each line which penetrates the containment except for lines utilized during the accident. An Isolation Valve Seal Water System provides a water seal at the isolation valves thus sealing the pipes penetrating the contair

2. Reducing the fission product concentration in the containment atmosphere This is accomplished by:

   a. Containment Air recirculation filters which provide for rapid removal of particles and iodine vapor from the containment atmosphere.

   b. Chemically treated spray which removes elemental iodine vapor from the containment atmosphere by washing action.

3. Reducing the containment pressure and thereby limiting the driving potential for fission product leakage. This is accomplished by cooling the containment atmosphere by the following independent systems of equal heat removal capacity.

   a. Containment Spray System

   b. Containment Air Recirculation Cooling System

<u>Reliability and Testability of Engineered Safety Features</u>

Criterion: All engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public. (GDC 38)

A comprehensive program of plant testing is formulated for all equipment, systems and system control vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop, integrated tests of the system as a whole, and periodic tests of the actuation circuitry and mechanical components to assure reliable performance, upon demand, throughout the plant lifetime.

The initial tests of individual components and the integrated test of the system as a whole complement each other to assure performance of the system as designed and to prove proper operation of the actuation circuitry.

Routine periodic testing of the engineered safety features components is intended. In the event that one of the components should require maintenance as a result of failure to perform during the test according to prescribed limits, the redundant component is immediately tested to confirm functional availability. A satisfactory performance test of the remaining redundant component(s) is proof of the availability of that safety feature, and it is not necessary to adjust plant load during the brief period that the malfunctioning component may be out for servicing. The necessary corrections or minor maintenance are made, and the repaired unit is retested immediately to confirm proper performance.

Missile Protection

Criterion: Adequate protection for those engineered safety features, the failure of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40)

A loss-of-coolant accident or other plant equipment failure might result in dynamic effects or missiles. For engineered safety features which are required to ensure safety in the event of such an accident or equipment failure, protection is provided primarily by the provisions which are taken in the design to prevent the generation of missiles. In addition, protection is also provided by the layout of plant equipment or by missile barriers in certain cases. References made to Section 5.1.2-5 for a discussion of missile protection.

Injection paths leading to unbroken reactor coolant loops are protected against damage as a result of the maximum reactor coolant pipe rupture by layout and structural design considerations. Injection lines penetrate the main missi barrier, which is the crane wall, and the injection headers are located in the missile-protected area between the crane wall and the containment wall. Individual injection lines, connected to the injection header, pass through the barrier and then connect to the loop Separation of the individua injection lines is provided to the maximum extent practicable. Movement of the injection line, associated with rupture of a reactor coolant loop, is accommodated by line flexibility and by the design of the pipe supports such that no damage outside the missile barrier is possible.

The containment structure is capable of withstanding the effects of missiles originating outside the containment and which might be directed toward it so that no loss-of-coolant accident can result.

All hangers, stops and anchors are designed in acc   :nce with USAS B31.1 Code for Pressure Piping and ACI 318 Building Code Requirements for Reinforced Concrete which provide minimum requirements on material, design and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment.

Engineered Safety Features Performance Capability

Criterion: Engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public. (GDC 41)

Each engineered safety feature provides sufficient performance capability to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public.

The extreme upper limit of public exposure is taken as the levels and time periods presently outlined in 10 CFR 100, i.e. 300 rem to the thyroid in two hours at the exclusion radius and 300 rem to the thyroid over the duration of the accident at the low population zone distance. The accident condition considered is the hypothetical case of a release of fission products per TID 14844. Also, the total loss of all outside power is assumed concurrently with this accident. With all engineered safety features system functioning at full capacity, the offsite exposure would be within 10 CFR 20 limits.

Under the above accident conditions, the Containment Air Recirculation Cooling and Filtration System and the Containment Spray System are designed and sized so that both systems, each operating with partial effectiveness, are able to supply the necessary post-accident iodine removal capacity and cooling capacity to assure the maintenance of containment integrity, that is, keeping the pressure below design pressure at all times, assuming that the core residual heat is released to the containment as steam. Partial effectiveness is defined as operation of a system with at least one active component failure.

Engineered Safety Features Components Capability

Criterion: Engineered safety features shall be designed so that the capabil of these features to perform their required function is not impaired by the effects of a loss-of-coolant accident to the extent of causing undue risk to the health and safety of the public. (GDC 42)

Instrumentation, pumps, fans, filters, cooling units, valves, motors, cable and penetrations located inside the containment are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure, and humidity expected during the required operati period.

The Safety Injection System pipes serving each loop are anchored at the crane wall which constitutes the missile barrier in each loop area to restr potential accident damage to the portion of piping beyond this point. The anchorage is designed to withstand, without failure, the thrust force of any branch line severed from the reactor coolant pipe and discharging fluic to the atmosphere, and to withstand a bending moment equivalent to that which produces failure of the piping under the action of free end discharge to atmosphere or motion of the broken reactor coolant pipe to which the injection pipes are connected. This prevents possible failure at any point upstream from the support point including the branch line connection into the piping header.

Accident Aggravation Prevention

Criterion: Protection against any action of the engineered safety features which would accentuate significantly the adverse after-effects of a loss of normal cooling shall be provided. (GDC 43)

The reactor is maintained subcritical following a pipe rupture accident. Introduction of borated cooling water into the core results in a net negative reactivity addition. The control rods insert and remain inserted.

The supply of water by the Safety Injection System to cool the core cladding does not produce significant metal-water reaction. (<1.0%).

The delivery of cold safety injection water to the reactor vessel following accidental expulsion of reactor coolant does not cause further loss of integrity of the Reactor Coolant System boundary.

Sharing of Systems

Criterion: Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public (GDC 4)

The residual heat removal pumps and heat exchangers serve dual functions. Although the normal duty of the residual heat exchangers and residual heat removal pumps is performed during periods of reactor shutdown, during all plant operating periods these residual heat removal pumps are aligned to perform the low head safety injection function. In addition, during the recirculation phase of a loss-of-coolant accident, the residual heat exchangers of this system perform the core cooling function and the containment cooling function as part of the Containment Spray System and the residual heat removal pumps which are part of the external recirculation loop provide back-up capability to the recirculation pumps which comprise part of the internal recirculation loop. Demonstration checking of the system, performed during each refueling period before plant startup, provides assurance of correct system alignment for the safety injection function of the components.

During the injection phase, the safety injection pumps do not depend on any portion of other systems. During the recirculation phase, if Reactor Coolant System pressure stays high due to a small break accident, suction to the safety injection pumps is provided by the internal recirculation pumps.

The Containment Air Recirculation and Filtration System also serves the dual function of containment cooling during normal operation and containment cooling after an accident. Since the method of operation for both cooling functions is the same, the dual aspect of this system does not affect its function as an engineered safety feature.

6.1.2 RELATED CRITERIA

The following are criteria which, although related to all engineered safety features, are more specific to other plant features or systems, therefore are discussed in other sections, as listed.

Name	Discussion
Quality Standards (GDC 1)	Section 4
Performance Standards (GDC 5)	Section 4
Records Requirements (GDC 2)	Section 4
Instrumentation and Control Systems (GDC 12)	Section 7
Engineered Safety Features Protection Systems (GDC 15)	Section 7
Emergency Power (GDC 39)	Section 8

6.2 SAFETY INJECTION SYSTEM

6.2.1 DESIGN BASIS

Emergency Core Cooling System Capability

Criterion: An Emergency Core Cooling System with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty. (GDC 44).

Adequate emergency core cooling is provided by the Safety Injection System (which constitutes the Emergency Core Cooling System) whose components operate in three modes. These modes are delineated as passive accumulator injection, active safety injection and residual heat removal recirculation.

The primary purpose of the Safety Injection System is to automatically deliver cooling water to the reactor core in the event of a loss-of-coolant accident. This limits the fuel clad temperature and thereby ensures that the core will remain intact and in place, with its essential heat transfer geometry preserved. This protection is afforded for:

a) All pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop, assuming unobstructed discharge from both ends.

b) A loss of coolant associated with the rod ejection accident.

c) A steam generator tube rupture.

The basic design criteria for loss of coolant accident evaluations are:

1. The cladding temperature is to be less than:
   a. The melting temperature of Zircaloy-4.
   b. The temperature at which gross core geometry distortion, including clad fragmentation may be expected.
2. The total core metal-water reaction will be limited to less than 1 percent.

These criteria will assure that the core geometry remains in place and substantially intact to such an extent that effective cooling of the core is not impaired.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the Safety Injection System adds shutdown reactivity so that with a stuck rod, no off-site power and minimum engineered safety features, there is no consequential damage to the Reactor Coolant System and the core remains in place and intact.

Redundancy and segregation of instrumentation and components is incorporated to assure that postulated malfunctions will not impair the ability of the system to meet the design objectives. The system is effective in the event of loss of normal station auxiliary power coincident with the loss of coolant, and is tolerant of failures of any single component or instrument channel to respond actively in the system. During the recirculation phase of a loss of coolant, the system is tolerant of a loss of any part of the flow path since back up alternative flow path capability is provided.

The ability of the Safety Injection System to meet its capability objectives is presented in Section 6.2.3. The analysis of the accidents is presented in Section 14.

Inspection of Emergency Core Cooling System

Criterion: Design provisions shall, where practical, be made to facilitate physical parts of the Emergency Core Cooling System, including reactor vessel internals and water injection nozzles (GDC 45).

Design provisions are made to the extent practical to facilitate access to the critical parts of the reactor vessel internals, pipes, valves and pumps for visual or boroscopic inspection for erosion, corrosion and vibration wear evidence, and for non-destructive test inspection where such techniques are desirable and appropriate.

Testing of Emergency Core Cooling System Components

Criterion: Design provisions shall be made so that components of the Emergency Core Cooling System can be tested periodically for operability and functional performance (GDC 46).

The design provides for periodic testing of active components of the Safety Injection System for operability and functional performance.

Power sources are arranged to permit individual actuation of each active component of the Safety Injection System.

The safety injection pumps can be tested periodically during plant operation using the minimum flow recirculation lines provided. The residual heat removal pumps are used every time the residual heat removal loop is put into operation. All remote operated valves can be exercised and actuation circuits can be tested during routine plant maintenance.

Testing of Emergency Core Cooling System

Criterion: Capability shall be provided to test periodically the operability of the Emergency Core Cooling System up to a location as close to the core as is practical (GDC 47).

An integrated system test can be performed when the plant is cooled down and the residual heat removal loop is in operation. This test would not introduce flow into the Reactor Coolant system but would demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry upon initiation of safety injection.

Level and pressure instrumentation are provided for each accumulator tank, and accumulator tank pressure and level are continuously monitored during plant operation. Flow from the tanks can be checked at any time using test lines.

The accumulators and the safety injection piping up to the final isolation valve is maintained full of borated water at refueling water concentration while the plant is in operation. The accumulators and injection lines will be refilled with borated water as required by using the safety injection pumps to recirculate refueling water through the injection headers. A small bypass line and a return line are provided for this purpose.

Flow in each of the high head injection branch lines in the main flow line for the residual heat removal pumps is monitored by a flow indicator. Pressure instrumentation is also provided for the main flow paths of the high head and residual heat removal pumps.

Testing of Operational Sequence of Emergency Core Cooling System

Criterion: Capability shall be provided to test initially, under conditions as close as practical to design, the full operational sequence that would bring the Emergency Core Cooling System into action, including the transfer to alternate power sources. (GDC 48)

The design provides for capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the Safety Injection System to demonstrate the state of readiness and capability of the system. Details of the operational sequence testing are presented in 6.2.5, Tests and Inspections.

Codes and Classifications

Table 6.2-1 tabulates the codes and standards to which the safety injection system components are designed.

Service Life

All portions of the system located within the containment are designed to operate without benefit of maintenance and without loss of functional performance for the duration of time the component is required.

6.2.2 SYSTEM DESIGN AND OPERATION

System Description

Adequate emergency core cooling following a loss-of-coolant accident is provided by the Safety Injection System shown in Figure 6.2-1. Figures 6.2-2, 6.2-3, 6.2-4, and 6.2-5 depict how this system concept is translated into plant layout design. The system components operate in the following possible modes:

a) Injection of borated water by the passive accumulators.

b) Injection by the safety injection pumps drawing borated water first from the boron injection tank (because of the N_2 over pressure) and then from the refueling water storage tank.

c) Injection by the residual heat removal pumps also drawing borated water from the refueling water storage tank.

d) Recirculation of spilled reactor coolant, injected water and Containment Spray System drainage back to the reactor from the recirculation sump by the recirculation pumps. (The residual heat removal pumps provide backup recirculation capability.)

The initiation signal for core cooling by the safety injection pumps and the residual heat removal pumps is the Safety Injection Signal which is actuated by any of the following:

a) Low pressurizer pressure in coincidence with low pressurizer water level (1/3 pairs - A pair is defined as a pressure signal and its assigned level signal).

b) High containment pressure (2/3, Hi level; also 2 pairs of 2/3, Hi Hi level)

c) High differential pressure between any two steam generators (2/3)

d) High steam flow in any two of the four steam lines (1/2 per line)

e) Manual Actuation.

Injection Phase

The principal components of the Safety Injection System which provide emergency core cooling immediately following a loss of coolant are the accumulators (one for each loop), the three safety injection (high head) pumps and the two residual heat removal (low head) pumps. The safety injection and residual heat removal pumps are located in the auxiliary building.

The accumulators, which are passive components discharge into the cold legs of the reactor coolant piping when pressure decreases to 660 psig, thus rapidly assuring core cooling for large breaks. They are located inside the containment, but outside the crane well, therefore each is protected against possible missiles.

The safety injection signal, opens the Safety Injection System isolation valves and starts the safety injection pumps and the residual heat removal pumps. (The items on Figure 6.2-1 marked with an "S" receive the safety injection signal.)

The safety injection pumps (high head) deliver borated water to two separate discharge headers. The flow from one header is injected into two hot legs of the reactor coolant system, and the flow from the other header is injected into two cold legs. If the two injection lines on a header remain intact, the flow from one safety injection pump is sufficient to meet design requirements for makeup of coolant following a small break which does not immediately depressurize the Reactor Coolant System to the accumulator discharge pressure. Since the small break may be an injection line, two safety injection pumps are required. Each pump delivers to a separate header, thus insuring delivery to an intact header.

For large breaks, the Reactor Coolant System would be depressurized and voided of coolant rapidly (about 10 seconds for the largest break) and a high flow rate is required to quickly recover the exposed fuel rods and limit possible core damage. To achieve this objective, one residual heat removal pump (high flow, low head) is required to deliver borated water to the cold legs of the reactor coolant loops. Two pumps are available in order to provide for an active component failure. Delivery from these pumps supplements the accumulator discharge. Since the Reactor Coolant System back pressure is relatively low (rapid depressurization for large breaks), only one header is provided. A broken injection line would not appreciably change the flows in the other injection lines delivery to the core.

The boron injection tank, located in the primary auxiliary building, is the source of borated water for the safety injection pumps. The tank contains boric acid at a nominal value of 20,000 ppm boron (12% boric acid solution) and is isolated from the safety injection pump suction line by redundant normally closed parallel valves. The valves open upon receipt of a safety injection signal. Overpressure in the boric acid injection tank in excess of the static head developed by the refueling water storage tank causes preferential flow from the former. When the boron solution in the tank is nearly depleted, two out of three low water level signals automatically recloses the isolation valves to isolate the tank and pressure source, allowing flow to the safety injection pumps to come from the refueling water storage tank. Sufficient liquid remains in the boron injection tank to assure that the gas over the liquid cannot enter the line before the isolation valves close. The residual heat removal pumps take suction from the refueling water storage tank. In addition, the charging pumps of the Chemical Volume and Control System are available but are not required to augment the flow of the Safety Injection System.

Because the injection phase of the accident is terminated before the refueling water storage tank is completely emptied, all pipes are kept filled with water before recirculation is initiated. Water level indication and alarm on the refueling water storage tank give the operator ample warning to terminate the injection phase. Additional level indicators and alarms are provided in the containment sump which also gives back-up indication when injection can be terminated and recirculation initiated.

Recirculation Phase

After the injection operation, coolant spilled from the break and water collected from the containment spray is cooled and returned to the Reactor Coolant System by the recirculation system.

When the break is large, depressurization occurs due to the large rate of mass and energy loss through the break to containment. In the event of a large break the recirculation flow path is within the containment. The system is arranged so that the recirculation pumps take suction from the recirculation sump in the containment floor and deliver spilled reactor coolant and borated refueling water back to the core through the residual heat exchangers. The system is also arranged to allow either of the residual heat removal pumps to take over the recirculation function. The residual heat removal pumps would only be used if backup capacity to the internal recirculation loop is required. Water is delivered from the containment to the residual heat removal pumps from a separate sump inside the containment.

For small breaks the depressurization of the Reactor Coolant System is augmented by steam dump and auxiliary feed water addition to the Steam System. For the smaller breaks in the Reactor Coolant System where recirculated water must be injected against higher pressures for long term core cooling, the system is arranged to deliver the water from the residual heat exchangers to the high-head safety injection pump suction and, by this external recirculation route, to the reactor coolant loops. Thus, if depressurization of the Reactor Coolant System proceeds slowly, the safety injection pumps may be used to augment the flow-pressure capacity of the recirculation pumps in returning the spilled coolant to the reactor.

The recirculation pumps, the residual heat exchangers, piping and valves vital to the function of the recirculation loop are located in a missile-shielded space inside the polar crane support wall on the west side of the reactor primary shield.

There are two sumps within the containment, the recirculation sump and the containment sump. Both sumps collect liquids discharged into the containment during the injection phase of the design basis accident.

The recirculation sump contains two screens through which the recirculated wa[illegible] must flow before entering the pumps. The first screen consists of a floor gr[illegible] (1" x 4") which covers the sump on the basement floor with the purpose of the grating being to prevent large particles from entering the sump. The second [illegible] is located in the sump and has the capability to exclude particles greater th[illegible] 1/4 inch in diameter from the recirculation pump suction. This floor grating [illegible] a total surface area of 100 ft^2. Since all recirculated water passes through [illegible] screens before entering the pumps, particles in excess of 1/4 inch diameter a[illegible] precluded from entering these lines. The water velocity through the sump is less than one foot per second.

The containment sump contains two screens for the purpose of preventing parti[illegible] greater than 1/4" in diameter from entering the residual heat removal pump suction. The first screen consists of 1" x 4" floor grating with an area of 13.6 ft^2; the second screen is located in the sump. The water velocity throu[illegible] the sump is less than one foot/second.

The low head external recirculation loop via the containment sump line and the residual heat removal pumps provides backup recirculation capability to the low head internal recirculation loop. The containment sump line has two remote motor operated normally closed valves located outside the containment The high head external recirculation flow path via the high head safety injection pumps is only required for the range of small break sizes for which the reactor coolant system pressure remains in excess of the shut-off head of the recirculation pumps (or residual heat removal pumps) at the end of the injection phase.

The external recirculation flow paths within the primary auxiliary building are designed so that external recirculation can be initiated immediately after the accident. Those portions of the Safety Injection System located outside of the containment which are designed to circulate, under post-accident conditions, radioactively contaminated water collected in the containment, meet the following requirements:

a) Shielding to maintain radiation levels within the guidelines set forth in 10 CFR 100.

b) Collection of discharges from pressure relieving devices into closed systems.

c) Means to detect and control radioactivity leakage into the environs, to the limits consistent with guidelines set forth in 10 CFR 100.

This criterion is met by minimizing leakage from the system. External recirculation loop leakage is discussed in Section 6.2.3.

One pump (either recirculation or residual heat removal) and one residual heat exchanger of the recirculation system provides sufficient cooled recirculated water to keep the core flooded with water by injection through the cold leg connections while simultaneously providing, if required, sufficient containment spray flow to prevent the containment pressure from rising above design because of the boiloff from the core. Only one pump and one heat exchanger are required to operate for this capability at the earliest time recirculation is initiated. With a recirculation (or residual heat removal) pump in operation, with a spray header valve open, no containme cooling fans (Section 6.4) are required. The design ensures that heat remove from the core and containment is effective in the event of a pipe or valve body rupture.

Cooling Water

The Service Water System (Section 9) provides cooling water to the component cooling loop which in turn cools the residual heat exchangers, both of which are part of the Auxiliary Coolant Systems (Section 9). Three conventional service water pumps are available to take suction from the river and dischar to the two component cooling heat exchangers. Three component cooling pumps are available to discharge through their heat exchangers and deliver to the two residual heat exchangers. Only one pump and one heat exchanger of each type are required to meet the core cooling function. All of this equipment, with the exception of the residual heat exchangers, is located outside containment.

Change-Over from Injection Phase to Recirculation Phase

Assuming that the three high head safety injection pumps, the two residual heat removal pumps, and the two containment spray pumps (Section 6.3) are running at their maximum capacity. the time sequence, from the time of the safety injection signal, for the change over from injection to recirculation in the core of a large rupture is as follows:

a) In approximately ten minutes, sufficient water has been delivered to provide the required NPSH to start the recirculation pumps.

b) In approximately fifteen to twenty minutes the first low level alarm on the RWST sounds. The alarm serves to alert the operator to prepare for switch over to the recirculation mode.

c) Approximately two minutes after the first low level alarm the RWST will be at the level for switch over to the recirculation phase.
10 The operator is given a second alarm and will also see on the control board that the level is at the set point for switch-over. Switch-over via the eight switch sequence is performed at this time.

d) With the completion of switch-over one spray pump continues to draw from the RWST for approximately a further twenty-five minutes. The remaining operating spray pump is stopped when the level in the RWST reaches the low low level point.

Recirculation pump motors are 2'2" above the highest water level after addition of the injected water to the spilled coolant.

The eight switch sequence which accomplishes the change over from injection to recirculation is listed below. This switch over takes place when the level indicators in the refueling water storage tank indicate that the fluid has been injected. The level indicators in the containment sump will verify that the level is sufficient within the containment. The sequence is followed regardless of which power supply is available. The time required to complete the switch-over is just the time for the switch gear to function and this will take less than 90 seconds. (The eight switch sequence automatically performs the pump shedding operations described in paragraph b above so that the operator can directly initiate the change-over when the first alarm sounds).

1. Terminate safety injection signal in order that the control logic permits manipulation of the system. (At any time following completion of the auto start sequence).

2. Close Switch One. (Remove and isolate unnecessary loads from the diesels).

   a) Trips one of three high head safety injection pumps if all three are operating, (no action if two are operating) and isolate the pump suction to the refueling water storage tank if the tripped pump is the middle safety injection pump.

   b) Trip one spray pump if both are operating (No action if one is operating).

   c) Closes containment isolation valves at the inoperative spray pump discharge.

3. Close Switch Two (Establish cooling flow for residual heat exchangers).

   a) Starts one service water pump, conventional her[illegible] (the second or third pump is given a start signal if the fi[illegible] pump fails to start).

   b) Starts one component cooling pump (the second or th[illegible] given a start signal if the first or second p[illegible] fails

4. Close Switch Three (Remove and isolate unnecessary loads from the diesels).

   a) Trips both residual heat removal pumps.

   b) Closes isolation valves at pump suction and discharge headers.

5. Close Switch Four (Initiate internal recirculation flow)

   a) Opens valves on discharge of recirculation pumps.

   b) Starts recirculation pump A (If A fails to start, use manual start on Pump B).
      (Pump B control switch is adjacent to switch four).

6. Close Switch Five (action nullified if only 2/3 diesels available) (establish additional cooling capability if power permits).

   a) Starts second service water pump, conventional header (the third pump is given the start signal if the second pump fails to start).

   b) If a) completed, starts second component cooling pump (the third pump is given the start signal if the second pump fails to start).

   c) If b) completed, starts recirculation pump B.

7. Check flow to Reactor Coolant System via the low head injection lines.

   A. If the flow on 3 out of 4 lines is less than 300 gpm, close Switch Six then go to Switch Eight (Provide recirculation at elevated system pressure).

      a) Aligns flow from residual heat exchanger to high head safety injection pumps. (The motor-operated valves on the outlet of the residual heat exchangers to the suction of the high-head safety injection pumps are opened. The motor-operated valves on the outlets of the residual heat exchangers to the low-head injection lines are closed together with the safety injection pump mini-flow and residual heat removal pump mini-flow.)

B. If the flow on 3 out of 4 lines is equal to or greater than 300 gpm, omit Switch Six, go to Switch Eight (Provides recirculatio at low system pressure).

a) Trips high head safety injection pumps.

8. Close Switch Eight (Complete the isolation of the safety injection system and containment spray system lines to the refueling water storage tank).

a) Close the valve on the spray test line.

b) Close the valve in the safety injection pumps suction line from the refueling water storage tank.

Although the listed switches are manual, each automatically causes the operations listed. An indicating lamp is provided to show the operator when the operations of a given switch have been performed and when he should proceed with the next switching operation. In addition, lamps indicating completion of the individual functions for a given switch are provided. These lamps are adjacent to the switches. The time required to complete the switch over is just the time for the switchgear to operate. Should an individual component fail to respond, the operator can take corrective action to secure appropriate response from controls within the control room.

Remote operated valves for the injection phase of the Safety Injection System (Figure 6.2-1) which are under manual control, (that is, valves which normally are in their ready position and do not receive a safety injection signal) have their positions indicated on a common portion of the control board. At any time during operation when one of these valves is not in the ready position for injection, it is shown visually on the board. Reference is made to Table 6.2-12 which is a listing of the instrumentation readouts on the control board which the operator can monitor during recirculation. In addition, an audible annunciation alerts the operator to the condition.

Location of the Major Components Required for Recirculation

The residual heat removal pumps are located in the residual heat removal pump room which is below the basement floor of the primary auxiliary building (El. -3'1"). The residual heat exchangers are located on a platform above the basement floor of the containment building (El. 66').

The recirculation pumps are located directly above the recirculation sump in the containment building (El. 46').

The component cooling pumps and heat exchangers are located in the primary auxiliary building (El. 68' and 80' respectively).

The service water pumps are located in the screenhouse and the redundant piping to the component cooling heat exchangers is run underground.

Steam Break Protection

A large break of a steam system pipe rapidly cools the reactor coolant causing insertion of reactivity into the core and depressurization of the system. Compensation is provided by injection of boric acid from the boron injection tank. The discharge line from the tank is aligned to the suction of the safety injection pumps. Redundant isolation valves open upon a safety injection signal, providing an injection of 20,000 ppm boron. This is sufficient to terminate the reactor power transient before any clad damage results. Before the boron injection tank is emptied, the suction for the safety injection pumps switches automatically to the refueling water storage tank. The analysis of the steam line rupture accident is presented in Section 14.2.5.

Components

All associated components, piping, structures, and power supplies, of the Safety Injection System are designed to Class I seismic criteria.

All components inside the containment are capable of withstanding or are protected from differential pressure which may occur during the rapid pressure rise to 47 psig in 10 seconds.

All motors, instruments, transmitters, and their associated cables locate inside the containment are designed to function under the post-accident temperature, pressure, and humidity conditions. In addition, this equipm is designed to withstand the pressure and temperature conditions associat with 1.5 times the design pressure 70.5 psig and 298°F) for one hour withc impairing operability.

Emergency core cooling components are austenitic stainless steel, and hen are quite compatible with the spray solution over the f ll range of expos in the post-accident regime. While this material is subject to crevice c by hot, concentrated caustic, the NaOH additive cannot enter the containm or Emergency Core Cooling Systems without first being diluted and partial neutralized with boric acid to a mild solution. Corrosion tests performe with simulated spray showed negligible attack, both generally and locally in stressed and unstressed stainless steel at containment and ECCS condit These tests are discussed in WCAP-7153.[1]

The quality standards of all safety injection system components are tabul in summary form in Table 6.2-13.

1. WCAP-7153, "Investigations of Chemical Additives for Reactor Containme Sprays," M. J. Bell, et al, March 1968 (W Confidential).

Accumulators

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal plant operation each accumulator is isolated from the Reactor Coolant System by two check valves in series. Should the Reactor Coolant System pressure fall below the accumulator pressure, the check valves open and borated water is forced into the Reactor Coolant System. Mechanical operation of the swing-disc check valves is the only action required to open the injection path from the accumulators to the core via the cold leg.

The level of borated water in each accumulator tank is adjusted remotely as required during normal plant operations. Refueling water is added using a safety injection pump. Water level is reduced by draining to the reactor coolant drain tank. Samples of the solution in the tanks are taken at the sampling station for periodic checks of boron concentration.

The accumulators are passive engineered safety features because the gas forces injection; no external source of power or signal transmission is needed to obtain fast-acting, high-flow capability when the need arises. One accumulator is attached to each of the cold legs of the Reactor Coolant System.

The design capacity of the accumulators is based on the assumption that flow from one of the accumulators spills onto the containment floor through the ruptured loop. The flow from the three remaining accumulators provides sufficient water to fill the volume outside of the core barrel below the nozzles, the bottom plenum, and one-half the core.

The accumulators are carbon steel, clad with stainless steel and designed to ASME Section III, Class C. Connections for remotely draining or filling the fluid space, during normal plant operation, are provided.

Redundant level and pressure indicators are provided with read outs on the control board. Each indicator is equipped with high and low level alarms.

The accumulator design parameters are given in Table 6.2-1.

Boron Injection Tank

The boron injection tank constructed of stainless steel, is located in the auxiliary building. Nitrogen is supplied to the space over the 12% boric acid solution. Nitrogen overpressure, which is maintained automatic provides the motive force to eject the boric acid solution into the safety injection pump suction header when the isolation valves open.

Redundant electric tank heaters and redundant line heat tracing are provid to ensure that the solution will be stored at a temperature in excess of t solubility limie (130°F at a concentration of 20,000 ppm boron). The heat elements are located near the bottom of the tank.

The design paremeters are presented in Table 6.2.-2.

Refueling Water Storage Tank

In addition to its normal duty to supply borated water to the refueling canal for refueling operations, this tank provides borated water to the safety injection pumps, the residual heat removal pumps and the containmen spray pumps for the loss-of-coolant accident. During plant operation, it is aligned to these pumps.

The capacity of the refueling water storage tank is based on the requireme for filling the refueling canal and a minimum of 350,000 gallons is availa for delivery. This capacity provides an amount of borated water to assure

a) A volume sufficient to refill the reactor vessel above the nozzles (21,000 gal.)

b) The volume of borated refueling water needed to increase the concentration of initially spilled water to a point that assures no return to criticality with the reactor at cold shutdown and all control rods, except the most reactive RCC assembly, inserted into the core (50,000 gal.)

c) A sufficient volume of water on the floor to permit the initiation of recirculation (125,000 gal.)

The water in the tank is borated to a concentration which assures reactor shutdown by approximately 10% $\delta k/k$ when all RCC assemblies are inserted and when the reactor is cooled down for refueling. The maximum boric acid concentration is approximately 1.4 weight percent boric acid. At 32°F the solubility limit of boric acid is 2.2%. Therefore the concentration of boric acid in the refueling water storage tank is well below the solubility limit at 32°F. Steam heating is provided for the tank and the outside lines to maintain the temperature above freezing.

Two level indications with low level alarms are provided.

The design parameters are presented in Table 6.2-3.

Pumps

The three (high head) safety injection pumps for supplying borated water to the Reactor Coolant System are horizontal centrifugal pumps driven by electrical motors. Parts of the pump in contact with borated water are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on each pump discharge to recirculate flow to the refueling water storage tank in the event the pumps are started with the normal flow paths blocked. Each safety injection pump is sized at 50% of the capacity required to meet the design criteria outlined in Section 6.2.1. The design parameters are presented in Table 6.2-4 and Figure 6.2-7 gives the performance characteristics of these pumps.

The two residual heat removal (low head) pumps of the Auxiliary Coolant System are used to inject borated water at low pressure to the Reactor Coolant System. The two recirculation pumps are used to recirculate fluid from the recirculation sump and send it back to the reactor, the spray headers or to suction of the safety injection pumps. All four of these pumps are of the vertical centrifugal type, driven by electric motors. Parts of the pumps which contact the borated water and sodium hydroxide solution during recirculation are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on the discharge of the residual heat exchangers to recirculate cooled fluid to the suction of the residual heat removal pumps should these pumps be started with their normal flow paths blocked. A minimum flow bypass, discharging back into the recirculation sump, is provided to protect the recirculation pumps should these flow paths be blocked. Figures 6.2-8 and 6.2-9 give the performance characteristics of these pumps. The design parameters are presented in Table 6.2.4.

The safety injection pump bearings are cooled by booster pumps using component cooling water as a heat sink. The booster pumps are directly connected to the injection pump motor shaft. The pump seals are designed to operate at accident conditions without cooling water.

The recirculation pump motors are enclosed fan cooled. The air is cooled by coils utilizing component cooling water and two auxiliary component cooling pumps located outside the containment. During recirculation the sump water cools the pump bearings. The two auxiliary component cooling pumps are started during the injection phase and either is capable of protecting the recirculation pump motors from the containment atmosphere. The fans are directly connected to the recirculation pump motor shafts. The auxiliary component cooling pumps are a part of the component cooling water system and pump data is provided in Section 9. The component cooling water volume constitues a large heat sink so that the main component cooling pumps are not needed during the injection phase.

Details of the component cooling pumps and service water pumps, which serve the Safety Injection System, are presented in Section 9.

The pressure containing parts of the high head safety injection pumps are castings conforming to ASTM A-296 Grade CA-15. The pressure containing parts of the Residual Heat Removal Pumps and the Recirculation Pumps are castings conforming to ASTM-296 Grade CF-8a* (*chromium content 21.0 to 22.5) and ASTM-296 Grade CF-8 respectively. Stainless steel forgings are procured per ASTM A-182 Grade F304 or F316 or ASTM A-336, Class F8 or F8M, and stainless plate is constructed to ASTM A-240, Type 304 or 316. All bolting material conforms to ASTM A-193. Materials such as weld-deposited Stellite or Colmonoy are used at points of close running clearances in the pumps to prevent galling and to ensure continued performance ability in high velocity areas subject to erosion.

All pressure containing parts of the pumps are chemically and physically analyzed and the results are checked to ensure conformance with the applicable ASTM specification. In addition, all pressure containing parts of the pump are liquid penetrant inspected in accordance with Appendix VIII of Section VIII of the ASME Boiler and Pressure Vessel Code. The acceptance standard for the liquid penetrant test is USAS B31.1, Code for Pressure Piping, Case N-10.

The pump design is reviewed with special attention to the reliability and maintenance aspects of the working components. Specific areas include evaluation of the shaft seal and bearing design to determine that adequate allowances have been made for shaft deflection and clearances between stationary parts.

Where welding of pressure containing parts is necessary, a welding procedure including joint detail is submitted for review and approval by Westinghouse. The procedure includes evidence of qualification necessary for compliance with Section IX of the ASME Boiler and Pressure Vessel Code Welding Qualifications. This requirement also applies to any repair welding performed on pressure containing parts.

The pressure-containing parts of the pump are assembled and hydrostatically tested to 1.5 times the design pressure for 30 minutes.

Each pump is given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps are run at design flow and head, shut-off head and three additional points to verify performance characteristics. Where NPSH is critical, this value is established at design flow by means of adjusting suction pressure.

Heat Exchangers

The two residual heat exchangers of the Auxiliary Coolant System cool the recirculated sump water. These heat exchangers are sized for the cooldown of the Reactor Coolant System. Table 6.2-5 gives the design parameters of the heat exchangers.

The ASME Boiler and Pressure Vessel Code has strict rules regarding the wall thicknesses of all pressure containing parts, material quality assurance provisions, weld joint design, radiographic and liquid penetrant examination of materials and joints, and hydrostatic testing of the unit as well as requiring final inspection and stamping of the vessel by an ASME Code inspector.

The designs of the heat exchangers also conform to the requirements of TEMA (Tubular Exchanger Manufacturers Association) for Class R heat exchangers.

Class R is the most rugged class of TEMA heat exchangers and is intended for units where safety and durability are required under severe service conditions. Items such as: tube spacing, flange design, nozzle location, baffle thickness and spacing, and impingement plate requirements are set forth by TEMA Standards.

In addition to the above, additional design and inspection requirements were imposed to ensure rugged, high quality heat exchangers such as: confined-type gaskets, main flange studs with two nuts on each end to ensure permanent leak tightness, general construction and mounting brackets suitable for the plant seismic design requirements, tubes and tube sheet capable of withstanding full shell side pressure and temperature with atmospheric pressure on the tube side, ultrasonic inspection in accordance with Paragraph N-324.3 of Section III of the ASME Code of all tubes before bending, penetrant inspection in accordance with Paragraph N-627 of Section III of the ASME Code of all welds and hot or cold formed parts, a hydrostatic test duration of not less than thirty minutes, the witnessing of hydro and penetrant tests by a qualified inspector, a thorough final inspection of the unit for good workmanship and the absence of any gouge marks or other scars that could act as stress concentration points, a review of the radiographs and of the certified chemical and physical test reports for all materials used in the unit.

The Residual Heat Exchangers are conventional vertical shell and U-tube type units. The tubes are seal welded to the tube sheet. The shell connections are flanged to facilitate shell removal for inspection and cleaning of the tube bundle. Each unit has a SA-285 Grade C carbon steel shell, a SA-234 carbon steel shell end cap, SA-213 TP-304 stainless steel tubes, SA-240 type-304 stainless steel channel, SA-240 Type 304 stainless steel channel cover and a SA-240 Type 304 stainless steel tube sheet.

Valves

All parts of valves used in the Safety Injection System in contact with borated water are austenitic stainless steel or equivalent corrosion resistant material. The motor operators on the injection line isolation valves are capable of rapid operation. All valves required for initia- of safety injection or isolation of the system have remote position indication in the control room.

Valving is specified for exceptional tightness and, where possible, such as instrument valves, packless diaphragm valves are used. All valves, except those which perform a control function, are provided with backseats which are capable of limiting leakage to less than 1.0 cc per hour per inch of stem diameter, assuming no credit taken for valve packing. Those valves which are normally open, are backseated. Normally closed globe valves are installed with recirculation flow under the seat to prevent leakage of recirculated water through the valve stem packing. Relief valves are totally enclosed. Control and motor-operated valves, 2 1/2" and above, which are exposed to recirculation flow, are provided with double-packed stuffing boxes and stem leakoff connections which are piped to the Waste Disposal System.

The check valves which isolate the Safety Injection System from the Reactor Coolant System are installed immediately adjacent to the reactor coolant piping to reduce the probability of a safety injection line rupture causing a loss-of-coolant accident.

A relief valve is installed in the safety injection pump discharge header discharging to the pressurizer relief tank, to prevent overpressure in the lines which have a lower design pressure than the Reactor Coolant System. The relief valve is set at the design pressure of the safety injection piping.

The gas relief valves on the accumulators protect them from pressures in excess of the design value.

Motor Operated Butterfly Valves (Containment Sump Valves)

The pressure containing parts (body, discs) of the valve employed in the Safety Injection System are designed per criteria established by the USAS B16.5 or MSS SP67 specifications. The materials of construction for these parts are procured per ASTM A182, F316 or A351, Gr CF8M or CF8. All material in contact with the primary fluid, except the packing and the liner is austenitic stainless steel or equivalent corrosion resisting material. The liner is EPT-NORDEL (Dupont). The pressure containing cast components are radiographically inspected as outlined in ASTM E-71 Class 1 or Class 2. The body and disc are liquid penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code Section VIII, Appendix VIII. The liquid penetrant acceptable standard is as outlined in USAS B31.1 Case N-10.

The entire assembled unit is hydrotested as outlined in MSS SP-67 with the exception that the test in maintained for a minimum period of 30 minutes. The motor operator has to work only against the frictional component of the hydraulic unbalance on the disc and the packing box friction.

The shaft material is ASTM A276 Type 316 condition B or precipitation hardened 17-4 PH stainless procured and heat treated to Westinghouse Specifications. These materials are selected because of their corrosion resistance, high tensile properties, and their resistance to surface scoring by the packing.

