5 CONTAINMENT
SYSTEM

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5 CONTAINMENT SYSTEM

The containment for this station consists of two systems which are:

a. The Reactor Building to provide biological and missile shielding and to contain the energy and material that could be released by an accident.

b. The engineered safeguards systems which limit the maximum value of the energy released by an accident.

5.1 REACTOR BUILDING

The Reactor Building will be a reinforced concrete structure composed of cylindrical walls with a flat foundation mat and a shallow dome roof. The foundation slab will be reinforced with conventional mild-steel reinforcing. The cylindrical walls will be prestressed with a post-tensioning system in the vertical and horizontal directions. The dome roof will be prestressed utilizing a three-way post tensioning system. The inside surface of the Reactor Building will be lined with a carbon steel liner to insure a high degree of tightness during operating and accident conditions. Liner plate thickness will be 3/8 in. for the cylinder and dome and 1/4 in. for the base.

The foundation mat will be bearing on sound rock and will be approximately 9 ft thick with a 2 ft thick concrete slab above the bottom liner plate. The cylinder portion will have an inside diameter of 130 ft, wall thickness of 3 ft 6 in., and a height of 157 ft from top of foundation slab to the spring line. The shallow dome roof will have a large radius of 110 ft, a transition radius of 20 ft 6 in., and a thickness of 3 ft. The Reactor Building is shown in Figure 5-1.

The experience and knowledge gained in the design at Rochester Gas and Electric Company's Brookwood Plant No. 1 (AEC Docket No. 50-244), as well as designs by others that are similar in functional requirements, will be utilized in the design of this Reactor Building.

5.1.1 DESIGN BASES

5.1.1.1 Postulated Accident Conditions

The Reactor Building is to provide biological shielding for normal and accident conditions. It will enclose the reactor and the reactor coolant system and will be designed to insure that an acceptable upper limit of leakage of radioactive material will not be exceeded under the maximum loss-of-coolant accident as described in Section 14, "Safety Analysis." The accident is based on a double-ended pipe break in the main coolant system and produces pressures and temperatures that are influenced by the safeguard system, heat sinks, and energy sources. This is described in Section 6, "Engineered Safeguards" and Section 14 "Safety Analysis."

5.1.1.2 Energy and Mass Releases

Additional energy and mass will be available for release into the Reactor Building from the following sources:

a. Stored heat in the reactor
b. Reactor decay heat
c. Stored heat in the reactor coolant system
d. Water-Metal reaction
e. Hydrogen combustion
f. Fission products in the core

The energy released from these sources is discussed in Section 14. The energy contribution from the secondary steam system is not included in the calculations for the Reactor Building design pressure and temperatures. The supports for the reactor coolant system will be designed to withstand the forces associated with a break in the reactor coolant pipe so that a rupture in the secondary system will not be considered to act simultaneously with a reactor coolant pipe break.

5.1.1.3 Contribution of Engineered Safeguard Systems

The contribution of the engineered safeguard system is discussed in Section 6. These systems will be actuated to minimize the accident conditions by removing heat from the Reactor Building and inserting negative reactivity into the reactor.

These safeguard systems will be:

a. A high pressure injection system
b. A low pressure injection system
c. A core flooding system
d. A Reactor Building emergency cooling system
e. A Reactor Building spray system
f. A Reactor Building isolation system

5.1.2 STRUCTURE DESIGN

5.1.2.1 Design Conditions

The Reactor Building will be a Class 1 structure as defined in Appendix 5A. The internal free volume will be at least 2,000,000 cubic feet. It will be designed for an internal pressure of 55 psig with a coincident temperature of 281 F at accident conditions and an external pressure differential of 2.5 psi at normal working conditions. Due consideration will be given to the dead load, live load, temperature gradients, and penetrations at accident and working conditions. The functional requirements for the liner are covered in detail in Appendix 5E, "Liner Plate Specification".

The design moments, shears, deflections, membrane forces, and stresses are being developed but have not been completed. For preliminary structural concept of the Reactor Building refer to Figure 5-1. The design criteria for the Reactor Building is covered in Appendix 5C, "Design Criteria for Concrete Reactor Building".

5.1.2.2 Design Leakage Rate

The Reactor Building will be designed to limit the leakage rate to 0.2 per cent by weight in 24 hours at the design pressure.

5.1.2.3 External Loadings

5.1.2.3.1 External Pressure

The Reactor Building will be designed for an external atmospheric pressure of 2.5 psi greater than that of the internal pressure.

5.1.2.3.2 Tornadoes

Structures will be designed to withstand short term tornado loadings including tornado generated missiles where such structures house systems and components whose failure could result in an inability to safely shut down and isolate the reactor. Structures that will be so designed include the following:

a. Reactor Building

b. Control, Relay, and Battery Rooms

c. Designated areas of Auxiliary Building

The tornado design requirements are:

a. Tangential wind velocity of 300 mph.

b. An external vacuum of 3 psig.

c. Missile equivalent to a utility pole 35 ft long, 14 inches in diameter, weighing 50 PCF and traveling at 150 mph.

d. Missile equivalent to a one-ton automobile traveling at 150 mph.

A 300 mph wind will be applied in accordance with standard wind design practice and utilizing applicable pressures, shape factors, and drag coefficients from ASCE Paper No. 3269, "Wind Forces on Structures." The vacuum of 3 psig is conservative considering that most measured pressure drops are in the order of magnitude of 1.5 psig.

Recognition is given to the fact that the steel superstructure of the Auxiliary Building and Spent Fuel Building could be damaged. All vital equipment and component in these areas will be located below the operating floor of the Auxiliary Building. The operating floor slab at elevation 329'-0' and all exposed portions of the Auxiliary Building below elevation 329'-0' will be designed for the loading conditions stated above. 6

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The structural concrete walls and base mat of the spent fuel pit will be designed to resist all tornado loads thereby ensuring no excessive loss of water due to structural failure.

5.1.2.3.3 Wind, Snow, or Ice

The Reactor Building will be designed to withstand a snow or ice load of 35 lbs/ft^2 and a wind velocity corresponding to a 100 year frequency wind. The wind velocity as a function of height and drag coefficients will be established on the basis of ASCE Paper No. 3269 (Wind Forces on Structures).

5.1.2.3.4 Ground Water and Floods

The top of the Reactor Building foundation mat will be approximately at elevation 280 ft. The ground water elevation will be approximately 280 ft. The foundation slab design will take into consideration ground water pressure. The plant site will be protected from any flood conditions. Fluctuations in the ground water due to flood and normal variation will be given due consideration in designing the walls and foundation slab. Refer to Sections 2.4, "Hydrology and Groundwater" and 2.5, "Geology."

5.1.2.3.5 Seismic Conditions

The site seismology and response spectra are described in the Seismology section of Appendix 2B.

The seismic design of the Reactor Building will be based on the response to a maximum horizontal component of ground acceleration of 0.06 g. In addition, the design will be checked to ensure no loss of function based on the response to a maximum horizontal component of ground acceleration of .12 g. The vertical component will be taken as 2/3 of the horizontal component and will be assumed to occur simultaneously with the horizontal component.

5.1.2.4 Codes

The Reactor Building will be designed under the following codes:

a. Regulations for Protection From Fire and Panic - Commonwealth of Pennsylvania
b. Building Code Requirements for Reinforced Concrete (ACI 318-63)
c. AISC Manual of Steel Construction
d. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessel, Sections VIII, Unfired Pressure Vessels, Section IX, Welding Qualifications (Applicable Portions).

For additional design criteria see Appendix 5A "Structural Design Bases."

5.1.2.5 Drawings

Figures 1-2 through 1-10 and Figure 5-1 are plans and elevations showing principal dimensions of the Reactor Building.

5.1.2.6 Penetrations

5.1.2.6.1 Piping, Duct, and Electrical Penetrations

Penetrations for process piping, ventilation ducts, fuel transfer, instrumentation lines, and electrical cables will be designed to withstand the following loads:

a. Incident pressure and temperature including response of the Reactor Building shell.
b. Pipe reactions based on thermal flexibility and seismic loads.
c. Expansion of containment shell under test conditions.
d. Pipe thrust loads to ensure the vapor barrier is not breeched due to a primary system pipe rupture which might result in subsequent failure of the primary coolant system.

All pipe penetrations will normally be anchored at the Reactor Building shell. Penetrations will be made of material which exhibits by test a transition temperature at least 30 F below the minimum service metal temperature.

Typical piping, duct, and electrical penetrations are shown in Figure 5-2. 6
All penetrations will be of the double barrier type. Where temperature changes require, the second barrier of piping penetrations will include expansion bellows. All penetrations are designed to provide a captive air space that can be pressurized to the Reactor Building design pressure for leak testing and accident conditions. All piping systems that may be open to reactor building atmosphere will be designed for adding a fluid block.

5.1.2.6.2 Personnel Access Locks

Two personnel access locks are provided, one of which penetrates the dished head of the equipment hatch. Each personnel hatch is a welded steel assembly with double doors equipped with double gaskets to provide an air space that can be pressurized to the Reactor Building design pressure for leak testing and accident conditions. The doors are interlocking to ensure that only one door is opened at a time. Remote indicating lights and annunciators will be provided in the Control Room to indicate if a door is open.

The personnel locks will be designed and fabricated so as to comply in all respects with the requirements of Section III of the ASME Boiler and Pressure Vessel Code for Class B vessels.