The motor operator is located above the maximum sump fluid level and therefore is never submerged. The motor operator is extremely rugged and is noted throughtout the power industry for its reliability. The unit incorporates a "hammer blow" feature that allows the motor to impact the discs away from the fore or backseat upon opening or closing. This "hammer blow" feature not only impacts the disc but allows the motor to attain its operational speed.

The valve is assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned and packaged per specifications. All manufacturing procedures employed by the valve supplier such as welding, repair welding, and testing are submitted to Westinghouse for approval.

The valve operator completes its cycle from one position to the other in 120 seconds.

Valves which must function against system pressure are designed such that they function with a pressure drop equal to full system pressure across the valve disc.

Motor Operated Gate Valves

The pressure containing parts (body, bonnet and discs) of the valves empl in the safety Injection system are designed per criteria established by t USAS B16.5 or MSS SP66 specifications. The materials of construction for these parts are procured per ASTM A182, F316 or A351, Gr CF8M or CF8. All material in contact with the primary fluid, except the packing, is austenitic stainless steel or equivalent corrosion resisting material. The pressure containing cast components are radiographically inspected as outlined in ASTM E-71 Class 1 or Class 2. The body, bonnet and discs are liquid penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code Section VIII, Appendix VIII. The liquid penetrant acceptable standard is as outlined in USAS B31.1 Case N-10.

When a gasket is employed the body-to-bonnet joint is designed per ASME Boiler and Pressure Vessel Code Section VIII or USAS B16.5 with a fully trapped, controlled compression, spiral wound asbestos gasket with provisions for seal welding, or of the pressure seal design with provisions for seal welding. The body-to-bonnet bolting and nut materials are procured per ASTM A193 and A194, respectively.

The entire assembled unit is hydrotested as outlined in MSS SP-61 with the exception that the test is maintained for a minimum period of 30
14 minutes. The seating design is of the Darling parallel disc design, the Crane flexible wedge design, or the equivalent. These designs have the feature of releasing the mechanical holding force during the first increment of travel. Thus, the motor operator has to work only against the frictional component of the hydraulic unbalance on the disc and the packing box friction. The discs are guided throughout the full disc travel to prevent chattering and provide ease of gate movement. The seating surfaces are hard faced (Stellite No. 6 or equivalent) to prevent galling and reduce wear.

The stem material is ASTM A276 Type 316 condition B or precipitation hardened 17-4 PH stainless procured and heat treated to Westinghouse Specifications. These materials are selected because of their corrosion resistance, high tensile properties, and their resistance to surface scoring by the packing. The valve stuffing box is designed with a lantern ring leak-off connection with a munimum of a full set of packing below the lantern ring and a maximum of one-half of a set of packing above the lantern ring; a full set of packing is defined as a depth of packing equal to 1-1/2 times the stem diameter. The experience with this stuffing box design and the selection of packing and stem materials has been very favorable in both conventional and nuclear power plants.

The motor operator is extremely rugged and is noted throughout the power industry for its reliability. The unit incorporates a "hammer blow" feature that allows the motor to impact the discs away from the fore or backseat upon opening or closing. This "hammer blow" feature not only impacts the disc but allows the motor to attain its operational speed.

The valve is assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned and packaged per specifications. All manufacturing procedures employed by the valve supplier such as hard facing, welding, repair welding and testing are submitted to Westinghouse for approval.

For those valves which must function on the safety injection signal, 10 seconds operation is required. For all other valves in the system, the valve operator completes its cycle from one position to the other in 120 seconds.

Valves which must function against system pressure are designed such that they function with a pressure drop equal to full system pressure across the valve disc.

Manual Valves

The stainless steel manual globe, gate and check valves are designed and built in accordance with the requirements outlined in the motor operated valve description above.

The carbon steel valves are built to conform with USAS B16.5. The materials of construction of the body, bonnet and disc conform to the requirements of ASTM A105 Grade II, A181 Grade II or A216 Grade WCB or WCC. The carbon steel valves pass only non-radioactive fluids and are subjected to hydrostatic test as outlined in MSS SP-61 except that the test pressure is maintained for at least 30 minutes per inch of wall thickness. Since, the fluid controlled by the carbon steel valves is not radioactive, the double packing and seal weld provisions are not provided.

Accumulator Check Valves

The pressure containing parts of this valve assembly are designed in accordance with MSS SP-66. All parts in contact with the operating fluid are of austenitic stainless steel or of equivalent corrosion resistant materials procured to applicable ASTM or WAPD specifications. The cast pressure-containing parts are radiographed in accordance with ASTM E-94 and the acceptance standard as outlined in ASTM E-71 Class 1 or Class 2. The cast pressure-containing parts, machined surfaces finished hard facings, and gasket bearing surfaces are liquid penetrant inspected per ASME B&PV Code, Section VIII and the acceptance standard is as outlined in USAS B31.1 Code Case N-10. The final valve is hydrotested per MSS SP-66 except that the test pressure is maintained for at least 30 minutes. The seat leakage is conducted in accordance with the manner prescribed in MSS SP-61 except that the acceptable leakage is 2cc/hr/in, nominal pipe diameter.

The valve is designed with a low pressure drop configuration with all operating parts contained within the body, which eliminates those problems associated with packing glands exposed to boric acid. The clapper arm shaft is manufactured from 17-4 PH stainless steel heat treated to Westinghouse Specifications. The clapper arm shaft bushings are manufactured from Stellite No. 6 material. The various working parts are selected for their corrosion resistant, tensile, and bearing properties.

The disc and seat rings are manufactured from a forging. The mating surfaces are hard faced with Stellite No. 6 to improve the valve seating life. The disc is permitted to rotate, providing a new seating surface after each valve opening.

The valves are intended to be operated in the closed position with a normal differential pressure across the disc of approximately 1700 psi. The valves shall remain in this position except for testing and safety injection. Since the valve will not be required to normally operate in the open condition, hence be subjected to impact loads caused by sudden flow reversal, it is expected that this equipment will not have difficulties performing its required functions.

When the valve is required to function a differential pressure of less than 25 psig will shear any particles that may attempt to prevent the valve from functioning. Although the working parts are exposed to the boric acid solution contained within the reactor coolant loop, a boric acid "freeze up" is not expected with this low a concentration.

The experience derived from the check valves employed in the Safety Injection System of the Carolina - Virginia Tube Reactor in a similar system indicates that the system is reliable and workable.

The CVTR Emergency Injection System, maintained at atmospheric conditions is separated from the main coolant piping by one six inch check valve. A leak detection pit is provided in CVTR to accumulate any leakage coming back through the check valve. A level alarm provides a signal on excessive leakage. There is a gas volume in the upper space of the loop. The pressure differential is 1500 psi and the system is stagnant. The valve is located 2 to 3 feet from the main coolant piping which results in some heatup and cooldown cycling. The CVTR went critical late in 1963. Since that time, the level alarm in the detection pit has never gone off due to check valve leakage.

Relief Valves

The accumulator relief valves are sized to pass nitrogen gas at a rate in excess of the accumulator gas fill line delivery rate. The relief valves will also pass water in excess of the expected leak rate, but this is not necessary because the time required to fill the gas space gives the operator ample opportunity to correct the situation. For an inleakage rate 15 times the manufacturing test rate, there will be about 1000 days before water will reach the relief valves. Prior to this, level and pressure alarms would have been actuated.

The safety injection test line relief valves are provided to relieve any pressure, above design, that might build up in the high head safety injection piping. The valve will pass a nominal 15 gpm (2.25×10^5 cc/hr), which is far in excess of the manufacturing design leak rate of 24 cc/hr.

Leakage Limitations of Valves

Valving is specified for exceptional tightness and where possible, such as instrument valves, packless diaphragm valves are used.

Normally open valves have backseats which limit leakage to less than one cubic centimeter per hour per inch of stem diameter assuming no credit for packing in the valve. Normally closed globe valves are installed with recirculation flow under the seat to prevent stem leakage from the more radioactive fluid side of the seat.

Motor operated valves, which are exposed to recirculation flow, are provided with double-packed stuffing boxes and stem leakoff connections which are piped to the waste disposal system.

The specified leakage across the valve disc required to meet the equipment specification and hydrotest requirements is as follows:

a) Conventional globe - 3cc/hr/in. of nominal pipe size

b) Gate valves - 3cc/hr/in. of nominal pipe size; 10/cc/hr/in for 300 and 150 pound USA Standard

c) Motor-operated gate valves - 3 cc/hr/in. of nominal pipe size: 10/cc/hr/in for 300 and 150 pound USA Standard

d) Check valves - 3 cc/hr/in. of nominal pipe size; 10/cc/hr/in for 300 and 150 pound USA Standard

e) Accumulator check valves - 2 cc/hr/in. of nominal pipe size; Relief valves are totally enclosed

Leakage from components of the recirculation loop including valves, is tabulated in Table 6.2-10.

Piping

All Safety Injection System piping in contact with borated water is austenitic stainless steel. Piping joints are welded except for the flanged connections at the safety injection pumps and recirculation pumps.

The piping beyond the accumulator stop valves is designed for Reactor Coolant System conditions (2485 psig, 650°F). All other piping connected to the accumulator tanks is designed for 700 psig and 400°F.

The safety injection pump and residual heat removal pump suction piping (210 psig at 300°F) from the refueling water storage meets NPSH (net positive suction head) requirements of the pumps.

The safety injection high pressure branch lines (1500 psig at 300°F) are designed for high pressure losses to limit the flow rate out of the branch line which may have ruptured at the connection to the reactor coolant loop. The system design incorporates the ability to isolate the safety injection pumps on separate headers such that full flow from at least one pump is ensured should a branch line break.

The piping is designed to meet the minimum requirements set forth in (1) the USAS B31.1 Code (1955) for the Pressure Piping, (2) Nuclear Code Case N-7, (3) USAS Standards B36.10 and B36.19 and (4) ASTM Standards with supplementary standards plus additional quality control measures.

Minimum wall thicknesses are determined by the USAS Code (1955) formula found in the power piping Section 1 of the USAS Code (1955) for Pressure Piping. This minimum thickness is increased to account for the manufacturer's permissible tolerance of minus 12-1/2 per cent on the nominal wall. Purchased pipe and fittings have a specified nominal wall thickness that is no less than the sum of that required for pressure containment, mechanical strength and manufacturing tolerance.

Thermal and seismic piping flexibility analyses are performed. Special attention is directed to the piping configuration at the pumps with the objective of minimizing pipe imposed loads at the suction and discharge nozzles. Piping is supported to accommodate expansion due to temperature changes during the accident.

Pipe and fittings materials are procured in conformance with all requirements of the ASTM and USAS specifications. All materials are verified for conformance to specification and documented by certification of compliance to ASTM material requirements. Specifications impose additional quality control upon the suppliers of pipes and fittings as listed below.

a) Check analyses are performed on both the purchased pipe and fittings.

b) Pipe branch lines 2 1/2 inch and longer between the reactor coolant pipes and the isolation stop valves conform to ASTM A376 and meet the supplementary requirement S6 ultrasonic testing. Fittings conform to the requirements of ASTM A403. Fittings 2 1/2 inch and above have requirements for UT inspection similar to S6 of A376.

Shop fabrication of piping subassemblies is performed by reputable suppliers in accordance with specifications which define and govern material procurement, detailed design, shop fabrication, cleaning, inspection, identification, packaging and shipment.

Welds for pipes sized 2-1/2" and larger are butt welded. Reducing tees are used where the branch size exceeds 1/2 of the header size. Branch connections of sizes that are equal to or less than 1/2 of the header size are of a design that conforms to the USAS rules for reinforcement set forth in the USAS B31.1 Code for Pressure Piping. Bosses for branch connections are attached to the header by means of full penetration welds.

All welding is performed by welders and welding procedures qualified in accordance with the ASME Boiler and Pressure Vessel Code Section IX, Welding Qualifications. The Shop Fabrication is required to submit all welding procedures and evidence of qualifications for review and approval prior to release for fabrication. All welding materials used by the Shop Fabricator must have prior approval.

All high pressure piping butt welds containing radioactive fluid, at greater than 600°F temperature and 600 psig pressure or equivalent, are radiographed. The remaining piping butt welds are randomly radiographed. The technique and acceptance standards are those outlined in UW-51 of the ASME B&PV Code Section VIII. In addition, butt welds are liquid penetrant examined in accordance with the procedure of ASME B&PV Code, Section VIII, Appendix VIII and the acceptance standard as defined in the USAS Nuclear Code Case N-10. Finished branch welds are liquid penetrant examined on the outside and where size permits, on the inside root surfaces.

A post-bending solution anneal heat treatment is performed on hot formed stainless steel pipe bends. Completed bends are then completely cleaned of oxidation from all affected surfaces. The shop fabricator is required to submit the bending, heat treatment and clean-up procedures for review and approval prior to release for fabrication.

General cleaning of completed piping subassemblies (inside and outside surfaces) is governed by basic ground rules set forth in the specifications. For example, these specifications prohibit the use of hydrochloric acid and limit the chloride content of service water and demineralized water.

Packaging of the piping subassemblies for shipment is done so as to preclude damage during transit and storage. Openings are closed and sealed with tight-fitting covers to prevent entry of moisture and foreign material. Flange facings and weld end preparations are protected from damage by means of wooden cover plates and securely fastened in position. The packing arrangement proposed by the shop fabricator is subject to approval.

Pump and Valve Motors

Motors Outside the Containment

Motor electrical insulation systems are supplied in accordance with USAS, IEEE and NEMA standards and are tested as required by such standards.

Temperature rise design selection is such that normal long life is achieved even under accident loading conditions.

Criteria for motors of the Safety Injection System require that under any anticipated mode of operation, the motor name plate rating is not exceeded. The motors have a 1.15 service factor for normal operation. Design and test criteria ensure that motor loading does not exceed the application criteria.

Motors Inside the Containment

The recirculation pump motors are designed to operate in an ambient condition of saturated steam of 271°F and 47 psig pressure for one day, followed by indefinite operation at 155°F and 5 psig in a steam atmosphere.

The recirculation pump's motors are of the same design as the containment fan cooler motors. Reference is therefore made to Section 6.4.2 for a description and evaluation of the motor design.

The motors for the valves inside containment are designed to withstand containment environment conditions following the loss of coolant accident so that the valves can perform the required function during the recovery period.

Periodic operation of the motors and tests of the insulation ensure that the motors remain in a reliable operating condition.

Although the motors which are provided only to drive engineered safety features equipment are normally run only for test, the design loading and temperature rise limits are based on accident conditions. Normal design margins are specified for these motors to make sure the expected lifetime includes allowance for the occurrence of accident conditions.

Valve Motor Operators

Tests to demonstrate the adequacy of valve motor operators to be functional after exposure to temperature, pressure and radiation are being conducted in two groups.

The first group is the exposure of valve motor operators to both temperature and pressures. Two suppliers, Philadelphia Gear Corporation Limitorque Division and Crane Co. Teledyne Division are conducting simulated containment pressure and temperature tests as follows with power-temperature similar to that predicted for the incident:

a) Operator is located inside a pressure vessel which is to exposed to approximately 330°F at 90 psig.

b) Operator will be cycled approximately three times under simulated valve operating loads.

c) Pressures and temperatures will be reduced in step changes to 285°F at 60 psig, 219°F at 20 psig and 152°F at atmosphere or less.

d) Operator will be cycled approximately three times at each of the levels of change. Full recordings of pertinent data will be taken throughout the tests.

e) Unit shall be examined after completion of test and operating data compared to data prior to exposure.

The second group test is the radiation test on a motor from the valve operator.

a) Two production line motors are being used for this test. One is to be exposed to 1.5×10^8 rads of gamma radiation for approximate period of one month. The other motor will be used for the final comparative analysis.

b) Both units will be tested for coil resistance, insulation meggering both before and after motor vibration and reversing operations.

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Electrical Supply

Detail of the normal and emergency power sources for the Safety Injection System are presented in Section 8.

Protection Against Dynamic Effects

The injection lines penetrate the containment adjacent to the primary auxiliary building.

For most of the routing, these lines are outside the crane wall, and hence are protected from missiles originating within these areas. Each line penetrates the crane wall near the injection point to the reactor coolant pipe. In this manner maximum separation and hence protection is provided in the coolant loop area.

Coolant loop supports are designed to restrict the motion in one loop due to rupture in another to about one-tenth of an inch, whereas the attached safety injection piping can sustain a 3 inch displacement without exceeding the working stress range. The analysis assumes the injection flow to the ruptured loop is spilled on the containment floor.

All hangers, stops and anchors are designed in accordance with USAS B31.1 Code for Pressure Piping and ACI 318 Building Code Requirements for Reinforced Concrete which provide minimum requirements on materials, design and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment. Specifically, these standards require the following:

(1) All materials used are in accordance with ASTM specifications which establish quality levels for the manufacturing process, minimum strength properties, and for test requirements which ensure compliance with the specifications.

(2) Qualification of welding processes and welders for each class of material welded and for types and positions of welds.

(3) Maximum allowable stress values are established which provide an ample safety margin on both yield strength and ultimate strength.

6.2.3 DESIGN EVALUATION

<u>Range of Core Protection</u>

The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of this performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met (Section 6.2.1). The range of core protection as a function of break diameter provided by the various components of the Safety Injection System is presented in Figure 6.2.6.

Figure 6.2.6 was developed from the results of the loss of coolant accident studies presented in Section 14.3.1. Simulations of a sufficient number of break sizes were performed to demonstrate that the Safety Injection components meet the emergency core cooling requirements. The injection from the following combination of components was analyzed as discussed below:

- Bar A. Two Safety Injection Pumps
- Bar B. Two Safety Injection Pumps, One Residual Heat Removal Pump and Two Accumulators
- Bar C. One Residual Heat Removal Pump and Three Accumulators
- Bar D. Two Safety Injection Pumps and Three Accumulators
- Bar E. Two Safety Injection Pumps, One Residual Heat Removal Pump and Three Accumulators

Note: For all of the cases, one recirculation pump is required for long term recirculation.

No credit is taken for the accumulator which is attached to the ruptured leg in the case of a cold leg break.

With minimum on-site emergency power available (two-of-three diesel generators) the emergency core cooling equipment available is represented by Bar E (two out of three safety injection pumps, one out of two residual heat pumps, and three out of four accumulators for a cold leg break and four accumulators for a hot leg break). With these systems, the calculated maximum fuel cladding temperature is limited to 2120°F, which meets the emergency core cooling design objectives for all break sizes up to and including the double-ended severance of the reactor coolant pipe. (Section 14.3.2).

The remaining four combinations (Bar A, B, C, and D) represent degraded cases with operation of less than the minimum design emergency core cooling equipment. These cases are shown only to present the capability of individual portions of the systems and to demonstrate the overall margins of the system. The operation of two safety injection pumps, (Bar A) provides core protection for break sizes up to an equivalent break diameter of approximately 4 inches. The operation of two safety injection pumps would allow flow spilling from a broken safety injection line to go uncorrected by operator action. Isolation of the broken line by operator action would increase the range of protection.

In Section 14.3.2 an analysis is presented that demonstrated the ability of the two high pumps to keep the core hot spot covered for all breaks up to a 4" equivalent diameter hole. When the hot spot remains covered, no core damage is expected.

An accident occuring with one accumulator isolated from the system is represented by (Bar B). From the detailed break size study used in arriving at the minimum safeguards case (Bar E) it was determined that the double ended cold leg break caused the highest clad temperature. This case was reanalyzed for the Bar B equipment, and the resulting clad temperature was 2550°F, which is well below the Zircaloy-4 melt temperature.

The cases represented in Bars C and D are presented to demonstrate the redundancy between the low and high head pumping systems to ground protection for large area rupture. For large area ruptures analyzed (14.3.2) the clad temperatures are turned around by the accumulator injection. The active pumping components serve only to complete the refill started by the accumulators. Either two safety injection pumps or one residual heat removal pump provides sufficient addition of water to continue the reduction of clad temperature initially caused by the accumulator.

System Response

To provide protection for large area ruptures in the Reactor Coolant System the Safety Injection System must respond to rapidly reflood the core following the depressurization and core voiding that is characteristic of large area ruptures. The accumulators act to perform the rapid reflooding function with no dependence on the normal or emergency power sources, and also with no dependence on the receipt of an actuation signal.

Operation of this system with three of the four available accumulators delivering their contents to the reactor vessel (one accumulator spilling through the break) prevents fuel clad melting and limits metal-water reaction to an insignificant amount (<1%).

The function of the safety injection or residual heat removal pumps is to complete the refill of the vessel and ultimately return the core to a subcooled state. As discussed on page 6.2-37 through 6.2-38, the flow from either two safety injection pumps or one residual heat removal pump is sufficient to complete the refill with no loss of level in the core. Moreover, there is sufficient excess water delivered by the accumulators to tolerate a delay in starting the pumps.

Initial response of the injection systems is automatic, with appropriate allowances for delays in actuation of circuitry and active components. The active portions of the injection systems are automatically actuated by the safety injection signal (Section 7). In addition, manual actuation of the entire injection system and individual components can be accomplished from the control room. In analysis of system performance, delays in reaching the programmed trip points and in actuation of components are conservatively established on the basis that only emergency on-site power is available.

The starting sequence of the safety injection and residual heat removal pumps and the related emergency power equipment is designed so that delivery of the full rated flow is reached within 34 seconds after the process parameters reach the set points for the injection signal.

The delay time consists of the time intervals:

		Seconds
a.	To initiate safety injection signal, including instrument lag	1
b.	To start two diesel-generators	19
c.	To start two safety injection pumps	8
d.	To start one residual heat removal pump	6
	Total	34

Motor control centers are energized and injection valves are opened at the same time as the pumps are started.

Although this delay is in excess of the 25 second delay which is assumed in the analyses of the loss-of-coolant accident as described in Section Section 14, it is emphasized that:

a) The delay internals which comprise the 34 second total delay are conservative estimates of what will be achieved in practice.

b) As described in Section 14, core recovery following a loss-of-coolant accident is achieved by the passive accumulator system prior to the operation of the safety injection system. An increase in the delay time to initiate safety injection flow of the order of 10 seconds has a negligible effect on the loss-of-coolant transient.

Single Failure Analysis

A single active failure analysis is presented in Table 6.2-7(a). All credible active system failures are considered. The analysis of the loss-of-coolant accident presented in Section 14 is consistent with the single failure analysis.

It is based on the worst single failure (generally a pump failure) in both the safety injection and residual heat removal pumping systems. The analysis shows that the failure of any single active component will not prevent fulfilling the design function.

In addition, an alternative flow path is available to maintain core cooling if any part of the recirculation flow path becomes unavailable. This is evaluated in Table 6.2-6(b).

The procedure followed to establish the alternate flow path also isolates the spilling line. A valve is provided in the containment recirculation line to the residual heat removal pumps, to isolate this line should it be required.

Failure analyses of the component cooling and service water system under loss-of-coolant accident conditions are described in Section 9.3 and 9.6, respectively.

Reliance on Interconnected Systems

During the injection phase, the high head safety injection pumps do not depend on any portion of other systems with the exception of the suction line from the refueling water storage tank. During the recirculation phase of the accident for small breaks, suction to the high head safety injection pumps is provided by the recirculation pumps.

The residual heat removal (low head) pumps are normally used during reactor shutdown operations. Whenever the reactor is at power, the pumps are aligned for emergency duty.

Shared Function Evaluation

Table 6.2-8 is an evaluation of the main components, which have been previously discussed, and a brief description of how each component functions during normal operation and during the accident.

Passive Systems

The accumulators are a passive safety feature in that they perform their design function in the total absence of an actuation signal or power source. The only moving parts in the accumulator injection train are in the two check valves.

The working parts of the check valves are exposed to fluid of relatively low boric acid concentration. Even if some unforeseen deposition accumulated, a reversed differential pressure of about 25 psi can shear any particles in the bearing that may tend to prevent valve functioning. This is demonstrated by calculation.

The isolation valve at each accumulator is only closed when the reactor is intentionally depressurized or momentarily for testing when pressurized. The isolation valve is normally opened and an alarm in the control room sounds if the valve is inadvertently closed. It is not expected that the isolation valve will have to be closed due to excessive leakage through the check valves.

The check valves operate in the closed position with a nominal differential pressure across the disc of approximately 1650 psi. They remain in this position except for testing or when called upon to function. Since the valves operate normally in the closed position and are therefore not subject to the abuse of flowing operation or impact loads caused by sudden flow reversal and seating, they do not experience any wear of the moving parts, and therefore function as required.

When the Reactor Coolant System is being pressurized during the normal plant heatup operation, the check valves are tested for leakage as soon as there is about 100 psi differential across the valve. This test confirms the seating of the disc and whether or not there has been an increase in the leakage since the last test. When this test is completed, the discharge line test valves are opened and the Reactor Coolant System pressure increase is continued. There should be no increase in leakage from this point on since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage.

The accumulators can accept leakage from the Reactor Coolant System without effect on their availability. Table 6.2-8 indicates what inleakage rates, over a given time period, require readjusting the level at the end of the time period. In addition, these rates are compared to the maximum allowed leak rates for manufacturing acceptance tests (20 cc/hr, i.e., 2/cc/hr/in).

In-leakage at a rate of 5 cc/hr/inch, 2-1/2 times test, would require that the accumulator water volume be adjusted approximately once every 30 months. This would indicate that level adjustments can be acheduled for normal refueling shutdowns and that this work can be done at the operator's convenience. At a leakrate of 30 cc/hr/inch (15 times the acceptance leakrate), the water level will have to be readjusted approximately once every 5 to 6 months. This readjustment will take about 2 hours maximum.

The accumulators are located inside the reactor containment and protected from the reactor coolant system piping and components by a missile barrier. Accidential release of the gas charge in the accumulators would cause an increase in the containment pressure of approximately 0.1 psi. This release of gas has been included in the containment pressure analysis for the loss-of-coolant accident, Section 14.

During normal operation, the flow rate through the reactor coolant piping is approximately five times the maximum flow rate from the accumulator during injection. Therefore, fluid impingement on reactor vessel components during operation of the accumulator is not restricting.

Emergency Flow to the Core

Special attention is given to factors that could adversely affect the accumulator and safety injection flow to the core. These factors are:

a) Steam binding in the core, including flow blockage due to loop sealing

b) Loss of accumulator water during blowdown

c) Short circuiting of the accumulator from the core to another part of the Reactor Coolant System

d) Loss of accumulator water through the breaks

All of the above are considered in the analysis, and are discussed quantitatively in Section 14.

External Recirculation Loop Leakage

Table 6.2-10 summarizes the maximum potential leakage from the leak sources of the external recirculation loop which goes through the residual heat removal pumps, a residual heat exchanger and the high head safety injection pumps. In the analysis, a maximum leakage is assumed from each leak source. For conservatism, three times the maximum expected leak rate from the pump seals was assumed, even though the seals are acceptance tested to essentially zero leakage, and a leakage of 10 drops per minute was assumed from each flange although each flange would be adjusted to essentially zero leakage. The total maximum potential leakage resulting from all sources is 999 cc/hr to the auxiliary building atmosphere and 21 cc/hr to the drain tank.

During external recirculation, significant margin exists between the design and operating conditions of the residual heat removal system components, as shown in Table 6.2-11. In addition, during normal plant cooldown, operation of the residual heat removal system is initiated when the primary system pressure and temperature have been reduced to 350 psig and 350°F respectively. Since the maximum operating conditions during recirculation are 150 psig and 213°F, significant margin also exists between normal operating and accident conditions. In view of the above margins, it is considered that the leakage rates tabulated in Table 6.2-10 are conservative.

Leakage detection exterior to containment is achieved through use of sump level detection. The Auxiliary Building sump pumps start automatically in the event that liquid accumulates in the sump and alarm in the control room indicates that water has accumulated in the sump. Valving is provided to permit the operator to isolate individually the residual heat removal pumps.

Pump NPSH Requirements

Residual Heat Removal Pumps

The NPSH of the residual heat removal pumps is evaluated for normal plant shutdown operation, and both the injection and recirculation phase operation of the design basis accident.

The residual heat removal pumps are used as back-up to the internal recirculation pumps in the event of failures to the normal recirculation path; this duty provides the pumps with the minimum NPSH condition. For the design case of two pumps recirculating through two heat exchangers paths; 60% NPSH margin is available. In the unlikely event of only one pump recirculating through one heat exchanger path the NPSH margin would be reduced to 15%; assuming saturated fluid and no operator action to throttle back the flow.

Safety Injection Pumps

The NPSH for the safety injection pumps is evaluated for both the injection and recirculation phase operation of the design basis accident. The end of injection phase operation gives the limiting NPSH requirement and the NPSH available is determined from the elevation head and vapor pressure of the water in the refueling water storage tank, and the pressure drop in the suction piping from the tank to the pumps. At the end of the injection phase, 20% NPSH margin is available assuming all three pumps running together with two RHR pumps at run-out condition. 10

Recirculation Pumps

The NPSH for the recirculation pumps is evaluated for recirculation operation. The NPSH available is determined from the elevation head of the water above the pump inlet in the sump.

The internal recirculation pumps are conventional vertical condensate pumps which in the past have been used with NPSH control. This type control used the NPSH available in the condenser hot well to control the discharge condition of the pump thus resulting in continuous pump cavitation. In the IPP application no approach to cavitating conditions are anticipated for the
10 normal case of two pumps operating. If, however, only one pump is delivering through two heat exchangers with fully saturated fluid, the operator is advised to throttle back via the heat exchanger butterfly valves to avoid long term operation at or near cavitating conditions.

6.2.4 MINIMUM OPERATING CONDITIONS

The Technical Specifications, Section 15, establishes limiting conditions regarding the operability of the system when the reactor is critical.

6.2.5 INSPECTIONS AND TESTS

Inspection -

All components of the Safety Injection System are inspected periodically to demonstrate system readiness.

The pressure containing components are inspected for leaks from pump seals, valve packing, flanged joints and safety valves during system testing.

In addition, to the extent practical, the critical parts of the reactor vessel internals, pipes, valves and pumps are inspected visually or by boroscopic examination for erosion, corrosion, and vibration wear evidence, and for non-destructive test inspection where such techniques are desirable and appropriate.

Pre-operational Testing

Component Testing

Pre-operational performance tests of the components are performed in the manufacturer's shop. The pressure-containing parts of the pump are hydrostatically tested in accordance with Paragraph UG-99 of Section VIII of the ASME Code. Each pump is given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps are run at design flow and head, shut-off head and at additional points to verify performance characteristics. NPSH is established at design flow by means of adjusting suction pressure for a representative pump. This test is witnessed by qualified Westinghouse personnel.

The remote operated valves in the safety injection systems will be motor-operated. Shop tests for each valve include a hydrostatic pressure test, leakage tests, a check of opening and closing time, and verification of torque switch and limit switch settings. The ability of the motor operator to move the valve with the design differential pressure across the gate is demonstrated by opening the valve with an appropriate hydrostatic pressure on one side of the valve.

The recirculation piping and accumulators are initially hydrostatically tested at 150 per cent of design pressure. The service water and component cooling water pumps are tested prior to initial operation.

The service water and component cooling water pumps are tested prior to initial operation.

System Testing

An initial functional test of the core cooling portion of the safety injection systems is conducted during the hot-functional testing of the reactor coolant system before initial plant startup. The purpose of the initial systems test is to demonstrate the proper functioning of instrumentation and actuation circuits and to evaluate the dynamics of placing the system in operation. This test is performed following the flushing and hydrostatic testing of the system.

The functional test is performed with the water level below the safety injection set point in the pressurizer and with the Reactor Coolant System initially cold and at low pressure. The safety injection system valving is set to initially simulate the system alignment for plant power operation.

To initiate the test, the safety injection block switch is moved to the unblock position to provide control power allowing the automatic actuation of the safety injection relays from the low water level and low-pressure signals from the pressurizer instrumentation. Simultaneously, the breakers supplying outside power to the 480 volt buses are tripped manually and

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operation of the emergency diesel system automatically commences. The high-head safety injection pumps and the residual heat removal pumps are started automatically following the prescribed diesel loading sequence. The valves are operated automatically to align the flow path for injection into the reactor coolant system.

The rising water level in the pressurizer provides indication of systems delivery. Flow into the reactor coolant system terminates with the filling of the pressurizer, and the operation of the safety injection systems is terminated manually in the control room.

This functional test provides information to confirm valve operating times, pump motor starting times, the proper automatic sequencing of load addition to the emergency diesels, and delivery rates of injection water to the reactor coolant system.

The functional test is repeated for the various modes of operation needed to demonstrate performance at partial effectiveness, i.e., to demonstrate the proper loading sequence with two of the three emergency diesels, and to demonstrate the correct automatic starting of a second pump should the first pump fail to respond. These latter cases are performed without delivery of water to the reactor coolant system, but include starting of all pumping equipment involved in each test.

The systems are accepted only after demonstration of proper actuation and after demonstration of flow delivery and shutoff head within design requirements.

Flow is introduced into the Reactor Coolant loops through the accumulator discharge line to demonstrate operability of the check valves and remotely actuated stop valve and confirm L/D ratios of accumulator discharge lines used in the calculation.

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Post-Operational Testing

Component Testing

Routine periodic testing of the safety injection system components and all necessary support systems at power is planned. No inflow to the Reactor Coolant System will occur whenever the reactor coolant pressure is above 1500 psi. If such testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance to be performed without shutting down or reducing load under conditions defined in the Technical Specifications. These conditions include such matters as the period within which the component should be restored to service and the capability of the remaining equipment to meet safety limits within such a period.

The operation of the remote stop valves in the accumulator tank discharge line may be tested by opening the remote test valves just downstream of the stop valve. Flow through the test line can be observed on instruments and the opening and closing of the discharge line stop valves can be sensed on this instrumentation. Test circuits are provided to periodically examine the leakage back through the check valves and to ascertain that these valves seat whenever the reactor system pressure is raised.

It is expected that this test will be routinely performed when the reactor is being returned to power after an outage and the reactor pressure is raised above the accumulator pressure. If leakage through a check valve should become excessive, the isolation valve would be closed. (The safety injection actuation signal will cause this valve to open should it be in the closed position at the time of a loss-of-coolant accident.) The performance of the check valves has been carefully studied and it is concluded that it is highly unlikely that the accumulator lines would have to be closed because of leakage.

No inflow to the Reactor Coolant System will occur whenever the reactor coolant pressure above 1500 psi.

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The recirculation pumps are normally in a dry sump. These pumps are started periodically and allowed to reach full speed. Minimum flow testing of these pumps can be performed during refueling operations by filling the recirculation sump and opening the minimum valve on the discharge of the pump and directing the flow back to the sump. Those service water and component cooling pumps not running during normal operation may be tested by alternating with the operating pumps.

The content of the accumulators, the boron injection tank and the refueling water storage tank are sampled periodically to determine that the required boron concentration is present.

System Testing

System testing can be conducted during plant shutdown to demonstrate proper automatic operation of the Safety Injection System. A test signal is applied to initiate automatic action and verification made that the safety injection and residual heat removal pumps attain required discharge heads. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry. Isolation valves in the injection line will be blocked closed so that flow is not introduced into the reactor coolant system. The test is considered satisfactory if control board indication and visual observations indicate all components have operated and sequenced properly.

The accumulators and the injection piping up to the final isolation valve are maintained full of borated water while the plant is in operation. The accumulators and the high head injection lines are refilled with borated water as required by using the safety injection pumps to recirculate refueling water through the injection lines. A small test line is provided for the purpose in each injection header.

Flow in each of the safety injection headers and in the main flow line for the residual heat removal pumps is monitored by a local flow indicator. Pressure instrumentation is also provided for the main flow paths of the safety injection and residual heat removal pumps. Accumulator isolation valves are blocked closed for this test.

The eight-switch sequence for recirculation operation may be tested following the above injection phase test to demonstrate proper sequencing of valves and pumps. The recirculation pumps are blocked from starting during this test.

The external recirculation flow paths are hydrotested during periodic re-tests at the operating pressures. This is accomplished by running each pump which could be utilized during external recirculation (safety injection and residual heat removal pumps) in turn at near shutoff head conditions and checking the discharge and recirculation test lines. The suction lines are tested by running the residual heat removal pumps and opening the flow path to the safety injection pumps in the same manner as described above.

During the above test, all system joints, valve packings, pump seals, leakoff connections or other potential points of leakage are visually examined. Valve gland packing, pump seals, and flanges are adjusted or replaced as required to reduce the leakage to acceptable proportions. For power operated valves, final packing adjustments are made, and the valves are put through an operating cycle before a final leakage examination is made.

The entire recirculation loop except the recirculation line to the residual heat removal pumps is pressurized during periodic testing of the engineered safety features components. The recirculation line to the residual heat removal pump is capable of being hydrotested during plant shutdown and it is also leak tested at the time of the periodic retests of the containment.

TABLE 6.2-1

SAFETY INJECTION SYSTEM - CODE REQUIREMENTS

Component	Code
Refueling Water Storage Tank	AWWA D100-65
Residual Heat Exchanger	
Tube Side	ASME Section III Class C
Shell Side	ASME Section VIII
Accumulators	ASME Section III Class C
Boron Injection Tank	ASME Section VIII
Valves	USAS B16.5 (1955)
Piping	USAS B31.1 (1955)

TABLE 6.2-2

ACCUMULATOR DESIGN PARAMETERS

Number	4
Type	Stainless steel lined/ carbon steel
Design pressure, psig	700
Design temperature, °F	300
Operating temperature, °F	100 - 150
Normal Operating pressure, psig	650
Minimum Operating pressure, psig	600
Total volume ft^3	1100
Minimum water volume at operating conditions, ft^3	700
Boron concentration (as boric acid), ppm	2000
Relief Valve set point psig*	700

*The relief valves have soft seats and are designed and tested to ensure zero leakage at the normal operating pressure.

TABLE 6.2-3

BORON INJECTION TANK DESIGN PARAMETERS

Number	1
Total volume, gal.	2700
Minimum Volume at operating conditions (solution), gal.	2150
Boron concentration (as boric acid) nominal, ppm	20,000
Design pressure, psig	100
Design temperature, °F	250
Operating pressure, psig	70
Operating temperature, °F	150 - 180
Material	Stainless steel
Number of immersion heaters	2
Heater capacity, each, kw	3

TABLE 6.2-4

REFUELING WATER STORAGE TANK DESIGN PARAMETERS

Number	1
Material	Stainless Steel
Total volume, ft^3	395,000
Minimum volume, (solution) gal.	350,000
Normal pressure, psig	atmospheric
Operating temperature, °F	above freezing
Design pressure, psig	atmospheric
Design temperature, °F	120
Boron concentration (as boric acid), ppm	2000
Type of Heating	Steam

TABLE 6.2-5

PUMP PARAMETERS

Safety Injection Pump Design Parameters

Number	3
Design pressure, discharge, psig	1700
Design pressure, suction, psig	250
Design temperature, °F	300
Design flow rate, gpm	400
Max. flow rate, gpm	650
Design head, ft	2,500
Shutoff head, ft	3,500
Material	Austenitic stainless steel
Motor H. P.	400
Type	Horizontal centrifugal

Recirculation Pump Design Parameters

Number of pumps	2
Type	Vertical centrifugal
Design pressure, discharge, psig	250
Design temperature, °F	300
Design flow, gpm	3000
Design head, ft.	350
Material	Austenitic stainless steel
Maximum flow rate, gpm	4000
Shutoff head, ft	495
Motor H.P.	350
Type	Vertical centrifugal

TABLE 6.2-5 (Continued)

Residual Heat Removal Pump Design Parameters

Number of pumps	2
Type	Vertical centrifugal
Design pressure, discharge, psig	600
Design temperature, °F	400
Design flow, gpm	3000
Design head, ft.	350
Material	Austenitic Stainless Steel
Maximum flow rate, gpm	5500
Shutoff head, ft.	390
Motor H.P.	400
Type	Vertical centrifugal

TABLE 6.2-6

RESIDUAL HEAT EXCHANGERS DESIGN PARAMETERS

Number		2
Design heat duty, Btu/hr (Normal)		30.8×10^6
(Accident)		56.4×10^6
Design UA, Btu/hr/°F		1.1×10^6
Design Cycles (85°F – 350°F)		200
Type		Vertical Shell and U-tube
	Tube-side	Shell-side
Design pressure psig	600	150
Design flow, lb/hr	1.44×10^6	2.46×10^6
Inlet temperature, °F (Normal)	135	88.3
(Accident)	213	94.2
Outlet temperature, °F (Normal)	113.5	100.8
(Accident)	134.7	125.9

TABLE 6.2-7(a)

SINGLE ACTIVE FAILURE ANALYSIS SAFETY INJECTION SYSTEM

Component	Malfunction	Comments
A. Accumulator (injection phase)	Deliver to broken loop	Totally passive system with one accumulator per loop. Evaluation based on three accumulators delivering to the core and one spilling from ruptured loop.
B. Pump: (injection phase)		
1) Safety injection	Fails to start	Three provided. Evaluation based on operation of two.
2) Residual heat removal	Fails to start	Two provided. Evaluation based on operation of one.
3) Component cooling*	Fails to start	A total of 1 of 3 required during recirculation
4) Conventional service water	Fails to start	A total of 1 of 3 required during recirculation
5) Recirculation*	Fails to start	Two provided. One required to operate during recirculation.
6) Auxiliary component cooling pump	Fails to start	Two provided. One required to operate during injection

*Recirculation phase

TABLE 6.2-7(a) (Continued)

C. Automatically Operated Valves: (Open on Safety Injection Signal)-(injection phase)		
1) Boron injection tank isolation	Fails to open	Two parallel lines, one valve in either line is required to open.
2) Safety injection line isolation valve at the loops	Fails to open	Three out of four are assumed to open.
3) Residual heat removal line isolation valve at residual heat exchanger discharge	Fails to open	Two parallel lines, one valve in either line is required to open.
4) Isolation valve on component cooling water line from residual heat exchangers	Fails to open	Two parallel lines, one valve in either line is required to open.
D. Automatically Operated Valves: (Close on Low Level in Boron Injection Tank)-(injection phase)		
1) Isolation valve on boron injection tank discharge line	Fails to close	Three provided. Operation of two required.
E. Valves Operated From Control Room for Recirculation: (recirculation phase)		
1) Containment sump internal recirculation isolation	Fails to open	Two lines in parallel, one valve in either line is required to open.

TABLE 6.2-7(a) (Continued)

2) Safety injection pump suction valve at residual heat exchanger discharge	Fails to open	Two parallel lines, one valve in either line required to open.
3) Isolation valve on the miniflow line returning to the refueling water storage tank	Fails to close	Two valves in series, one required to close.
4) Isolation at suction header from refueling water storage tank to safety injection pumps	Fails to close	Two valves in series, one required to close (one valve is a check valve).
5) Residual heat removal pump recirculation line	Fails to close	Two valves in series, one required to close.
6) Residual heat removal pump discharge line	Fails to close	Two valves in series, one required to close (one valve is a check valve).

The status of all active components of the safety injection system is indicated on the main control board. Reference is made to Table 6.2-A

TABLE 6.2-7(b)

LOSS OF RECIRCULATION FLOW PATH

Flow Path	Indication of Loss of Flow Path	Alternate Flow Path
Low Head Recirculation		
From recirculation sump to low head injection header via the recirculation pumps and the residual heat exchangers.	1. Insufficient flow in low head injection lines, (one flow monitor in each of the four low head injection lines†).	From recirculation sump to high head injection header via the recirculation pumps, one of the two residual heat exchangers and the safety injection pumps.*
	2. As 1 above.	a. From containment sump to discharge header of the residual heat exchangers via the residual heat removal pumps.
		b. If flow not established in low head injection lines - as (a), except path is from discharge of one residual heat exchanger to the high head injection header via the safety injection pumps.
High Head Recirculation		
From recirculation sump to high head injection header via the recirculation pumps, one of the two residual heat exchangers and the high head injection pumps.	1. No flow in high head injection header (four flow monitors, one in each injection line and one pressure monitor).	a. From containment sump to high head injection header via the residual heat removal pumps, one of the two residual heat exchangers and the high head injection pumps.

NOTE: As shown on Figure 6.2-1, there are valves at all locations where alternative flow paths are provided.

† If 3 out of 4 meters read ≥ 300 gpm, the supply of recirculated water will maintain the core flooded even in the event of a low head spilling line and one failed flow meter.

*Manual Start

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Flow Path	Indication of Loss of Flow Path	Alternative Flow Path
		b. If flow is not established in high head injection header - as (a) except path is from discharge of the residual heat removal pumps to the high head injection pump via the middle safety injection pump (by-passing the residual heat exchangers*).
	2. Flow in only one of the two high head injection branch headers (two flow monitors per branch header).	a. as 1(b) except that flow from the middle safety injection pump is only supplied to the unbroken branch header.

NOTE: As shown on Figure 6.2-1, there are valves at all locations where alternative flow paths are provided.

* In this recirculation mode, water is returned to the core without being cooled by the residual heat exchangers. Heat is removed from the core by boil-off of the water to the containment; heat is then removed from the containment by either the containment fan coolers or/and the containment spray system (using cooled water from the recirculation sump via the recirculation pumps and one residual heat exchanger).

TABLE 6.2-8

SHARED FUNCTIONS EVALUATION

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Boron Injection Tank	None	Lined up to suction of safety injection pumps	Source of high concentrated borated water for core	Lined up to suction of safety injection pumps
Refueling Water Storage Tank	Storage tank for refueling operations	Lined up to suction of safety injection, residual heat removal, and spray pumps	Source of borated water for core and spray nozzles	Lined up to suction of safety injection, residual heat removal, and spray pumps
Accumulators (4)	None	Lined up to cold legs of reactor coolant piping	Supply borated water to core promptly	Lined up to cold legs of reactor coolant piping
Safety Injection Pumps (3)	None	Lined up to hot and cold legs of reactor coolant piping	Supply borated water to core	Lined up to hot and cold legs of reactor coolant piping
Residual Heat Removal Pumps (2)	Supply water to core to remove residual heat during shutdowns	Lined up to cold legs of reactor coolant piping	Supply borated water to core	Lined up to cold legs of reactor coolant piping
Recirculation Pumps (2)	None	Lined up to cold legs of reactor coolant piping, spray headers and suction of safety injection pumps	Supply borated water to core and spray nozzles from recirculation sump.	Lined up to cold legs of reactor coolant piping spray headers, and suction of safety injection pumps
Conventional Service Water Pumps (3)	Supply river cooling water to component cooling heat exchangers	Two pumps in service	Supply river cooling water to component cooling heat exchangers	*One pump in service

TABLE 6.2-8 (Continued)

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Component Cooling Pumps (3)	Supply cooling water to station nuclear components	Two pumps in service	Supply cooling water to residual heat exchangers S.I. pump bearings and recirculation pump motor coolers	*One pump in service
Residual Heat Exchangers (2)	Remove residual heat from core during shutdown	Lined up for residual heat removal pump operation	Cool water in containment sump for core cooling and containment spray	Lined up to discharge of (recirculation pumps).
Component Cooling Heat Exchangers (2)	Remove heat from component coolant water	One heat exchanger in service	Cool water for residual heat exchangers	Both heat exchangers in service
Auxiliary component cooling pumps (2)	None	Lined up for pump operation	Provide component cooling water to recirculation pump motor coolers	Lined up for pump operation

TABLE 6.2-9

ACCUMULATOR INLEAKAGE*

Time Period Between Level Adjustments	Observed Leak Rate cc/hr	(Observed Leak Rate) (Max. Allowed Design)
1 month	1955	99.8
3 months	665	33.3
6 months	333	16.7
9 months	221	11.1
1 year	167	8.
10 years	16.7	0.8

* A total of 83.. cubic feet, added to the initial amount, can be accepted in each accumulator before an alarm is sounded.

TABLE 6.2-10

MAXIMUM POTENTIAL EXTERNAL RECIRCULATION LOOP LEAKAGE

Items	No. of Units	Type of Leakage Control and Unit Leakage Rate Used in the Analysis	Leakage to Atmosphere cc/hr	Leakage to Drain Tank cc/hr
Residual Heat Removal Pumps (Low Head Safety Injection)	2	Mechanical seal with leakoff - drop/min	0	6
High Head Safety Injection Pumps	3	Same as residual heat removal	0	9
Flanges:		Gasket - adjusted to zero leakage following any test - 10 drops/min/flange used in analysis		
a. Pump	15		450	0
b. Valves Bonnet to Body (larger than 2")	16		480	0
Valves - Stem Leakoffs	6	Backseated, double packing with leakoff - 1 cc/hr/in. stem diameter	0	6
Misc. Small Valves	23	Flanged body packed stems - 1 drop/min used	69	0
		TOTALS	999	21

TABLE 6.2-11

RESIDUAL HEAT REMOVAL SYSTEM
DESIGN, OPERATION AND TEST CONDITIONS

	Pumps	Heat** Exchangers	Valves	Pipes and Fittings
Design Conditions				
Pressure, PSIG	600	600	665	700
Temperature, °F	400	400	400	400
Operating Conditions (Max)*				
Pressure, PSIG	150	150	150	150
Temperature, °F	213	213	213	213
Test Pressure, PSIG	1200	900	1100	900
Allowable Pressure at Operating Temp. PSIG	>600	> 600	> 690	> 850

* During recirculation.

** Located inside containment.

TABLE 6.2-17

INSTRUMENTATION READOUTS ON THE CONTROL BOARD FOR OPERATOR MONITORING DURING RECIRCULATION

Valves

System	Valve Number
SIS	MOV 1802 A, B
SIS	MOV 885 A, B
SIS	MOV 889 A, B
SIS	MOV 888 A, B
SIS	MOV 866 A, B, C, D
SIS	MOV 851 A, B
SIS	MOV 856 A, B, C, D
SIS	882
SIS	AOV 842
SIS	AOV 843
ACS	MOV 744
ACS	MOV 745, A, B
ACS	MOV 746
ACS	MOV 747
	Channel Number
SIS	FI 945 A, B
SIS	FI 946 A, B
SIS	FI 924
SIS	FI 923
SIS	FI 926
SIS	FI 927
SIS	LI 938
SIS	LI 939
SIS	LIA 940
SIS	LIA 941
SIS	PI 922
SIS	PI 923
SIS	PI 947
ACS	PI 635
ACS	FI 640
ACS	LIT 628
ACS	HCV 638
ACS	HCV 640
ACS	TR 639
RCS	LRCA 459
RCS	LICA 406
RCS	LICA 461
RCS	LI 462
	Pump
SIS	Safety Injection
	Service Water
ACS	Component Cooling
LS	Containment Spray
RS	Recirculation
ACS	Residual Heat Removal

TABLE 6.2-13

QUALITY STANDARDS OF SAFETY INJECTION SYSTEM COMPONENTS
RESIDUAL HEAT EXCHANGER

A. Tests and Inspections

1. Hydrostatic Test
2. Radiograph of longitudinal and girth welds (tube side only)
3. UT of tubing or eddy current tests
4. Dye penetrant test of welds
5. Dye penetrant test of tube to tube sheet welds
6. Gas leak test of tube to tube sheet welds before hydro and expanding of tubes

B. Special Manufacturing Process Control

1. Tube to tube sheet weld qualifications procedure
2. Welding and NDT and procedure review
3. Surveillance of supplier quality control and product

COMPONENT COOLING HEAT EXCHANGER

A. Tests and Inspections

1. Hydrostatic Test
2. Dye penetrant test of welds

B. Special Manufacturing Process Control

1. Welding and NDT and procedure review
2. Surveillance of supplier quality control and product

SAFETY INJECTION, RECIRCULATION AND RESIDUAL HEAT REMOVAL PUMPS

A. Tests and Inspections

1. Performance Test
2. Dye penetrant of pressure retaining parts*
3. Hydrostatic test

* Except Internal Recirculation Pump

TABLE 6.2-13(Cont'd)

B. Special Manufacturing Process Control

1. Weld, NDT and inspection procedures for review
2. Surveillance of suppliers quality control system and product

ACCUMULATORS

A. Tests and Inspections

1. Hydrostatic test
2. Radiography of longitudinal and girth welds
3. Dye penetrant/magnetic particle of weld

B. Special Manufacturing Process Control

1. Weld, fabrication, DNT and inspection procedure review
2. Surveillance of suppliers quality control and product

VALVES

A. Tests and Inspections

(a) 200 psi and 200°F or below (cast or bar stock)

1. Dye Penetrant Test
2. Hydrostatic Test
3. Seat Leakage Test

(b) Above 200 psi and 200°F

(i) Forged Valves (2 1/2" inch and larger)

1. UT of billet prior to forging
2. Dye penetrant 100% of accessible areas after forging
3. Hydrostatic Test
4. Seat Leakage Test

(ii) Cast Valves

1. Radiograph 100%*
2. Dye Penetrant all accessible areas*
3. Hydrostatic Test
4. Seat Leakage

TABLE 6.2-13 (Cont'd)

(c) Functional Tests Required for:

1. Motor Operated Valves
2. Auxiliary Relief Valves

B. Special Manufacturing Process Control

1. Weld, NDT, performance testing, assembly and inspection procedure review
2. Surveillance of suppliers quality control and product
3. Special Weld process procedure qualification (e.g. hard facing)

* For valves with radioactive service only

TABLE 6.2-13 (Cont'd)

PIPING

A. Tests and Inspections

Class 1501 and below

Seamless or welded. If welded 100% radiography is required. Shop Fabricated and field fabricated pipe weld joints are inspected as follows:

2501R - 601R - 100% radiographic inspection and penetrant examination
301R - 302R - 20% random radiographic inspection
151R - 152R - 100% liquid penetrant examination

B. Special Manufacturing Process Control

Surveillance of suppliers quality control and product.

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Figure 6.2-2, Titled "Primary Auxiliary Building Safety Injection System Piping Plans"

The above was **redacted** by NRC staff as **sensitive information** to allow release of this document to the public.

Indian Point
Final Facility Description and Safety Analysis Report
Dated: October 15, 1968

Ex 4

Figure 6.2-3, Titled “Primary Auxiliary Building Safety Injection System Piping Elevations”

The above was **redacted** by NRC staff as **sensitive information** to allow release of this document to the public.

TAB

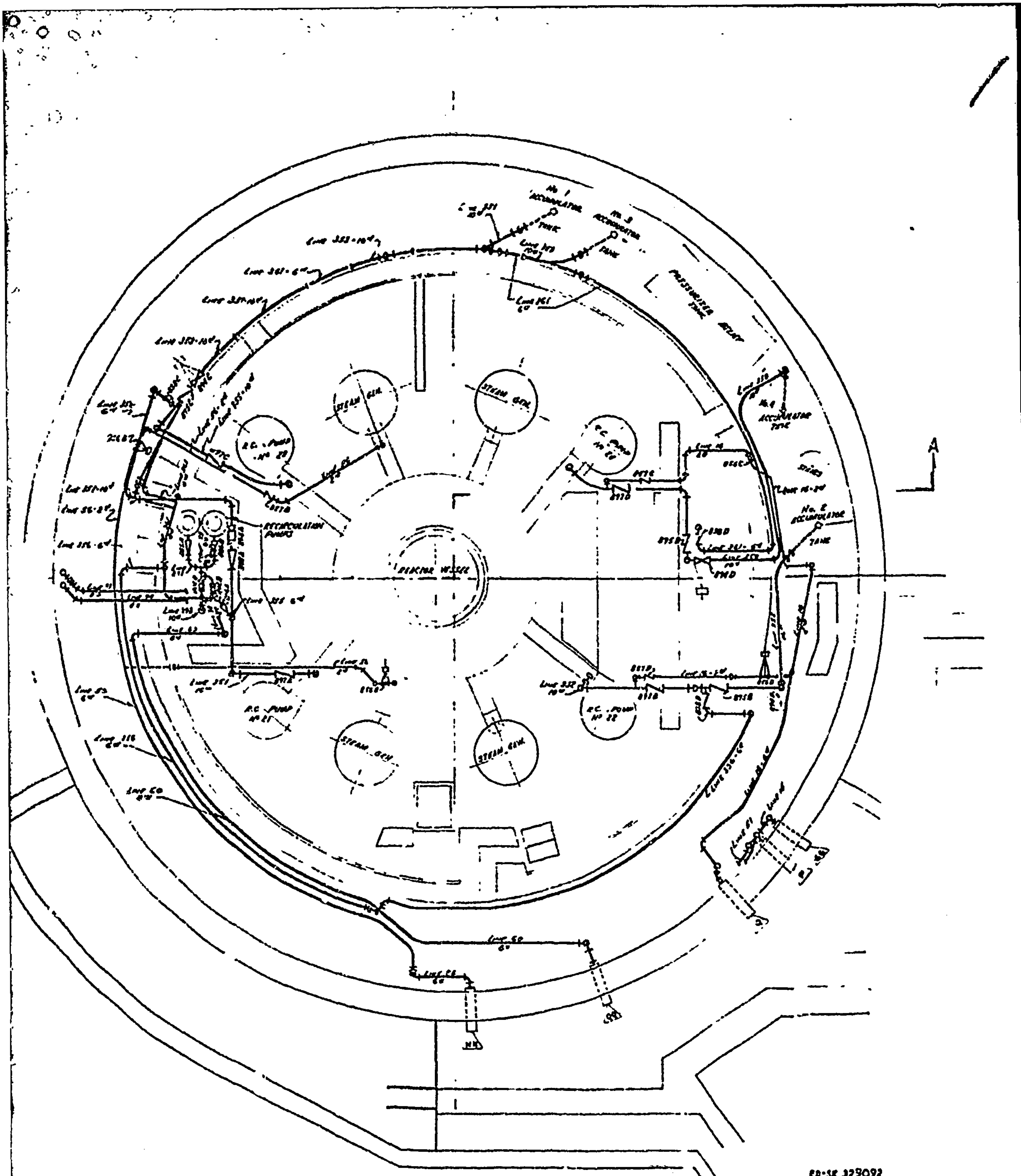
CONTAINMENT PIP
(P

		Penetration and System	Dia-gram	Valve No. Or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Positi
	51.	Recirculation Pump Discharge Sample Line	5.2-20	A B	Globe Globe	Manual Manual	No No	Closed Closed
15	52.	Pressurizer Steam Sample Line	5.2-21	A B	Globe Globe	Air Air	Yes Yes	Closed Closed
15	53.	Pressurizer Liquid Space Sample	5.2-21	A B	Globe Globe	Air Air	Yes Yes	Closed Closed
15	54-56.	Containment Pressure Instrumentation	5.2-21	A	Globe	Manual	No	Open
	57.	Post Accident Containment Sampling System Supply and Return Lines*	5.2-22	A B	Dia. Dia.	Manual Manual	No No	Close Close
	58.	Oxygen Supply to Containment Atomsphere*	5.2-23	A B	Gate Globe	Manual Manual	No Yes	Closed Closed
	59.	H_2 Supply to H_2 Recombiner*		A B	Gate Globe	Manual Solenoid	No Yes	Closed Closed

*Isolation valves are opened intermittently post accident for samplings.

F-1(

Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Sealing Method	Used After Accid.	Fluid Gas Or Water	Temp. Hot >200°F Cold <200°F
Open	Closed	As is	-	Water (A)	No	W	Hot
Open	Closed	As is	-	Water (A)	No	W	Hot
Closed	Closed	FC	T	Water (A)	No	W	Hot
Closed	Closed	FC	T	Water (A)	No	W	Hot
Closed	Closed	FC	T	Water (A)	No	W	Hot
Closed	Closed	FC	T	Water (A)	No	W	Hot
Open	Open	As is	-	-	Yes	G	Cold
Closed	Both	As is	-	-	Yes	G	Cold
Closed	Both	As is	-	-	Yes	G	Cold
Closed	Both	As is	-	-	Yes	G	Cold
Closed	Both	FC	-	-	Yes	G	Cold
Closed	Both	As is	-	-	Yes	G	Cold
Closed	Both	FC	-	-	Yes	G	Cold



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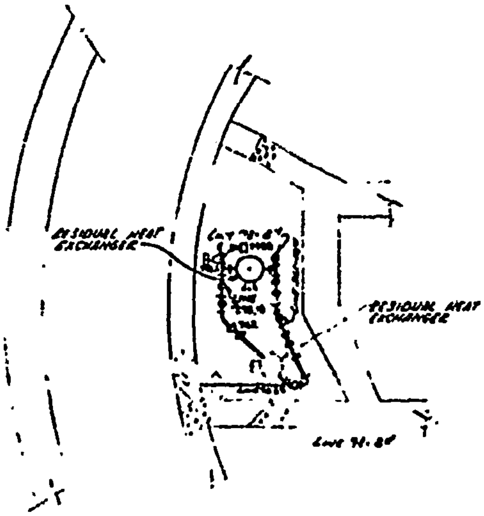
Containment Building Safety Injection System
Piping Plan Figure 6.2-4



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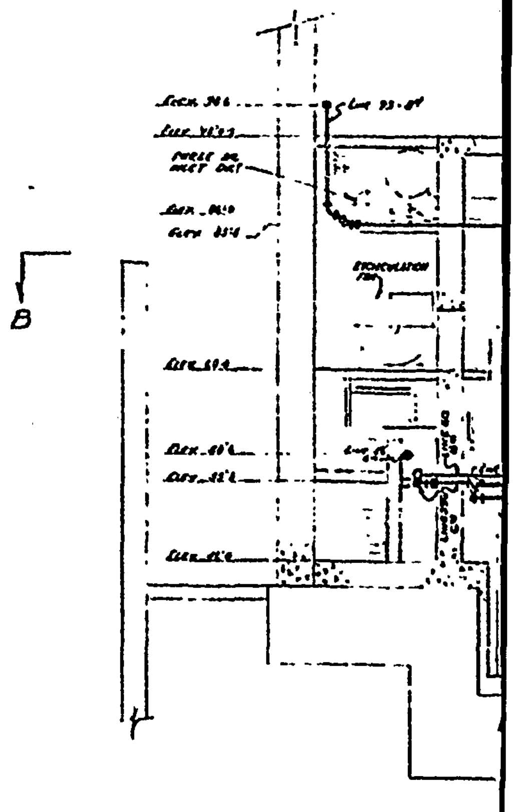
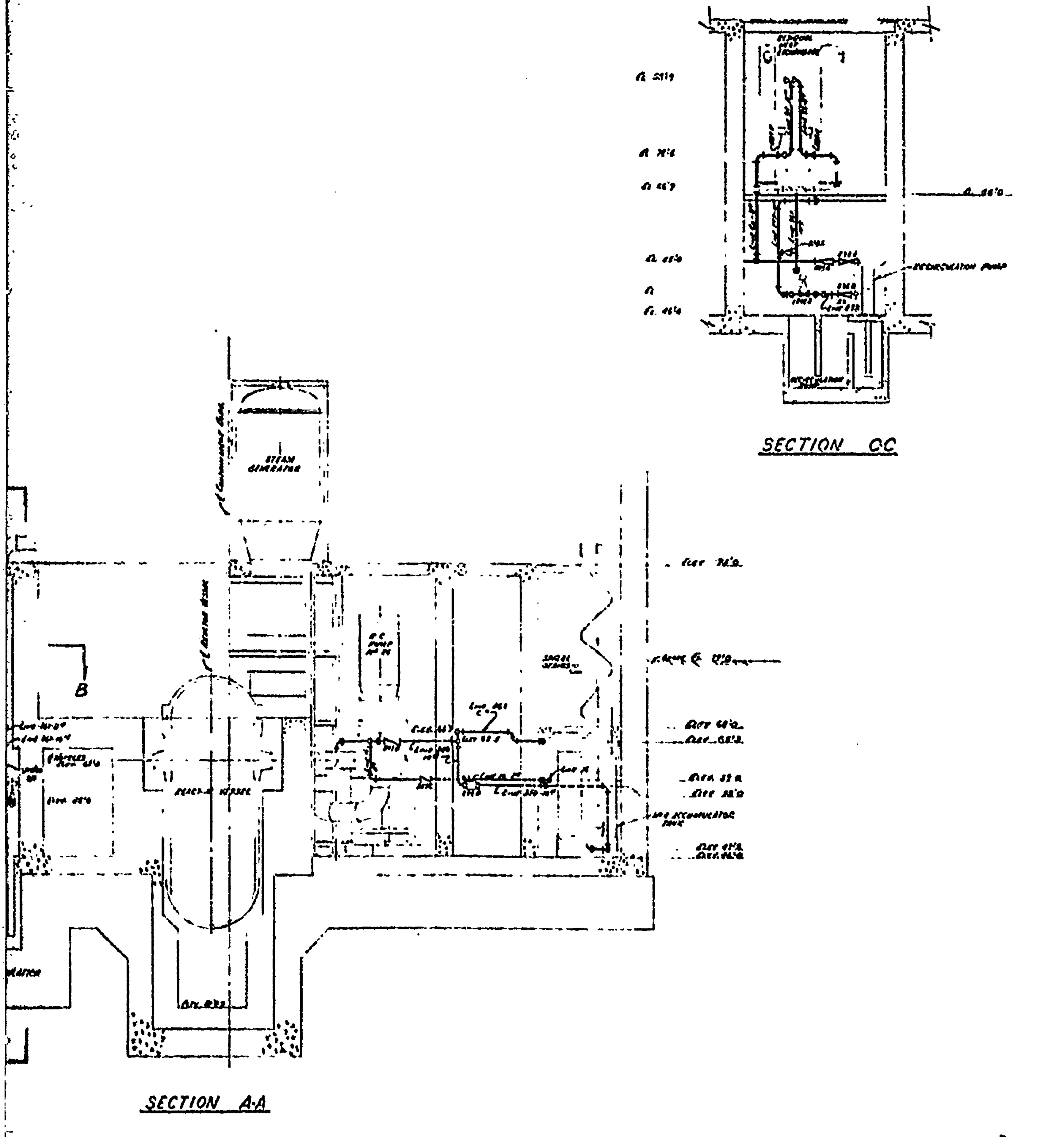




Containment Building S_fety Injection System
Piping Elevation Figure 6.2-5

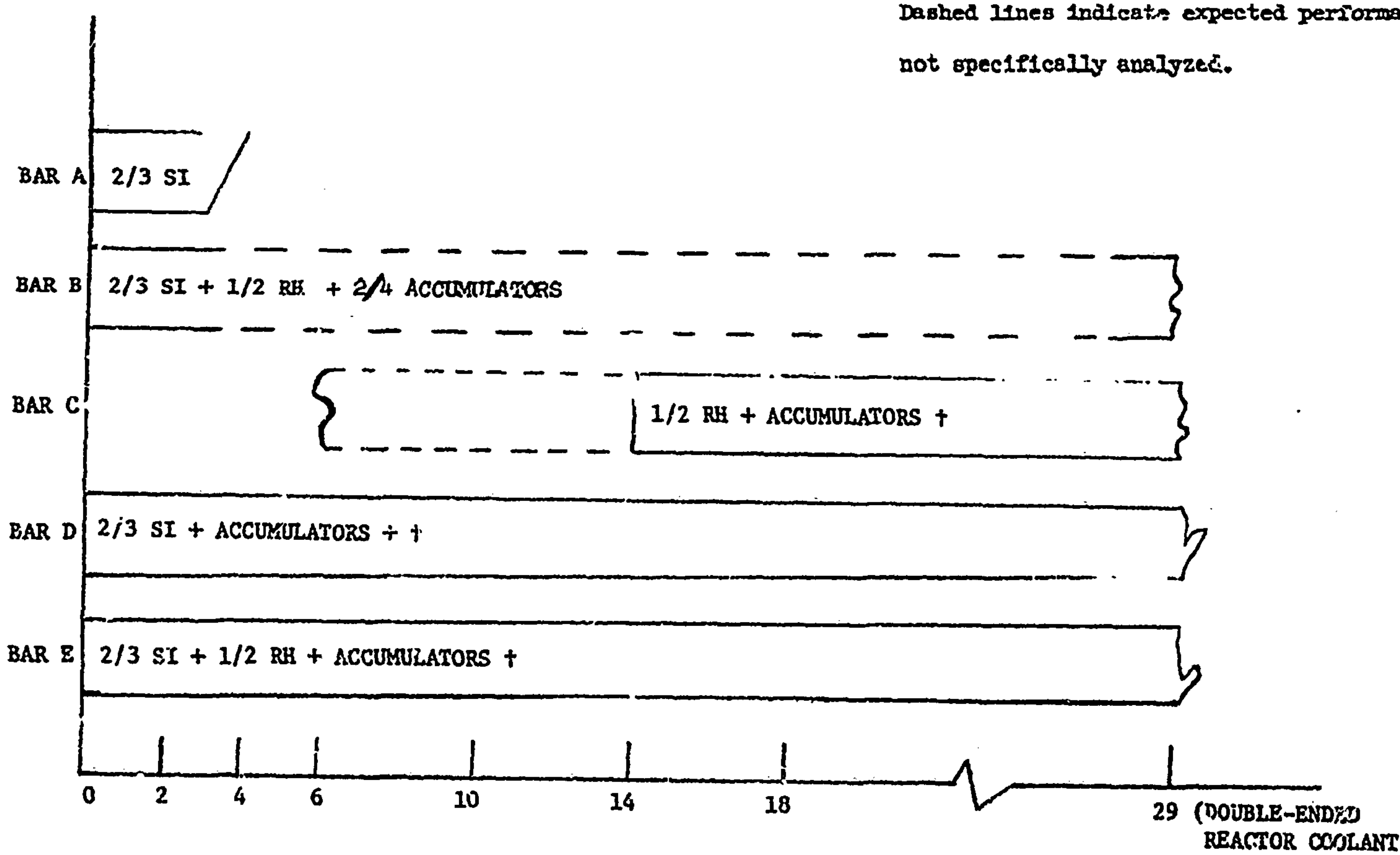
2/3 SI = TWO OF THREE SAFETY INJECTION PUMPS

1/2 RH = ONE OF TWO RESIDUAL HEAT REMOVAL PUMPS

CORE PROTECTION

Solid bar indicates capacity to meet core cooling criterion of no clad melting.

Dashed lines indicate expected performance not specifically analyzed.



NOTE: FOR ALL CASES ONE OF TWO RECIRCULATION PUMPS REQUIRED FOR RECIRCULATION

† NO CREDIT IS TAKEN FOR THE ACCUMULATOR WHICH IS ATTACHED TO THE RUPTURED LEG IN THE CASE OF A COLD LEGBREAK

RANGE OF PROTECTION BY SAFETY INJECTION SYSTEM

FIGURE 6.2-6

SAFETY INJECTION PUMP PERFORMANCE

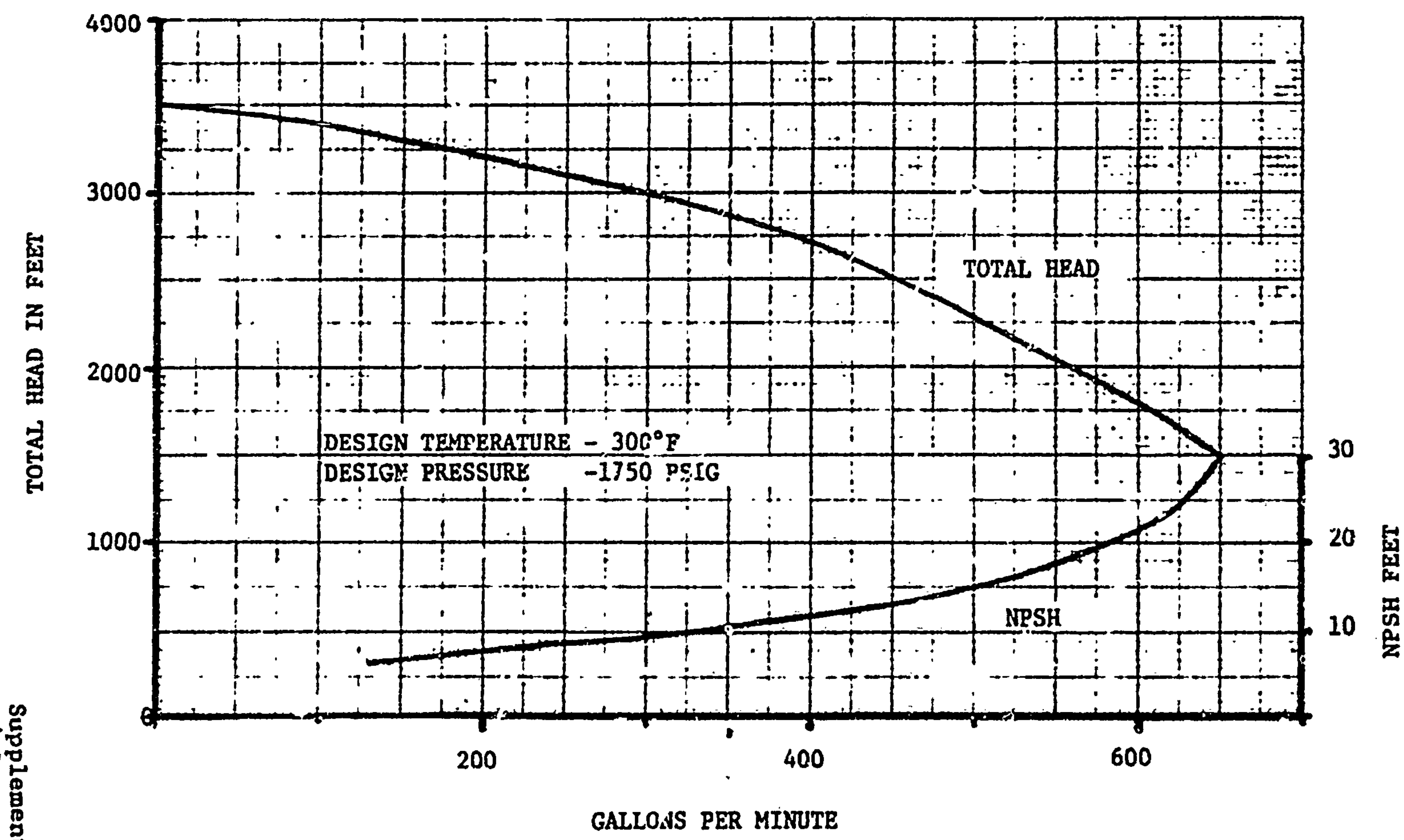


FIGURE 6.2-7

RESIDUAL HEAT REMOVAL PUMP PERFORMANCE

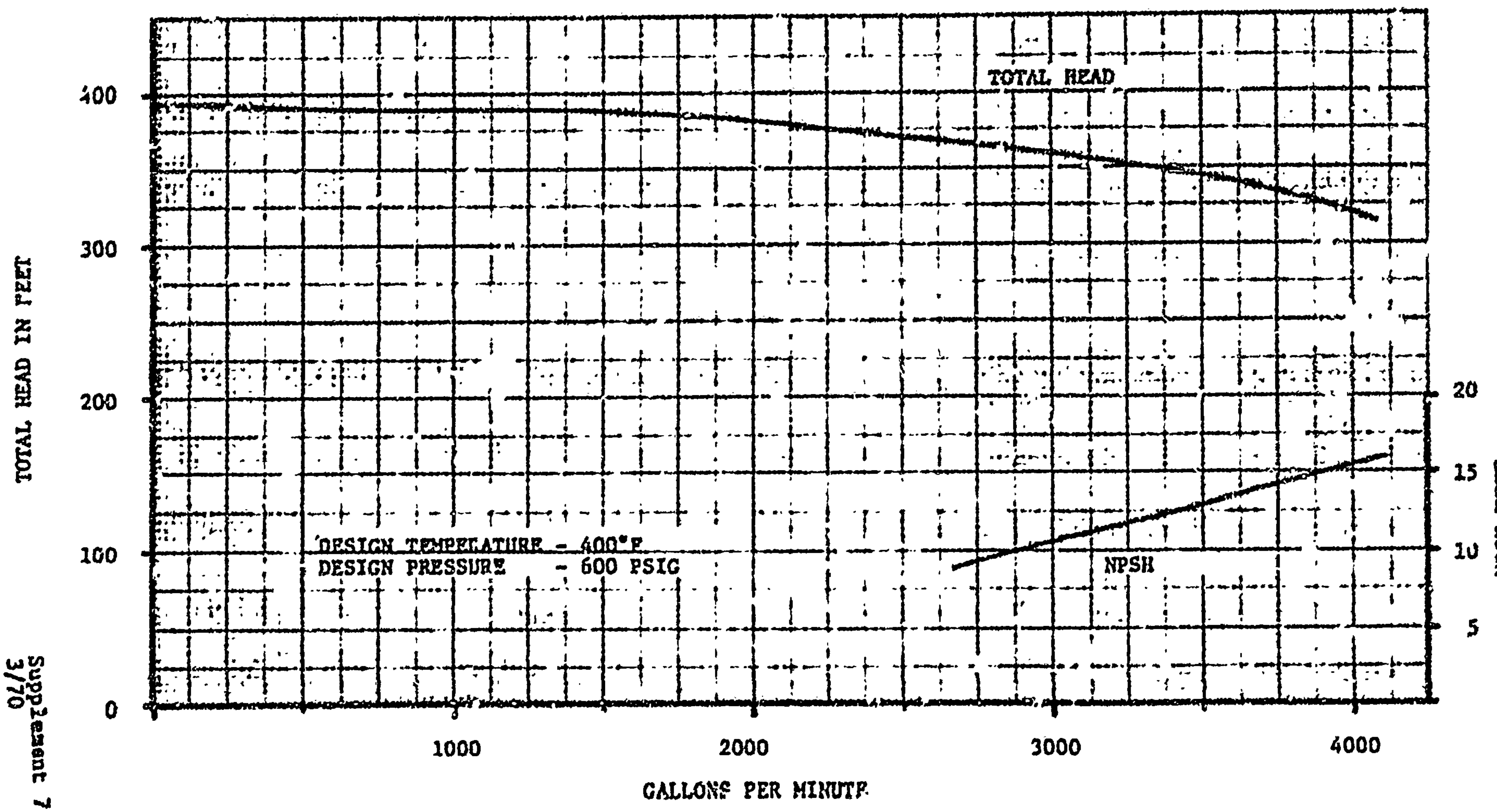


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FIGURE 6.2-8

RECIRCULATION PUMP PERFORMANCE

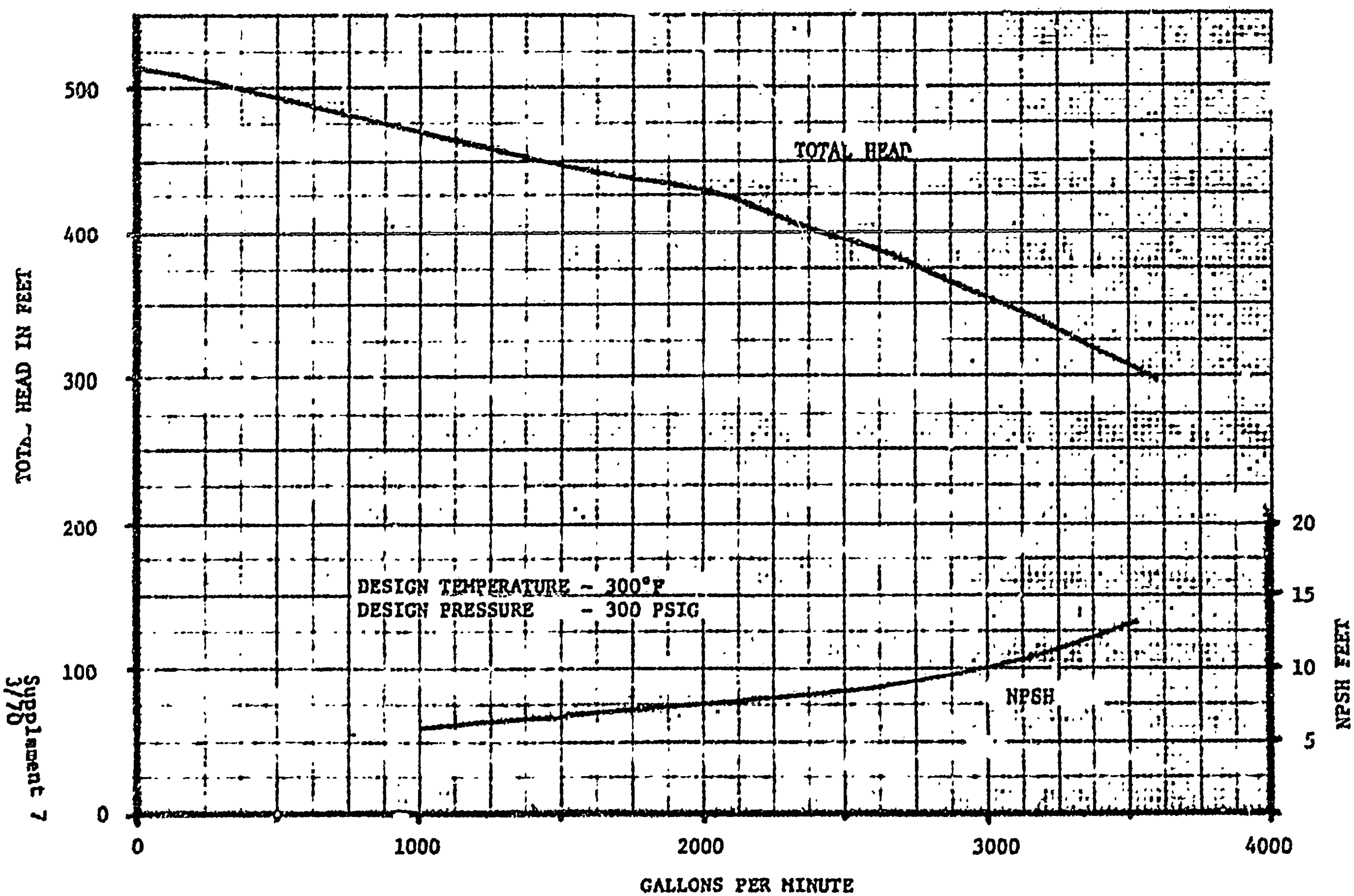

FIGURE 6.2-9

6.3 CONTAINMENT SPRAY SYSTEM

6.3.1 DESIGN BASES

Containment Heat Removal Systems

Criterion: Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure, this system shall perform its required function, assuming failure of any single active component. (GDC 52)

Adequate containment heat removal capability for the Containment is provided by two separate, full capacity, engineered safety feature systems. The Containment Spray System, whose components operate in the sequential modes described in 6.3.2, and the Containment Air Recirculation Cooling and Filtration System which is discussed in Section 6.4.