5.1.2.6.3 Equipment Access Hatch

An equipment hatch with an inside diameter of 22 ft 4 in. is provided to enable passage of large equipment and components into the Reactor Building during a plant shutdown. The items brought into the vessel include in part the reactor coolant pumps and motors, and reactor vessel O-rings.

5.1.2.7 Missile Protection Features

5.1.2.7.1 Reactor Building

Missile protection for the Reactor Building liner:

a. The building and liner will be protected from loss of function due to damage by such missiles as might be generated in a loss-of-coolant accident for break sizes up to and including the double-ended severance of a main coolant pipe.

b. The engineered safeguard components required to maintain containment integrity and to meet the site criteria of 10 CFR 100 will be protected against loss of function due to damage by the missiles defined below.

During the detailed plant design, the missile protection necessary to meet the above criteria will be developed and implemented using the following considerations:

a. The reactor coolant system will be surrounded by reinforced concrete and steel structures designed to withstand forces associated with double-ended rupture of a main coolant pipe and any missiles that may be generated.

b. The structural design of the missile shielding will take into account both static and impact loads and will be based upon a barrier cross section with energy absorption capacity at least 25 per cent greater than that required when considering a potential missile.

c. Components of the reactor coolant system will be examined to identify and classify potential missiles according to size, shape, kinetic energy, and driving force for purposes of analyzing their effects.

d. The types of missiles for which missile protection will be provided are:

   1. All valve stems up to and including the largest size to be used.

   2. All valve bonnets

   3. All instrument thimbles

   4. Various sizes of nuts and bolts

   5. Reactor vessel head bolts

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5.1.2.7.2 Main Steam Turbine Missiles

The turbine-generator supplier has made a study of various failures of turbine-generator rotating elements and has concluded that they are of two general types: (a) Failure of rotating components operating at or near normal operating speed. (b) Failure of components that control admission of steam to the turbine resulting in destructive shaft rotating speed.

a. Turbine Rotor Failure at or Near Operating Speed

Turbine and generator rotor failures at or near rated speed have resulted from the combination of severe strain concentrations in relatively brittle materials. To minimize the probability of brittle fracture in rotors, wheels, and shafts, new alloys, and processes have been developed and adopted. Careful control of chemistry and detailed heat treating cycles has greatly improved the mechanical properties of all of these components. Transition temperatures (the temperature at which the character of the fracture in the steel changes from brittle to ductile, often referred to as FATT) has been reduced on the low temperature wheel and rotor applications for nuclear units to well below start-up temperatures. Improved steel mill practices in vacuum pouring and alloy addition have resulted in forgings which are more uniform and defect free than ever before. More comprehensive vendor and manufacturer tests involving improved ultrasonic and magnetic particle testing techniques are better able to discover surface and internal defects than in the past. Laboratory investigation has revealed some of the basic relationships between structure strength, material strength, FATT, and defect size and location so that the reliability of the rotor as a structure has been significantly improved over the past few years.

New starting and loading instructions have been developed to reduce the severity of surface and bore thermal stress cycles incurred during service. The new practices include:

1. Better temperature sensors

2. Better control devices for acceleration and loading

3. Better guidance for station operators in the control speed, acceleration, and loading rates to minimize rotor stresses.

Progress in design, better material and quality control, more rigorous acceptance criteria, and improved machine operation have substantially reduced the likelihood of burst failures of turbine-generator rotors operation near rated speed.

b. Turbine-Generator Overspeed Failure

The improvements of rotor quality discussed above, while reducing the chance of failures at operating speed, tend to increase the hazard level associated with unlimited overspeed because of the greater missile energy associated with higher bursting speed. Thus, the turbine overspeed protection systems will be as follows:

1. Main and secondary steam inlets will have the following valves in series:

   a. Control valves - controlled by the speed governor and tripped closed by emergency governor and back-up overspeed trip

   b. Stop valves or trip throttle valve - actuated by the emergency governor and back-up overspeed trip. Emergency main stop valves of the stem sealed design have been used on General Electric steam turbines 10,000 kw and larger since 1948. There have been over 650 turbines shipped and put in service during this period and there has been no report of the main steam stop valve failing to close when required to protect the turbine. Impending sticking has been disclosed by means of the daily full closed test feature so that a planned shutdown could be made to make the necessary correction. This almost always involves the removal of the oxide layer which builds up on the stem and bushing and which would not occur on a low temperature nuclear application.

   c. Combined stop and intercept valves in cross around systems where required to control overspeed to the valves mentioned above. These are actuated by the speed governor, emergency and backup overspeed trips. These valves also include the daily testing features described above.

2. Uncontrolled Extraction lines to Feedwater Heaters

   If the energy stored in an uncontrolled extraction line is sufficient to cause dangerous overspeed, the positive closing non-return valves are provided, to be actuated by the emergency governor and back-up overspeed trip. These are designed for remote manual periodic tests to insure proper operation. The station piping, heater, and check valve system are reviewed during the design stages to make sure the entrained steam cannot overspeed the unit beyond safe limits.

   Special field tests are made of new components both to obtain design information and to confirm proper operation. Such tests include the capability of controls to prevent excessive overspeed on loss of load.

   Careful analysis of all past failures have led to design, inspection, and testing procedures to substantially eliminate destructive overspeed as a possible cause of failure in modern design units.

   The turbine-generator supplier has made a study of the major missiles that might escape the turbine-generator exciter housing as a result of a hypothetic failure. The last stage wheel is considered to have the worst combination of weight, size, and energy. The latest analysis indicates that the last stage wheel could fail at an overspeed of 169 percent of the rated speed. Properties and penetration into the Reactor Building for the last stage wheel missiles are shown in Table 13.2-1. (Supplement 3) The maximum penetration will occur with the 120 degree segment as shown in Figure 13.2-1. (Supplement 3) 6

A comparison of the penetration depths of the 120 degree segment at a 69 6
percent overspeed, as stated above, and the 134 degree segment at an 86 percent overspeed (as shown in section 5.1.2.7.2 of the PSAR) indicates that the difference in penetration depths is negligible and that the difference in percent of overspeed does not appreciably influence the penetration depth. For example, penetration in concrete is 12.8 in. at 69 percent overspeed as opposed to 13.2 in. at 86 percent overspeed.

Missile penetrations when equal to or less than one-half the shield thickness will not produce a breech of the shield due to a local material failure. In the case of the Reactor Building this means that penetrations less than 1.75 feet will not breech the vapor barrier.

5.1.2.8 Corrosion Peotection

The containment structure is protected against external corrosive influences 3
by the following means:

1. A water proof tendon access galle y at the base of the cylinder, as shown on Figure 5-1.
2. A retaining wall around the vessel which precludes contact of ground water with the shell.
3. A cover of concrete in excess of that for normal construction as exemplified by code requirements. (Refer to the answer to Question 7.12.1).
4. Use of a water-tight steel conduit for tendons with added precaution of a thicker walled, rigid confuit in the base mat and extending immediately above into the cylinder.
5. An inboard-oriented haunch which results in only nominal tensile stresses of the outer fibers; these stresses are within the capacity of concrete in tension.

The exposed surface of the liner will be given a protective coating of Carbonzinc #11 Gray, as manufactured by the Carboline Company. The outer surface of the steel will be in direct contact with the concrete, which provides adequate corrosion protection because of the alkaline properties of the concrete.

The tendons will be inserted in steel conduit embedded in the concrete which will provide an additional water barrier, as well as an electrical shield against stray currents. The inner surface of the steel conduit, as well as the tendons including anchorages, will be protected with NO-OX-ID "CM" Casing Filler, as manufactured by the Dearborn Chemical Division of W.R. Grace and Company. This material is composed essentially of a selected paraffin-base refined mineral oil, blended with a microcrystalline petroleum-derived base (petrolatum) which has a definite melting point and penetration range. Additives consisting of lanolin, and sodium petroleum sulphonates are incorporated as water-displacing, surface-active agents and corrosion inhibitors.

*Nav Docks P-51, Design Criteria - Structural Theory For Protective Construction, Bureau of Yards and Docks, Department of Navy, April 1951

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The proportion of oil to microcrystalline wax in the formulation is adjusted to give a pour or gelling point within the range of 110 to 120 F. The oil and wax are highly refined long-chain, saturated, paraffinic, petroleum derivatives, resistant to oxidation and chemical or physical degradation within the temperature ranges to which they will be exposed in this service. The lanolin is a polar substance which enhances inhibitor performance and wetting of the metal surface by the microwax blend. The petroleum sulphonate is a surface-active, water displacing corrosion inhibitor of long tested merit. The physical properties and tests performed on this material are more completely described in the Fourth Supplement to the Preliminary Facility Description and Safety Analysis Report for Brookwood Nuclear Station Unit No. 1 (Docket No. 50-244). The end anchorages will be covered with a metal container and provision made to control the humidity to a safe limit so that the temperature of the air within the enclosure is maintained above the dew point for all operating conditions. 3

The retaining wall and drainage system around the Reactor Building provides excellent protection for the liner and tendons against ground corrosion,and therefore no cathodic protection system will be provided. All metallic components including the liner plate and tendon conduit will be electrically connected to prevent stray current corrosion. The tendons will be enclosed in a metallic tube so as to isolate them from outside electrical influences.

Permanent reference electrode stations will be installed to facilitate measurements of structure potentials. These stations will consist of plastic pipe extending from the ground surface to the point at which the structure potential measurement is required. The plastic pipe will function as a salt bridge. Standard reference half cells placed into the pipe,down to the ground water level, will be used to make structure potential measurements.

5.1.2.9 Insulation

The Reactor Building liner plate will not be insulated. Insulation and/or cooling coils will be provided when required for hot line penetrations through the Reactor Building so as to limit concrete temperatures during normal operation to no greater than 150 F.

5.1.2.10 Shielding

The Reactor Building completely encloses the reactor coolant system and is designed to contain all radioactive material which might be released following a loss of coolant accident. The building provides adequate biological shielding for accident and normal operating conditions. The interior walls provide adequate shielding for limited access during normal operating conditions.

5.2 ISOLATION SYSTEM

5.2.1 DESIGN BASES

The general design basis governing isolation valve requirements is:

> Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss of isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the Reactor Building, and various types of isolation valves.

Reactor Building isolation occurs on a signal of approximately 4 psig in the Reactor Building. Valves that isolate penetrations that are directly open to the Reactor Building, such as the Reactor Building purge valves and sump drain valves, will also be closed on a high radiation signal.

The isolation system closes all fluid penetrations not required for operation of the engineered safeguards in order to prevent leakage of radioactive materials to the environment. Fluid penetrations serving engineered safeguards also meet this design basis.

All remotely operated Reactor Building isolation valves are provided with position limit indicators in the control room.

5.2.2 SYSTEM DESIGN

The fluid penetrations that require isolation after an accident may be classed as follows:

Type I. Each line connecting directly to the reactor coolant system has two Reactor Building isolation valves. One valve is external and the other is internal to the Reactor Building. These valves may be either a check valve and a remotely operated valve or two remotely operated valves depending on the direction of normal flow.

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Type II. Each line connecting directly to the Reactor Building atmosphere has two isolation valves. At least one valve is external and the other may be internal or external to the Reactor Building. These valves may be either a check valve and a remotely operated valve or two remotely operated valves depending on the direction of normal flow.

Type III. Each line not directly connected to the reactor coolant system or not open to the Reactor Building atmosphere has one valve, either a check valve or a remotely operated valve. This valve is located external to the Reactor Building.

Type IV. Lines that penetrate the Reactor Building and are connected to either the building or the reactor coolant system, but which are never opened during reactor operation, have provisions for locking in a closed position.

There are additional subdivisions in each of these major groups. The individual system flow diagrams show the manner in which each Reactor Building isolation valve arrangement fits into its respective system. For convenience, each different valve arrangement is shown in Table 5-3 and Figures 5-4 and 5-5 of this section. The symbols on these figures are identified on Figure 9-1. This table lists the mode of actuation, the type of valve, its normal position, and its position under Reactor Building isolation conditions. The specific system penetrations to which each of these arrangements is applied are also presented. Each valve will be tested periodically during normal operation or during shutdown conditions to insure its operability when needed.

The accident analysis for failure or malfunction of each valve is presented with the respective system evaluation of which that valve is a part, for example, chemical addition and sampling system, etc.

There is sufficient redundancy in the instrumentation circuits of the safeguards actuation system to minimize the possibility of inadvertent tripping of the isolation system. This redundancy and the instrumentation signals that trip the isolation system are discussed further in Section 7.

The system abbreviations in column three of Table 5-3 are defined as follows:

MU Makeup and Purification System
DH Decay Heat Removal System
RB Reactor Building Cooling System
SF Spent Fuel Cooling System
WD Waste Disposal System
CA Chemical Addition and Sampling System
RBS Reactor Building Spray System
IC Intermediate Cooling System
SW Nuclear Services Cooling Water System
RW River Water
SP Secondary Plant

5.3 VENTILATION SYSTEM

5.3.1 DESIGN BASES

5.3.1.1 Governing Conditions

The Reactor Building ventilation system performs the following functions:

a. Removes or adds sensible heat under normal conditions of operation to maintain the building at or below a predetermined maximum temperature and at or above a predetermined minimum temperature.

b. Removes sensible and latent heat under emergency accident conditions to maintain the building at or below predetermined maximum values of temperature and pressure.

c. Deleted 1

d. Filters all air.

e. Filters contaminated air through charcoal filters before discharging it to atmosphere.

f. Purges the Reactor Building with outside air whenever desired.

5.3.1.2 Sizing

To provide normal cooling in the Reactor Building, the ventilation system will be sized to control the interior air temperature to 110 F maximum in accessible areas during operation and 60 F minimum during shutdown. Air from the system will be distributed over and around all heat producing equipment. The anticipated total normal cooling load is 4.3×10^6 Btu/hr. 6

To provide for emergency accident cooling in the Reactor Building, the ventilation system will be additionally selected and sized to meet the abnormal conditions of temperature, pressure, and latent heat which will be imposed upon it is such an event. It is anticipated that these conditions will be 281 F, 55 psig, and 240×10^6 Btu/hr. 1

Deleted

To provide a flow of outside air through the Reactor Building, the purge components of the ventilation system will be selected and sized for a purge rate of 50,000 cfm. All air entering the Reactor Building will be filtered and heated as required. All air leaving will be passed through HEPA and charcoal filters before being discharged to the atmosphere through the vent. 1

The cooling which is required during normal operation will be obtained from water in a closed circuit which is cooled in an evaporative-type industrial cooler to a temperature of 85 F maximum. 1

The cooling which is required during an emergency condition will be obtained from water from the nuclear services cooling water system which will be available at a maximum temperature of 95 F. 1

The normal cooling water circuit will be entirely independent of the emergency cooling water circuit.

5.3.2 SYSTEM DESIGN

Figure 5-5 is a flow diagram of the ventilation system for the Reactor Building showing the cooling and purging components.

The cooling components consist of three fan assemblies, each of which will contain a normal coil, emergency coil, filter, direct-driven fan, and suitable casing. To insure adequate air distribution to the major head loads, auxiliary fans and ducts will be employed. 6

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The purge exhaust air component consists of two fan assemblies each of which will contain a HEPA filter, a charcoal filter, fan, and casing. Each assembly will handle one half the total air quantity (25,000 cfm each).

The purge exhaust air component consists of two fan assemblies each of which will contain a HEPA filter, a charcoal filter, fan, ard casing. Each assembly will handle one half the total air quantity (25,000 cfm each).

All of the purge components of the ventilation system, except the interior ducts and isolation valves as shown on Figure 5-5, will be located outside the Reactor Building.

The purge discharge to the station vent will be monitored and alarmed to prevent release exceeding acceptable limits.

5.3.2.1 Isolation Valves

As the cooling component of the ventilation system is contained completely within the Reactor Building, it will not include provisions for any isolation valves other than on cooling water lines.

The purge component of the ventilation system will be provided with double automatic isolation valves (or dampers) in both the supply and discharge ducts. These valves will be normally closed and will be opened only for the purging operation. They will be electrically actuated inside the Reactor Building and pneumatically actuated outside.

The isolation signal and controls are discussed in section 5.2. The closure times and sequence will be developed during detailed design and safety analyses. Operability testing of the isolation valves is accomplished each time the purge system is put into operation.

5.4 LEAKAGE MONITORING SYSTEM

Provision for monitoring of leakage through the penetrations and penetration sleeve to liner welds will be provided.

The normal barrier to leakage from the Reactor Building will be a steel liner three-eighth of an inch thick in the cylinder and dome and one-quarter of an inch thick in the base. All penetrations through the containment shell for pipes, ducts, electrical conductors, and access will be welded to the liner plate before the concrete is placed. Insofar as possible the penetrations and reinforcing plates will be shop welded assemblies. All penetrations in the Reactor Building will be of the double barrier type and will be equipped for continuous leakage monitoring with means provided to isolate and locate a leaking penetration. All penetrations as shown typically in Figure 5-2 will be designed, fabricated, and tested so as to ensure leak tightness. All penetration sleeve to liner plate welds will be covered by a test channel to permit verification of leak tightness.

The liner plate will be anchored to the concrete shell so as to ensure elastic stability under all loading conditions and composite action between the liner and the concrete. Elastic stability will be ascertained by analyzing the liner as a flat plate between supports subjected to the biaxial stresses. Should liner stresses under the factored load conditions exceed yielding compression, the analysis will assume plastic behavior at a stress of 1.2 times the minimum guaranteed yield stress. The Reactor Building will be tested for leak tightness under conditions more severe for possible liner leakage than under the conditions of the hypothetical accident and other simultaneously occurring loads. Surveillance of the quality of all materials and workmanship will be maintained so as to ensure that the containment shell including the liner, penetrations, and reinforced concrete does meet the intent of the design in all respects.

The details of a program to demonstrate integrity of the structure and the leak tightness of the vapor barrier are described in Section 5.6 and summarized as follows:

a. The test to demonstrate structural integrity will be at 115 per cent of design pressure. This test is established to duplicate, insofar as possible, the incident loads plus the most severe effects of the design earthquake.

b. The integrated leak rate test will be at design pressure and will demonstrate that the leak rate is no greater than 0.2 per cent of the contained volume in 24 hours at the design pressure. The conditions for this test will impose greater tensile or lower compressive membrane stresses on the liner than would exist under accident conditions, thereby ensuring that the test is performed under the most severe condition insofar as leakage is concerned. The pressure load in combination with the thermal load results in the lesser probability of leakage. Leak rate measurements will also be made at one or more lower pressures (i.e. below design pressure) to demonstrate leak tightness and to establish a reference for future testing.