The primary purpose of the Containment Spray System is to spray cool water into the containment atmosphere when appropriate in the event of a loss-of-coolant accident and thereby ensure that containment pressure does not exceed its design value which is 47 psig at 271°F. (100% R.H.) This protection is afforded for all pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop. Pressure and temperature transients for a loss-of-coolant accident are presented in Section 14. Although the water in the core after a loss-of-coolant accident is quickly subcooled by the Safety Injection System, the Containment Spray System design is based on the conservative assumption that the core residual heat is released to the containment as ·am.

Any of the following combinations of equipment will provide sufficient heat removal capability to maintain the post-accident containment pressure below the design value, assuming that the core residual heat is released to the containment as steam.

1) Both containment spray pumps (and one of the two spray valves in the recirculation path).

2) All five containment cooling fans (to be discussed in Section 6.4).

3) One containment spray pump and three of the five containment cooling fans.

Inspection of Containment Pressure Reducing System

Criterion: Design provisions shall be made to the extent practical to facilitate the periodic physical inspection of all important components of the containment pressure reducing systems, such as pumps, valves, spray nozzles and sumps. (GDC 58)

Where practicable, all active components and passive components of the Containment Spray System are inspected periodically to demonstrate system readiness. The pressure containing components are inspected for leaks from pump seals, valve packing, flanged joints and safety valves. During operational testing of the containment spray pumps, the portions of the system subjected to pump pressure are inspected for leaks. Design provisions for inspection of the Safety Injection System, which also function as part of the Containment Spray System, are described in Section 6.2.5.

Testing of the Containment Pressure Reducing Systems Components

Criterion: The containment pressure reducing systems shall be designed, to the extent practical so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance. (GDC 59)

All active components in the Containment Spray System are adequately tested both in pre-operational performance tests in the manufacturer's shop and in-place testing after installation. Thereafter, periodic tests are also performed after any component maintenance. Testing of the components of the Safety Injection System which are used for containment spray purposes are described in Section 6.2.5.

The component cooling water pumps and the conventional service water pumps which supply the cooling water to the residual heat exchangers are in operation on a relatively continuous schedule during plant operation. Those pumps not running during normal operation may be tested by changing the operating pump(s).

Testing of Containment Spray Systems

Criterion: A capability shall be provided to the extent practical to test periodically the operability of the containment spray system at a position as close to the spray nozzles as is practical. (GDC 60)

Permanent test lines for the containment spray loops are located so that all components up to the isolation valves at the spray nozzles may be tested. These isolation valves are checked separately.

The air test lines, for checking that spray nozzles are not obstructed, connect downstream of the isolation valves. Air flow through the nozzles will be monitored by means of the helium filled balloon method.

Testing of Operational Sequence of Containment Pressure Reducing Systems

Criterion: A capability shall be provided to test initially under conditions as close as practical to the design and the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources. (GDC 61)

Capability is provided to test initially to the extent practical the operational start-up sequence of the Containment Spray System including the transfer to alternate power sources.

Performance Objectives

The Containment Spray System is designed to spray at least 5000 gpm of borated water, to which sodium hydroxide has been added, into the containment whenever the coincidence of two sets of two out of three (Hi Hi) containment pressure (approximately 50% of design value) signals occurs or a manual signal is given. Either of two subsystems containing a pump and associated valving and spray header are independently capable of delivering one-half of this flow, or 2500 gpm.

The design basis is to provide sufficient heat removal capability to maintain the post-accident containment pressure below 47 psig, assuming that the core residual heat is released to the containment as steam.

A second purpose served by the Containment Spray System is to remove elemental iodine from the containment atmosphere should it be released in the event of a loss-of-coolant accident. The analysis showing the systems ability to limit off-site thyroid dose to within 10 CFR 100 limits after a hypothetical loss-of-coolant accident is presented in Section 14. If all engineered safety features operate at design capacity, off-site doses will be limited to within the limits of 10 CFR 20.

The spray system is designed to operate over an extended time period, following a Reactor Coolant System failure, as required to restore and maintain contain-ment conditions at or near atmospheric pressure. It has the capability of reducing the containment post-accident pressure and consequent containment leakage.

Portions of other systems which share functions and become part of the Containment Spray System when required, are designed to meet the criteria of this section. Neither a single active component failure in such systems during the injection phase nor an active/passive failure during the re-circulation phase will degrade the design heat removal capability of con-tainment cooling.

System piping located within the containment is redundant and separable in arrangement unless fully protected from damage which may follow any Reactor Coolant System loop failure.

System isolation valves relied upon to operate for containment cooling are redundant, with automatic actuation.

Service Life

All portions of the system located within containment are designed to with-stand, without loss of functional performance, the post-accident containment environment and operate without benefit of maintenance for the duration of time to restore and maintain containment conditions at near atmospheric pressure.

Codes and Classifications

Table 6.3-1 tabulates the codes and standards to which the Containment Spray System components are designed.

6.3.2 SYSTEM DESIGN AND OPERATION

System Description

Adequate containment cooling and iodine removal are provided by the Containment Spray System shown in Figure 6.2-1 whose components operate in sequential modes. These modes are:

a) Spray a portion of the contents of the refueling water storage tank into the entire containment atmosphere using the containment spray pumps. During this mode, the contents of the spray additive tank (sodium hydroxide) are mixed into the spray stream to provide adequate iodine removal from the containment atmosphere by a washing action.

b) Recirculation of water from the containment sump by the diversion of a portion of the recirculation flow from the Safety Injection System to the spray headers inside the containment after injection from the refueling water storage tank has been terminated.

The bases for the selection of the various conditions requiring system actuation is presented in Section 14.

The principal components of the Containment Spray System which provides containment cooling and iodine removal following a loss-of-coolant accident consists of two pumps, one spray additive tank, spray ring headers and nozzles, and the necessary piping and valves. The containment spray pumps and the spray additive tank are located in the primary auxiliary building and the spray pumps take suction directly from the refueling water storage tank.

The Containment Spray System also utilizes the two 100% capacity recirculation pumps, two residual heat exchangers and associated valves and piping of the Safety Injection System for the long term recirculation phase of containment cooling and iodine removal after the refueling water storage tank has been exhausted.

The spray water is injected into the containment through spray nozzles connected to four 360° ring headers located in the containment dome area. Each of the spray pumps supplies two of the ring headers.

Injection Phase

The spray system will be actuated by the coincidence of two sets of two out of three high containment pressure signals. This starting signal will start the pumps and open the discharge valves to the spray header. The valves associated with the spray additive tank will be opened on the same signal.

After the containment spray signal is actuated the operator has the capability to stop the timer if he determines that actuation of the sodium hydroxide addition is not warranted. The operator also has the capability to reinitiate the sodium hydroxide addition, if required. Emergency procedures will set forth guidelines for the action period. If required, the operator can manually actuate the entire system from the control room, and periodically, the operator will actuate system components to demonstrate operability.

During injection, approximately 100 gpm of pump discharge flow is diverted from the spray pump discharge line through the spray eductors. The liquid from the tank then mixes with the liquid entering the suction of the pumps via the eductors. The result is a solution suitable for the removal of iodine from the containment atmosphere.

During the injection phase, the safety injection and residual heat removal pumps inject borated water into the reactor and containment. Since these flow paths do not inject NaOH solution, the ratio of the total volume of fluid injected via the spray pumps to the total volume injected via the safety injection and residual heat removal pumps determines the sump pH after the injection phase.

To meet the containment sump pH conditions, the injection phase will be terminated after the injection of approximately 250,000 gallons of the refueling water storage tank capacity. Recirculation will commence after 250,000 gallons of injection and the remaining 80,000 gallons will be injected into the containment via one containment spray pump. By this procedure it can be assumed that even in the event of a failure to one spray pump train a sump pH which assures continued iodine removal and retention effectiveness will be obtained.

Recirculation Phase

When the refueling water storage tank is exhaused recirculation spray flow will be initiated. The operator can remotely open the stop valves on either of the two spray recirculation lines. Throttle valves in the injection lines to the core split the recirculation flow so that at least 600 gpm is delivered to the core and the remainder to the spray headers. With this split flow, decay heat can be removed by boiloff and the containment pressure maintained below design. This mode of operation will be continued for a period of at least two hours following the accident in order to complete iodine removal from the containment atmosphere.

After the two hour containment scrubbing operation it is expected that spray flow could be discontinued while maintaining containment pressure with the containment fan cooler units, and returning all of the recirculated water to the core. In this mode the bulk of the core residual heat is transferred directly to the sump by the spilled coolant to be eventually dissipated through the residual heat exchanger once the sump water becomes heated. The heat removal capacity of three of the five fan coolers is sufficient to remove the corresponding energy addition to the vapor space resulting from steam boiloff from the core assuming flow into the core from one recirculation pump at the beginning of recirculation without exceeding containment design pressure; hence it is not expected that continued spray operation for containment heat removal would be required. If, however, the containment pressure was observed to increase, then recirculation to the spray header may be resumed by operator action as described above.

Cooling Water

The cooling water for the residual heat exchangers has been described in Section 6.2.

Change-Over

The sequence for the change-over from injection to recirculation has been described in Section 6.2.

Remote operated valves of the Containment Spray System which are under manual control (that is, valves which normally are in their ready position and do not receive a containment spray signal) have their positions indicated on a common portion of the control board. At any time during operation when one of these valves is not in the ready position for injection, it is shown visually on the board In addition, an audible annunciation alerts the operator to the condition.

Charcoal Filter Dousing

A dousing system is provided for the charcoal filter bank of each fan cooler unit of the Containment Air Recirculation Cooling and Filtration System. Each dousing system can be supplied with water from the containment spray headers as shown in Figure 6.2-1. The dousing system is designed to be started manually by the operator following high temperature indication in a charcoal filter bank if high temperature conditions were to occur as a result of a failure of a fan. Further details of the dousing system are given in Section 6.4.

Components

All associated components, piping, structures, and power supplies of the Containment Spray System are designed to Class I seismic criteria.

All components inside containment are capable of withstanding or are protected from differential pressures which may occur during the rapid pressure rise to 47 psig in ten (10) seconds. The lines of the system are protected from missile damage by the concrete crane wall and operating floor.

Parts of the system in contact with borated water, the sodium hydroxide spray additive, or mixtures of the two are stainless steel or an equivalent corrosion reistant material.

The Containment Spray System shares the refueling water storage tank capacity with the Safety Injection System. For a detailed description of this tank see Section 6.2.

Pumps

The two containment spray pumps are of the horizontal centrifugal type driven by electric motors.

The design head of the pump is sufficient to continue at rated capacity, with a minimum level in the refueling water storage tank, against a head equivalent to the sum of the design pressure of the containment, the head to the uppermost nozzles, and the line and nozzle pressure losses. Pump motors are direct-coupled and large enough for maximum power requirement of the pump. The materials of construction are suitable for use in sodium hydroxide and mild boric acid solutions, such as stainless steel or equivalent corrosion resistant material. Design parameters are presented in Table 6.3-1 and the containment spray pump characteristics are shown on Figure 6.3-1.

The containment spray pumps are designed in accordance to the specifications discussed for the pumps in the Safety Injection System, Section 6.2.

The recirculation pumps of the Safety Injection System, which provide flow to the Containment Spray System during the recirculation phase, are described in Section 6.2.

Details of the component cooling pumps and service water pumps, which serve the Safety Injection System, are presented in Section 9.

Heat Exchangers

The two residual heat exchangers of the Safety Injection System which are used during the recirculation phase are described in Section 6.2.

Spray Nozzles

The spray nozzles, which are of the hollow cone, ramp bottom design (Spray Engineering Company - Model 1713-A), are not subject to clogging by particles 1/4 inch in maximum dimension, and are capable of producing a surface area averaged drop diameter of approximately 1000 microns at 15 gpm and 40 psi differential pressure. With the spray pump operating at design conditions and the containment at design pressure the pressure drop across the nozzles will exceed 40 psi.

During spray recirculation operation, the water is screened through a 1/4 inch mesh before leaving the containment sump. The spray nozzles are stainless steel and have a 3/8 inch diameter orifice. The nozzles are connected to four 360° ring headers (alternating headers connected) of radii 8'2" (El. 228.5'), 25'4" (El. 223.5), 42'3" (El. 218.5) and 59'6" (El. 213.5). There are 315 nozzles distributed on the four headers. This nozzle and header arrangement results in maximum area coverage with either branch of the system operating alone, while assuring minimum overlap of spray trajectories in the minimum flow case (Section (Section 14).

Spray Additive Tank

The capacity of the stainless steel tank is sufficient to contain enough sodium hydroxide solution which, upon mixing with the refueling water from the refueling water storage tank, the boric acid from the boron injection tank, the borated water contained within the accumulators and reactor coolant, will bring the concentration in the final mixture in the containment sump assures the continued iodine removal and retention effectiveness of the containment sump water during the recirculation phase of operation after the supply of borated water in the refueling water storage tank has been exhausted.

A level indicating alarm is provided in the control room if, at any time, the solution tank contains less than the required amount of sodium hydroxide solution. Periodic sampling confirms that proper sodium hydroxide concentration exists in the tank. The design parameters are presented in Table 6.3-2.

Spray Additive Eductors

The means of adding NaOH to the spray liquid is provided by a liquid jet eductor, a device which uses the kinetic energy of a pressure liquid to entrain another liquid, mix the two, and discharge the mixture against a counter pressure. The pressure liquid in this case is the spray pump discharge which is used to entrain the NaOH solution and discharge the mixture into the suction of the spray pumps. The two eductors are designed to provide enough NaOH in the mixture so as not to exceed pH 10 during the injection phase.

Valves

The valves for the Containment Spray System are designed in accordance to the specifications discussed for the valves in the Safety Injection System (Section 6.2).

Piping

The piping for the Containment Spray System is designed in accordance to the specifications discussed for the piping in the Safety Injection System (Section 6.2).

The system is designed for 150 psig at 300°F on the suction side and 300 psig at 300°F on the discharge side of the spray pumps.

Motors for Pumps and Valves

The motors inside and outside containment for the Containment Spray System are designed in accordance to the specifications discussed for motors in the Safety Injection System. (Section 6.2)

Electrical Supply

Details of the normal and emergency power sources are presented in the discussion of the Electrical System, Section 8.

Environmental Protection

The spray headers are located outside and above the reactor and steam generator concrete shield. A shield which is removable for refueling also provides missile protection for the area immediately above the reactor vessel. The spray headers are therefore protected from missilies orginating within the Reactor Coolant System.

Material Compatibility

Parts of the system in contact with borated water, the sodium hydroxide spray additive, or mixtures of the two are stainless steel or an equivalent corrosion resistant material.

All exposed surfaces within the containment have coatings which are not subject to interaction under exposure to the containment spray solution with the exception of small amounts of aluminum associated with the nuclear flux instrumentation.

6.3.3 DESIGN EVALUATION

Range of Containment Protection

For the first 15 to 22 minutes following the maximum loss-of-coolant accidents (i.e., during the time that the containment spray pumps take their suction from the refueling water storage tank) this system provides the design heat removal capacity for the containment. After the injection phase, one spray pump continues to spray into the containment for approximately an additional thirty minutes. The single pump operation is continued primarily to guarantee that even under failure conditions sufficient sodium hydroxide will be present in the containment sump water. This continued spray injection is also sufficient to maintain the containment pressure below the design value even if no containment fans were operating.

With the completion of containment spray injection the operator sets up recirculation to one spray header and to the core; flows are adjusted so that sufficient cooled recirculated water is delivered to keep the core flooded as well as providing flow to one spray header. Flow is maintained to the spray header at this stage primarily to complete the iodine scrubbing operation, i.e. for at least two hours after the accident; the flow, however, is also sufficient to maintain the containment pressure below the design value.

Any of the following combinations of equipment will provide sufficient heat removal capability to maintain the post-accident containment pressure below the design value, assuming that the core residual heat is released to the containment as steam.

1) Both containment spray pumps (and one of the two spray valves in the recirculation path).

2) All five containment cooling fans (To be discussed in Section 6.4).

3) One containment spray pump and three of the five containment cooling fans.

During the injection and recirculation phases the spray water is raised to the temperature of the containment in falling through the steam-air mixture. The minimum fall path of the droplets is approximately 118 ft. from the lowest spray ring headers to the operating deck. The actual fall path is longer due to the trajectory of the droplets sprayed out from the ring header. Heat transfer calculations, based upon 1000 micron droplets, show that thermal equilibrium is reached in a distance of approximately five feet. Thus the spray water reaches essentially the saturation temperature.

At containment design pressure, 47 psig, 2500 gpm of sodium hydroxide solution is injected into the containment atmosphere by one spray pump. At containment design temperature, 271°F, the total heat absorption capability of one spray pump is 218×10^6 Btu/hr based on addition of 100°F refueling water.

When recirculation is initiated, approximately 80,000 gallons of refueling water have been left in the refueling water storage tank for spray pump usage. The supply is reserved to insure operation of a spray pump to complete sodium hydroxide addition. When the refueling water storage tank is empty, the recirculation pumps supply the flow to the spray headers. Spraying 2400 gpm of water from the sump into the containment atmosphere with one recirculation pump, after cooling to 134.7°F with a residual heat exchanger, results in a heat removal rate of 1.63×10^8 Btu/hr at design temperature. This heat removal balances decay heat after 5000 seconds. Performance of the Containment Spray System in containment pressure reduction is discussed in Section 14.

In addition to heat removal, the spray system is effective in scrubbing fission products from the containment atmosphere. However, quantitative credit is taken only for absorption of reactive and/or soluble forms of inorganic iodine in the analysis of the hypothetical accident (Section 14.3). Experimental work done to date is not considered extensive enough to assess accurately the effect of the spray on particulates and non-reactive iodine under the conditions which would exist in the containment after such an accident. A discussion of the effectiveness of containment spray as fission product trapping process is contained in Appendix 6A.

Any of the combinations of equipment (spray pumps and fans) requir·· for containment heat remova' ..ill provide sufficient iodine trapping capability to ensure post-accid- .t fission product leakage (based on TID - 14844 release fractions) which w... not result in exceeding the dose limits of 10 CFR 100. This is evaluated in Section 14.3.

System Response

The starting sequence of the containment spray pumps and their related emergency power equipment is designed so that delivery of the minimum required flow is reached in 43 seconds.

The starting sequence is:		Seconds
a)	Initiation of signal, including instrument lag	1
b)	Starting of diesel-generator	19
c)	Starting of two containment spray pumps	23
	Total	43

Motor control centers are energized and valves are opened at the same time as the pumps are started. As described in Section 14.3 a delay of 60 seconds is assumed for the starting of the containment cooling.

Single Failure Analysis

A failure analysis has been made on all active components of the system to show that the failure of any single active component will not prevent fulfilling the design function. This analys.s is summarized in Table 6.3-4.

In addition, each spray header is supplied from the discharge from one of the two res.dual heat removal heat exchangers. As described in Section 6.2.3, these two hea. exchangers are redundant and can be supplied with recirculated water via separate and redundant flow paths. The analysis of the loss-of-coolant accident presented in Section 14 reflects the single failure analysis.

Reliance on Interconnected Systems

For the injection phase the Containment Spray System operates independently of other engineered safety features following a loss-of-coolant accident except that it shares the source of water in the refueling water storage tank with the Safety Injection System. The system acts as a backup to the Containment Air Recirculation Cooling and Filtration System for both the cooling and iodine removal functions. For extended operation in the recirculation mode, water is supplied through recirculation pumps.

During the recirculation phase some of the flow leaving the residual heat exchangers may be diverted to the containment spray headers or the high head safety injection pumps. Minimum flow requirements are set for the flow being sent to the core and for the flow being sent to the containment spray headers such that at least 600 gpm is sent to the core. Sufficient flow instrumentation is provided so that the operator can perform appropriate flow adjustments with the remote throttle valves in the flow path as shown in Figure 6.2-1.

Normal and emergency power supply requirements are discussed in Section 8.

Shared Function Evaluation

Table 6.3-4 is an evaluation of the main components which have been discussed previously and a brief description of how each component functions during normal operation and during the accident.

Containment Spray Pump NPSH Requirements

The NPSH for the containment spray pumps is evaluated for injection operation. The end of the injection phase gives the limiting NPSH requirement. The

NPSH available is determined from the evaluation head and vapor pressure of the water in the RWST and the pressure drop in the piping to the pump. At the end of the injection phase an 80% margin is available.

6.3.4 MINIMUM OPERATING CONDITIONS

The Technical Specifications, Section 15, establish limiting conditions regarding the operability of the system when the reactor is critical.

6.3.5 INSPECTIONS AND TESTS

<u>Inspections</u>

All components of the Containment Spray System are inspected periodically to demonstrate system readiness.

The pressure containing systems are inspected for leaks from pumps seals, valves packing, flanged joints and safety valves during system testing. During the operational testing of the containment spray pumps, the portions of the system subjected to pump pressure are inspected for leaks.

<u>Pre-Operational Testing</u>

<u>Component Testing</u>

All active components in the Containment Spray System are tested both in pre-operational performance tests in the manufacturer's shop and in-place testing after installation.

A representative sample of the spray nozzles are tested in the manufacturer's shop to demonstrate consistency of nozzle performance. After installation, the containment spray nozzles are tested by blowing air through the nozzles. The air test lines for checking the spray nozzles connect downstream of the isolation valves.

During the initial pre-operation tests of the spray system the performance of the eductor and spray additive system is checked by running the pumps near shut-off with the spray additive tank filled with water.

System Testing

The functional test of the Safety Injection System described in Section 6.2-5 demonstrates proper transfer to the emergency diesel generator power source in the event of a loss of power. A test signal simulating the containment spray signal is used to demonstrate the operation of the spray system up to the isolation valves on the pump discharge using the test pumps. The isolation valves are blocked closed for the test. These isolation valves are checked separately.

Post Operational Testing

Component Testing

Routine periodic testing of the Containment Spray System components and all necessary support systems at power is planned. If such testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance to be performed without shutting down or reducing load under conditions defined in the Technical Specifications. These conditions would include such matters as the period within which the component should be restored to service and the capability of the remaining equipment to meet safety limits within such a period.

The containment spray pumps are tested singly by opening the valves in the miniflow line. Each pump in turn is started by operator action and checked for flow establishment. The spray injection valves are tested with the pumps shutdown.

The spray eductors are tested singly by opening the valves in the pump miniflow lines, the valve in the eductor bed line from the RWST and running the respective pump. The operator observes the eductor suction flow.

The spray additive tank isolation valves can be opened periodically for testing. The contents of the tank are periodically sampled to determine that the required solution is present.

The valves in the dousing lines to the charcoal filter units may be exercised during a shutdown after the spray header drains are opened to ensure that the header is empty.

During these tests the equipment is visually inspected for leaks. Leaking seals, packing or flanges are tightened to eliminate the leak. Valves and pumps are operated and inspected after any maintenance to ensure proper operation.

System Testing

The post-operational testing of the safety injection system described in 6.2-5 demonstrates proper transfer to the emergency diesel generator power source in the event of a loss of power.

TABLE 6.3-1

CONTAINMENT SPRAY SYSTEM-CODE REQUIREMENTS

Component	Code
Spray Additive Tank	ASME Section III Class C
Valves	USAS B16-5
Piping (including headers and spray nozzles)	USAS B31.1

TABLE 6.3-2

CONTAINMENT SPRAY SYSTEM DESIGN PARAMETERS

PUMPS	
Quantity	2
Design pressure, discharge, psig	300
Design pressure, suction, psig	150
Design temperature, °F	300
Design flow rate, gpm	2600
Design head, ft.	450
Maximum flow rate, gpm	3120
Shutoff heat, ft.	490
Motor HP	400
Type	Horizontal-Centrifugal
EDUCTORS	
Quantity	2
Eductor Inlet (motive)	Injection Phase
Operating Fluid	Water (with 2000 ppm boron)
Operating Pressure, psig	195
Operating Temperature	Ambient
Flow Rate, gpm, (operating conditions)	112
Discharge Head (including static pressure, friction loss, and discharge elevation), psig	0.4 to 16.5
Eductor Suction	
Fluid	30% NaOH (solution)
Specific Gravity	1.3
Viscosity (design), cp	10
Suction Pressure, psia	9.3 to 11.0
Operating Temperature	Ambient
Suction Capacity (required), gpm	29.5

TABLE 6.3-3

SPRAY ADDITIVE TANK DESIGN PARAMETERS

Number	1
Total volume (empty), gal.	5100
Minimum volume at operating conditions (solution), gal.	4000
NaOH concentration, w/o	30
Design temperature, °F	300
Design pressure, psig	300
Operating temperature, °F	110
Operating pressure, psig	- 1*
Material	Carbon steel with stainless steel cladding

* During normal conditions there is a 1 to 2 psig N_2 gas blanket. During the accident the tank pressure will fall below atmospheric pressure; vacuum breakers are provided for this purpose.

TABLE 6.3-4

SINGLE FAILURE ANALYSIS - CONTAINMENT SPRAY SYSTEM

	Component	Malfunction	Comments and Consequences
A.	Spray Nozzles	Clogged	Large number of nozzles (315) renders clogging of a significant number of nozzles as incredible.
B.	Pumps		
	1) Containment Spray Pump	Fails to start	Two provided. Evaluation based on operation of one pump in addition to three out of five containment cooling fans operating during injection phase.
	2) Recirculation Pump	Fails to start	Two provided. Evaluation based on operation of one pump and no containment cooling fans operating during recirculation phase.
	3) Conventional Service Water	Fails to start	Three provided. Operation of one pump during recirculation required.
	4) Component Cooling	Fails to start	Three provided. Operation of one pump during recirculation required.
	5) Auxiliary Component Cooling Pump	Fails to start	Two provided. One required to operate.
C.	Automatically Operated Valves: (Open on coincidence of two - 2/3 high containment pressure signals)		
	1) Containment spray pump discharge isolation valve	Fails to open	Two provided. Operation of one required.

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TABLE 6.3-4 (Continued)

	Component	Malfunction	Comments and Consequences
	2) Spray additive tank outlet isolation valve	Fails to open	Two provided. Operation of one required.
	3) Isolation valve on component cooling water lines from residual heat exchangers	Fails to open	Two parallel lines, one valve in either line is required to open
D.	Valves Operated From Control Room for Recirculation		
	1) Containment sump recirculation isolation	Fails to open	Two lines in parallel, one valve in either line is required to open
	2) Containment spray header isolation valve from residual heat exchangers	Fails to open	Two valves provided. Operation of one required.
	3) Residual heat removal pump recirculation line	Fails to close	Two valves in series, one required to close.
	4) Residual heat removal pump discharge line	Fails to close	Two valves in series, one required to close (one valve is a check valve)
E.	Automatically Operated Valves (Close from control room on injection to recirculation changeover)		
	1) Isolation valves at spray pump discharge	Fails to close	Check valve in series with two parallel valves provided. Operation of one of the two valve arrangements in series required.

TABLE 6.3-4 (Continued)

	Component	Malfunction	Comments and Consequences
F.	Valves Operated from Control Room for Charcoal Filter Dousing		
	1) Isolation valves at filter unit	Fails to open	Two valves provided for each of the five units. Operation of one valve per unit required.

TABLE 6.3-5

SHARED FUNCTIONS EVALUATION

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Spray Additive Tank	None	Lined up for spray water diversion	Source of sodium hydroxide for spray water	Lined up for spray water diversion
Containment Spray Pumps (2)	None	Lined up to spray headers	Supply spray water to containment atmosphere	Lined up to spray headers

NOTE: Refer to Section 6.2 for a brief description of the refueling water storage tank, recirculation pumps, conventional service water pumps, component cooling pump, residual heat exchangers, component cooling heat exchangers and the auxiliary component cooling pumps which are also associated either directly or indirectly with the Containment Spray System.

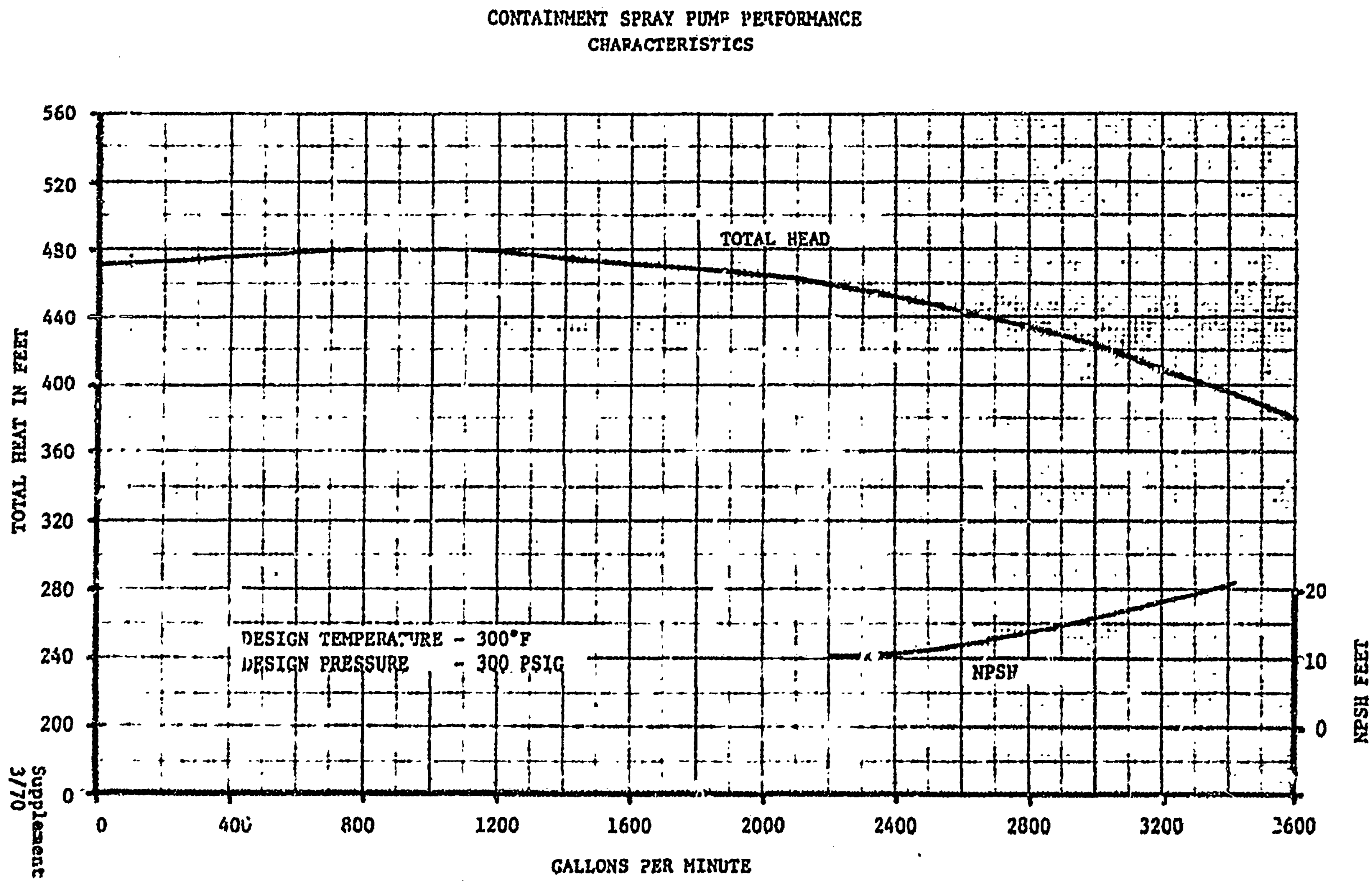

FIGURE 6.3-1

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CONTAINMENT SPRAY PUMP PERFORMANCE CHARACTERISTICS

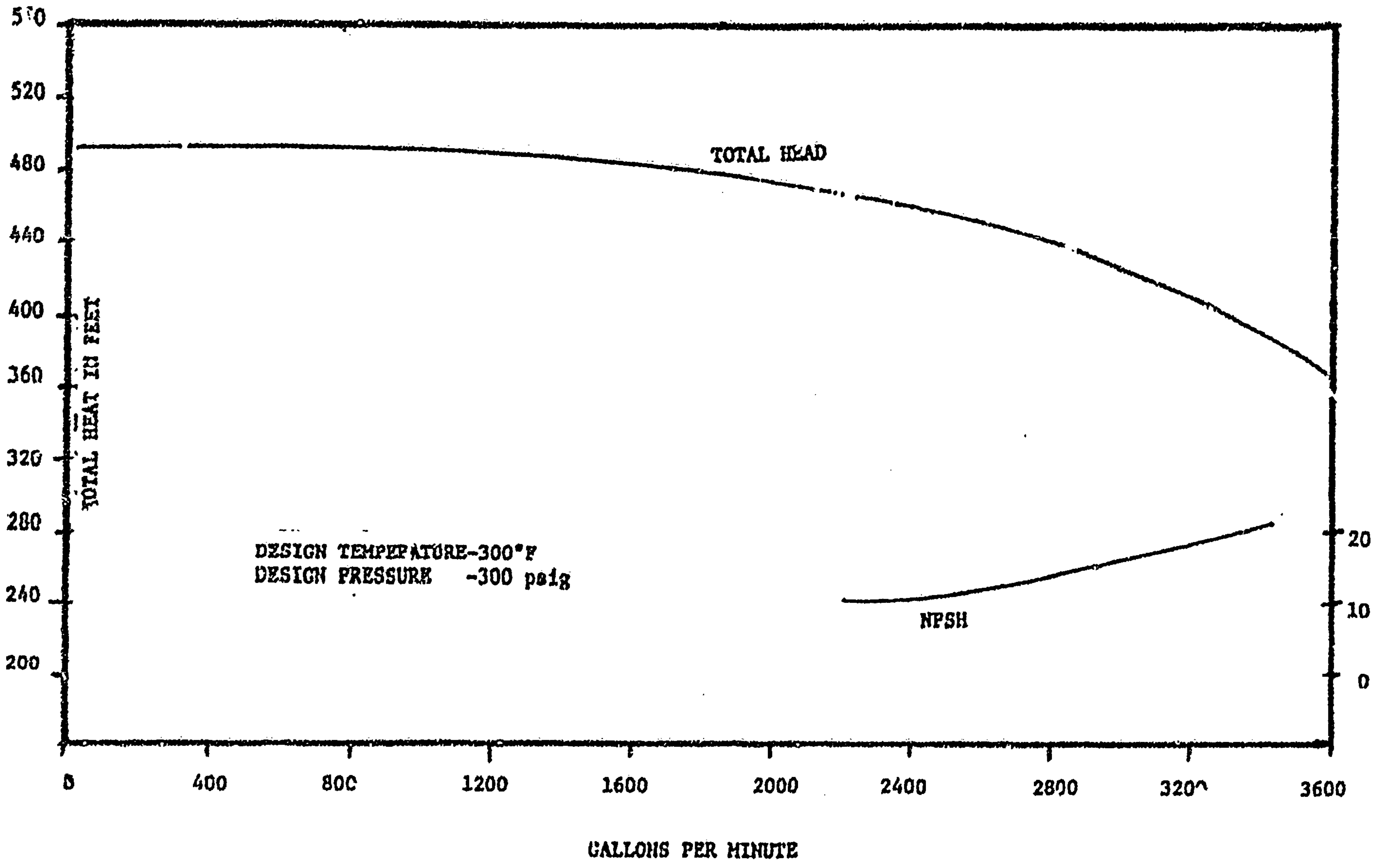


FIGURE 6.3-2

6.4 CONTAINMENT AIR RECIRCULATION COOLING AND FILTRATION SYSTEM

6.4.1 DESIGN BASES

Containment Heat Removal Systems

Criterion: Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure, this system shall perform its required function, assuming failure of any single active component. (GDC 52)

Adequate heat removal capability for the Containment is provided by two separate, full capacity, engineered safety features systems. These are the Containment Spray System, whose components are described in Section 6.3 and the Containment Air Recirculation Cooling and Filtration System, whose components operate as described in Section 6.4.2. These systems are of different engineering principles and serve as independent backups for each other.

The Containment Air Recirculation Cooling and Filtration System is designed to recirculate and cool the containment atmosphere in the event of a loss-of-coolant accident and thereby ensure that the containment pressure will not exceed its design value of 47 psig at 271°F (100% relative humidity). Although the water in the core after a loss-of-coolant accident is quickly subcooled by the Safety Injection System, the Containment Air Recirculation Cooling and Filtration System is designed on the conservative assumption that the core residual heat is released to the containment as steam.

Any of the following combinations of equipment will provide sufficient heat removal capability to maintain the post-accident containment pressure below the design value, assuming that the core residual heat is released to the containment as steam.

1) All five containment cooling fans

2) Both containment spray pumps (and one of the two spray valves in the recirculation path).

3) Three of the five containment cooling fans and one containment spray pump.

Inspection of Containment Pressure-Reducing Systems

Criterion: Design provisions shall be made to extent practical to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as pumps, valves, spray nozzles, torus, and sumps. (GDC 58)

Design provisions are made to the extent practical to facilitate access for periodic visual inspection of all important components of the Containment Air Recirculation Cooling and Filtration System.

Testing of Containment Pressure-Reducing Systems Components

Criterion: The containment pressure-reducing systems shall be designed to the extent practical so that components, such as pumps and valves, can be tested periodically for operability and required functional performance. (GDC 59)

The Containment Air Recirculation Cooling and Filtration System is designed to the extent practical so that the components can be tested periodically, and after any component maintenance, for operability and functional performance.

The air recirculation and cooling units, and the service water pumps, which supply the cooling units, are in operation on an essentially continuous schedule during plant operation, and no additional periodic tests are required.