Under all normal operating conditions and even including the most severe loss-of-coolant accident, there is virtually no possibility that leakage would occur that would be a hazard to the public or to the station personnel. The complete safety analyses are presented in Section 14.

As described in Section 5.1.2.6, the personnel access hatches will be interlocked so as to ensure one barrier will always remain intact and that indicators and annunciators will be provided in the Control Room to indicate when a door is open. The permanent equipment access hatch cannot be opened except by extensive deliberate action which cannot be taken except at plant shut down.

All liner seams will be tested individually to demonstrate their leak tightness. Where the liner will be inaccessible for future inspection (i.e. where it will be totally encased in concrete), test channels will be provided on all weld seams. These channels will ensure leak tightness of the plate to plate weld and the channel to plate weld and will be segmented so as to permit pressurization at specified locations. On liner plate weld seams where test channels are not provided, testing will consist of the use of soap film and a vacuum box. The penetration sleeve to liner plate welds are also covered with test channels to permit testing similar to that for inaccessible weld seams. Each shop assembled unit including single and multiple penetration units will be shop tested to demonstrate leak tightness. More complete details of testing procedures on the liner and components are included in Appendix 5E.

Where the liner abuts concrete, it will be in an environment which will minimize any possible corrosion. As described in Section 5.1.2.8 cathodic protection will be provided to minimize any ground corrosion influences. Where the liner is exposed, it will be protected with a coating similar to Carbo Zinc as manufactured by the Carboline Corporation. The concrete shell will protect the liner from weather influences and potential externally generated missiles and insulate the liner plate from low temperatures. The liner surface on the interior of the vessel will be protected from internally generated missiles by concrete and in instances a combination of concrete and steel shields.

Because the vapor barrier is protected in this manner, and once the adequacy of the liner has been initially established, there is no reason to anticipate deterioration of the liner, which would jeopardize its effectiveness as a vapor barrier. Nevertheless periodic inspections will be made of the exterior of the Reactor Building and of interior spaces which are accessible during full power operation.

A significant number of steel lined, reinforced concrete containment vessels are being constructed presently; these will provide a considerable background of experience prior to the construction of this vessel. Full advantage of this knowledge will be taken in all phases of design, fabrication, installation, inspection, testing, and operation. Appropriate action will also be taken to minimize the possibility of reoccurrence of further leakage, include such redesign as might be necessary.

5.5 SYSTEM DESIGN EVALUATION

The Reactor Building, including extensions of the containment boundary as described in Sections 5.2 and 5.3, will with the functioning of the additional engineered safeguards prevent an uncontrolled release of radioactivity to either the plant site or the surrounding areas during normal operation or any accident conditions up to the most severe hypothetical accident. Containment integrity is maintained when there is nuclear fuel in the core coincident with either a reactor coolant system pressurized above 300 psig or a reactor coolant temperature above 200 F.

5.6 TESTS AND INSPECTION

5.6.1 PREOPERATIONAL TESTING AND INSPECTION

5.6.1.1 During Construction

Test, code, and cleanliness requirements will accompany each specification or purchase order for materials and equipment. Hydrostatic, leak, metallurgical, electrical, and other tests to be performed by the supplying manufacturers will be enumerated in the specifications together with the requirements, if any, for test witnessing by an inspector. Fabrication and cleanliness standards, including final cleaning and sealing, will also be described together with shipping procedures. Standards and tests will be specified in accordance with applicable regulations, recognized technical society codes and current industrial practices.

Inspection will be performed in the shops of vendors and subcontractors as necessary to verify compliance with specifications.

5.6.1.1.1 Concrete

Testing of concrete materials and concrete as placed is described in Appendix 5D, "Quality Control." An experienced full-time concrete inspector will continuously check concrete batching and placing operations.

5.6.1.1.2 Prestressing

Testing and inspection of all prestressing materials and special installation equipment is described in Appendix 5D, "Quality Control." Full-time supervision of the prestressing operations will be provided by an inspector experienced in prestressing as well as by the aforementioned concrete inspector.

5.6.1.1.3 Reinforcing Steel

Testing and inspection of reinforcing steel is described in Appendix 5D, "Quality Control." The concrete inspector will check the condition and placement of the bars in the forms for compliance with drawings and specifications, including welded splices.

5.6.1.1.4 Liner Plate

Testing and inspection of the liner plate is described in Appendix 5E, "Liner Plate Technical Specifications."

5.6.1.2 <u>Structural Test</u>

Following completion of construction and prior to the initial fuel loading, the Reactor Building will be pressureized at 115 per cent of the design pressure for one hour to establish the structural integrity of the building. The response of the building will be compared with the calculated behavior to confirm the design by means of instrumentation described in Appendix 5F, "Instrumentation."

5.6.1.3 <u>Initial Integrated Leakage Rate Test</u>

The purpose of the integrated leakage rate test is to measure the percentage by weight of air which can leak out of the reactor building per day at the design pressure of 55 psig. The specified design leakage of 0.2 per cent or less per day will be measured using the Absolute Method. During the test, air pressure, water vapor pressure, and air temperature will be measured using high accuracy instruments. The reactor building ventilation system will be used continuously throughout the test to achieve complete air mixing and control of air temperature. Duration of the test will be a minimum of 24 hours.

The leakage rate test will be repeated at reduced pressures of approximately 50 per cent and 25 per cent of the design pressure to establish the variation of leakage rate with pressure. The results of these reduced pressure tests may allow a basis for carrying out future integrated leak rate tests at pressures below the design pressure.

5.6.2 POSTOPERATIONAL LEAK MONITORING

5.6.2.1 <u>Leakage Monitoring</u>

Periodic integrated leak rate tests of the Reactor Building will be carried out to verify its continued leak tight integrity. The postoperational leak rate test will be conducted at a pressure established from the program of initial leakage rate tests. The acceptable leakage rate at this pressure will be determined from the measured variation of leakage rate with pressure.

TABLE 5-1

MISSILE ENERGIES

DELETED

TABLE 5-2

MISSILE PENETRATIONS

DELETED

6

Table 5-3
Reactor Building Isolation Valve Information

Penetration No.	Service	System	Flow Direction	Valve Arrgt.	Location Referred to R.B.	Valve Type	Line Size, in.	Method of Actuation	Signal	Normal Valve Position	Position Indication	Post-Accident Position
437 (a)	Pressurizer and Reactor Coolant Sample Lines	CA	Out	5	Inside	Gate	3/4	EMO*	ES	Closed	Yes	Closed
(b)						Gate	3/4	EMO	ES	Closed	Yes	Closed
					Outside	Gate	3/4	Air	ES	Closed	Yes	Closed
319	Intermediate Cooling Water Supply Line	IC	In	4	Inside	Check	6			Open	No	Closed
					Outside	Gate	6	--	--	Open	No	Closed
320	Intermediate Cooling Water Outlet Line	IC	Out	3	Inside	Gate	6	EMO	ES	Open	Yes	Closed
					Outside	Gate	6	Air	ES	Open	Yes	Closed
					Outside	Gate	4	Air	ES	Closed	Yes	Closed
305	Let Down Line to Purification Demineralizers	MU	Out	5	Inside	Gate	1-1/2	EMO	ES	Open	Yes	Closed
						Gate	1-1/2	EMO	ES	Closed	Yes	Closed
					Outside	Gate	2-1/2	Air	ES	Open	Yes	Closed
306	Reactor Coolant Pump Seal Return Line	MU	Out	3	Inside	Gate	3	EMO	ES	Open	Yes	Closed
					Outside	Gate	3	Air	ES	Open	Yes	Closed
307	Reactor Coolant Pump Seal Water Supply	MU	In	1	Inside	Stop Ck.	1	--	--	Open	No	Closed
					Outside	Globe	4	Air	ES	Throttled	Yes	Closed**
						Globe	4	Air	ES	Closed	Yes	Closed**
308	Normal Makeup to the Reactor Coolant System	MU	In	2	Inside	Check	4	--	--	--	No	Open
					Outside	Globe	1-1/2	Air	--	Throttled	Yes	Open
					Outside	Globe	1-1/2	Manual	--	Closed	Yes	Open
					Outside	Gate	4	Air	ES	Closed	Yes	Open
[illegible]	High Pressure Injection Line	MU	In	7	Inside	Check	4	--	--	--	No	Open
					Outside	Gate	4	Air	ES	Closed	Yes	Open
314 439	Fuel Transfer Tubes	SF	In/Out	8	Inside	Special Closure	30	--	--	Closed	--	Closed
					Outside	Gate	30	Manual	--	Closed	No	Closed
317	Reactor Bldg Spray Inlet Line	RBS	In	7	Inside	Check	8	--	--	--	No	Open
					Outside	Gate	8	Air	ES	Closed	Yes	Open