Testing of Operational Sequence of Containment Pressure-Reducing Systems

Criterion: A capability shall be provided to test initially under conditions as close as practical to the design and the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources. (GDC 61)

Means are provided to test initially to the extent practical the full operational sequence of the Air Recirculation System including transfer to alternate power sources.

Inspection of Air Cleanup Systems

Criterion: Design provisions shall be made to the extent practical to facilitate physical inspection of all critical parts of containment air cleanup systems, such as, ducts, filters, fans, and dampers. (GDC 62)

Access is available for periodic visual inspection of the Containment Air Recirculation Cooling and Filtration System components.

Testing of Air Cleanup Systems Components

Criterion: Design provisions shall be made to the extent practical so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance. (GDC 63)

The charcoal filters of the Filtration System are bypassed during normal operation by closed butterfly valves. The valves in a non-operating unit can be periodically tested by actuating the controls and verifying deflection by instruments in the Control Room. Since the fans are normally in operation, no additional periodic fan tests are necessary.

Testing Air Cleanup Systems

Criterion: A capability shall be provided to the extent practical for in site periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits. (GDC 64)

Representative sample elements in each of the activated charcoal filter plenums will be removed periodically during shutdowns and tested on the site to verify their continued efficiency. After reinstallation the filter units will be tested in place by aerosol injection to determine integrity of the flow path.

Testing of Operational Sequence of Air Cleanup Systems

Criterion: A capability shall be provided to test initially under conditions as close to design as practical, the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability. (GDC 63)

Means are provided to test initially under conditions as close to design as is practical the full operational sequence that would bring the Containment Air Recirculation Cooling and Filtration System into action, including transfer to the emergency diesel-generator power source.

Performance Objectives

The Containment Ventilation System, Section 5, which all of the components of the Containment Air Recirculation Cooling and Filtration System (with the exception of the charcoal filters) are a part of, is designed to remove the normal heat loss from equipment and piping in the reactor containment during plant operation and to remove sufficient heat from the reactor containment, following the initial loss-of-coolant accident containment pressure transient, to keep the containment pressure from exceeding the design pressure. The fans and cooling units continue to remove heat after the loss-of-coolant accident and reduce the containment pressure close to atmospheric within the first 24 hours.

A second function of the Containment Air Recirculation Cooling and Filtration System is to remove fission products from the containment atmosphere should they be released in the event of an accident.

The filtration capacity of the system is sufficient to reduce the concentration of fission products in the containment atmosphere following a loss of reactor coolant to levels ensuring that the 2 hour and 30 day thyroid doses will be be limited to within the guidelines of 10 CFR 100 limits. Details of the site boundary dose calculation are given in Section 14.

The air recirculation filtering capacity used to satisfy the design basis is determined for the following conditions:

a) Containment leak rate of 0.1% per day.

b) Conservative meteorology corrected for building wake effects.

c) 5% efficiency for filtration of organic iodine. This assumes credit for the demonstrated ability to filter organic forms of iodine at high relative humidity with impregnated charcoal.(1),(2), (3)

d) Fission product release to the containment per TID 14844 at a power level of 3216 MWt. This assumes no credit for safety injection in limiting fission product release.

e) Partial effectiveness of the filtration equipment. This assumes two of the five installed charcoal filter units are unavailable at the time of the loss of coolant.

In addition to the design bases specified above, the following objectives are met to provide the engineered safety features functions:

a) Each of the five fan-cooler units is capable of transferring heat at the rate of 21,200 Btu/sec. (76.32×10^6 Btu/hr) from the containment atmosphere at the post-accident design conditions, i.e., a saturated air-steam mixture at 47 psig 271°F. This heat transfer rate is that assigned to the fan-cooler units in the analysis of containment and related heat removal system capability in Section 14.

The establishment of basic heat transfer design parameters for the cooling coils of the fan-cooler units, and the calculation by computer of the overall heat transfer capacity are discussed in Section 14. Among the topics covered are selection of the tube side fouling factor, effect of air side pressure drop, effect of moisture entrainment in the air steam mixture entering the fan-coolers, and calculation of the various air side to water side heat transfer resistances.

b) In removing heat at the design basis rate, the coils are capable of discharging the resulting condensate without impairing the flow capacity of the unit and without raising the exit temperature of the service water to the boiling point. Since condensation of water from the air-steam mixture is the principal mechanism for removal of heat from the post-accident containment atmosphere by the cooling coils, the coil fins will operate as wetted surfaces under these conditions. Entrained water droplets added to the air-steam mixture, such as by operation of the containment spray system, will therefore have essentially no effect on the heat removal capability of the coils.

c) Each of the five air handling units is equipped with moisture separators and high efficiency particulate air (HEPA) filters rated for full unit flow. The latter are capable of 99.97% removal efficiency for 0.3μ particles at the post-accident conditions.

d) Each of the five air handling units is capable of supplying air to separate carbon-bed filter units following an accident for fission product iodine removal. The design flow rate through each carbon filter unit is 8,000 cfm, at a face velocity of approximately 50 fpm. The remainder of the flow by-passes the carbon filters via baffle plates having the same pressure drop as the carbon filters. The carbon filter units are designed to remove at least 5% of the incident radioactive iodine in the form of methyl iodide (CH_3I).[(1), (2), (3)] These are the iodine removal efficiencies assumed in the analysis of containment capability to retain fission product iodine under the post-accident design conditions in Section 14.

In addition to the above design bases, the equipment is designed to operate at the post-accident conditions of 47 psig and 271°F for three hours, followed by operation in an air-steam atmosphere at 20 psig, 219°F for an additional 21 hours. The equipment design will permit subsequent operation in an air-steam atmosphere at 5 psig, 152°F for an indefinite period.

All components are capable of withstanding or are protected from differential pressures which may occur during the rapid pressure rise to 47 psig in ten (10) seconds.

Portions of other systems which share functions and become part of this con-tainment cooling system when required are designed to meet the criteria of this section. Neither a single active component failure in such systems during the injection phase nor an active/passive failure during the recirculation phase will degrade the heat removal capability of containment cooling.

Where portions of these systems are located outside of containment, the following features are incorporated in the design for operation under post-accident conditions:

a) Means for isolation of any section

b) Means to detect and control radioactivity leakage into the environs, to the limits consistent with guidelines set forth in 10 CFR 100.

6.4.2 SYSTEM DESIGN AND OPERATION

The Flow Diagram of the Containment Air Recirculation Cooling and Filtration System is shown in Figure 6.4-1.

Individual system components and their supports meet the requirement for Class I (Seismic) structures and each component is mounted to isolate it from fan vibration.

Containment Cooling System Characteristics

The air recirculation system consists of five 20% capacity air handling units, each including motor, fan, cooling coils, moisture separators, roughing filters and HEPA filters, duct distribution system, instrumentation and controls. The units are located on the intermediate floor between the containment wall and the primary compartment shield walls. In addition each of the five air-handling units is equipped with an activated charcoal filter unit, normally isolated from the main air recirculation stream. The air flow (air-steam mixture) is bypassed through the charcoal filter units to remove volatile iodine following an accident.

Each fan is designed to supply 65,000 cfm at approximately 22.8" s.p., 271°F, 0.175 lb/ft^3 density. The fans are direct driven, centrifugal type, and the coils are plate fin-tube type. Each air handling unit is capable of removing 76.32×10^6 Btu/hr from the containment atmosphere under accident conditions. 2000 gpm of service (cooling) water is supplied to each unit during accident conditions. The design maximum river water inlet temperature is 85°F which results in a maximum outlet temperature of 161°F.

Air operated, tight closing, 125 lb USAS butterfly valves isolate any inactive air handling unit from he duct distribution system. Duct work distributes the cooled air to the various containment compartments and areas. During normal operation, the flow sequence through each air handling unit is as follows: cooling coils, moisture separators, HEPA filters, fan, discharge header. Roughing filters are installed up-stream of the cooling coils during plant clean-up and any time the reactor is down.

In an event of an accident, the flow sequence would be the same except that the fan di harge would be automatically diverted by air operated butterfly valves . compartment containing the charcoal filters before entering the discharge header for distribution. Figure 6.4-2 is an arrangement drawing showing the details of the charcoal filter unit.

Figure 6.4-3 is an engineering layout drawing of an air handling unit, showing the arrangement of the above components in the unit. Figure 6.4-4 shows the location of the five units on the intermediate floor (elevation 68'-0").

Actuation Provisions

The butterfly valves used to route air flow through the charcoal filters have only two positions, full open and full closed. These valves are air operated and spring loaded. Upon loss of control signal or control air, the spring actuates the valve to the accident position (fail-safe operation).

Upon either manual or automatic actuation of the safety injection safe uards sequence, the butterfly valves are tripped to the accident position. Accident position is also the "fail-safe position."

Redundant electrically operated three-way solenoid valves are used with each butterfly valve to control the instrument air supply (control air). These valves are arranged so that failure of a single solenoid valve to respond to the accident signal will not prevent actuation of the but erfly valve to the accident position (fail-safe operation).

The containment pressure is sensed by six separate pressure transmitters located outside the containment. Containment pressure is communicated to the transmitters through three 1" stai , steel lines penetrating the containment vessel. A high containment pressure signal automatically actuates the safety injection safeguard sequence (Reference is made to Section 6.2.2) which trips the valves to the accident position.

The fans are part of the engineered safety features and either all five, or at least three of five fans will be started after an accident, depending on the availability of emergency power. Reference is made to Section 8.

Overload protection for the fan motors is provided at the switchgear by overcurrent trip devices in the motor feeder breakers. The breakers can be operated from the control room and can be reclosed from the control room following a motor overload trip..

Redundant flow switches in the system, operating both normally and post-accident, indicate whether air is circulating in accordance with the design arrangement. Abnormal flow alarms are provided in the control room.

Flow Distribution and Flow Characteristics

The location of the distribution ductwork outlets, with reference to the location of the air handling unit return inlets, ensures that the air will be directed to all areas requiring ventilation before returning to the units. The arrangement is shown in Figure 6.4-1.

In addition to ventilating areas inside the periphery of the shield wall, the distribution system also includes two branch ducts located at opposite extremes of the containment wall for ventilating the upper portion of the containment. These ducts are provided with nozzles and extend upward along the containment wall as required to permit the throw of air from nozzles to reach the dome area and assure that the discharge air will mix with the atmosphere.

The air discharge inside the periphery of the shield wall will circulate and rise above the operating floor through openings around the steam generators where it will mix with air displaced from the dome area. This mixture will return to the air handling units through floor grating located at the operating floor directly above each air handling unit inlet. The temperature of this air will be essentially the ambient existing in the containment vessel.

The steam-air mixture from the containment entering the cooling coils during the accident will be at approximately 271°F and have a density of 0.175 pounds per cubic foot. Part of the water vapor will condense on the cooling coil, and the air leaving the unit will be saturated at a temperature slightly below 271°F. The fluid will leave the cooling coils and enter the moisture separators at approximately 271°F and saturated (100% R.H.) condition. The purpose of the moisture separators is to remove the entrained moisture.

The fluid will remain in this condition as it flows through the HEPA filter and into the fan, but will pick up some sensible heat from the fan and fan motor before flowing through the charcoal filters and into the distribution header. This sensible heat will increase the dry-bulb temperature slightly above 271°F and will decrease the relative humidity slightly below 100%.

With a flow rate of 65,000 cfm from each fan under accident conditions and the containment free volume of 2,610,000 ft^3 the recirculation rate with five fans operating is approximately 7.5 containment volumes per hour.

Charcoal Filter High Temperature Detection and Dousing System

The five charcoal filter units are provided with high-temperature detectors, and associated alarms in the control room. Each charcoal filter unit is also provided with a spray system for water dousing, upon a signal of high temperature.

Capability for detecting and alarming the presence of fires and localized hot spots in the charcoal filters is provided by a system of temperature switches. Each charcoal filter plenum (containing one bank of 12 adsorber units in an 62" wide by 49 1/2" high array) is provided with 6 temperature switches. These switches are uniformly distributed for good coverage. The temperature switches are set to close at 400°F, (which is significantly below the charcoal ignition temperature of 680°F) and are wired in parallel to a common alarm in the control room. Thus closing of a single switch will actuate the alarm to indicate a high temperature condition in the filter plenum.

The water dousing system provided with each charcoal filter plenum is designed to drench the absorbers thoroughly in the extremely unlikely event of a charcoal fire during the post-accident recovery. Water for this system is obtained from the main headers of the containment spray system through a separate 2 inch stainless steel line to each filter plenum. There are two normally closed motor operated valves in parallel in each 2 inch line.

The Containment Spray System is automatically actuated and will be running in the event of a loss-of-coolant accident (injection phase). In the event of a high temperature alarm in a filter unit, the operator manually initiates filter dousing by actuating the parallel-connected isolation valves for each filter assembly. Because of the piping arrangement either of the two spray pumps can be used to feed the dousing lines. The dousing flow (approximately 12 gpm per fan cooler unit) is sized to completely wet the charcoal and remove the decay heat of the adsorbed iodine thereby preventing heating to the ignition temperature. The system is designed so containment spray at slightly reduced flow can continue simultaneously with filter dousing.

During the recirculation phase of core cooling, operation of the dousing system is the same as above except that water to the spray headers is supplied from the discharge of the residual heat removal heat exchangers.

Cooling Water for the Fan Cooler Units

The cooling water requirements for all five fan cooling units during a major loss of primary coolant accident and recovery are supplied by two of the three nuclear service water pumps. The Service Water System is described in Section 9.

The cooling water discharges from the cooling coils to the discharge canal and is monitored for radioactivity by routing a small bypass flow from each through a common radiation monitor. Upon indication of radioactivity in the effluent, each cooler discharge line is monitored individually to locate the defective cooling coil, which when identified would remain isolated, and operation would continue with the remaining units. The service water system pressure at locations inside the containment is 15 to 20 psig, which is below the containment design pressure of 47 psig. However, since the cooling coils and service water lines are completely closed inside the containment, no contaminated leakage is expected into these units.

Local flow and temperature indication is provided outside containment, for service water flow to each cooling unit. Abnormal flow alarms are provided in the control room.

During normal plant operation, flow through the cooling units is throttled for containment temperature control purposes by a valve on the common discharge header from the cooling units. Two independent, full flow, isolation valves open automatically in the event of a high containment pressure signal or safety injection signal to bypass the control valve. Both valves fail in the open position upon loss of air pressure and either valve is capable of passing the full flow required for all five fan cooling units.

Environmental Protection

All system control and instrumentation devices required for containment accident conditions are located to minimize the danger of control loss due to missile damage. Flow switches in the ductwork system, operating both normally and post-accident, indicate whether air is circulating in accordance with the design arrangement. Abnormal flow alarms are provided in the control room.

All fan parts, valve shaft and disc seating surfaces and ducts in contact with the containment fluid are protected against corrosion. The fan motor enclosures, electrical insulation and bearings are designed for operation during accident conditions.

All of the air handling units are located on the intermediate floor between the Containment Vessel and the primary compartment shield wall. The distribution header and service water cooling piping are also located outside the shield wall. This arrangement provides missile protection for all components.

Components

Moisture Separators

The moisture separators are designed to remove a minimum of 99.9% of the entrained water in the air-steam entering the air handling units following a loss-of-coolant accident. With an air entrained moisture content of 0.35 lb $H_2O/1000\ ft^3$ the water flow rate entering the moisture separator section is approximately 23 lb/min, and the moisture separator effluent has essentially zero moisture content.

Each bank is designed for horizontal air flow and is composed of forty (40) elements. Each element or separator is 24 in. x 25 in. x 2 in. (minimum) thick and is mounted in a steel support frame.

A steel drain trough is incorporated for each horizontal tier of separators to collect and remove the water that is recovered from the air stream. Further, the design enables the separators to be removed from the upstream side of the support frame.

In order to prevent the bypass of air around the bank, air-tight seals are provided between the floor, walls, plenum, and around the perimeter of each moisture separator. The tight seal is accomplished by gaskets, adhesive, and pressure-sealing tape, all of which can withstand a temperature of 300°F. The thickness of the gaskets is 1/4 in. for the separator elements and 3/8 in. for the perimeter sealing of the support frame; and they do not extend into the media area when installed.

The moisture separator elements are of fire resistant construction, and consist of mats of fiberglass pads reinforced with stainless steel wire mesh. Non-stainless steel parts used in the construction are protected against corrosion by painting with one (1) three-mil shop coat of Carbo Zinc No. 11 or equal. The separator frames are fabricated of Type 304L stainless steel, with welded joints.

Roughing Filters

The roughing filters remove the large particles from the air stream before it contacts the cooling coils. The roughing filters are in operation during plant clean up and any time the reactor is down. These are efficient for removing large particles. Under normal air flow, they offer a resistance to air flow of 0.2 inches of water.

As in the case for all components of the air handling recirculating system, the bank will be designed for horizontal air flow. The bank contains forty (40) filters, each of which has dimensions of 24 in. x 24 in. x 2 in. thick.

All other details of the mounting frame, sealing and materials of construction, other than the filters themselves, are the same as described for the moisture separators.

The filter is of fire resistant construction with the media composed of a glass fiber mat reinforced with stainless steel wire cloth.

HEPA (absolute) Filters

The high efficiency particulate air (HEPA) filters are capable of 99.97% removal efficiency for 0.3μ particles at the post accident design conditions. All materials of construction of these filters are compatible with the sodium hydroxide/boric acid solution in the post accident environment.

The filter media is made of glass fiber with asbestos and can withstand the incident ambient steam/air temperature conditions and 100% relative humidity. Filter frames are made of stainless steel, and asbestos separators resistant to moisture and high temperature are used.

Fan-Motor Units

The five containment cooling fans are of the centrifugal, non-overloading, direct drive type.

Each fan can provide a minimum flow rate of 65,000 cfm when operating against the system resistance of approximately 22.8" SP existing during the accident condition (0.175 lb/ft^3 density, a containment pressure of 47 psig, and temperature of 271°F).

The reactor containment fan cooler motors are Westinghouse, totally enclosed water cooled, 350 horse power, induction type, 3 phase, 60 cycle 1200 RPM, 440 volt with ample insulation margin. Significant motor details are as follows:

a. Insulation - Class F (NEMA rated total temperature 155°C) Westinghouse Thermalastic. It is impregnated and coated to give a homogeneous insulation system which is highly impervious to moisture. Internal leads and the terminal box-motor interconnection are given special design consideration to assure that the level of insulation matches or exceeds that of the basic motor system. At incident ambient and load conditions, (270°F and 350 HP) the motor insulation hot spot temperature is not expected to exceed 127°C.

b. Heat Exchanger

An air to water, heat exchanger is connected to the motor to form an entirely enclosed cooling system. Air movement is through the heat exchanger and is returned to the motor. A vent valve permits incident ambient (increasing containment pressure) to enter the motor air system so the bearings will not be subjected to differential pressure. The cooling coil condensate drain line will enable pressure equalization as the containment pressure is reduced by the motor heat exchanger. Water connections are welded throughout and supply and discharge are common with the containment cooler water system, i.e., supplied from the nuclear service water header. The drain will be piped to the containment cooler drain system.

c. Bearings

The motors are equipped with high temperature grease lubricated ball bearings as would be required if the bearings were subjected to incident ambient temperatures. Continuous bearing monitoring is provided which will alarm in the control room.

Conduit (Connection) Box

The motor leads are brought out of the frame through a seal and into a sealed conduit box.

Factory Tests

In addition to the usual quality control tests which are performed to give assurance that the motors meet design specifications, special tests have been performed to demonstrate that insulation margins are built in as expected. The completely wound stators have been given a special high potential test to ground. The stators were immersed in water, meggered and given a high potential test while immersed. After passing the water tests, the motor was baked and given a final coating dip. The stator and rotor were then again baked.

Charcoal Filters

The charcoal filters are fabricated with stainless steel frames filled with impregnated, activated charcoal. The cell construction insures compacted carbon beds of uniform density and thickness.

The design flow rate through each carbon filter unit is 8,000 cfm, at a face velocity of approximately 50 fpm. These units are designed to remove at least 5% of the incident radioactive iodine in the form of methyl iodide (CH_3I).[(1), (2)]

Each of the five charcoal filter units consists of an airtight plenum containing a single bank of charcoal filter cells. Air flow enters the plenum through one end, passes through ductwork into the main distribution header. Only 8,000 cfm of the total 65,000 cfm actually enters the 12 carbon cells. The remainder is by-passed through perforate plates having the same pressure drop as the carbon cells. An open flow area in the plenum of ~16% will balance the pressure drop between the perforated plate and the carbon cells used. The perforated plate will be 13 gage with a 15-1/2% open area formed by 5/16 in. holes on 0.756 in. staggered centers. The plates will be cut to the size of the carbon cell seal flange and held in place by the fasteners used to hold the carbon cell. (See Figure 6.4-2.) Seal plates are used to block off flow area to achieve the proper flow areas.

The individual filter cells are of the "flat-bed" type of construction, with two 2-inch thick horizontal charcoal elements separated by a 2-inch air gap. The sides and back of the cell are enclosed by solid (unperforated) stainless steel sheet metal; the larger (horizontal) surfaces are enclosed by perforated stainless steel sheets. An unperforated stainless steel sheet seals the front edge; this sheet is slightly larger than the basic filter dimensions to provide flanges for clamping in the mounting frame. Several rectangular slots are cut in the front face to permit air flow. Each filter cell provides approximately 11.2 sq. ft. of active surface area for air flow. The charcoal used is MSA type 85851. The volume of impregnated charcoal per cell is 2.05 cubic feet consisting of a minimum charcoal weight of 44.7 lbs. and 2.0 lbs (4.5% by weight) of impregnated iodine per cell. 12

During operation, air flows vertically downward through the top surface of the filter and upward through the bottom surface, enters the air space between the two charcoal elements, and is discharged through the slots in the front face.

Each filter bank consists of 12 cells in a 2 ft. wide by 6 ft. high array. The downstream mounting racks arrangement permits removal of individual cells from the side of the plenum. Perforate and seal plates fill the remainder of the flow area.

The duct connections are flexible to prevent transmissions of duct vibration to the filter units.

The filter units are designed to withstand the maximum differential pressure developed by the fans under accident conditions without developing internal leaks or being dislodged from their frame seals.

<u>Charcoal Filter Dousing System</u>

The spray water dousing system inside the charcoal filter plenums is of stainless steel and copper construction. This system provides three individual injection lines, terminating in 3/8 inch brass nozzles, which spray into the air space between each pair of vertically adjacent adsorber units. The nozzles discharge horizontally, to assure complete wetting of both upper and lower adsorber surfaces. The design flow of approximately 1/3 gpm per nozzle results in a design pressure drop of 60 psi. The nozzle orifice is not subject to clogging with particle size less than 0.045 inches.

Cooling Coils

The coils are fabricated of copper plate fins vertically oriented on copper tubes. The heat removal capability of the cooling coils is 76.32×10^6 Btu/hr per air handling unit at saturation conditions (271°F, 47 psig).

The design internal pressure of the coil is 150 psig at 300°F and the coils can withstand an external pressure of 70.5 psig at a temperature of 300°F without damage.

Each recirculating unit will consist of ten (10) coil units mounted in two banks of five (5) coils high. These banks will be located one behind the other for horizontal series air flow, and the tubes of the coil will be horizontal.

Each coil assembly consists of one bank of four row deep coils, and one bank of six row deep coils. Each bank contains four Westinghouse Sturtevant designation WC-36108 (36" high by 108" long) coils, and one Westinghouse Sturtevant WC-30108 (30" high by 108" long) coil. The coils are stacked five high to a bank. The total coil assembly (two banks of coils) is 42" wide. There are 10 rows of tubes in the horizontal flow direction and a total of 116 rows of tubes in the vertical direction. Cooling water flow will be 1/3 velocity through the first coil bank (6 rows of tubes in the horizontal) and half velocity through the second coil bank (4 rows of tubes in the horizontal). Tube supports will be provided on 15 in. center lines to permit free expansion and contraction of the tubes.

For normal operation, 8 fins/inch are required to remove 2,000,000 Btu/hr using 108 tubes.

Local flow and temperature indication of service water are provided at each air handling unit. Alarms indicating abnormal service water flow, and radioactivity are provided in the control room.

The coils are provided with drain pans and drain piping to prevent flooding during accident conditions. This condensate is drained to the Containment Sump. Reference is made to Section 6.7.

Ducting

The ducts are designed to withstand the sudden release of reactor coolant system energy and energy from associated chemical reactions without failure due to shock or pressure waves by incorporation of dampers along the ducts which open at slight overpressure, 1.0 psi. The ducts are designed and supported to withstand thermal expansion during an accident.

Where flanged joints are used, joints are provided with gaskets suitable for temperatures to 300°F.

Ducts are constructed of corrosion resistant material.

Butterfly Valves

The spring loaded air operated valves are tight sealing when closed. This prevents leakage of air into the charcoal filter compartment during normal operation thereby preventing charcoal deterioration. These valves fail to the open position to assure flow through the charcoal filters during the accident condition.

Electrical Supply

Details of the normal and emergency power sources are presented in Section 8.

Further information on the Components of the Containment Air Recirculation Cooling and Filtration System is given in Section 5.

6.4.3 DESIGN EVALUATION

Range of Containment Protection

The Containment Air Recirculation Cooling and Filtration System provides the design heat removal capacity and the design iodine removal capability for the containment following a loss-of-coolant accident assuming that the core residual heat is released to the containment as steam. The system accomplishes this by continuously recirculating the air-steam mixture: 1) through coiling coils to transfer heat from containment to service water, and 2) through activated charcoal filters to transfer methyl iodide to the filters from the air-steam mixture.

The performance of the Containment Recirculation Cooling and Filtration System in pressure reduction and iodine removal is discussed in Section 14..

Any of the following combinations of equipment will provide sufficient heat removal capability to maintain the post-accident containment pressure below the design value assuming that the core residual heat is released to the containment as steam.

1) All five containment cooling fans

2) Both containment spray pumps (and one of the two spray valves in the recirculation path).

3) Three of the five containment cooling fans and one containment spray pump.

System Response

The starting sequence of the last of the five containment cooling fans (at design conditions five of the fans and two of the nuclear service water pumps operate during normal power operations for containment ventilation) and the related emergency power equipment are designed so that delivery of the minimum required air flow to the charcoal filters and cooling water flow is reached in 58 seconds. In the analysis of the containment pressure transient, Section 14.3, a delay time of 60 seconds was assumed.

The starting sequence is:		Seconds
a)	Initiation of safety injection signal, including instrument lag.	1
b)	Starting of diesel generators	19
c)	Starting of last containment cooling fan	38
		Total 58

The valves are actuated to safeguards position by the safety injection signal.

Single Failure Analysis

A failure analysis has been made on all active components of the system to show that the failure of any single active component will not prevent fulfilling the design function. This analysis is summarized in Table 6.4-1.

The analysis of the loss-of-coolant accident presented in Section 14 is consistent with the single failure analysis.

Loss of a fan motor in a unit should not result in ignition of the charcoal. Ignition should be prevented by backflow induced by the operating fans. If an increase in the charcoal filter temperature were to occur, the high temperature detectors would initiate an alarm and the operator would cause the affected bank to be sprayed.

Reliance on Interconnected Systems

The Containment Air Recirculation Cooling and Filtration System is dependent on the operation of the electrical and service water systems. Cooling water to the coils is supplied from the service water system. Three nuclear service water pumps are provided, only two of which are required to operate during the post-accident period.

Shared Function Evaluation

Table 6.4-2 is an evaluation of the main components which have been discussed previously and a brief description of how each component functions during normal operation and during the accident.

Reliability Evaluation of the Fan Cooler Motor

The basic design of the motor and heat exchanger as described herein is such that the incident environment is prevented, in any major sense, from entering the motor winding or when entering in a very limited amount (equalizing motor interior pressure) the incoming atmosphere is directed to the heat exchanger coils where moisture is condensed out. If some quantity of moisture should pass through the coil, the changed motor interior environment would "clean up" in that interior air continually recirculates through the heat exchanger.

It will be noted that the motor insulation hot spot temperature is not expected to exceed 127°C even under incident conditions. Rated life could be expected with a continuous hot spot of 155°C.

During the lifetime of the plant, these motors perform the normal heat removal service and as such are only loaded to approximately 120-150 HP.

The bearings are designed to perform in the incident ambient temperature conditions. However, it will be noted that the interior bearing housing details are cooled by the heat exchanger. It is expected that bearing temperatures would be 125°C to 140°C, under incident conditions.

The insulation has high resistance to moisture and tests performed indicate the insulation system would survive the incident ambient moisture condition without failure. The heat exchanger system of preventing moisture from reaching the winding keeps the winding in much more favorable conditions. In addition, it will be noted that at the time of the postulated incident, the load on the fan motor would increase, internal motor temperature would increase, and would, therefore, tend to drive any moisture if present, out of the winding. Additionally, the motors are furnished with insulation voltage margin beyond the operating voltage of 440 V.

Following the incident rise in pressure, it is not expected that there will be significant mixing of the motor (closed system) environment and the containment ambient.

The heat exchanger has been designed using a very conservative fouling factor.

To prove the effectiveness of the heat exchanger in inhibiting large quantities of the steam air mixture from impinging on the winding and bearings, a full scale motor of the exact same type as described, was subjected to prolonged exposure of accident conditions. The test exposed the motor to a steam air mixture as well as boric acid and alkaline spray at 80 psig and saturated temperature conditions. Insulation resistance, winding and bearing temperature, relative humidity, voltage and current as well as heat exchanger water temperature and flow were recorded periodically during the test.

Following the test the motor was disassembled and inspected to further assure that the unit performed as designed. The post-testing inspection showed no degradation of the motor components (Details are found in WCAP 9003, Westinghouse proprietary document).

6.4.4 MINIMUM OPERATING CONDITIONS

The Technical Specifications establish limiting conditions regarding the operability of the air recirculation units when the reactor is critical.

6.4.5 INSPECTION AND TESTING

Inspection

Access is available for visual inspection of the containment fan-cooler and recirculation filtration components including fans, cooling coils, butterfly valves, filter units and ductwork. Provision has been made for ready removal of a section of the filter banks for inspection and testing.

Testing

Component Testing

The HEPA filters used in the containment fan cooler system are specified to operate in the post-accident containment environment. Each filter is subjected to standard manufacturer's efficiency and production tests prior to shipment.

These include flow resistance tests and the standard Efficiency Penetration test r[illegible] that penetration does not exceed 0.03 percent for 0.3 micr[illegible] homogeneous dioctylphthalate (DOP) particles.

Evaluati[illegible]s are performed on sample filters constructed from the filter medium to demonstrate retention of strength under wet conditions as follows:

(1) The filter is exposed to a flow of wet steam and water spray in a test facility which will simulate the actual filter installation. The water is injected ahead of the filter with a nozzle designed to produce a fine spray. Free (unentrained) moisture will be removed by means of a moisture separator upstream of the filter but no provision will be made for removal of entrained moisture entering the filter.

(2) Following the wet flow test in (1) above, the filter will be dried and tested to demonstrate that its resistance to flow has not significantly increased.

(3) Following test (2) above, the filter will be subjected to the NBS Dust Loading Test followed by an Ultimate Strength Test with the deposited dust still on the filter.

Only filters of a type which have been certified to have passed these tests are accepted for initial use or replacement in the fan coolers application.

Any of the activated charcoal filter absorbers in the air handling units can be removed and tested periodically for effectiveness in removing methyl iodine forms. In addition, periodic, inplace testing of the filtration assemblies will be made by injection of a freon aerosol in the air stream at the filter inlet to verify the leak-tightness of individual filter elements and their frame seals.

The butterfly valves on each air handling unit can be operated periodically to assure continued operability. The degree of leak tightness of the valves will be established by test at the time of installation.

System Testing

Each fan cooling unit will be tested after installation for proper flow and distribution through the duct distribution system. Four of the fan cooling units are used during normal operation. (Five will only be required for normal operation at design conditions i.e., when the service water inlet temperature is 85°F and this condition is expected to exist only for relatively short periods, if at all). The fan not in use can be started from the control room to verify readiness. The butterfly valves directing flow through the charcoal filter banks will be tested only when the fan is not running.

After reinstallation, following testing, the filter charcoal units will be tested in place by aerosol injection to determine integrity of the flow path.

<u>Operational Sequence Testing</u>

The test described in 6.2.5 will demonstrate proper transfer and sequencing of the fan motor supplies from the diesel generators in the event of loss of power. A test signal will be used to demonstrate proper valve motion and fan starting prior to installation of the charcoal filters. This test will verify proper functioning of the vane-switch flow indicators.

References

(1) "Connecticut-Yankee Charcoal Filter Tests", CYAP 101, (December, 1966).

(2) Ackley, R. D. and R. E. Adams, "Trapping of Radioactive Methyl Iodide from Flowing Steam-Air : Westinghouse Test Series", ORNL-TM-2728, (December, 1969).

(3) Nuclear Safety Quarterly Report - August, September, October, 1969, Engineered Safety System Studies, BNWL-1266.

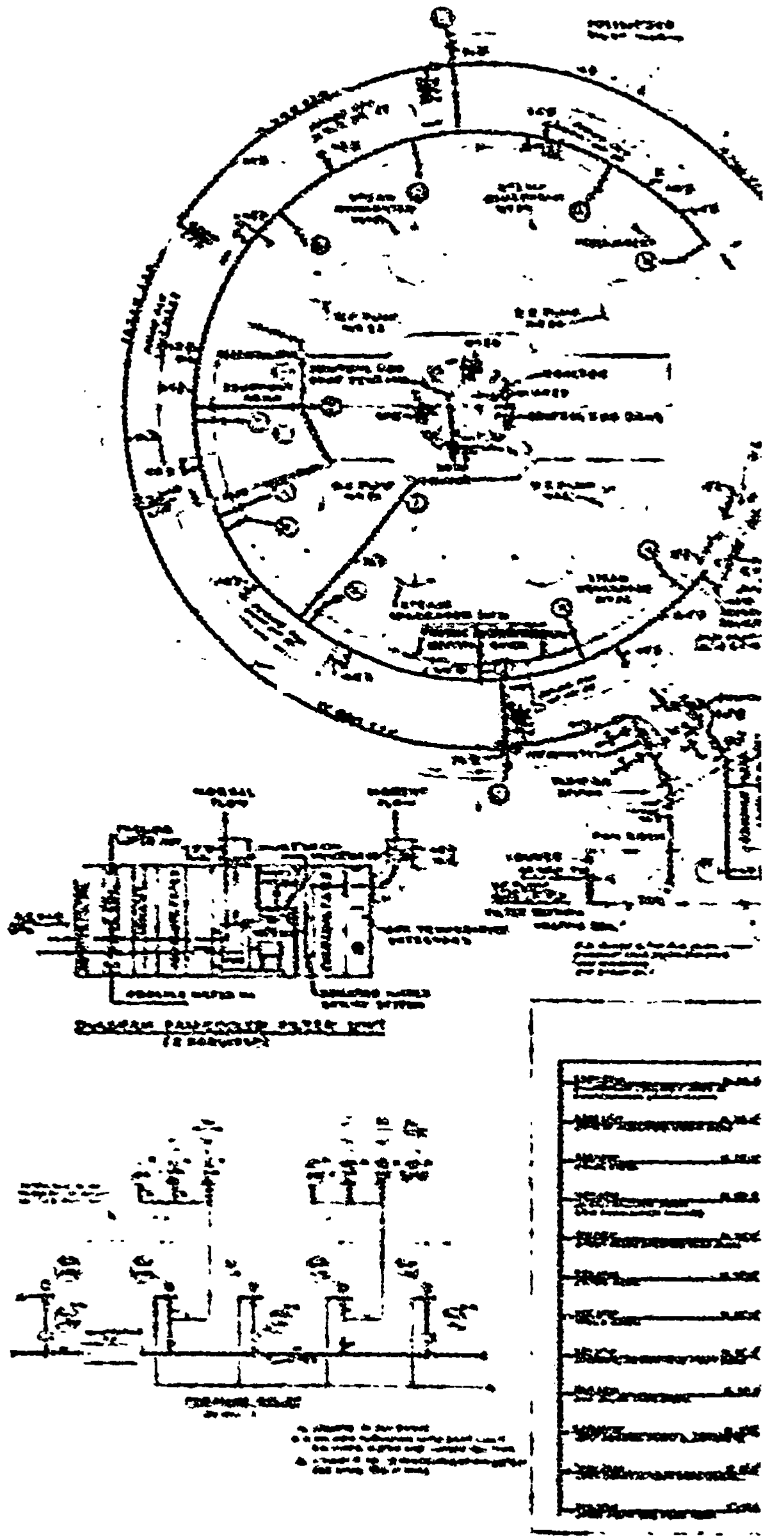
TABLE 6.4-1

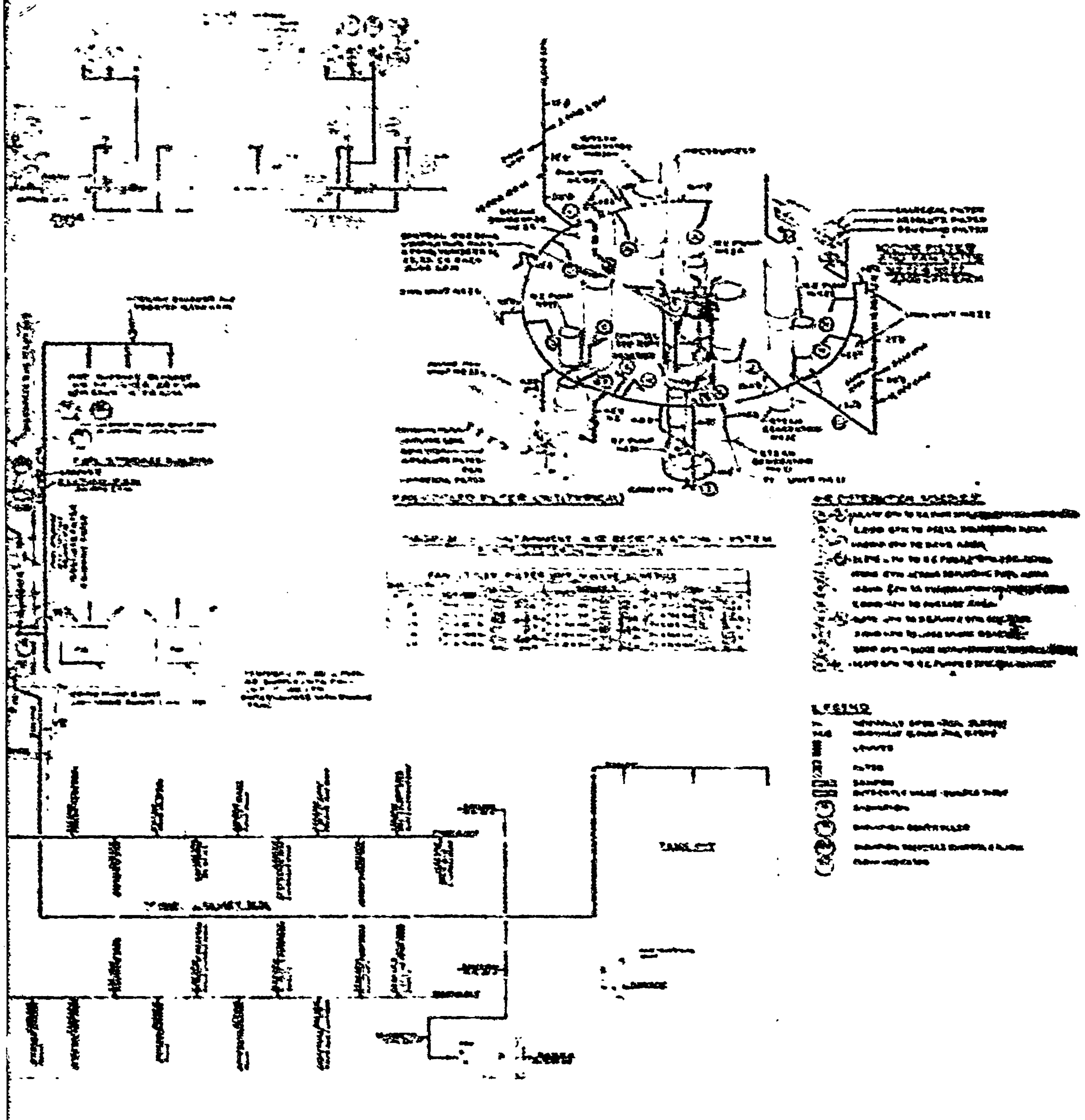
SINGLE FAILURE ANALYSIS - CONTAINMENT AIR RECIRCULATION COOLING AND FILTRATION SYSTEM

Component	Malfunction	Comments and Consequences
A. Containment Cooling Fan	Fails to start	Five provided. Evaluation based on three fans in operation and one containment spray pump operating during the injection phase.
B. Nuclear Service Water Pumps	Fails to start	Three provided. Two required for operation.
C. Automatically Operated Valves: (Open on automatic safeguards sequence)		
1) Charcoal filter compartment butterfly valves	Fails to open	Five filters provided. Evaluation based on three filters in operation and one containment spray pump in operation during the injection phase.
2) Nuclear service water discharge line isolation valve	Fails to open	Two provided. Operation of one required.

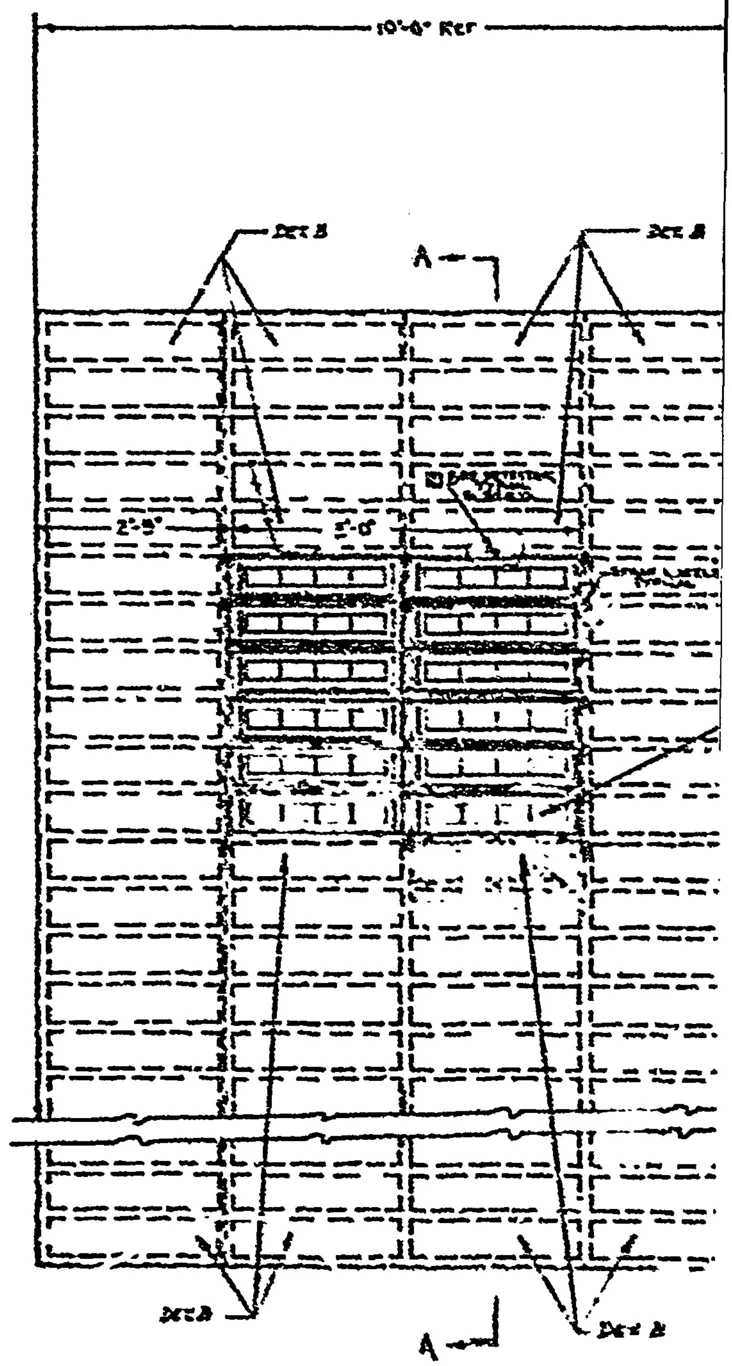
TABLE 6.4.2 - Shared Function Evaluation

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Containment Cooling Fan Units (5)	Circulate and cool containment atmosphere	Up to five fan units in service	Circulate and cool containment atmosphere	Five fan units in service
Nuclear Service Water Pumps (3)	Supply river cooling water to fan units	Two pumps in service	Supply river cooling water to fan units	Two pumps in service
Charcoal Filter Units (5)	None	Isolated from normal fan discharge flow	Remove iodine from containment atmosphere	Lined up to receive fan discharge flow

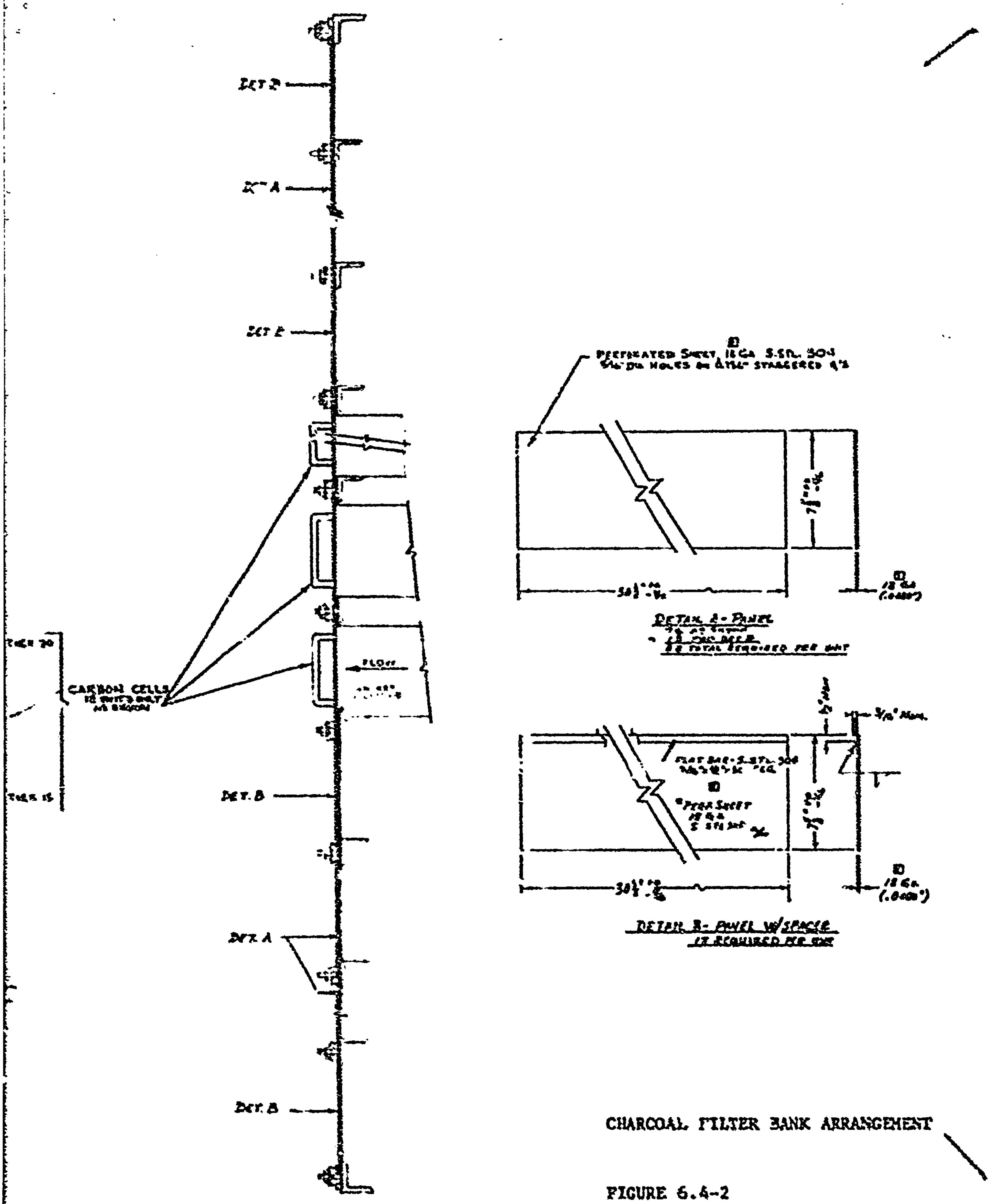




CONTAINMENT AIR RECIRCULATION AND FILTRATION SYSTEM FLOW DIAGRAM
FIGURE 6.4-1

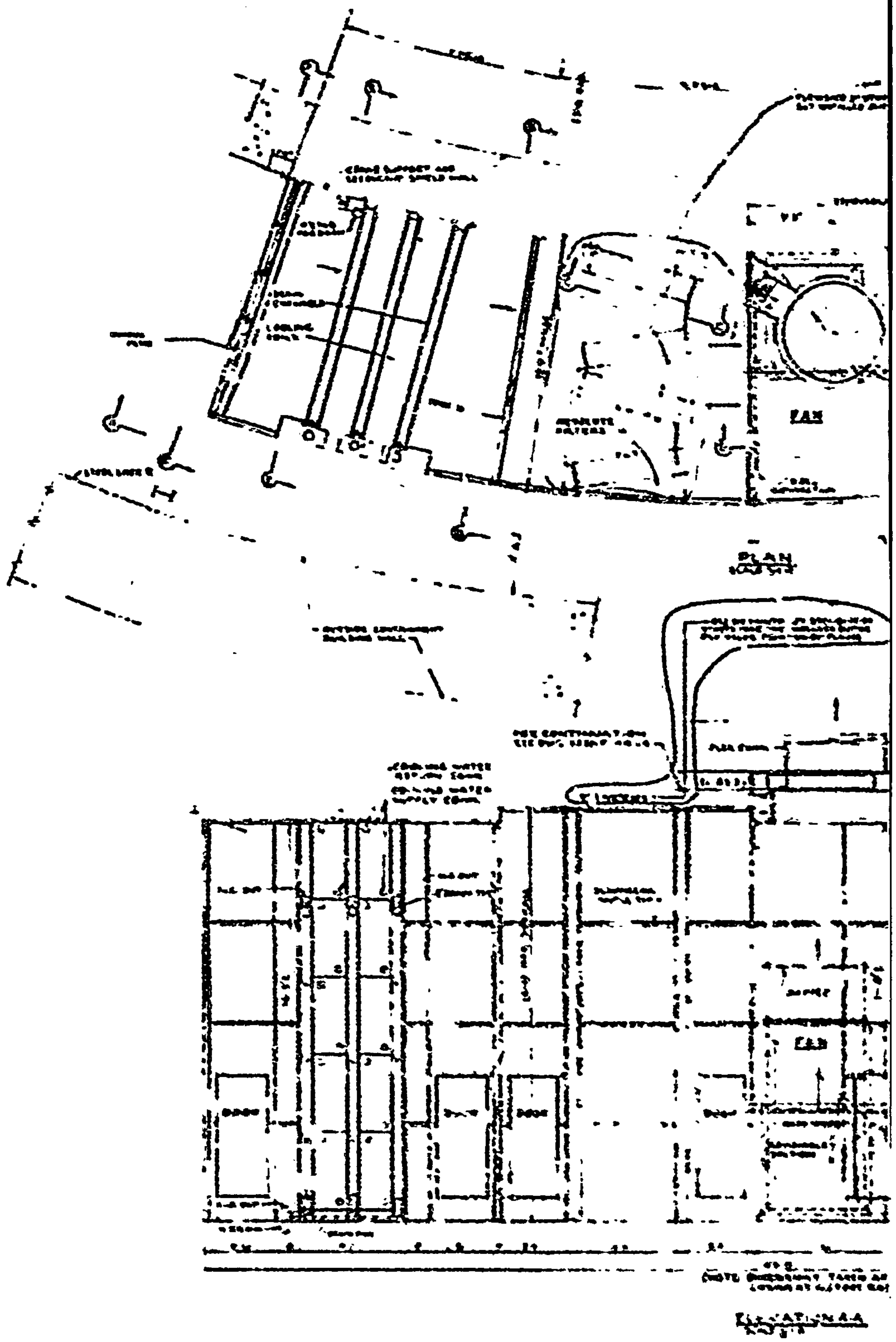


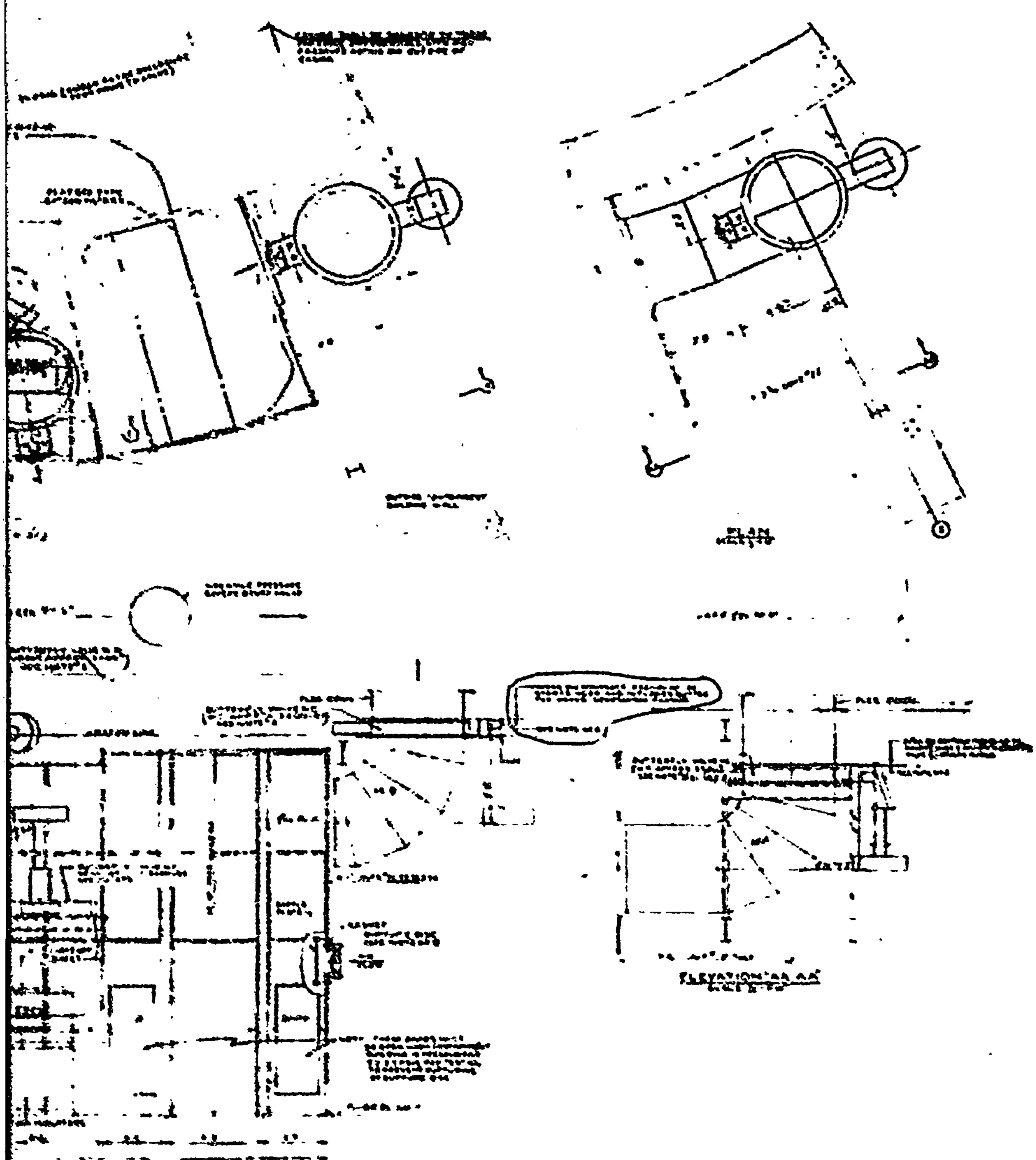
10'-0" Ref
Det B
Det B
A
2'-8"
5'-0"
Det B
Det B
A



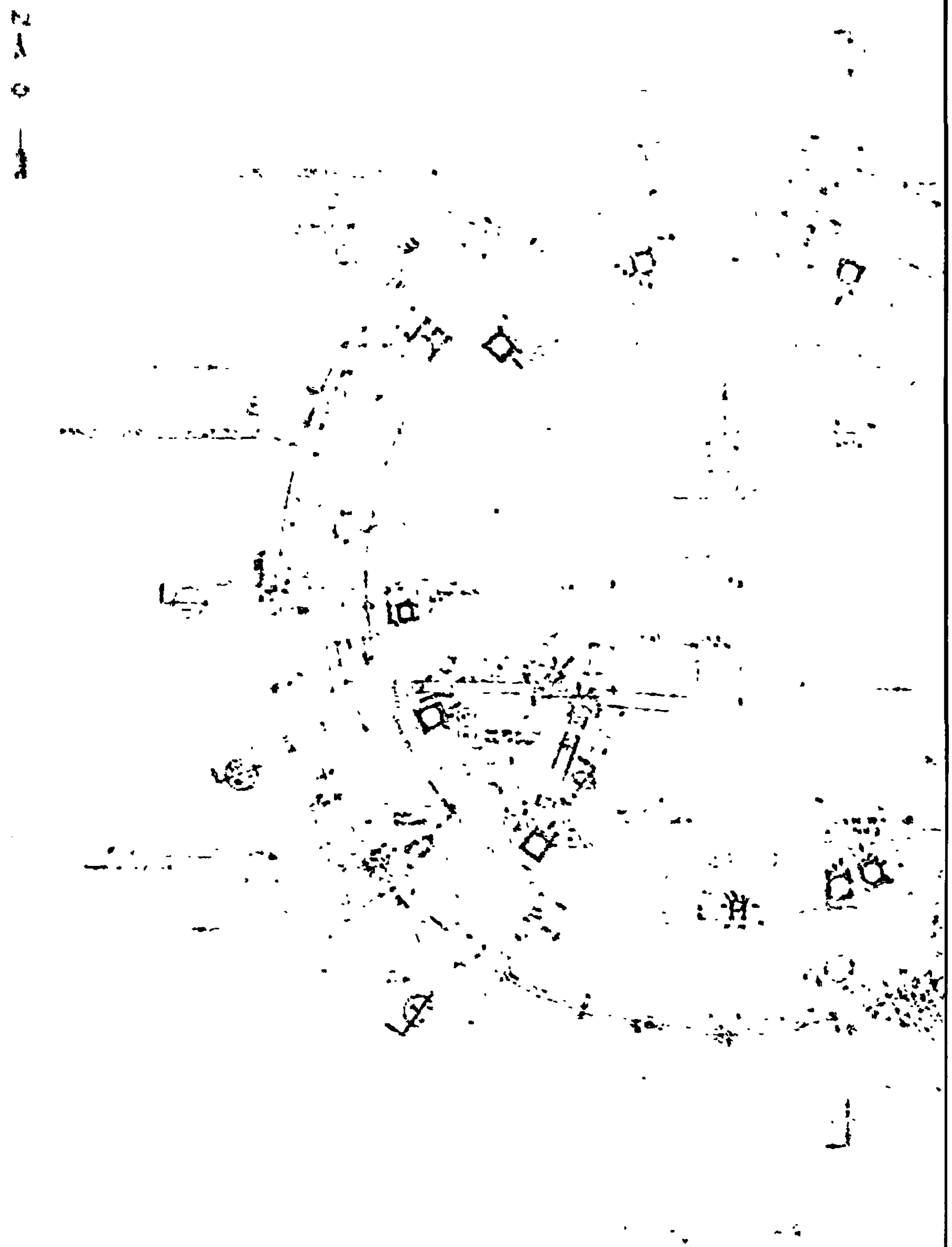
CHARCOAL FILTER BANK ARRANGEMENT

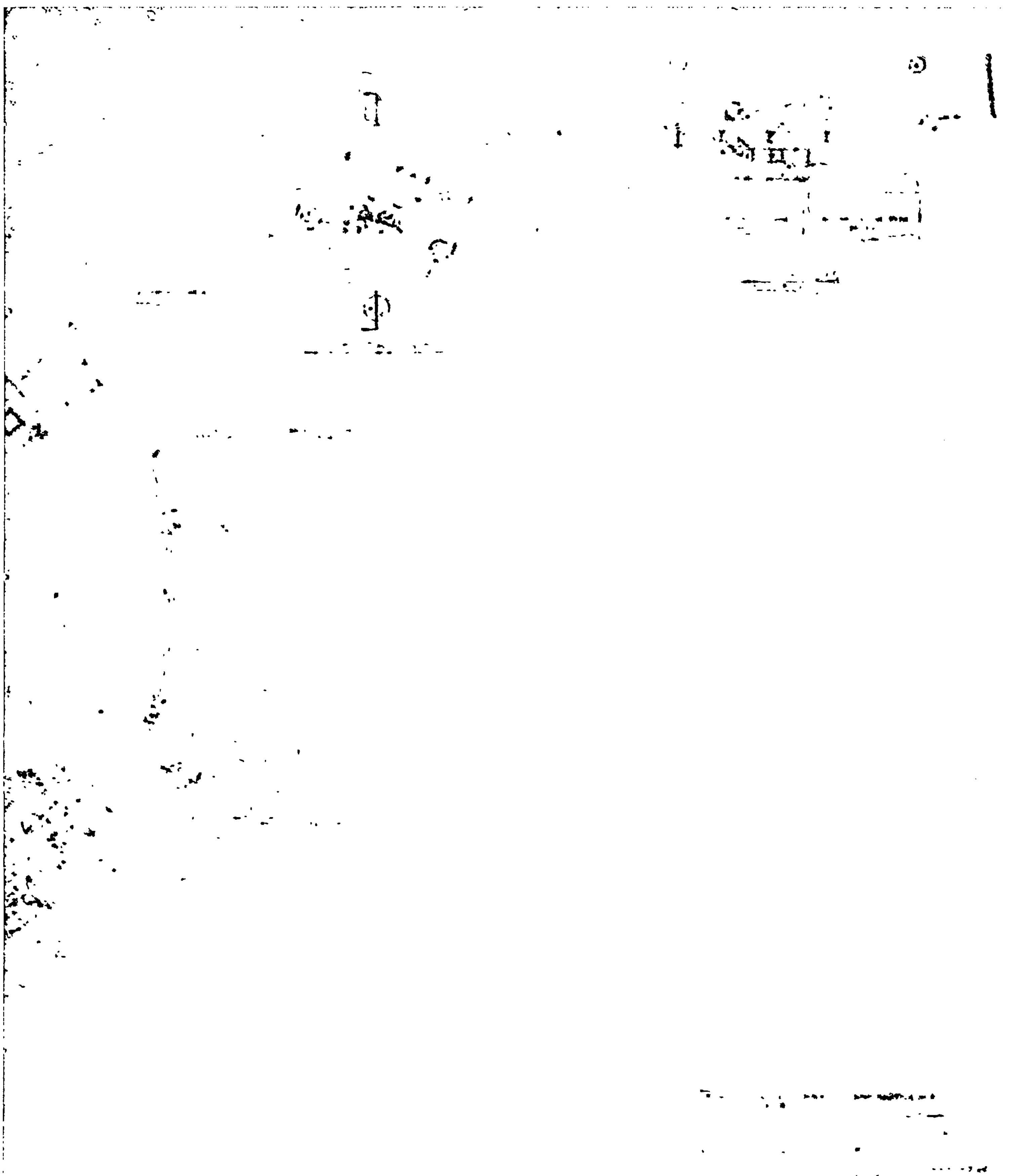
FIGURE 6.4-2





CONTAINMENT BUILDING AIR RECIRCULATION
FAN, COOLER, FILTER UNIT PLAN AND SECTION
FIGURE 6.4-3





CONTAINMENT BUILDING AIR RECIRCULATION SYSTEM PLAN (ABOVE EL. 68'-0")

FIGURE 6.4-4

6.5 ISOLATION VALVE SEAL WATER SYSTEM

6.5.1 DESIGN BASES

The Isolation Valve Seal Water System assures the effectiveness of those containment isolation valves that are located in lines connected to the Reactor Coolant System, or that could be exposed to the containment atmosphere during any condition which requires containment isolation, by providing a water seal (and in a few cases a gas seal) at the valves. The system provides a simple and reliable means for injecting seal water between the seats and stem packing of the globe and double disc types of isolation valves, and into the piping between closed diaphragm type isolation valves. 15
This system operates to limit the fission product release from the containment.

Although no credit is taken for operation of this system in the calculation of off-site accident doses, it does provide assurance that the containment leak rate is lower than that assumed in the accident analysis should an accident occur.

Design provisions for inspection and testing of the Isolation Valve Seal Water System are discussed in Section 6.5.5.

See Section 5.2, Containment Isolation System for containment isolation 15
diagrams (Figures 5.2-1 through 5.2-25), tabulation of isolation valve parameters (Table 5.2-1) and a description of the derivation of "Phase A" and "Phase B" containment isolation signals. Section 5.2.2 discusses the containment isolation valves that are sealed, post-accident, by air from the Penetration and Weld Channel Pressurization System.

6.5.2 SYSTEM DESIGN AND OPERATION

System Description

The Isolation Valve Seal Water System flow diagram is shown in Figure 6.5-1.

System operation is initiated either manually or by any automatic safety injection signal. When actuated, the Isolation Valve Seal Water System interposes water inside the penetrating line between two isolation points located outside the containment. The resulting water seal blocks leakage of the containment through valve seats and stem packing. The water is introduced at a pressure slightly higher (approximately 52 psig) than the containment design pressure of 47 psig. The high pressure nitrogen supply used to maintain pressure in the seal water tank does not require any external power source to maintain the required driving pressure. The possibility of leakage from the containment or Reactor Coolant System past the first isolation point is thus prevented by assuring that if leakage does exist, it will be from the seal water system into the containment.

The following lines would be subjected to pressure in excess of the Isolation Valve Seal Water System design pressure (150 psig) in the event of an accident, due to operation of the recirculation pumps:

1. Residual heat removal loop inlet line
2. Residual heat removal loop outlet line
3. Bypass line from residual heat exchanger outlet to safety injection pumps suction
4. Residual heat removal pumps miniflow line

15

5. Residual heat removal loop sample line
6. Recirculation pump discharge sample line

Lines 1, 2 and 3 are isolated by double disc gate valves, while 5 and 6 are each isolated by two globe valves in series. Line 4 is isolated by a globe and a gate valve in series. These valves can be sealed by nitrogen gas from the high pressure nitrogen supply of the Isolation Valve Seal Water System. A self contained pressure regulator operates to maintain the nitrogen injection pressure slightly higher than the maximum expected line pressure. All of these
15 valves, except those in line 4, are closed during power operation, and the nitrogen gas injection is manually initiated.

The system includes one seal water tank capable of supplying the total requirements of the system. The tank is pressurized from the system's own supply of high pressure nitrogen cylinders through pressure control valves. Design pressure of the tank and injection piping* is 150 psig, and relief valves are provided to prevent overpressurization of the system if a pressure control valve fails, or if a seal water injection line communicates with a high pressure line due to a valve failure in the seal water line.

In lines approximately three inches and larger, double disc gate valves are used for isolation. A drawing of this valve is presented in Figure 6.5-2. Redundant isolation barriers are provided when the valve is closed. The upstream and downstream discs are forced against their respective seats by the closing action of the valve. Seal water is injected through the valve bonnet and pressurizes the space between the two valve discs. The seal water pressure in excess of the potential accident pressure eliminates any outleakage past the first isolation point.

For smaller lines, isolation is provided by two globe valves in series with the seal water injected into the pipe between the valves. The valves are oriented such that the seal water wets the stem packing. When the valves are closed for containment isolation, the first isolation point is the valve plug in the valve closest to containment, and the water seal is applied between the valve plug and stem packing. In a number of the smaller lines, isolation is provided by two diaphragm (Saunders Patent) valves in series, with the seal water injected into the pipe between the valves.

The maximum acceptable leakage across both the seat and stem packing of any gate or globe valve is 10 cc/hr/inch of nominal pipe diameter. Tests on these valves have indicated that much lower leakage rates can be expected. However, design of the Isolation Valve Seal Water System is based on the conservative assumption that all isolation valves are leaking at five times the acceptable value, or 50 cc/hr/inch of nominal pipe diameter. In addition, should one of the isolation valves fail to seat, flow through the failed valve will be limited to approximately 100 times the maximum acceptable leakage value,

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* The injection piping runs and nitrogen supply piping are fabricated using 3/8 inch O.D. tubing, which is capable of 2500 psig service.

or 1000 cc/hr/inch of nominal pipe diameter, by the resistance of the seal water injection path. A water seal at the failed valve is assured by proper slope of the potential line, or a loop seal, or by additional valves on the side of the isolation valves away from the containment.

The seal water tank is sized to provide at least a 24 hour supply of seal water under the most adverse circumstances, i.e. isolation valves leaking at the design rate of 50 cc/hr/inch, plus the failure of the largest contain-
15 ment isolation valve to seat and leaking at the maximum rate of 1000 cc/hr/inch. The seal water volume required to satisfy these conditions is approximately
15 144 gallons. A 176 gallon seal water tank is provided. If all of the isolation valves seat properly, as expected, the tank volume is sufficient for approximately 2 1/2 days of operation at design seal water flow rates before makeup is required. Two separate sources of makeup water are provided to ensure that an adequate supply of seal water is available for long term operation.

Seal Water Actuation Criteria

Containment Isolation (Section 5.2 and seal water injection are accomplished automatically for certain penetrating lines requiring early isolation, and manually for others, depending on the status of the system being isolated and the potential for leakage in each case. Generally, the following criteria determine whether the isolation and seal water injection are automatic or manual.

Automatic containment isolation and automatic seal water injection are required for lines that could communicate with the containment atmosphere and be void of water following a loss of coolant accident. These lines include:

15 Reactor coolant pump cooling water supply and return lines (phase B isolation*)
Reactor coolant pump seal water return line (phase B isolation)
Excess letdown heat exchanger cooling water supply and return lines
Letdown line
Reactor Coolant System sample lines
Containment vent header
Containment air sample inlet and outlet lines (air pressurization)
Reactor coolant drain tank gas analyzer line

Auxiliary steam supply and condensate return lines
Service air and service water lines
Steam jet ejector return line to containment (air pressurization)

Automatic containment isolation and automatic seal water injection are also provided for the following lines, which are not connected directly to the Reactor Coolant System, but terminate inside the containment at certain components. These components can be exposed to the reactor coolant or containment atmosphere as the result of leakage or failure of a related line or component. The isolated lines are not required for post-accident service. These lines include:

Pressurizer relief tank gas analyzer line
Pressurizer relief tank makeup line
Safety Injection System test line
Reactor coolant drain tank pump discharge line
Steam generator blowdown lines
Steam generator blowdown sample lines
Accumulator sample line
Containment sump pump discharge

15

Manual containment isolation and manual seal water injection are provided for lines that are normally filled with water and will remain filled following the loss-of-coolant accident, and for lines that must remain in service for a time following the accident. The manual seal water injection assures a long term seal. These lines include:

Reactor coolant pump seal water supply lines
Charging Line
Safety injection headers
Containment spray headers

Manual containment isolation and manual seal gas injection are provided for lines that are filled with water during the accident but which are at a pressure higher than that provided by the Isolation Valve Seal Water System. These lines must remain in service for a period of time following the accident, or may be planced in service on an intermittent basis following the accident. These lines are as follows:

Residual heat removal loop inlet line
15 Bypass line from residual heat exchanger outlet to safety injection pumps suction
Residual heat removal loop sample line
Recirculation pump discharge sample line

Seal water injection is not necessary to insure the integrity of isolated lines in the following categories:

Lines that are connected to non-radioactive systems outside the containment and in which a pressure gradient exists which opposes leakage from the containment. These include nitrogen supply lines to the pressurizer relief tank, accumulators, and reactor coolant drain tank, the instrument air header, the weld channel pressurization air lines, and the pressurizer deadweight tester line.

Lines that do not communicate with the containment or Reactor Coolant System and are missile protected throughout their length inside containment. These lines are not postulated to be severed or otherwise opened to the containment atmosphere as a result of a loss of coolant accident. These include the steam and feedwater headers and the containment ventilation system cooling water supply and return lines.

Lines that are designed for post-accident service as part of the engineered safety features. The only line in this category is the containment sump recirculation line. This line is connected to a closed system outside containment.

Special lines such as the fuel transfer tube, containment purge ducts, and the containment pressure relief line. The zone between the two gaskets sealing the blind flange to the inner end of the fuel transfer tube is pressurized to prevent leakage from the containment in the event of an accident. The zone between the two butterfly valves in each containment purge duct is pressurized above incident pressure while the valves are closed during power operation, as are the two spaces between the three butterfly valves in the containment pressure relief line.

Components

All associated components, piping, and structures of the Isolation Valve Seal Water System are designed to Class I seismic criterion.

There are no components of this system located inside containment.

The piping and valves for the system including the air-operated valves, are designed in accordance with the USAS Code for Pressure Piping (Power Piping Systems), B31.1.

6.5.3 DESIGN EVALUATION

The Isolation Valve Seal Water System provides an extremely prompt and reliable method of limiting the fission product release from the containment isolation valves in the event of a loss-of-coolant accident.

The employment of the system during a loss-of-coolant accident, while not considered for analysis of the consequences of the accident, provides an additional means of conservatism in ensuring that leakage is minimized. No detrimental effect on any other safeguards system will occur should the seal water system fail to operate.

System Response

Automatic containment isolation will be completed within approximately two seconds following generation of the phase A containment isolation signal.

This is the estimated closing time of the air operated containment isolation valves (Section 5.2). Since the Isolation Valve Seal Water System is actuated by this signal, automatic seal water injection will be in effect within this time period.

Subsequent generation of the phase B isolation signal on containment high pressure (spray actuation signal) will close a number of motor operated isolation valves with an estimated closing time of 10 seconds. Automatic seal water injection flow will have been initiated in advance of this signal by the phase A signal.

Single Failure Analysis

A single failure analysis is presented in Table 6.5-2. The analysis shows that the failure of any single active component will not prevent fulfilling the design function of the system.

Reliance on Interconnnected Systems

The Isolation Valve Seal Water System can operate and meet its design function without reliance on any other system. Electric power is not required for system operation, although instrument power is required to provide indication in the control room of seal water tank pressure and level.

Shared Function Evaluation

Table 6.5-3 is an evaluation of the main components discussed previously and a brief description of how each component functions during normal operation and during an accident.

6.5.4 MINIMUM OPERATING CONDITIONS

The Technical Specifications, Section 15, establish limiting conditions regarding the operability of the system when the reactor is critical.

6.5.5 INSPECTIONS AND TESTS

Inspections

The system components are all located outside the containment and can be visually inspected at any time.

Component Testing

Each automatic isolation valve can be tested for operability at times when the penetrating line is not required for normal service. Lines supplying automatic seal water injection can be similarly tested.

System Testing

Containment isolation valves and the Isolation Valve Seal Water System can be tested periodically to verify capability for reliable operation. The seal water tank pressure and water level can be observed locally, and these parameters are also displayed continuously in the control room.

The system will not be in service during the containment leak rate test.

Operational Sequence Testing

The capacity of the system to deliver water at the required rate will be verified initially during the pre-operational test period of plant construction and startup. Prior to plant operation a containment isolation test signal will be used to ensure proper sequence of isolation valve closure and seal water addition. 15

TABLE 6.5-1

ISOLATION VALVE SEAL WATER TANK

Number	1	
Total Volume, ft^3	23.6	
Minimum Volume, gal.	120	15
Material	ASTM A-240	
Design Pressure, psig	150	
Design temperature, °F	200	
Operating pressure, psig	50-100	
Operating temperature, °F	Ambient	
Code	ASME UPV (Sect. VIII)	

TABLE 6.5-2

SINGLE FAILURE ANALYSIS - ISOLATION

VALVE SEAL WATER SYSTEM

Component	Malfunction	Comments
A. Automatically Operated Valves (Open on Phase A Containment Isolation Signal)		
1) Isolation valve for automatic injection headers	Fails to open	Two provided. Operation of one required.
B. Instrumentation		
1) Level transmitter	Fails	Local level indicator at tank also provided
2) Pressure transmitter	Fails	Local pressure indicator at tank also provided

TABLE 6.5-3

SHARED FUNCTIONS EVALUATION

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Isolation Valve Seal Water Storage Tank (1)	NONE	Lined up to seal water injection piping	Source of water for sealing isolation valves	Lined up to seal water injection piping
N_2 Supply Bottles (3)	NONE	Lined up to seal water tank	Source of N_2 to maintain seal water	Lined up to seal water tank

REFERENCES-

1 PROCESS FLOW DIAGRAM ½ DWG. (LATER)
2 DEFINITION OF SYMBOLS
 ⅓ E.SPEC. G675176 REV. 2
3. INSTRUMENTATION AND CONTROL STANDARDS
 ⅓ SYMBOLS AND APPLICATIONS FOR INSTRUMENT DIAGRAMS. SECTION 1.1 ISSUED AUGUST 12, 1966
 ⅓ INSTRUMENT INSTALLATION. SECTION 3.0 ISSUED (LATER)
4. MATERIAL SPEC. AND FITTINGS
 ⅓ E.SPEC. G569866 REV. 2 AND
 ⅓ E.SPEC. G676398 REV. 0

REFERENCE DRAWINGS-

RCS - REACTOR COOLANT SYSTEM
 ⅓ DWG. 540 F 898
ACS - AUXILIARY COOLANT SYSTEM
 ⅓ DWG. 684 J 793
CVCS - CHEMICAL & VOLUME CONTROL SYSTEM
 SHEET #1 ⅓ DWG. 684 J 627
 SHEET #2 ⅓ DWG. 684 J 871
 SHEET #3 ⅓ DWG. 541 F 236
WDS - WASTE DISPOSAL SYSTEM
 SHEET #1 ⅓ DWG. 684 J 729
 SHEET #2 ⅓ DWG. 684 J 918
SIS - SAFETY INJECTION SYSTEM
 ⅓ DWG. 684 J 730
SS - SAMPLING SYSTEM
 ⅓ DWG. 540 F 902
BD - STEAM GENERATOR BLOWDOWN SYSTEM
 UE&C DWG. 9321-F-2559
COMP - COMPOSITE PIPING IN TRENCHES
 UE&C DWGS. 9321-F-2677, 2678, 2679
 - ISOLATION VALVE SEAL WATER PIPING
 UE & C DWGS. 9321-F-2768 AND 2769

LEGEND-

PW - PRIMARY WATER SYSTEM
CW - CITY WATER SYSTEM
RFD - RADIOACTIVE FLOOR DRAINS
VH - VENT HEADER (WDS)
V - ATMOSPHERIC VENT
F.C. - FAIL CLOSED
L.O. - LOCKED OPEN
D - LOCAL DRAIN

NOTES:

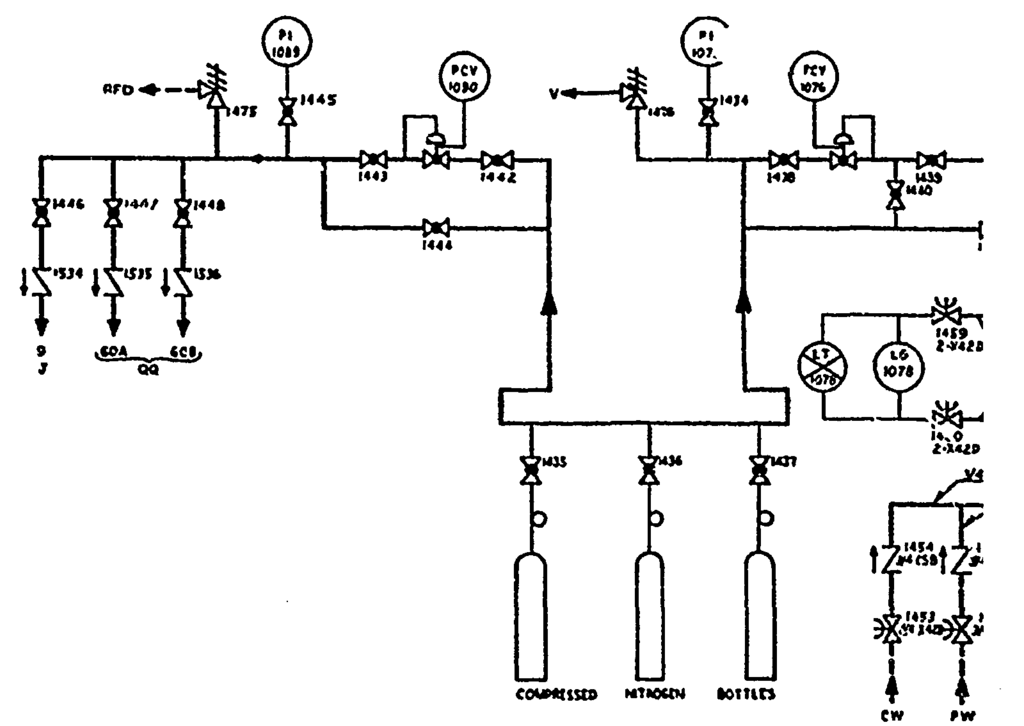
1. ALL PIPING IS 3/8" O.D. STAINLESS STEEL TUBING (LINE DESIGNATION 3/8 - IV - 2535) EXCEPT AS DETAILED.
2. ALL GLOBE VALVES INSTALLED WITH FLOW UNDER SEAT. EXCEPT NOS. 1402-1405, 1415-1418, 1421, AND 1463-1469.
3. ALL GLOBE VALVES ARE TYPE 3/8 - T58 EXCEPT NOS. 1434 - 1448 (NITROGEN SUPPLY), WHICH ARE TYPE 3/8 - 158S (SOFT SEATS AND LARGE ORIFICES).
4. ALL CHECK VALVES ARE TYPE 3/8 - C58, EXCEPT NUMBERS 1454, 1456 AND 1533 TO 1536. VALVES 1533 TO 1536 ARE TYPE 3/8 - C58S (SOFT SEATS).
5. ALL GLOBE AND CHECK VALVES ARE EQUIPPED WITH SWAGELOCK FITTINGS. EXCEPT CHECK VALVES 1454 AND 1456.
6. ADDITIONAL VENTS AND DRAINS MAY BE REQUIRED BY THE PIPING LAYOUT.
7. THE INDIVIDUAL SEAL WATER LINES ARE ROUTED TO THE CONTAINMENT ISOLATION VALVES AT THE NUMBERED LINES INDICATED.
8. THE LETTERS BELOW THE NUMBERS INDICATING THE PENETRATING LINES IDENTIFY THE PENETRATION SLEEVES.