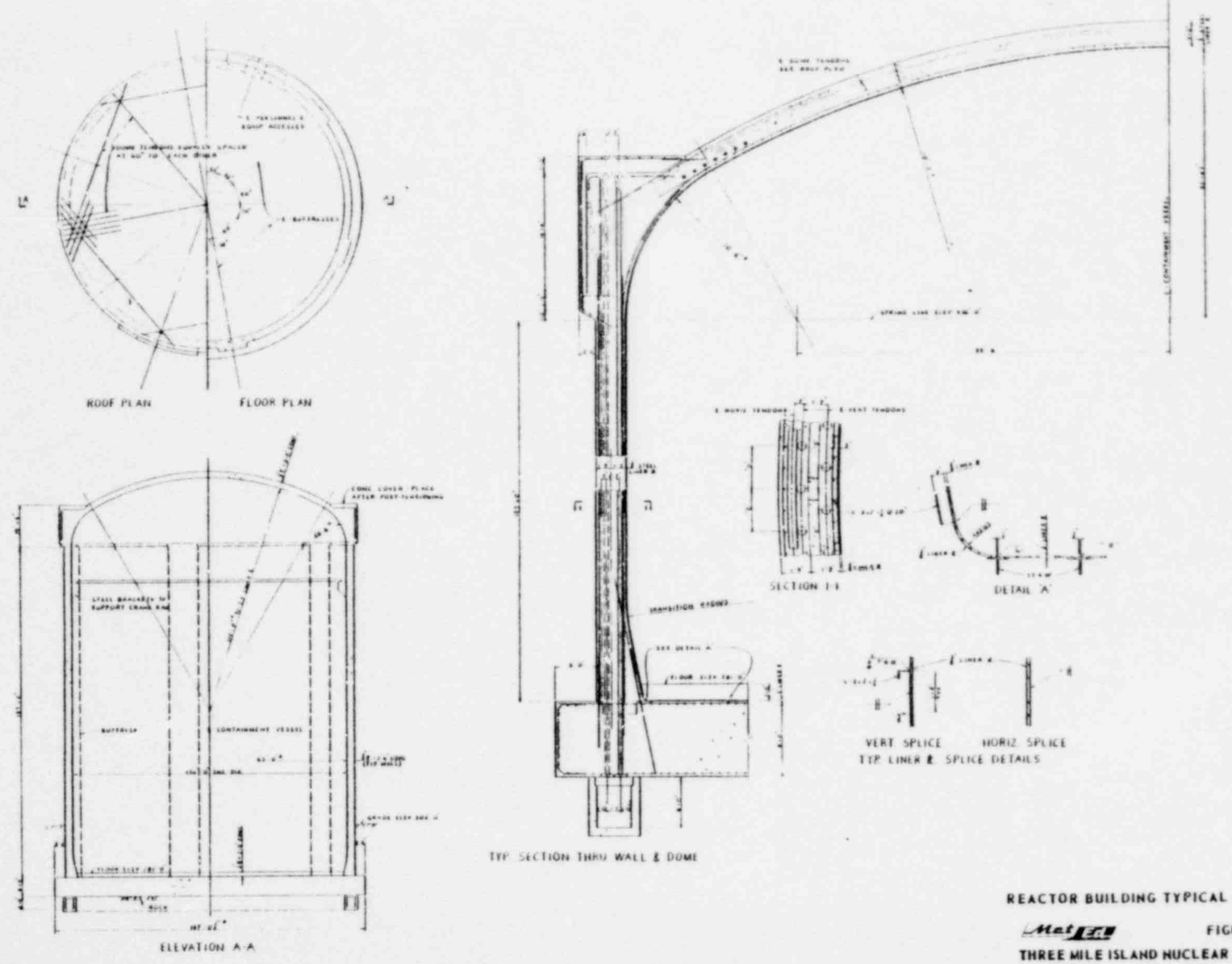
*All valves with electric motor operators are also equipped with hand wheels.

**Postaccident reactor coolant system pressure causes valve to close.

Table 5-3 (Cont'd)

Pene-tration No.	Service	System	Flow Direc-tion	Valve Arrgt.	Location Referred to R.B.	Valve Type	Line Size, in.	Method of Actuation	Signal	Normal Valve Position	Position Indication	Post-Accident Position
318	Reactor Bldg Spray Inlet Line	RBS	In	7	Inside	Check	8	--	--	Closed	No	Open
					Outside	Gate	8	Air	ES	Closed	Yes	Open
312	Decay Heat Coolant Return	DH	In	6	Inside	Check	12	--	--	Closed	No	Open
					Outside	Gate	12	Air	ES	Closed	Yes	Open
313	Decay Heat Coolant Return	DH	In	6	Inside	Check	12	--	--	Closed	No	Open
					Outside	Gate	12	Air	ES	Closed	Yes	Open
310	Decay Heat Coolant Let-Down	DH	Out	9	Inside	Gate Gate(2)	10 10	EMO EMO	-- Remote Manual	Closed Closed	No Yes	Closed Closed
					Outside	Gate	10	EMO	Remote Manual	Closed	Yes	Closed
352	Reactor Bldg Sump Recircula-tion Lines	DH	Out	10	Outside	Gate	14	EMO	Remote Manual	Closed	Yes	Open
353	Reactor Bldg Sump Recircula-tion Lines	DH	Out	10	Outside	Gate	14	EMO	Remote Manual	Closed	Yes	Open
337	Reactor Coolant Drain Tank	WD	Out	15	Outside	Gate	2	Air	ES	Closed	Yes	Closed
208	Reactor Bldg Inlet Purge Line	-	In	11	Inside	Butterfly	48	EMO	ES	Closed	Yes	Closed
					Outside	Butterfly	48	Air	ES	Closed	Yes	Closed
325	Reactor Bldg Outlet Purge Line	-	Out	12	Inside	Butterfly	48	EMO	ES	Closed	Yes	Closed
					Outside	Butterfly	48	Air	ES	Closed	Yes	Closed
332 333	Reactor Coolant Pump Motors and Lub Oil Coolers	SW	In and Out	13	Outside	Check	6	-	-	Open	No	Closed
					Outside	Gate	6	EMO	ES	Open	Yes	Closed
326	Reactor Bldg Normal Air Coolers	-	In and Out	13	Outside	Stop Check	6	Manual		Open	No	Closed
327					Outside	Gate	6	Air	ES	Open	Yes	Closed
202, 328, 329 203, 330, 331	Reactor Bldg Emergency Air Coolers	SW	In and Out	13	Outside	Stop Check	8	Manual	-	Closed	No	Open
					Outside	Gate	8	Air	ES	Closed	Yes	Open
101 103, 201	Feedwater and Steam Lines	-	In and Out	14	Outside	Swing Check	20	-	-	Open	No	Closed
		-			Outside	Stop Check	24	EMO	ES	Open	Yes	Closed
401 104, 105	Feedwater and Steam Lines	-	In and Out	14	Outside	Swing Check	20	-	-	Open	No	Closed
		-			Outside	Stop Check	24	EMO	ES	Open	Yes	Closed

6

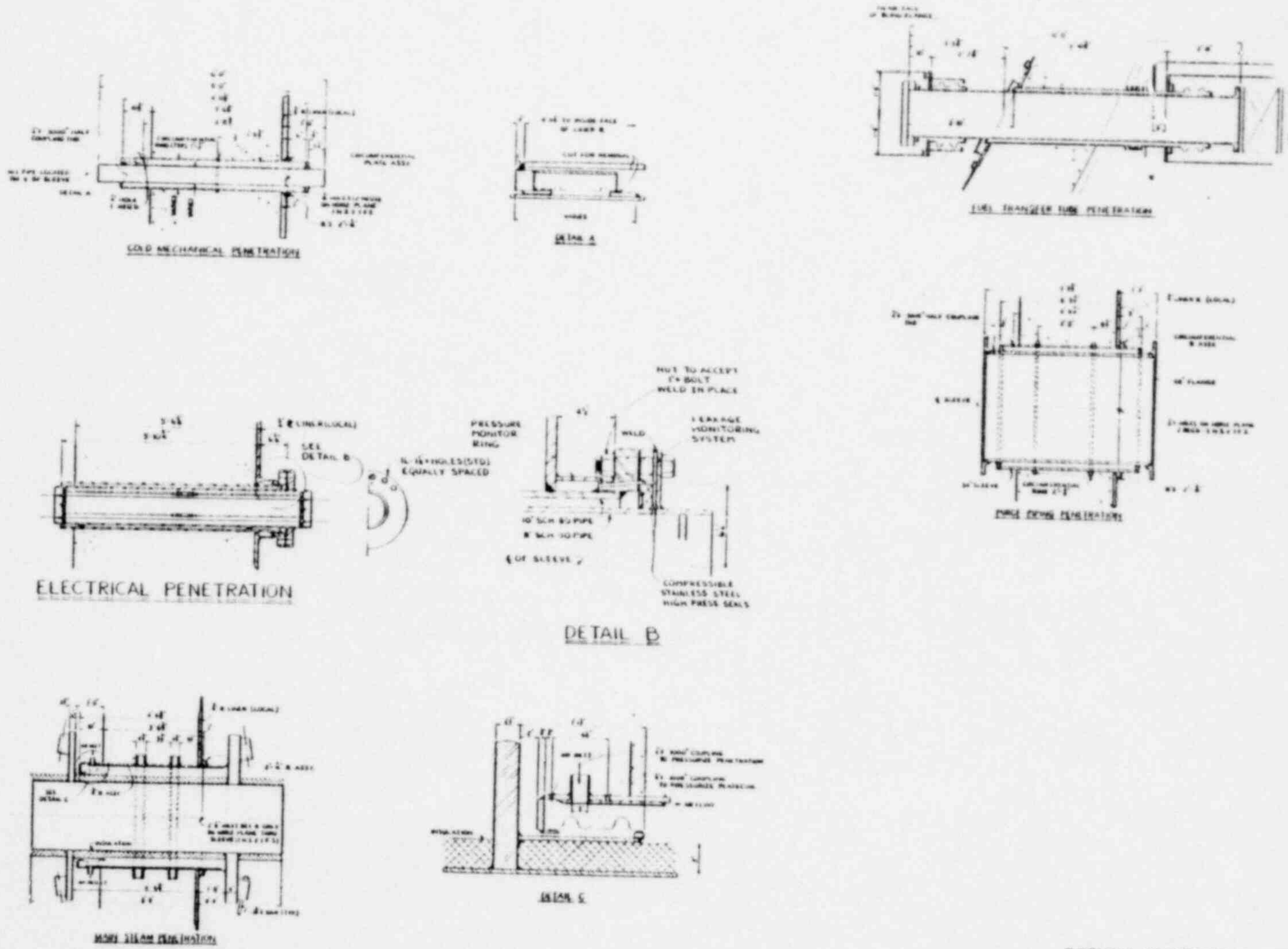


REACTOR BUILDING TYPICAL DETAILS

Met Ed FIGURE 5-1

THREE MILE ISLAND NUCLEAR STATION

0001 034



TYPICAL ELECTRICAL AND PIPING PENETRATIONS

Met Ed

FIGURE 5-2

THREE MILE ISLAND NUCLEAR STATION

AMEND. 6 (1-8-68)