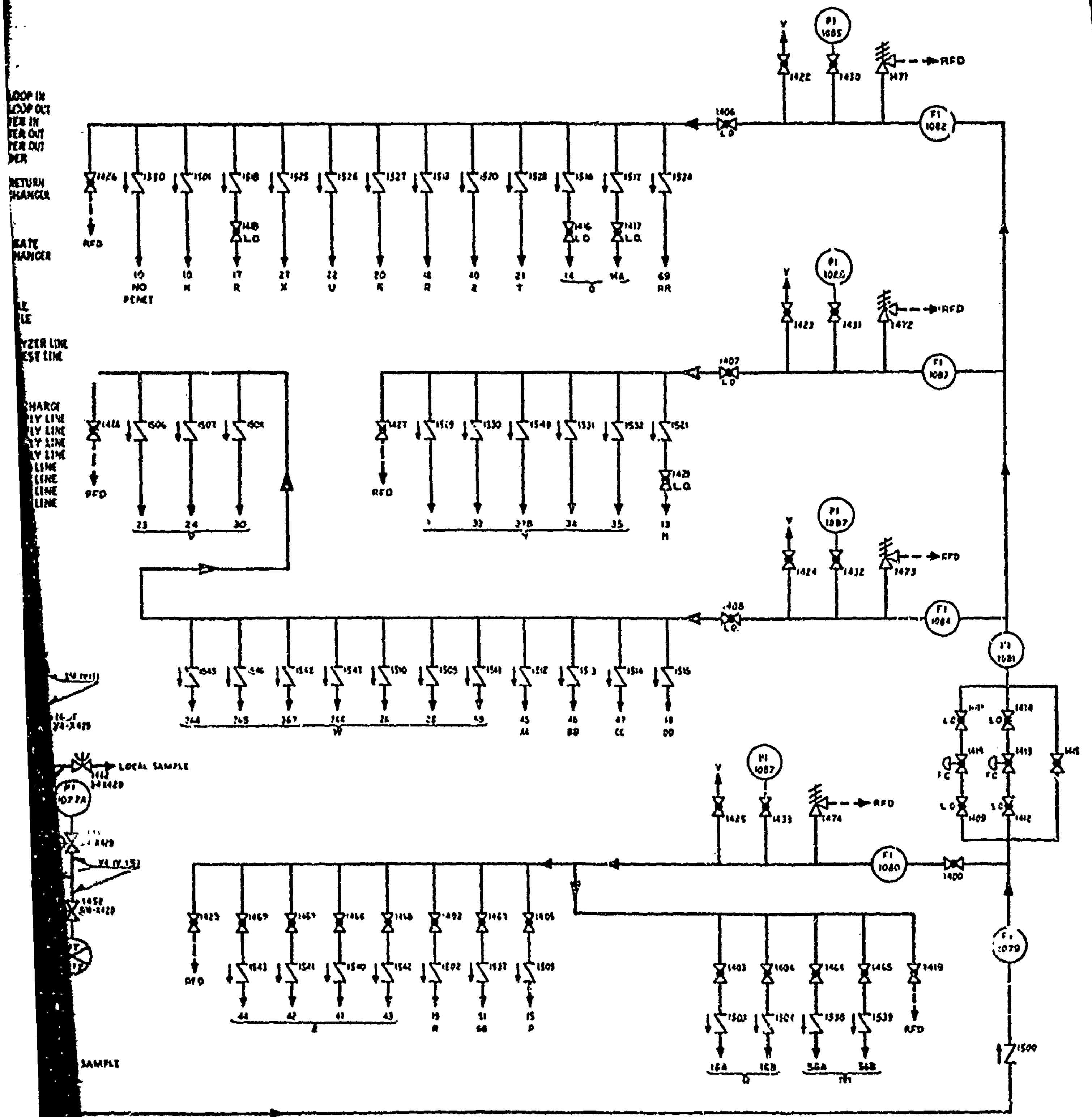
PENETRA[TION LIST]

9	RES
10	RES
13	R.C
14	R.C
14A	R.C
15	CON
16	SAF
17	R C
18	EXC COO
19	CHA
20	AUX
21	AUX
22	EXC COO
23	CON
24	P.R
25	PRE
26	PRE
27	LET
30	R.C
31	SAF
33	P.R
34	SER
35	CIT
40	R C
41	R.C
42	R C
43	R C
44	R C
45	STE
46	STE
47	STE
48	STE
51	CON
55	RES
56	SAF
59	REA
60	RES TO S
~~65~~	~~CO~~
~~66~~	~~CO~~
69	ACC

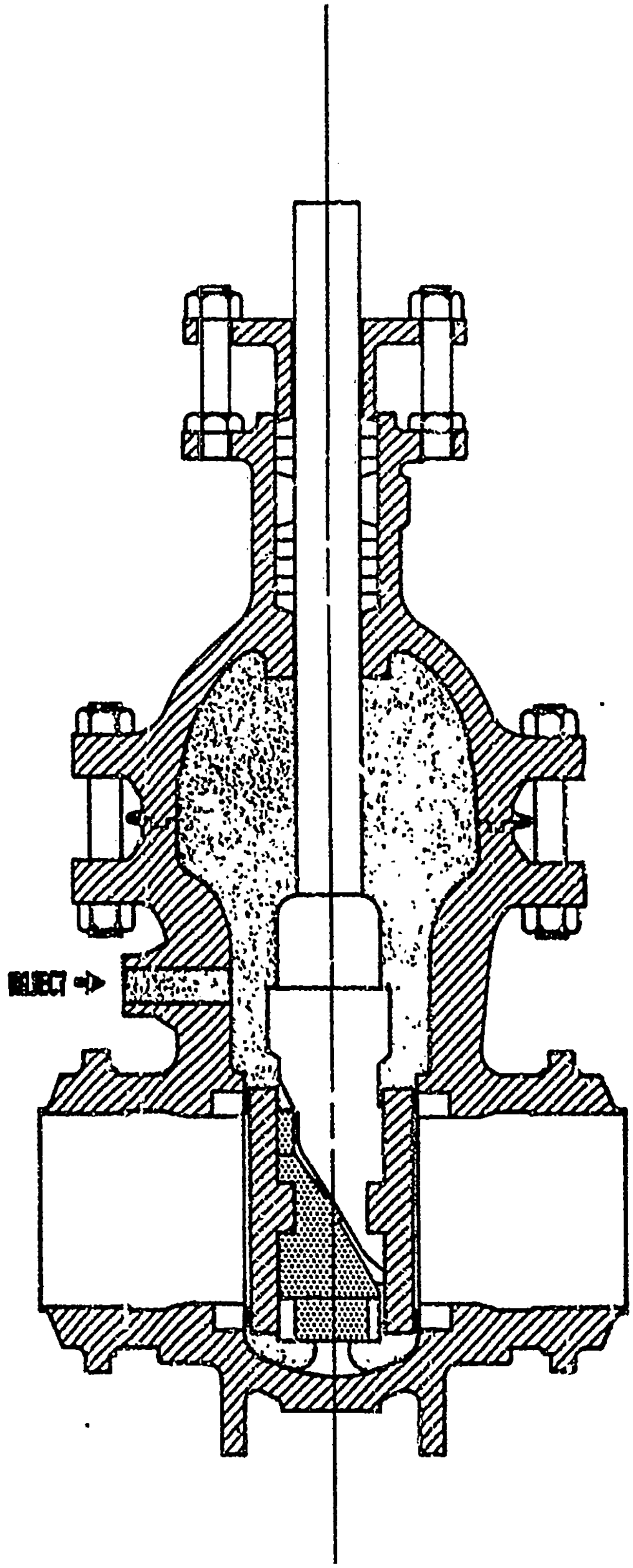
PENETRATION LIST (CONTINUED)

~~28~~	~~AIR EJECTOR DISCHARGE TO CONTAINMENT~~
338	SUMP PUMP DISCHARGE
364	STEAM GENERATOR BLOWDOWN SAMPLE
365	STEAM GENERATOR BLOWDOWN SAMPLE
366	STEAM GENERATOR BLOWDOWN SAMPLE
367	STEAM GENERATOR BLOWDOWN SAMPLE





Isolation Valve Seal Water System Flow Diagram



DOUBLE DISK ISOLATION VALVE WITH SEAL WATER INJECTION

FIGURE 6.5-2

6.6 CONTAINMENT PENETRATION AND WELD CHANNEL PRESSURIZATION SYSTEM

6.6.1 DESIGN BASES

The Contanment Penetration and Weld Channel Pressurization System provides means for continuously pressurizing the positive pressure zones incorporated into the containment penetrations and the channels over the welds in the steel inner liner in the event of a loss-of-coolant accident. Although no credit is taken for system operation in calculation of off-site accident doses, it is designed as an engineered safety feature and does provide assurance that the containment leak rate in the event of an accident is lower than that assumed in the accident analysis.

The system is designed to provide a means for determining the leak-tightness of the containment during power operation, thereby reducing the frequency for performing postoperational integrated leakage rate tests.

6.6.2 SYSTEM DESIGN AND OPERATION

System Description

The Containment Penetration and Weld Channel Pressurization System is shown in Figure 6.6-1. A regulated supply of clean and dry compressed air from either of the plant's 100 psig compressed air systems located outside the containment is supplied to all containment penetrations and inner liner weld channels. The system maintains a pressure in excess of containment design pressure continuously during all reactor operations thereby ensuring that there will be no out-leakage of the containment atmosphere through the penetrations and liner welds during an accident. Typical piping and electrical penetrations are described in Section 5.

The primary source of air for this system is the instrument air system (Section 9). Two instrument and control air compressors are used, although only one is required to maintain pressurization at the maximum allowable leakage rate of the pressurization system.

The plant air compressors act as a backup to the instrument and control air compressors (Section 9) for added reliability. One plant air compressor is available.

A standby source of gas pressure for the system is provided by a bank of nitrogen cylinders. The associated nitrogen system will automatically deliver nitrogen at a slightly lower pressure (approximately 50 psi) than the normal regulated air supply pressure of approximately [illegible] psig. Thus, in the event of failure of the normal and backup air supply systems during periods when the system is in operation, the penetration and weld channel pressure requirements will be automatically maintained by the nitrogen supply. This assures reliable pressurization under both normal and accident conditions.

Containment penetrations and liner weld channels are grouped into four independent zones to simplify the process of locating leaks during operation. Each such zone is served by its own air receiver. In the event that all normal and backup air supplies are lost, each of the four pressurization system zones continues to be supplied with air from its respective air receiver. Each of the air receivers, Table 6.6-1, is sized to supply air to its pressurized zone for a period of at least four hours, based on a leakage rate of 0.2% of the containment free volume per day (0.1% leakage into the containment and 0.1% leakage to the environment).

If the receivers become exhausted before normal and backup air supplies can be restored, nitrogen from the bank of pressurized cylinders will be supplied to the affected zones. The nitrogen bank is sized to provide a 24 hour supply of gas to the system, again based on a total leakage rate from the pressurization system of 0.2% of the containment free volume in 24 hours. There are three nitrogen cylinders in the bank, each 24" OD by 24'-0" long. The nitrogen supply will also automatically assume the pressurization gas load in the event an air receiver fails.

A pressure relief valve set at 150 psig (sized for 167 scfm at 10% accumulation) protects the system f failure of the pressure reducing valve in the line to each zone from the bank of nitrogen cylinders. Each zone of piping is also protected by a rupture disc, designed to open at 175 psig. Pressure control valves, isolation valves and check valves are located outside of the containment for ease of inspection and maintenance. Failure of any of these components does not lead to loss of pressure in the system since backup systems automatically augment the normal air supply.

The line to each of the four pressurized zones is equipped with a critical pressure drop orifice (installed in the pressure control valve body) to assure that air consumption will be within the capacity of the system. High air consumption in one zone cannot affect the operation of the other zones under any circumstances.

Means for assuring that all the weld channels and penetrations are pressurized is provided by flow-through test lines, connected to the pressurized weld channel zones and penetrations at points as far away from the supply points as possible. Pressurization of the zone is verified by closing off the air supply line and opening the flow-through test line valve to observe the escape of the pressurizing medium.

Pressure Indication

In order to ensure that the station operators are aware at all times that all penetrations and liner weld seam channels are pressurized, the following instrumentation is provided as described below.

The following pressurized zones are equipped with local pressure gages, mounted outside the containment for ready accessibility and available for regular reading. The accuracy of these gages is within 0.5% of the full scale reading.

a) Each piping penetration

b) Each electrical penetration

c) The spaces between the two isolation (butterfly) valves in the purge supply and exhaust ducts

d) The two spaces between the three isolation (butterfly) valves in the containment pressure relief line

e) The double-gasketed space on the outside hatch of each of the two personnel air locks

The pressurized zones located entirely inside the containment, and those zones located in inaccessible areas, are equipped to actuate pressure switches to provide remote low pressure alarms in the central control room. Examples of the zones so equipped are:

a) Each liner seam weld channel
b) The double-gasketed space on each inside hatch of the personnel air locks
c) The double-gasketed space on the equipment door flange
d) The pressurized zones in the spent fuel transfer tube
e) Shroud rings over penetration-to-containment liner weld-piping and electrical penetrations

The actuating pressure for each pressure switch is set just above incident pressure and just below the nitrogen supply regulator setting. Should pressure in any of these zones fall below the pressure switch set point, a light and an alarm in the control room will be activated. Each penetration and each section of liner weld joint channel so alarmed will be represented by a separate light and identified.

Personnel Air Lock Interlock

Continuous pressurization of air lock door double-gasketed barriers and the protection of the pressurization header against air loss are assured by a set of interlocks. One interlock on each air-lock door prevents opening of the door until the pressurization line is isolated and pressure in the double-gasketed closure is relieved to atmosphere. This prevents excessive leakage from the pressurization system. The pressurization line to this zone is also equipped with a restricting orifice to assure that air consumption, even upon failure of the interlock, will be within the capacity of the pressurization system, and will not result in loss of pressure in other zones connected to the same pressurization header. Another set of interlocks prevents opening of one air lock door until the double-gasketed zone on the other door is re-pressurized.

Containment Purge Line Interlock

The containment ventilation purge penetration butterfly valves are also interlocked to prevent the opening of either valve until the pressurization line to the space between the valves has been isolated. Isolation of the pressurization line to each purge duct pressurized zone can be accomplished remotely from the central control room. Alarm lights, prominently displayed on a panel indicating the isolation status of the containment, remain lit identifying an open purge duct isolation valve or a low pressurization zone pressure. Restricting orifices are installed in each pressurization line to the ventilation purge ducts to assure that air consumption, even on failure of an interlock, will not result in loss of pressure to the other zones connected to the same pressurization header.

The containment pressure relief line isolation valves (three butterfly valves in series), and the two pressurized spaces formed between these valves, are provided with similar interlocking to prevent the opening of any of the butterfly valves until the adjacent inter-valve space has been depressurized. The pressurization lines to these spaces are also equipped with flow restricting orifices, and alarm lights in the containment identify open valves or low inter-valve space pressure.

Containment Inleakage

With a continuous inleakage to the containment from the penetration and liner weld joint channel pressurization system of 0.1% of the containment volume per day, the calculated time for the containment pressure to rise by 1 psi is approximately 14 days and therefore is not considered to be an operating or safety problem. From the standpoint of allowable pressure, a much greater inleakage would be permitted. With the ability to limit the activity of the air in the containment during normal operation through the use of the two containment auxiliary charcoal filter units, each complete with roughing filters, HEPA filters, and charcoal filters (Section 5), containment overpressure can be relieved as required through the pressure relief duct and exhaust fan, passing up the discharge duct, along with the exhaust air from the Auxiliary Building.

Components

All associated components, pipings, and structures, of the Containment Penetration and Weld Channel Pressurization System are designed to Class I Seismic criteria.

The piping and valves for the system are designed in accordance with the USAS Code for Pressure Piping (Power Piping Systems), B31.1.

For a description of the instrument and control air compressors and the plant air compressors, refer to Service Air System, Section 9.

The three nitrogen cylinders used are designed in accordance with Section VIII (Unfired Pressure Vessels) of the ASME Boiler and Pressure Vessel Code, for 2200 psig maximum pressure, and contain a total of 22,000 scf of nitrogen.

6.6.3 DESIGN EVALUATION

The employment of this system following a loss-of-coolant accident, while not considered in the analysis of the consequences of the accident, provides an additional means for ensuring that leakage is minimized if not altogether eliminated. No detrimental effect on any other safety features system will be felt should the pressurization system fail to operate.

<u>System Response</u>

Since the Containment Penetration and Weld Channel Pressurization System is continuously pressurized above the containment design pressure during all reactor operations, there is no response time required for the system to operate.

<u>Single Failure Analysis</u>

A single failure analysis is presented in Table 6.6-2. The analysis shows that the failure of any single active component will not prevent fulfilling the design function of the system.

<u>Reliance on Interconnected Systems</u>

The Containment Penetration and Weld Channel Pressurization System can operate and meet its design function without reliance on any other system, except as limited by air compressor availability following depletion of all reserves in the systems' air receivers and backup nitrogen cylinders. Electric power is not necessary for operation of the system, although instrument power is required in order to provide indications in the control room of system operation.

Shared Functions Evaluation

Table 6.6-3 is an evaluation of the main components discussed previously and a brief description of how each component functions during normal operation and during an accident.

6.6.4 MINIMUM OPERATING CONDITIONS

The Technical Specifications, Section 15. establish limiting conditions regarding the operability of the system when the reactor is critical.

6.6.5 INSPECTIONS AND TESTS

Inspections

The system components located outside the containment can be visually inspected at any time. Components inside the containment can be inspected during shutdown. All pressurized zones have provisions for either local pressure indication outside the containment or remote low pressure alarms in the control room.

Testing

Since the system is in operation continuously during all reactor operations to maintain the penetrations and liner weld channels pressurized above containment design pressure, no special testing of system operation or components is necessary.

Should one zone indicate a leak during operation, the specific penetration or weld channel containing the leak can be identified by isolating the individual air supply line to each component in the zone and injecting leak test gas through a capped tube connection installed in each line.

Total leakage from penetrations and weld channels is measured by summing the recorded flows in each of the four pressurization zones. A leak would be expected to build up slowly and would therefore be noted before design leakage limits are exceeded. Therefore, remedial action can be taken before the limit is reached.

In order to provide facility for testing the larger penetrations, branch pressurizing lines are provided from one of the zones to:

a) The double-gasketed space on each hatch of the personnel air lock.
b) The double-gasketed space at the equipment hatch flange.
c) The pressurized zones in the spent fuel transfer tube.
d) The spaces between the two butterfly valves in the purge supply and exhaust ducts.
e) The two spaces between the three butterfly valves in the containment pressure relief line.
f) The spaces between double containment isolation valves in the steam jet air ejector return line to containment and in the containment radiation monitor inlet and outlet lines.

The make up air flow to the penetrations and liner weld joint channels during normal operation is recognized to be only an indication of the potential leakage from the containment. However, it does indicate the leakage from the pressurization system, and the degree of accuracy will be increased when correlated with the results of the full scale containment leak rate tests. The criteria for selection of operating limits for air consumption of the pressurization system are based upon the integrated containment leak rate test acceptance criterion and upon the maintenance of suitable reserve air supplies in the static reserves consisting of the air receivers and nitrogen cylinders. A summary of these operating limits is as follows:

1. A base-line air consumption rate shall be established for each of the four pressurization headers at the time of successful completion
14 of the preoperational integrated containment leakage rate tests. Unexplained increases from this consumption rate shall be considered as reason for concern and normal practice will require routine investigation and location of the point of leakage.

2. The upper limit for long-term uncorrected air consumption for the pressurization system shall be 0.2% of the containment volume per
14 day (sum of four headers) at the system operating pressure, contingent on the following:

   a) Pressure in all pressurization zones is maintained above incident pressure.

   b) Air supply is maintained from the compressed air systems with compressors running.

   c) The full complement of standby nitrogen cylinders (3) is charged.
14 This is consistent with maintenance of a 24 hours' supply.

A variable area flow sensing device is located in each of the headers supplying make up air to the four pressurization zones. Signal output from each of the four flow sensors is applied to an integrating recorder located in the control room. Output from all sensors is also applied to a summing amplifier which drives a total flow recorder. High flow alarms 14 are also derived in the recording channel, to alert the operator in the control room. With this instrumentation, the flow measurement accuracy is within $\pm$ 1% and the reproducibility of 0.3%. Since a flow of 0.2% of the containment volume per day at 47 psig is approximately 3.6 ft^3/minute, the sensitivity of the flowmeters is well within the maximum leakage of the pressurization system.

TABLE 6.6-1

CONTAINMENT PENETRATION AND WELD CHANNEL PRESSURIZATION AIR RECEIVERS

Number	4
Volume, (each) ft^3	360
Material	ASTM A-285-C
Design pressure, psig	140
Design temperature, ^oF	200
Operating pressure, psig	100
Operating temperature, ^oF	100
Code	ASME UPV (Sect. VIII)

TABLE 6.6-2

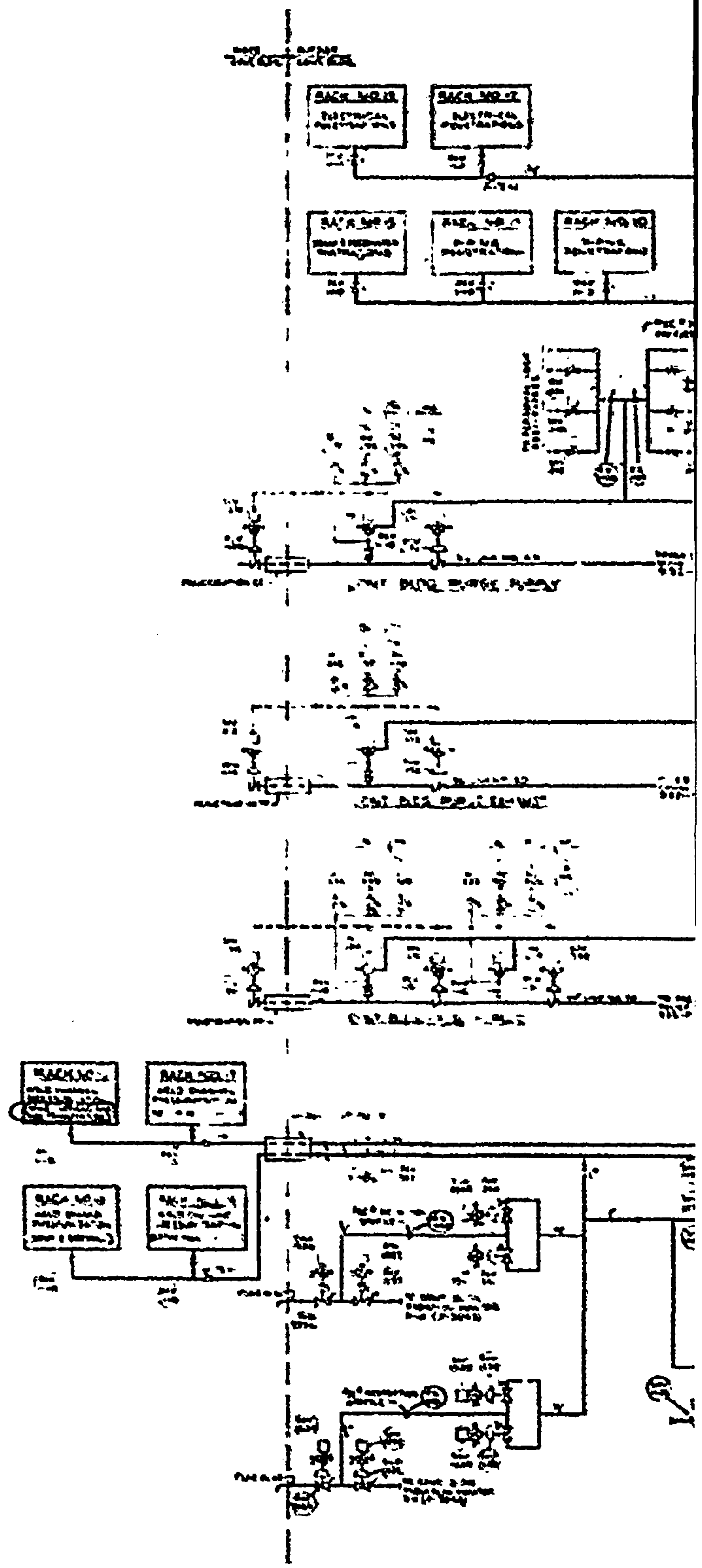
SINGLE FAILURE ANALYSIS - CONTAINMENT
PENETRATION AND WELD CHANNEL PRESSURIZATION SYSTEM

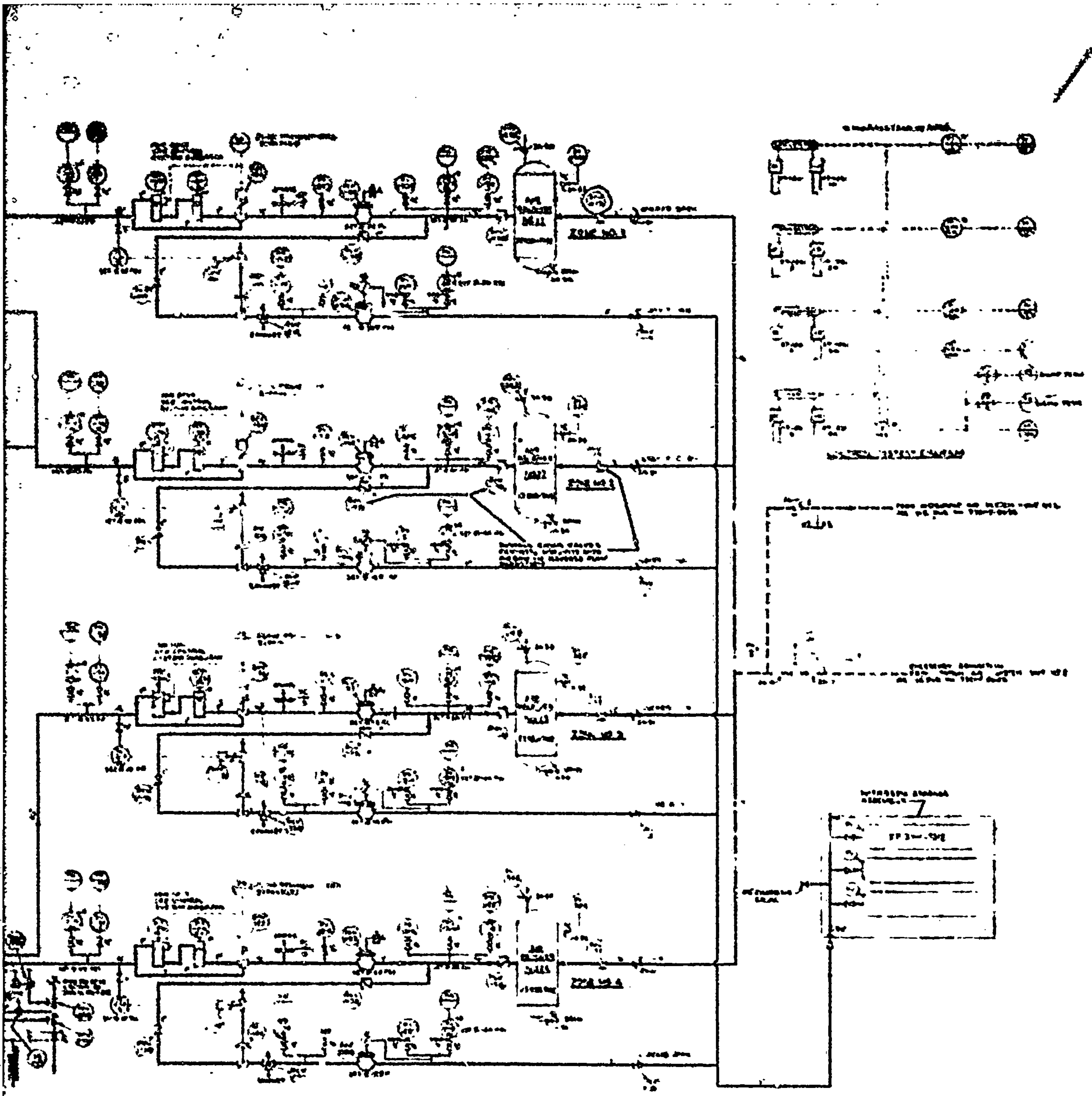
Component	Malfunction	Comments
Instrument and Control Air Compressor	Fails to maintain pressure	One of two instrument and control air compressors required to operate.
Pressure Reducing Valve for each zone	Fails to maintain pressure	On valve failure, flow is limited to acceptable value (75 scfm) by the critical pressure drop orifice. Under low flow conditions, over-pressurization of system downstream of valve is prevented by a rupture disc.

TABLE 6.6-3

SHARED FUNCTIONS EVALUATION

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Instrument and Control Air Compressors (2)	Supply air to plants' instruments and controls and to penetrations and weld channels	2 air compressors in operation	Supply air to penetrations and weld channels	1 air compressor in operation
Plant Air Compressors (1)	Supply air to station air headers	1 air compressor in operation	"	1 air compressor in operation
N_2 Cylinders (3)	Backup source of N_2 to maintain penetration and weld channel pressure	Lined up to Penetration and Weld Channel Pressurization System	Backup source of N_2 to maintain penetration and weld channel pressure	Lined up to Penetration and Weld Channel Pressurization System
Air Receivers (1) and Dryers (3)	Primary source of air for penetrations and weld channels	Lined up to Penetrations and Weld Channel Pressurization System	Primary source of air for penetrations and weld channels	Lined up to Penetration and Weld Channel Pressurization System





Containment Penetration and Weld Channel Pressurization System
Figure 6.6-1

6.7 LEAKAGE DETECTION AND PROVISIONS FOR THE PRIMARY AND AUXILIARY COOLANT LOOPS

6.7.1 LEAKAGE DETECTION SYSTEMS

The leakage detection systems reveal the presence of significant leakage from the primary and auxiliary coolant loops.

6.7.1.1 DESIGN BASES

Monitoring Reactor Coolant Leakage

Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (GDC 16)

Positive indications in the control room of leakage of coolant from the Reactor Coolant System to the containment are provided by equipment which permits continuous monitoring of containment air activity and humidity, and of runoff from the condensate collecting pans under the cooling coils of the containment air recirculation units. This equipment provides indication of normal background which is indicative of a basic level of leakage from primary systems and components. Any increase in the observed parameters is an indication of change within the containment, and the equipment provided is capable of monitoring this change. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, condensate runoff and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

Monitoring Radioactivity Releases

Criterion: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive. (GDC 17)

The containment atmosphere, the ventilation exhausts from the residual heat removal pumps compartments, the containment fan-coolers service water discharge, the component cooling loop liquid, the liquid phase of the secondary side of the steam generator and the condenser air ejector exhaust are monitored for radioactivity concentration during normal operation, anticipated transients and accident conditions.

Principles of Design

The principles for design of the leakage detection systems can be summarized as follows:

1. Increased leakage could occur as the result of failure of pump seals, valve packing glands, flange gaskets or instrument connections. The maximum leakage rate calculated for these type of failures is 50 gpm which would be the anticipated flow rate of water through the pump seal if the entire seal were wiped out and the area between the shaft and housing were completely open.

2. The leakage detection systems shall not produce spurious annunciation from normal expected leakage rates but shall reliably annunciate increasing leakage.

3. Increasing leakage rate shall be annunciated in the control room. Operator action will be required to isolate the leak in the offending system.

6.7.1.2 SYSTEMS DESIGN AND OPERATION

Various methods are used to detect leakage from either the primary loop or the auxiliary loops. Although described to some extent under each system description, all methods are included here for completeness.

Reactor Coolant System

During normal operation and anticipated reactor transients the following methods are employed to detect leakage from the Reactor Coolant System.

Containment Air Particulate Monitor

This channel takes continuous air samples from the containment atmosphere and measures the air particulate gamma radioactivity. The samples, drawn outside the containment, are in a closed, sealed system and are monitored by a scintillation counter - filter paper detector assembly. The filter paper collects all particulate matter greater than 1 micron in size on its constantly moving surface, which is viewed by a hermetically sealed scintillation crystal (NaI) - photomultiplier combination. After passing through the gas monitor, the samples are returned to the containment.

The filter paper has a 25-day minimum supply at normal speed. The filter paper mechanism, an electromagnetic assembly which controls the filter paper movement, is provided as an integral part of the detector unit.

The detector assembly is in a completely closed housing. The detector output is amplified by a preamplifier and transmitted to the Radiation Monitoring System cabinet in the control room. Lead shielding is provided to reduce the background radiation level to where it does not interfere with the detector's sensitivity.

The activity is indicated on meters and recorded by a multipoint recorder. High-activity alarm indications are displayed on the control board annunciator in addition to the Radiation Monitoring cabinets. Local alarms provide operational status of supporting equipment such as pumps, motors and flow and pressure controllers.

The containment air particulate monitor is the most sensitive instrument of those available for detection of reactor coolant leakage into the containment. This instrument is capably of detecting particulate radioactivity in concentrations as low as 10^{-9} μc/cc of containment air. The measuring range is 10^{-9} to 10^{-6} μc/cc.

The sensitivity of the air particulate monitor to an increase in reactor coolant leak rate is dependent upon the magnitude of the normal base-line leakage into the containment. The sensitivity is greatest where base-line leakage is low as has been demonstrated by the experience of Indian Point Unit No. 1 (See Appendix 6B), Yankee Rowe and Dredsen Unit 1. Where containment air particulate activity is below the threshold of detectibility, operation of the monitor with stationary filter paper would increase leak sensitivity to a few cubic centimeters per minute. Assuming a low background of containment air particulate radioactivity, a reactor coolant corrosion product radioactivity (Fe, Mn, Co, Cr) of approximately 0.4 μc/cc (a value consistent with little or no fuel cladding leakage), and complete dispersion of the leaking raidioactive solids into the containment air, the air particulate monitor is capable of detecting an increase in coolant leakage rate as small as approximately 0.025 gpm (100 cc/minutes within twenty minutes after it occurs. If only ten per cent of the particulate activity is actually dispersed in the air, leakage rate increases of the order of 0.25 gpm (1000 cc/minute) are detectable within the same time period.

For cases where base-line reactor coolant leakage falls within the detectable limits of the air particulate monitor, the instrument can be adjusted to alarm on leakage increases of from two to five times the base-line volume.

The containment air particulate monitor together with the other radiation monitors mentioned in this section are further described in Section 11.2.

Containment Radioactive Gas Monitor

This channel measures the gaseous gamma radioactivity in the containment by taking the continuous air samples from the containment atmosphere, after they pass through the air particulate monitor, and drawing the samples through a closed, sealed system to a gas monitor assembly.

Each sample is constantly mixed in the fixed, shielded volumes, where it is viewed by Geiger-Mueller tubes. The samples are then returned to the containment.

The detector is in a completely enclosed housing containing a gamma sensitive Geiger-Mueller tube mounted in a constant gas volume container. Lead shielding is provided to reduce the background radiation level to a point where it does not interfere with the detector's sensitivity. A preamplifier and impedance matching circuit is mounted at the detector.

The detector outputs are transmitted to the Radiation Monitoring System cabinets in the control room. The activity is indicated by meters and recorded by a multipoint recorder. High-activity alarm indications are displayed on the control board annunciator in addition to the Radiation Monitoring System cabinets. Local alarms annunciate the supporting equipment's operational status.

The containment radioactive gas monitor is inherently less sensitive (threshold at 10^{-6} μc/cc) than the containment air particulate monitor, and would function in the event that significant reactor coolant gaseous activity exists from fuel cladding defects. The measuring range is 10^{-6} to 10^{-3} μc/cc. Assuming the design valve of reactor coolant gaseous activity (1% fuel cladding defects), the occurence of a coolant leak of one gpm would double the background in about two hours. For coolant gaseous activity consistent with minimal cladding defects, a one gpm coolant leak would double the background in approximately two minutes. In these circumstances this instrument is a useful backup to the air particulate monitor.

The containment air particulate and radioactive gas monitors have assemblies that are common to both channels. They are described as follows:

i) The flow assembly includes a pump unit and selector valves that provide a representative sample (or a "clean" sample) to the detector.

ii) The pump unit consists of:

1. A pump to obtain the air sample.

2. A flowmeter to indicate the flow rate.

3. A flow control valve to provide flow adjustment.

4. A flow alarm assembly to provide low and high flow alarm signals.

iii) Selector valves are used to direct the desired sample to the detector for monitoring and to blow flow when the channel is in maintenance or "purging" condition.

iv) A pressure sensor is used to protect the system from high pressure. This unit automatically closes an inlet and outlet valve upon a high pressure condition.

v) Purging is accomplished with a valve control arrangement whereby the normal sample flow is blocked and the detector purged with a "clean" sample. This facilitates detector calibration by establishing the background level and aids in verifying sample activity level.

vi) The flow control panel in the control room Radiation Monitoring System racks permits remote operation of the flow control assembly. By operating a sample selector switch on the control panel the containment sample may be monitored.

vii) A sample flow rate indicator is claibrated linearly (from 0 to 14) cubic feet per minute.