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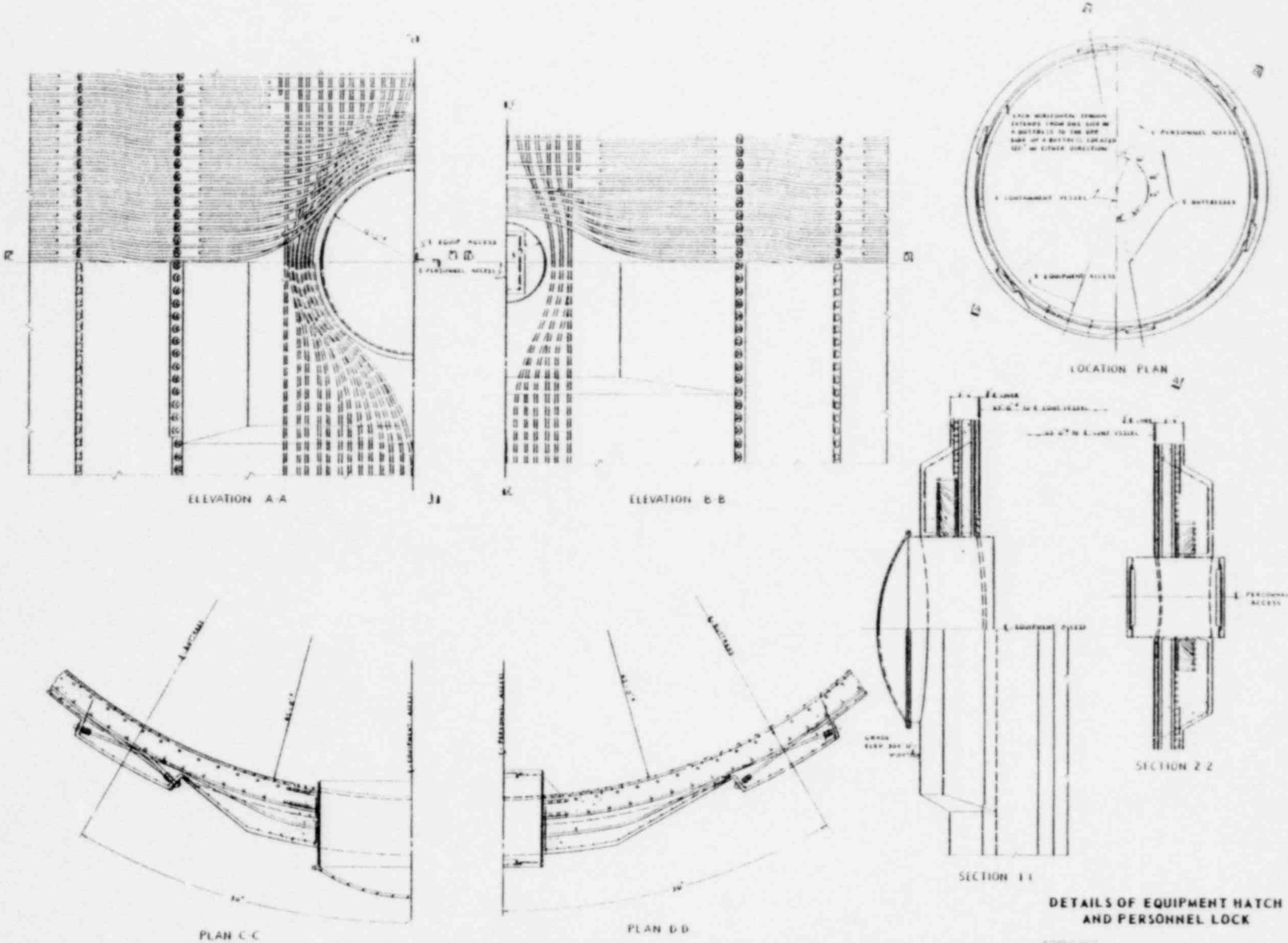
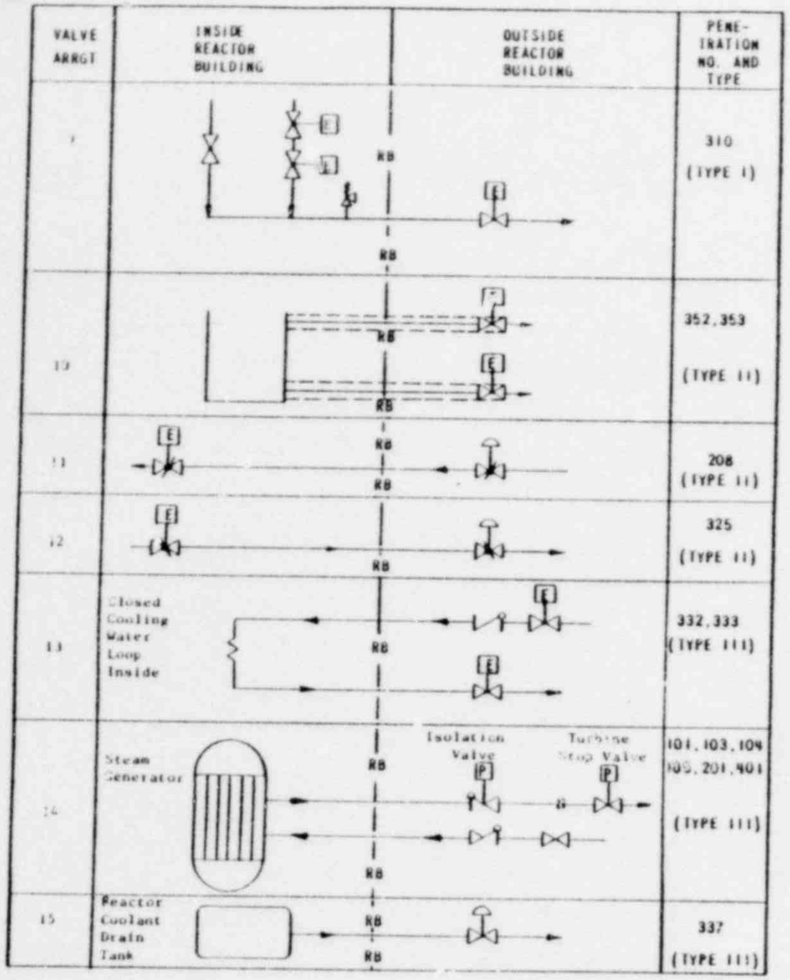
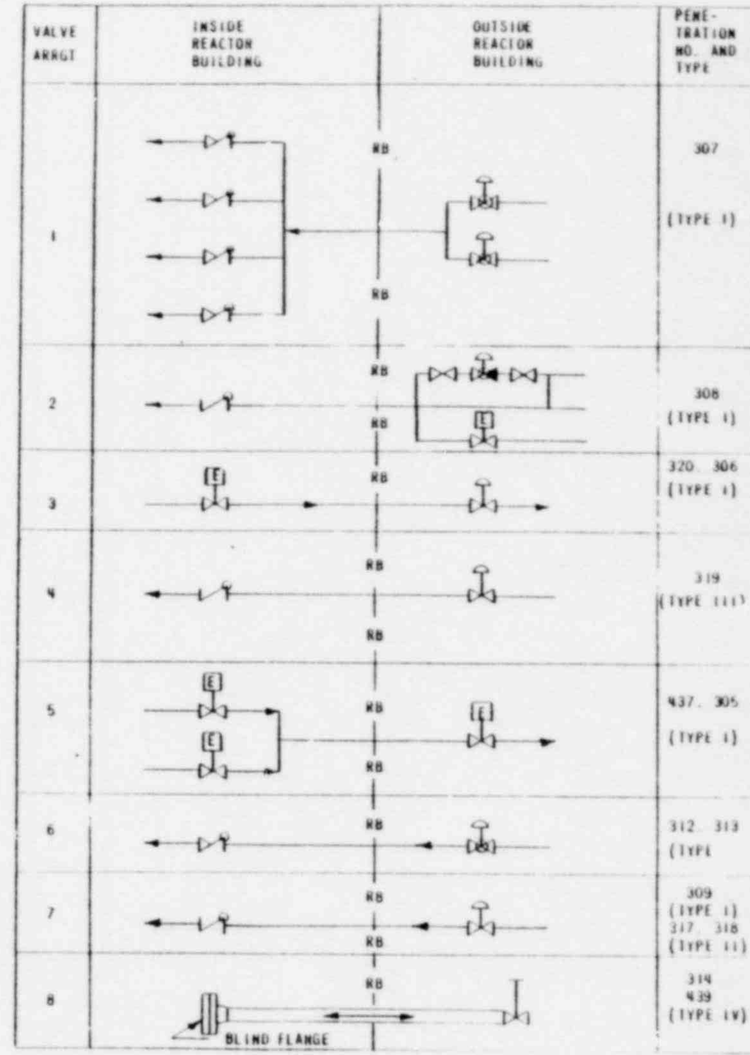


DETAILS OF EQUIPMENT HATCH
AND PERSONNEL LOCK

Met Ed FIGURE 5-3

THREE MILE ISLAND NUCLEAR STATION





REACTOR BUILDING ISOLATION VALVE ARRANGEMENT

Met Ed FIGURE 5-4

THREE MILE ISLAND NUCLEAR STATION

AMEND. 6 (1-8-68)

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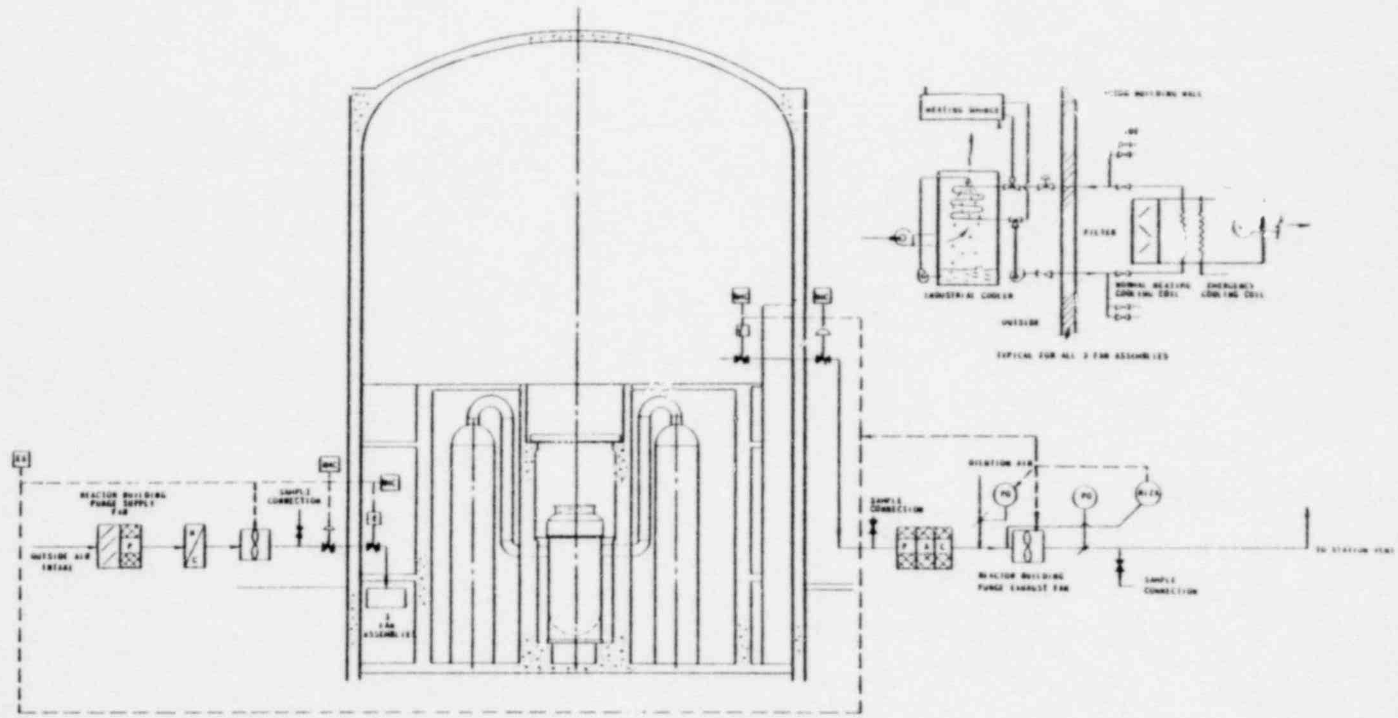


NOTE:
FOR LEGEND NOMENCLATURE SEE FIGURE [illegible]

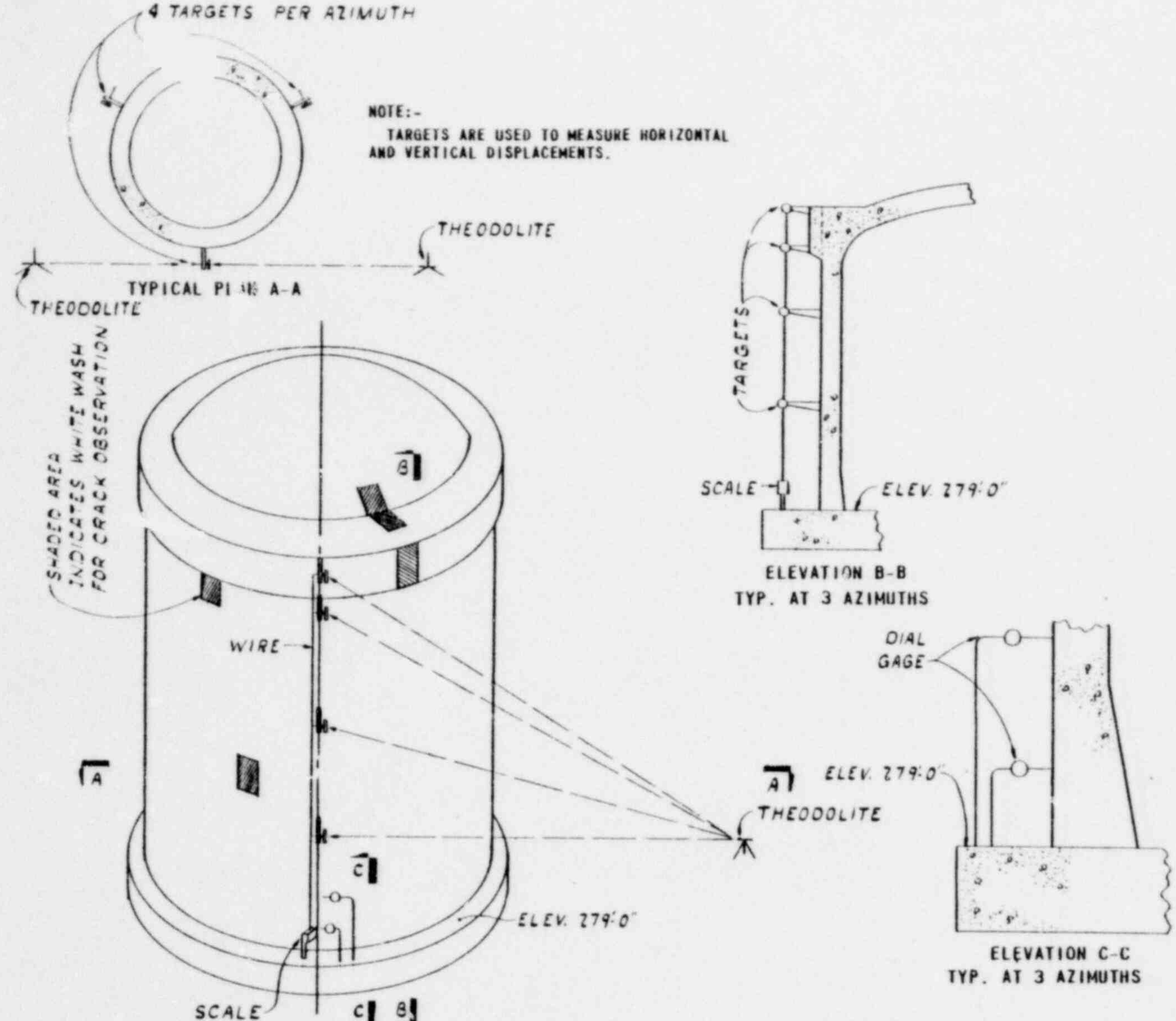
REACTOR BUILDING ISOLATION VALVE ARRANGEMENT

FIGURE 5-5

THREE MILE ISLAND NUCLEAR STATION

AMEND. 2 (10-2-67)

0001 038



1000 039

REACTOR BUILDING INSTRUMENTATION

Met Ed FIGURE 5-6

THREE MILE ISLAND NUCLEAR STATION

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(At rear of Section)

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6 ENGINEERED SAFEGUARDS

Engineered safeguards are provided to fulfill three functions in the unlikely event of a serious loss-of-coolant accident:

a. Protect the fuel cladding.

b. Insure reactor building integrity.

c. Remove fission products from the reactor building atmosphere and reduce the driving force for building leakage.

Emergency injection of coolant to the reactor coolant system satisfies the first function above, while building atmosphere cooling and washing satisfy the latter two functions. Each of these operations is performed by two or more systems which, in addition, employ multiple components to insure operability. All equipment requiring electrical power for operation is supplied by the emergency electrical power sources as described in 8.2.3.

The engineered safeguards include a core flooding system, high pressure injection equipment, the decay heat removal system, the reactor building cooling system, and the reactor building spray system. Figures 6-1 and 6-5 show the operation of these systems in the engineered safeguards mode, together with associated instrumentation and piping.