Alarm lights are actuated by the following:

1. Flow alarm assembly (low or high flow)

2. The pressure sensor assembly (high pressure)

3. The filter paper sensor (paper drive malfunction)

4. The pump power control switch (pump motor on)

Humidity Detector

The humidity detection instrumentation offers another means of detection of leakage into the containment. Although this instrumentation has not nearly the sensitivity of the air particulate monitor, it has the characteristics of being sensitive to vapor originating from all sources within the containment, including the reactor coolant and steam and feedwater systems. Plots of containment air dew point temperature variations above a base-line maximum established by the cooling water temperature to the air coolers should be sensitive to incremental increases of water leakage to the containment atmosphere on the order of .25 gpm per F degree of dewpoint temperature increase.

The sensitivity of this method depends on cooling water temperature, containment air temperature vatiation and containment air recirculation rate.

Condensate Measuring System

This method of leak detection is based on the principle that, under equilibrium conditions, the condensate flow draining from the cooling coils of the containment air handling units will equal the amount of water (and/or steam) evaporated from the leakage system. Reasonably accurate measurement of leakage from the Reactor Coolant System by this method is possible, because containment air temperature and humidity promote complete evaporation of any leakage from hot systems. The ventilation system is designed to promote good mixing within the containment. During normal operation the containment air conditions will be maintained near 120°F DB and 92°F WB (approximately 36% Relative Humidity) by the fan-coolers.

When the water from a leaking system evaporates into this atomosphere, the humidity of the fan-cooler intake air will begin to rise. The resulting increase in the condensate drainage rate is given by the equation

$$D = L\ [1-\exp\ (-\frac{Q}{V}\ t)\ ]$$

Where

D = Condensate drainage rate (gpm)

L = Evaporated leakage (gpm)

Q = Containment ventilation rate (CFM)

V = Containment free volume (ft^3)

t = Time after start of leak (min.)

Therefore, if four fan cooler units are operating (Q = 280,000 CFM), the condensation rate would be within 5% of a new equilibrium value in approximately 200 minutes after the start of the leak. Detection of the increasing condensation rate, however, would be possible within 5 to 10 minutes.

The condensate measuring device consists essentially of a vertical 6 inch diameter standpipe with a triangular weir cut into the upper portion of the pipe, to serve as an overflow. Each fan cooler is provided with a standpipe which is installed in the drain line from the fan-cooler unit. A differential pressure transmitter near the bottom of the standpipe is used to measure the water level. Each unit can be drained by a remote operated valve.

A wide range of flow rates can be measured with this device. Flows less than 1 gpm are measured by draining the standpipe and observing the water level rise as a function of time. Condensate flows from 1 gpm to 30 gpm may be measured by observing the height of the water level above the crest notch of the weir. This water head can be converted to a proportional flow rate by means of a calibration curve. A high level alarm, set above the established normal (base-line) flow is provided for each unit, to warn the operator when operating limits are approached.

All indicators, alarms, and controls are located in the control room.

Component Cc..ing Liquid Monitor

Thic channel continuously monitors the component cooling loop of the Auxiliary Coolant System for activity indicative of a leak of reactor coolant from either the Reactor Coolant System, the recirculation loop, or the residual heat removal loop of the Auxiliary Coolant System. A scintillation counter is located in an in-line well at the component cooling pump suction header. The detector assembly output is amplified by a preamplifier and transmitted to the Radiation Monitoring System cabinets in the control room. The activity is indicated on a meter and recorded by a multipoint recorder. High-activity alarm indications are displayed on the control board annunciator in addition to the Radiation Monitoring System cabinets.

The measuring range of this monitor is 10^{-5} to 10^{-2} μc/cc.

Condenser Air Ejector Gas Monitor

This channel monitors the discharge from the air ejector exhaust header of the condensers for gaseous radiation which is indicative of a primary to secondary system leak. The gas discharge is routed to the turbine roof vent.

The detector output is transmitted to the Radiation Monitoring System cabinets in the control room. The activity is indicated by a meter and recorded by a multipoint recorder. High-activity alarm indications are displayed on the control board annunciator in addition to the Radiation Monitoring cabinets.

A gamma sensitive Geiger-Mueller tube used to monitor the gaseous radiation level. The detector is inserted into an in-line fixed volume container which includes adequate shielding to reduce the background radiation to where it does not interfere with the detectors maximum sensitivity. This monitor has a maximum sensitivity of 10^{-6} microcuries per cubic centimeter.

Steam Generator Liquid Sample Monitor

This channel monitors the liquid phase of the secondary side of the steam generator for radiation, which would indicate a primary-to-secondary system leak, providing backup information to that of the condenser air ejector gas monitor. Samples from the bottom of each of the four steam generators are mixed to a common header and the common sample is continuously monitored by a scintillation counter and holdup tank assembly. Upon indication of a high-radiation level, each steam generator is individually sampled in order to determine the source. This sampling sequence is achieved by manually selecting the desired unit to be monitored and allotting sufficient time for sample equilibrium to be established (approximately 1 minute).

A remote indicator panel, mounted at the detector location, indicates the radiation level and high-radiation alarm.

The measuring range of this monitor is 10^{-5} to 10^{-2} microcuries per cubic centimeter.

A photomultiplier tube - scintillation crystal (NaI) combination, mounted in a hermetically sealed unit, is used to monitor liquid effluent activity. Lead shielding is provided to reduce the background level so it does not interfere with the detector's maximum sensitivity. The in-line, fixed-volume container is an integral part of the detector unit.

During cold shutdown personnel can enter the containment and make a visual inspection for leaks. The location of any leak in the Reactor Coolant System woul' 'e determined by the presence of boric acid crystals near the leak. The leaking fluid transfers the boric acid crystals outside the Reactor Coolant System and the process of evaporation leaves them behind.

If an accident involving gross leakage from the Reactor Coolant System occurred it could be detected by the following methods.

Pump Activity

During normal operation only one charging pump is operating. If a gross loss of reactor coolant to another closed system occurred which was not detected by the methods previously described, the speed of the charging pump would indicate the leakage.

The leakage from the reactor coolant will cause a decrease in the pressurizer liquid level that is within the sensitivity range of the pressurizer level indicator. The speed of the charging pump will automatically increase to try to maintain the equivalence between the letdown flow and the combined charging line flow and flow across the reactor coolant pump seals. If the pump reaches a high speed limit, an alarm is actuated.

A break in the primary system would result in reactor coolant flowing into the containment and/or recirculation sumps. Gross leakage to these sumps would be indicated by the frequency of operation of the containment or recirculation pumps. Since the building floor drains perferentially to the containment sump, the activity of the containment sump pumps would be more likely to indicate the leak than the activity of the recirculation sump pump.

Liquid Inventory

Gross leaks might be detected by unscheduled increases in the amount of reactor coolant makeup water which is required to maintain the mormal level in the pressurizer.

A large tube side to shell side leak in the non-regenerative (letdown) heat exchanger would result in reactor coolant flowing into the component cooling water and a rise in the liquid level in the component cooling water surge tank. The operator would be alerted by a high water alarm for the surge tank and high radiation and temperature alarms actuated by monitors at the component cooling water pump suction header. In addition a low flow alarm would be actuated by a monitor on the outlet line of the Chemical and Volume Control System from the non-regenerative heat exchanger.

A high level alarm for the component cooling water surge tank and high radiation and temperature alarms actuated by monitors at the component cooling pump suction header could also indicate a thermal barrier cooling coil rupture in a reactor coolant pump. However, in addition to these alarms, high temperature and high flow on the component cooling outlet line from the pump would activate alarms.

Gross leakage might also be indicated by a rise in the normal containment and/or recirculation sump levels. High level in either of these sumps will actuate an alarm. Since the building floor drains preferentially to the containment sump, the containment sump level transmitter would most likely actuate alarm prior to the level transmitter in the recirculation sump.

Residual Heat Removal Loop

The residual heat removal loop removes residual and sensible heat from the core and reduces the temperature of the Reactor Coolant System during the second phase of plant shutdown.

During normal operation the containment air particulate and radioactive gas monitors, the humidity detector and the condensate measuring system provide means for detecting leakage from the section of the residual heat removal loop inside the reactor containment. These systems have been described previously in this section (see description of leak detection from the Reactor Coolant System). Leakage from the residual heat removal loop into the component cooling water loop during normal operation would be detected outside the containment by the component cooling loop radiation monitor (see analysis of detection of leakage from the Reactor Coolant System in this section). 13

The physical layout of the two residual heat removal pumps is within separate shielded and isolated rooms outside of the containment. This will permit the detection of a leaking residual heat removal pump by means of radiation monitors located in the ventilation exhaust ducts from each compartment. Supplemental radiation monitoring will be provided by a plant vent gas monitoring system. Alarms in the control room will alert the operator when the activity exceeds a preset level. Small leaks to the environment could be detected with these systems within a short time after they occurred.

When the plant is shutdown personnel can enter the containment to check visually for leaks. Detection of the location of significant leaks would be aided by the presence of boric acid crystals near the leak.

In case of an accident which involves gross leakage from the part of the residual heat removal loop inside the containment, this leakage would be indicated by a rise in the containment and/or recirculation sump levels. Both of these sumps have level transmitters which activate an alarm if the level exceeds a preset level. As the building floor drains preferentailly to the containment sump, the level transmitter in this sump would most likely actuate an alarm first.

Should a large tube side to shell side leak develop in a residual heat exchanger or the seal of a residual heat removal pump break, the water level in the component cooling surge tank would rise, and the operator would be alerted by a high water alarm. Radiation and temperature monitors at the component cooling water pump suction header will also signal an alarm. In addition, in the case of a residual heat removal pump seal failure, a flow monitor on the component cooling outlet line from the pump will actuate an alarm.

Leakage from both of the residual heat removal pumps is drained to a common sump equipped with a sump pump. The operation of the sump pump will be indicated in the control room as a means of detection of gross leakage (i.e., a seal failure) from a residual heat removal pump.

Recirculation Loop

If a break occurs in the Reactor Coolant System, the recirculation loop provides long-term protection by recirculating spilled reactor coolant and injected refueling water.

The containment air particulate and radioactive gas monitors, the humidity detector and the condensate measuring system (see section discussing leak detection for the Reactor Coolant System) provide means of detecting small leaks in the part of the recirculation loop inside the reactor containment.

Leakage from the residual heat exchanger would be detected by a radiation monitor (discussed in the section on leak detection from the Reactor Coolant System) at the component cooling water pump suction header.

Entry to the containment is permissible during cold shutdown, and personnel could check for leaks at this time by looking for the presence of boric acid crystals.

Gross leakage from the recirculation loop inside the containment might be indicated by a rise in the level of the containment and/or recirculation sumps. Both of these sumps have a level transmitter which will actuate an alarm if the level goes above a preset level.

A rise in the liquid level in the component cooling surge tank would result if a large tube side to shell side leak developed in a residual heat exchanger. The operator would be alerted by a high level alarm in the component cooling water surge tank and a high radiation and temperature alarm actuated by monitors at the component cooling water pump suction header.

If the external recirculation loop is used, leakage from the section outside the containment would be directed by floor drains to the auxiliary building sump. From here it is transferred by sump pumps to the waste holdup tank. The operation of these auxiliary building sump pumps is indicated in the control room and would alert the operator to the leakage.

Component Cooling Loop

Leakage from the component cooling loop inside the reactor containment could be detected by the humidity detector and/or the condensate measuring system (see section on Reactor Coolant System leak detection for a description of these systems).

Visual inspection inside the containment is possible during cold shutdown.

Gross leakage from the component cooling loop would be indicated inside the containment by a rise in the liquid level of the containment and/or recirculation sumps. Both of these sumps have a high level alarm. Since the building floor drains preferentially to the containment sump the level transmitter in the sump would be more likely to signal an alarm.

If the leakage is from a part of the component cooling loop outside the containment, it would be directed by floor drains to the auxiliary building sump. The auxiliary building sump pumps then transfer the leakage to the waste holdup tank. Operation of the sump pumps is indicated in the control room and would thus serve as a means of leak detection for this part of the system.

Service Water System

During a loss-of-coolant accident the containment fan coolers service water monitors check the containment fan service water discharge line for radiation indicative of a leak from the containment atmosphere into the service water. A small bypass flow from each of the heat exchangers is mixed in a common header and monitored by redundant scintillation detectors mounted in separate holdup tank assemblies. Upon indication of a high radiation level, each heat exchanger is individually sampled to determine which unit is leaking. This sampling sequence is achieved by manually selecting the desired unit to be monitored and allotting sufficient time for sample equilibrium to be established (approximately 1 minute).

The measuring range of this monitor is 10^{-5} to 10^{-2} curies per cubic centimeter.

Gross leakage into the service water due to a faulty cooling coil in the Containment Air Recirculaiton Cooling and Filtration System can be detected by stopping the fans and continuing the cooling water flow. Any significant cooling water leakage would be seen as flow into a collecting pan.

Leakage from a component in the service water system will be directed by floor drains to the auxiliary building sump. Pumps will then transfer this leakage to the waste holdup tank. Operation of the auxiliary building sump pump is indicated in the control room and would serve as a means of leak detection.

6.7.2 LEAKAGE PROVISIONS

Provisions are made for the isolation and containment of any leakage.

6.7.2.1 DESIGN BASIS

The provisions made for leakage are designed to prevent uncontrolled leaking of reactor coolant or auxiliary cooling water. This is accomplished by (1) isolation of the leak by valves, (2) designing relief valves to accept the maximum flow rate of water from the worst possible leak, (3) supplying redundant equipment which allows a standby component to be p[illegible] in operation while the leaking component is repaired and (4) routing the leakage to various sumps and holdup tanks.

6.7.2.2 DESIGN AND OPERATION

Various provisions for leakage avert uncontrolled leakage from the primary and auxiliary coolant loops.

<u>Reactor Coolant System</u>

When significant leakage from the Reactor Coolant System is detected, action is taken to prevent the release of radioactivity to the atmosphere outside the plant.

If either the containment air particulated gamma activity or the radioactive gas activity exceed pre-set levels on the containment air particulate and radioactive gas monitors, respectively, the containment purge supply and exhaust duct valves and pressure relief line valves are closed.

On high radiation alarm signaled by the condenser air ejector monitor, the condenser exhaust gases are diverted from the turbine roof vent to the containment through a blower.

A high radiation alarm actuated by the steam generator liquid sample monitor initiates closure of the isolation valves in the blowdown lines and sample lines.

If the component cooling loop radiation monitor signals a high radiation alarm, the valve in the component cooling surge tank vent line automatically closes to prevent gaseous activity release.

If a leak from the Reactor Coolant System to the component cooling loop was a gross leak or if the leak could not be isolated from the component cooling loop before the inflow completely filled the surge tank, the relief valve on the surge tank would rise. The discharge from this valve is routed to the waste holdup tank in the auxiliary building.

A large leak in the Reactor Coolant System pressure boundary, which does not flow into another closed loop, would result in reactor coolant flowing into the containment sump and/or the recirculation sump.

Experience with the detection of Primary System leakage into the containment vessel of Indian Point Unit I is discussed in Appendix 6B.

Residual Heat Removal Loop

High containment air particulated gamma activity of high radioactive gas activity will result in an alarm being activated by either the containment air particulate or radioactive gas monitors, respectively. The containment purge supply and exhaust duct valves and pressure relief line valves are closed. This prevents the release of radioactivity to the atmosphere outside the nuclear plant.

If leakage from the residual heat removal loop into the component cooling loop occurs, the component cooling radiation monitor will actuate an alarm and the valve in the component cooling surge tank vent line is automatically closed to prevent gaseous radioactivity release. If the leaking component (i.e., a residual heat exchanger) could not be isolated from the component cooling loop before the inflow completely filled the surge tank, the relief valve on the surge tank would lift and the effluent would be discharged to the auxiliary building waste holdup tank.

Gross leakage from the section of the residual heat removal loop inside the containment, which does not flow into another closed loop, would result in reactor coolant flowing into the containment sump and/or the recirculation sump.

Other leakage provisions for the the residual heat removal loop are discussed in Section 9.3.

Recirculation Loop

The containment purge supply and exhaust duct valves and pressure relief line valves are closed when either the containment air particulate or the radioactive gas monitors read above a pre-set level. This prevents radioactivity from escaping to the outside atmosphere.

Leakage from the recirculation loop into the component cooling loop results in a radiation alarm and the automatic closing of the component cooling surge tank vent line to prevent gaseous radioactivity release. If the leak was gross and filled the surge tank before the leaking component could not be isolated from the component cooling loop, the relief valve on the surge tank would lift and the effluent would be discharge to the waste holdup tank in the auxiliary building.

Gross leakage from the internal recirculation loop which does not flow into another closed loop will flow into the containment sump and/or the recirculation sump. Gross leakage from the external recirculation loop which does not flow into another closed loop will be drained to the auxiliary building sump. From here it is pumped to the waste holdup tank.

Component Cooling Loop

Gross leakage from the section of the component cooling loop inside the containment which does not flow into another closed loop will flow into the containment sump and/or the recirculation sump. Outside the containment major leakage would be drained to the primary auxiliary building sump. From here it is pumped to the waste holdup tank.

Other provisons made for leakage from the component coo'ing loop are discussed in Section 9.3.

Service Water System

Gross leakage from the service water system will be directed by floor drains to the primary auxiliary sump. Pumps will then transfer this leakage to the waste holdup tank.

APPENDIX 6A

IODINE REMOVAL EFFECTIVENESS EVALUATION OF THE CONTAINMENT SPRAY SYSTEM

1.0 PURPOSE OF CHEMICAL MODIFICATION

The containment spray system in this pressurized water reactor facility is one of the engineered safety features systems employed following a loss-of-coolant inside the containment to reduce the pressure and temperature of the containment atmosphere. The flow rate and inlet subcooling of the spray are sufficient to provide thermal capacity for condensing steam produced by dissipation of heat in the reactor and its associated systems. Minimum operability of these systems with on-site power and under a single-component failure contingency will prevent pressurization of containment above the design pressure with a substantial capacity margin.

The spray system, by virtue of the large surface area provided between the liquid droplets and the containment atmosphere, affords an excellent means of absorbing soluble components from the gas phase. If the solubility of the component is sufficiently high, the rate of absorption is limited only by the mass transfer rate of the absorbing species through the gas film. In the case of I_2 vapor, elimination of all but the gas film resistance would permit the absorption by sprays to proceed with a removal half-life of less than two minutes, as will be shown later. However the solubility of I_2 in the refueling water used as spray is limited, as indicated by the partition coefficient [1]

$$K_c = 0.0125 \ \frac{\text{mol/liter gas}}{\text{mol/liter liquid}}$$

in acidic solution. While this coefficient corresponds to an equilibrium favoring solution of 60-80% of the iodine by the liquid (considering the gas/liquid volume ratios of conventional PWR containments) it is expected that the liquid phase mass transfer resistance would severely limit the removal rate. Assuming a liquid film coefficient of 0.01 cm/sec and a gas film coefficient (to be derived later) of the order of 10 cm/sec, the overall mass transfer coefficient, V_T, is obtained as follows:

$$\frac{1}{V_T} = \frac{1}{V_G} + \frac{Kc}{V_L} = \frac{1}{10} + \frac{0.0125}{0.01} = 0.1 + 1.25 \qquad (1)$$

$$V_T = 0.74 \text{ cm/sec}$$

If the I_2 were infinitely soluble (K_c ~ 0), the value of V_T would approach 10 cm/sec in this example.

To obtain the advantages of an order-of-magnitude improvement of absorption rate and nearly complete removal of I_2 at equilibrium, the chemistry of the spray solution is modified by adding NaOH, raising the pH to 9.5. According to the known behavior of elemental iodine in highly dilute solutions the hydroxysis reaction

$$I_2 + OH^- \longrightarrow HIO + I^-$$

proceeds nearly to completion[2] at pH >8. The iodide form is highly soluble, and HIO readily oxidixes to IO_3^- in the oxygenated medium, this form being likewise soluble:

$$HIO + O2 \longrightarrow IO_3^- + H^+ \qquad (2)$$

Griffith[1] suggested that the use of chemical additives which undergo ionic reactions with aqueous I_2 would improve the absorption rate to the point where the gas film mass transfer resistance became limiting, implying that $K_C/V_L << 10^{-1}$. His paper called attention to sodium thiosulfate ($Na_2S_2O_3$) as a likely reagent for this purpose and mentioned NaOH as another candidate. Subsequent experiments in a spray medium have shown that both additives bring about absorption rates indicative of gas film control, verifying the desired rate capability.

The selection of NaOH instead of $Na_2S_2O_3$ for this application followed an evaluation program which revealed the following disadvantages for $Na_2S_2O_3$: The results of this evaluation program are detailed in the Proprietary Westinghouse report, WCAP-7153.

By contrast, the same testing program revealed no instability of the solution formed by adding NaOH alone to the borated spray. Corrosion rates of copper and copper-alloy heat exchanger tubing were reduced by more than an order of magnitude compared with high pH $Na_2S_2O_3$ and were acceptably low ($<$0.01 mils/ month at 200°F) for the application. These tests showed that pitting or local corrosion did not occur.

For engineering reasons, therefore, further testing was centered on the use of NaOH as the spray additive leading to the development of a technical basis for its inclusion in the plant engineered safety features as a means of "fixing" absorbed iodine, enhancing the natural rate of deposition of I_2, and thus lowering the calculated off-site thyroid dose resulting from a postulated release of fission products to the containment atmosphere. In summary, this work supports the following conclusions comprising the technical basis for spray absorption process for iodine removal:

1) The conversion of absorbed I_2 to I^- and IO_2^- in pH 9.5 borate solution is quite rapid, such that the absorption process is gas film diffusion controlled.

2) Mass transfer follows the Ranz-Marshal rate equation [3] for soluble gases, as demonstrated by containment simulation tests performed with a nozzle design atmospheric conditions, iodine concentration and spray chemical composition in dose approximation to the design basis accident.

3) Under a range of conditions bracketing the possible accident modes, the spray modification shows irreversible and effective iodine removal, (i.e., K_c is not reduced with time if pH is maintained) compatibility with the vital materials and processes of the containment system and high mechanical reliability.

2.0 TECHNICAL BASIS FOR IODINE REMOVAL FACTOR

2.1 ANALYTICAL MODEL AND ASSUMPTIONS

The removal of a soluble component by a reactive spray under conditions of a constant mass transfer rate coefficient is exponential:

$$C = C_o e^{-\lambda_s t} \qquad (3)$$

The removal constant λ_s can be expressed as the product of a mass transfer coefficient, v_G, and the effective absorbing surface area, A.

In addition to the basic assumption that the absorption is gas-film resistance controlled, the following idealizations are made to simplify the physical model:

1) All droplets behave as spheres of diameter equal to that of the surface-mean diameter droplet, d.

2) Droplets fan at their terminal velocity, u_t, from the spray nozzle to the operating deck, a distance h.

3) Iodine concentration in the gas in uniform.

The effective absorbing area is then

$$A = \frac{6\ F\ h}{\bar{u}_t V_c d} \qquad (4)$$

where A = absorbing area per unit volume
F = volumetric spray flow rate
V_c = containment free volume

For a given droplet size, the terminal velocity and the mass transfer coefficient are temperature and pressure dependent. In the expression for λ_s, the, these variables can be treated as a dimensionless ratio:

$$\lambda_s = v_G A = \left(\frac{v_G}{\bar{u}_t}\right) \frac{6 F h}{V_c d} \tag{5}$$

When the remaining parameters are expressed in engineering units, λ_s in reciprocal hours is given by

$$\lambda_s = 1470 \left(\frac{v_G}{\bar{u}_t}\right) \frac{Fh}{V_c d} \tag{6}$$

where F = spray flow gal/min
h = fall height, ft.
V_c = volume, cu. ft.
d = droplet diameter, cm.

For the various classes of Westinghouse PWR containments, the following values of the physical parameters are conservatively approximated as follows:

	F	h	V_c	d	$\frac{Fh}{V_c d}$
Two Loop	1250	70	970,000	0.1	0.90
Three Loop	880	77	2,100,000	0.1	0.32
Four Loop	2340	104	2,600,000	0.1	0.94

The value of $v_G/\bar{u}_t$ for 0.1 cm droplets in a saturated air-steam atmosphere of pressure P^* is plotted in Figure 6A-1. These data are obtained from ORNL-TM-1911[4]. It is apparent that as pressure decreases during the post-accident period, the value of $v_G/\bar{u}_t$ and hence the removal coefficient λ_s will increase. The removal rate is underestimated, therefore, by assuming for purposes of analysis the value at this ratio at the design condition of the containment. The results, calculated from equation (6), are then:

* Defined as the mixture of air and steam produced by adding steam to the dry air at an initial temperature of 30°C and one atmosphere pressure, at constant volume.

	$v_G/\bar{u}_t$	λ_s, hr^{-1}
Two loop (60 psig, 286 °F)	0.0224	29.6
Three loop (42 psig, 264 °F)	0.0235	11.1
Four loop (47 psig, 270 °F)	0.0231	32.0

The "half-life" for removal of elemental iodine is obtained from the following expression for exponential decay:

$$T_H = \frac{0.693}{\lambda_s} \times 3600 \quad \text{(in seconds)} \tag{7}$$

The "dose reduction facto " applicable to elemental iodine is the ratio of the average two-hour inventory of I_2 without removal to the average with removal. It is given by the following expression:

$$DRF_2 \quad \frac{2\lambda_s}{1 - e^{-2\lambda_s}} \tag{8}$$

The calculated values of λ_s for the three plant types yield the following values of I_2 half-life and I_2 dose reduction factor:

	T_H sec	DRF_2
Two Loop	84.	59.
Three Loop	225.	22.
Four Loop	78.	64.

Concerning other airborne forms of iodine, the removal mechanism can be characterized in the following way:

HI - Hydrogen iodide may constitute an important fraction of the liberated iodine if oxygen is excluded from the reactor during the melt. The higher diffusivity of HI, compared with I_2, and the fact that favorable partition between vapor and liquid does not require that the absorbed HI molecule undergo chemical reaction, would lead to removal of HI by sprays no less rapid than I_2.

CH_3I - A small fraction (probably less than 5% of the available iodine) will exist as organic iodides of which methyl iodide is the most important. There is preliminary evidence that absorption and chemical decomposition of CH_3I occurs in the reference spray solution. The rate of absorption, which is expected to be liquid-film diffusion or liquid-phase reaction rate controlled, is so slow that the reduction of the two-hour average inventory of CH_3I vapor is less than the probable error in predicting that inventory. No credit for removal is taken in calculating the two-hour dose due to organic iodide leakage.

Particles - Spray may have an important effect on particle removal by increasing the rate of steam condensation. When the bulk flow of steam to the condensing surface is great enough to mask the diffusive motion of particles (as would be the case when cold droplets contact the containment atmosphere during the high-steam period), sub-micron particles are efficiently captured by the spray.[5] Larger particles would be removed by high efficiency particulate air (HEPA) filters or would agglomerate and settle out by gravity, reducing their importance as a potential leakage source, if indeed they could penetrate the leakage path at all. In evaluating the potential benefit of sprays in reducing post-accident iodine leakage, no quantitative consideration is given here to particle removal by condensation, because the phenomenon is independent of the chemical modification of the spray solution.

2.2 EXPERIMENTAL VERIFICATION

The droplet size assumed in the spray calculations summarized above was 0.10 cm or 1000 u. The spray pattern produced by a 3/8 in. aperture ramp bottom nozzle of the type used in these facilities was measured photographically at various operating nozzle pressures. A ramp bottom nozzle with a 3/8-inch orifice diameter was selected because it is capable of producing the smallest mean droplet size without employing impingement baffles, swirl vanes, or other features which might trap particles of debris. Note that a 1/4-inch mesh screen is used to trap debris which might otherwise enter the recirculation pump suction from the containment sump.

Should some of the spray nozzles become plugged, considerable performance margin is available. For example, at the time recirculation from the sump would be employed, the condensing capacity of one spray above exceeds the residual heat rate of 25%. Similarly, at the same point in time, there is a greatly reduced dependence on sprays for continued iodine removal because most of the absorbable iodine has been removed prior to recirculation. Theory predicts that over half-lives for removal of elemental iodine have elapsed during the period when clean spray water is being delivered from the storage tanks. One may conclude, therefore, that plugging of about one fifth of the nozzles in one spray system, complete outage of the other spray system, and disability of all five fan coolers could be tolerated at the time of recirculation without losing the ability to transfer residual heat from the containment atmosphere. A statistical analysis of the droplet images produced the following results:

Nozzle Pressure, psi	Flow Rate, gpm	Number-avg diameter μ	Surface-avg diameter μ
20	10.9	960	1340
30	12.9	830	1126
40	15.2	735	1012
50	16.9	685	961

In this table, the "number average diameter" is defined as

$$D_{NA} = D_{G} \exp \frac{\ln^2 S_G}{2}$$

where D_G = 50% undersize diameter

S_G = standard deviation, or ratio of 84% to 50% undersize diameter.

The "surface average diameter" is the diameter at which a uniformly sized array of drops would present the same surface to volume ratio as the observed array of drops. It is this parameter which is significant to the absorption rate model.

The spray system is designed to deliver rated flow with a minimum available nozzle pressure of 40 psi above the containment design pressure. Generally, the nozzle pressure will be more than 40 psi above the actual containment pressure, making the assumption 1000 u a realistic one.

A more meaningful demonstration of effective droplet size and verification of the overall mass transfer model is obtained from the spray test program at Nuclear Safety Pilot Plant (NSPP). Data from these tests are published through the regular ORNL reporting channels.[6,7] The data treatment in this program uses the same basic analytical model as has been presented here, and the results are entirely consistent with the premise that I_2, absorption by Na-OH - H_3BO_3 spray is gas-film controlled.

Applying the removal expression (Eq. 8) to the NSPP system for a typical test condition of 44 psig, 266°F the values corresponding to our plant parameters are

	F	h	V_c	d	$\frac{F\,h}{V_c d}$
NSPP	15	17	1330	0.100	1.92

The ratio $v_G/\bar{u}_t$ for 0.100 cm droplets in a 44 psig steam-air atmosphere, (Figure 1) is 0.0234, giving λ_s = 66 hr^{-1} and a half-line of 38 sec.

As reported by the NSPP investigators [7,8] a run made at these conditions (run No. 30) with $NaOH-H_3BO_3$ spray resulted in an iodine removal half life of 24 to 44 seconds depending on the method of sampling. Generally speaking, NSPP results at a variety of conditions have shown that analytical models based on the gas-film controlled drop-wise absorption theory (Ranz-Marshall equation) tend to underpredict the experimental absorption rate.

Further spray testing is being conducted at the Containment Systems Experiment (CSE). Preliminary results of tests at room temperature but with fall heights (35-50 feet) more comparable to full scale than those in NSPP have been published.[9] These tests have shown that substantial decontamination of the containmant atmosphere with respect to I_2 occurs with a half-life in good agreement with those calculated by the Ranz-Marshall equation.

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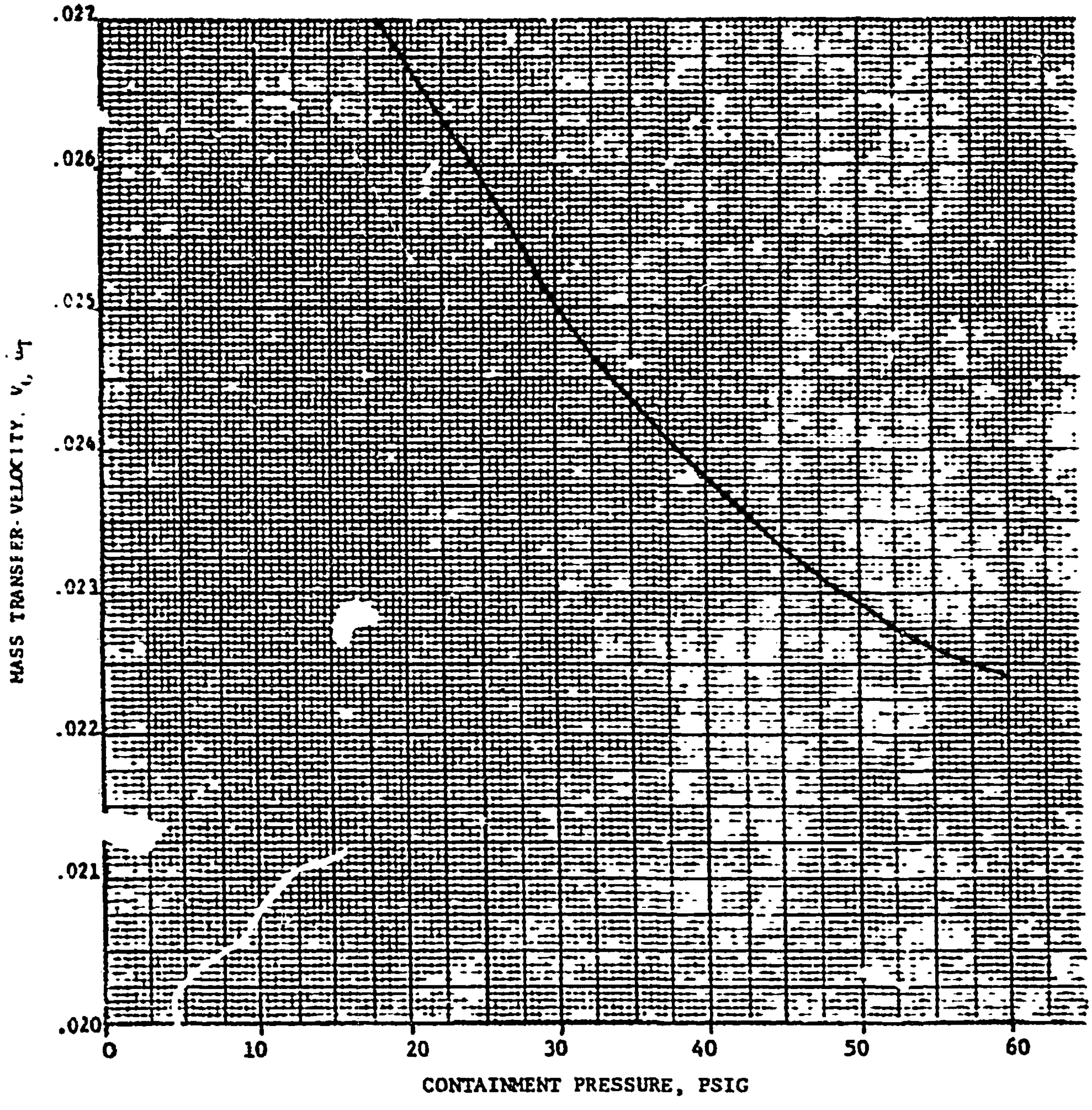

MASS TRANSFER-VELOCITY RATIO OF
SPRAY DROPLETS AS A FUNCTION OF
CONTAINMENT PRESSURE
FIGURE 6A-1

APPENDIX 6B

Primary System Leak Detection into Containment Vessel

Indian Point Unit I

Small leaks developed in the primary system pressure boundary can be detected by several continuously recording instruments available to the plant operators. The most sensitive of these detectors is the radioactive air particulate monitor which continuously samples the air in the containment cooling system. The purpose of the containment cooling system is to maintain proper ambient temperatures for equipment in the containment vessel. This system takes air from the upper elevations of the vessel and recirculates it through cooling coils on the suction side of the supply fan. This air is then discharged at a rate of 40,000 CFM through steam coils. The turnover rate of the containment vessel as a result of this system is approximately once every hour. By sampling air from the discharge of the containment cooling system supply fan, leak rates as small as 0.3 GPH (20 cc/minute) can be detected.

Another detector, the radiogas monitor, sampling air from the same position as the air particulate monitor continuously analyzes air from the containment cooling system for gaseous radioactivity. This monitor is capable of detecting a leak rate of about 100 GPH (6,500 cc/minute).

In addition to measuring changes in the radioactivity of the containment vessel, dew point sensors continuously sample the air from the suction side of the containment cooling system supply fans. These instruments can detect a primary coolant leak rate of approximately 4 GPH (250 cc/minute) by measuring changes in the moisture content of the containment vessel.

By the use of the above instruments, plant operators can continuously monitor the containment vessel for primary system leakage and take any steps necessary to safely operate the facility. Measurements made by the New York University Medical Center, Institute of Environmental Medicine, have shown that the samples analyzed by these instruments are representative of the containment vessel and that manual samples taken to back up these detectors are accurate to within a factor of 2.

Other methods for detecting and locating primary system leakage include visual inspection for escaping steam or water, Boric acid crystal formation, component and primary relief tank levels, hydrogen concentration and radioactivity, containment sump level, and manual samples for tritium radioactivity in condensed moisture from the containment vessel.

Although primary system leakage in Unit No. 1 has been minimal, a combination of all the above mentioned instrumentation has been used to detect several leaks ranging in size from 0.1 to 3 GPH. However, due to the magnitude of these leaks, positive identification has only resulted from visual inspection during containment entries made after the plant has been shut down.

ASSUMPTIONS:

1. Uniform mixing in containment occurs within one hour after a leak, based upon one containment cooling fan in service at 40,000 cfm.

2. The smallest significant changes in plant instrumentation are:

   a - Radiogas monitor on the containment cooling system:
   1 cps is equivalent to 3×10^{-7} μc/cc

   b - Particulate monitor -
   8 cps is equivalent to 8×10^{-9} μc/cc

   c - Dewpoint
   4°F

An eight hour period is used to evaluate these changes which provides time for checking instrumentation and determining the cause of the changes. The eight hour evaluation period is predicted on determination of the magnitude of small leaks. Large leaks would of course be evaluated much sooner.

Basic Data Used for Calculations

1 - Sphere Volume

1.8×10^{6} ft^3 (5.05×10^{10} cc)

2 - Sphere Environment

a - average temperature - 120°F
b - dewpoint - 70°F

3 - Normal Containment Cooling Radioactivity

a - radiogas 2.5 cps (7.5×10^{-7} μc/cc)
b - particulate 16 cps (1.6×10^{-3} μc/cc)

4 - Normal Primary Coolant Radioactivity after one hour

a - radiogas activity 5×10^{-3} μc/ml H_2O
b - particulate 5×10^{-2} μc/ml H_2O

Sample Calculations

1 - Dewpoint in Containment Cooling System

a - At 120°F and 70°F dewpoint - the water content of the sphere would be 0.016 lbs of water per lb of dry air
b - At a dewpoint of 74°F the water content of the sphere would be 0.018 lbs of water per lb of dry air

let X = the leak rate into the sphere in gallons per hour

$$X = \left[\frac{(0.018 - 0.016) \text{ lbs } H_2)/\text{lb dry air} \times 1.8 \times 10^{6} ft^3 \times 0.081 \times \frac{109}{121} \text{ lbs}/ft^3}{8 \text{ hrs} \times 8.3 \text{ lbs/gal}} \right]$$

= 3.95 GPH or 100 gallons per day

2 - Radioactivity in Containment Cooling System

a - Radiogas activity

1 - Increase in activity 1 CPS on installed monitor
The radiogas activity increase ≈ 3×10^{-7} μc/cc

2 - Let Y = leak rate into a sphere in gal per hour

$$Y = \left[\frac{3.0 \times 10^{-7} \text{ μc/cc air} \times 5.05 \times 10^{10} \text{ cc air}}{8 \text{ hrs} \times 5 \times 10^{-3} \text{ μc/ml } H_2O \times 3.8 \times 10^{3} \text{ ml/gal}} \right]$$

= 99.8 GPH or 2,400 gallons per day

b - Particulate activity in containment cooling system
8 CPS on the installed monitor
Radioactivity increase ≈ 8×10^{-9} μc/cc air
Let Z = leak rate into the sphere in gallons per hour

$$Z = \left[\frac{8 \times 10^{-9} \text{ μc/cc air} \times 5.05 \times 10^{10} \text{ cc air}}{8 \text{ hrs} \times 5 \times 10^{-2} \text{ μc/ml } H_2O \times 3.8 \times 10^{3} \text{ ml/gal}} \right]$$

= 0.265 GPH or 6 gallons per day