Applicable codes and standards for design, fabrication, and testing of components used as safeguards are listed in the introduction to Section 9, and seismic requirements are given in Section 2. The safety analysis presented in Section 14 demonstrates the performance of installed equipment in relation to functional objectives with assumed failures.

The engineered safeguards functions noted above are accomplished with the postaccident use of equipment serving normal functions. The design approach is based on the belief that regular use of equipment provides the best possible means for monitoring equipment availability and conditions. Because some of the equipment used serves a normal function, the need for periodic testing is minimized. In cases where the equipment is used for emergencies only, the systems have been designed to permit meaningful periodic tests. Additional descriptive information and design details on equipment used for normal operation are presented in Section 9. This Section 6 will present design bases for safeguards protection, equipment operational descriptions, design evaluations of equipment, failure analysis, and a preliminary operational testing program for systems used as engineered safeguards.

6.1 EMERGENCY INJECTION

6.1.1 DESIGN BASES

The principal design basis for the emergency injection is as follows:

Emergency core injection is provided to prevent clad melting for the entire spectrum of reactor coolant system failures ranging from the smallest leak to the complete severance of the largest reactor coolant pipe.

High pressure injection is provided to prevent uncovering of the core for small coolant piping leaks at high pressure and to delay uncovering of the core for intermediate-sized leaks. The core flooding system and the decay heat removal system (which provides low pressure injection) are provided to recover the core at intermediate-to-low pressures so as to maintain core integrity during leaks ranging from intermediate to the largest size. This equipment has been conservatively sized to limit the temperature transient to a clad temperature of 2,300 F or less. 1

6.1.2 DESCRIPTION

Figure 6-1 is the schematic flow diagram for the emergency injection and associated instrumentation.

Emergency injection fluid, pumped to the reactor coolant system during safeguards operations, is supplied in each case from the borated water storage tank. This tank contains the volume of borated water necessary to fill the fuel transfer canal during refueling operations and is connected to the injection pump suction headers by two lines. Additional 6 coolant for emergency injection supply is contained in core flooding tanks which inject without fluid pumping as described later in this section.

Emergency injection into the reactor coolant system will be initiated in the event of an abnormally low reactor coolant system pressure of 1,800 psi during power operation. The low pressure signals will automatically increase high pressure injection flow to the reactor coolant system with the following changes in the operating mode of the makeup and purification system described in Section 9: (a) the standby makeup pumps will start 6 and come on the line, (b) the stop-check valves in each injection supply line to the makeup and decay heat pumps will open, and (c) the injection valve in each of four injection lines will open. Emergency high pressure injection will continue until reactor coolant system pressure has dropped to the point where core flooding tanks begin emergency injection. The flow characteristic curves for each makeup pump are given in Figure 6-2

The core flooding system is composed of two flooding tanks, each directly connected to a reactor vessel nozzle by a line containing two check valves and one isolation valve; the system provides for automatic flooding injection with initiation of flow when the reactor coolant system pressure reaches approximately 585 psi. This injection provision does not require any electrical power, automatic switching, or operator action to insure supply of emergency coolant to the reactor vessel. Operator action is required only during reactor cooldown, at which time the isolation valves in the core flooding lines are closed to contain the contents of the core flooding tanks. The combined coolant content of the two flooding tanks is sufficient to recover the core hot spot assuming no liquid is contained in the reactor vessel, while the gas overpressure and flooding line sizes are sufficient to insure core reflooding within approximately 25 sec after the largest pipe rupture has occurred. 1

The decay heat removal system (described in Section 9) is normally maintained on standby during power operation and provides supplemental core flooding flow through the two core flooding lines after the reactor coolant system pressure reaches 135 psi. Emergency operation of this system will be initiated by a reactor coolant system pressure of 200 psi during any accident. The flow characteristics of each decay heat pump for injection are shown in Figure 6-3; these pumps are designed to deliver 3,000 gpm flow into the reactor vessel at a vessel pressure of 100 psi.

Low pressure injection, with supply from the borated water storage tank, using the decay heat pumps will continue until a low level signal is received from the tank (39 min at a combined low pressure injection and reactor building spray flow of 9,000 gpm). At this time, the operator will open the valves 6
controlling suction from the reactor building sump, and recirculation of coolant from the sump to the reactor vessel will begin. The decay heat coolers will cool the recirculated flow, thus removing heat from the reactor building fluid and preventing further building accumulation of decay heat generated by the core.

The decay heat removal pumps are located at an elevation below the reactor building sump with dual suction lines routed outside the reactor building to the pumps. In the event one suction line is unavailable for recirculation, the lines have been sized so that one line will be capable of handling the total potential recirculation flow of one 3,000-gpm decay heat removal pump, and one 1,500-gpm reactor building spray pump. The NPSH available has been conservatively calculated to be greater than the NPSH requirement of the decay heat removal pumps, and the reactor building spray pumps. 1 6

The heat transfer capability of each decay heat cooler as a function of recirculated water temperature is illustrated in Figure 6-4. The heat transfer capability at the saturation temperature corresponding to reactor building pressure is in excess of the heat generation rate of the core following storage tank injection.

Design data for core flooding system components are given in Table 6-1. Design data for other emergency injection components are given in Section 9 except for those shown in Figures 6-2, 6-3 and 6-4.

6.1.3 DESIGN EVALUATION

In establishing the required components for the emergency injection the following factors were considered:

a. The probability of a major reactor coolant system failure is very low.

b. The fraction of a given component lifetime for which the component is unavailable because of maintenance is estimated to be a small part of lifetime. On this basis, it is estimated that the

0001 046

probability of a major reactor coolant system accident occurring while a protective component is out for maintenance is two orders of magnitude below the low basic accident probability.

c. The equipment downtime for maintenance is a well-operated station often can be scheduled during reactor shutdown periods. When maintenance of an engineered safeguard component is required during operation, the periodic test frequency of the remaining equipment can be increased to insure availability.

d. Where the systems are designed to operate normally or where meaningful periodic tests can be performed, there is also a low probability that the required emergency action would not be performed when needed. That is, equipment reliability is improved by using it for other than emergency functions.

e. (DELETED) 6

f. Three makeup pumps are installed. One makeup pump is normally operating, and one pump can be down for maintenance. One pump is required for engineered safeguards. 1 6

Table 6-1
Core Flooding System Performance and Equipment Data

Core Flooding Tanks (*)	
Number	2
Design Pressure, psig	700
Normal Pressure, psig	600
Design Temperature, F	300
Operation Temperature, F	110
Total Volume, ft^3	1,410
Normal Water Volume, ft^3	940
Material of Construction	Carbon Steel-lined
Check Valves	
Number per Flooding Line	2
Size, in.	14
Material	SS
Design Pressure, psig	2,500
Design Temperature, F	650
Isolation Valves	
Number per Flooding Line	1
Size, in.	14
Material	SS
Design Pressure, psig	2,500
Design Temperature, F	650

(*) Designed to ASME Section III, Class C. 1

Table 6-1 (Cont'd)

Piping

Number of Flooding Lines	2
Size, in.	14
Material	SS
Design Pressure, psig	2,500
Design Temperature, F	650

1

6.1.3.1 Failure Analysis

The single failure analysis presented in Table 6-2 is based on the assumption that a major loss-of-coolant accident had occurred. It was then assumed that an additional malfunction or failure occurred either in the process of actuating the emergency injection systems or as a secondary accident effect. All credible failures were analyzed. For example, the analysis includes malfunctions or failures such as electrical circuit or motor failures, stuck check valves, etc. It was considered incredible that valves would change to the opposite position by accident if they were in the required position when the accident occurred. In general, failures of the type assumed in this analysis should be unlikely because a program of periodic testing and service rotation of standby equipment will be incorporated in the Station operating procedures.

The single failure analysis (Table 6-2) and the dynamic postaccident performance analysis (Section 14) of the engineered safeguards considered capacity reduction as a result of equipment being out for maintenance or as a result of a failure to start or operate properly. This amounts to adding another factor of conservatism to the analyses because good operating practice requires repairing equipment as quickly as possible. Station maintenance activities will be scheduled so that the required capacity of the engineered safeguards systems will always be available in the event of an accident. 1

The adequacy of equipment sizes is demonstrated by the postaccident performance analysis described in Section 14, which also discusses the consequences of achieving less than the maximum injection flows. There is sufficient redundancy in the emergency injection systems to preclude the possibility of any single credible failure leading to core melting.

6.1.3.2 Emergency Injection Response

The emergency high-pressure injection valves are designed to open within 10 sec. One makeup pump is normally in operation, and the pipe lines are filled with coolant. The four high-pressure injection lines contain thermal sleeves at their connections into the reactor coolant piping to prevent overstressing of the pipe juncture owing to the 90 F water being injected into these high temperature lines. The equipment normally operating is handling 125 F water, and hence will experience no thermal shock when 90 F water is introduced. 1 6

Injection response of the core flooding system is dependent upon the rate of reduction of reactor coolant system pressure. For a maximum hypothetical rupture, the core flooding system is capable of reflooding the core to the hot spot within a safe period after a rupture has occurred.

Emergency low pressure injection by the decay heat removal system will be delivered within 25 sec after the reactor coolant system reaches the actuating pressure of 200 psig. This anticipated delay time consists of these intervals:

a. Total instrumentation lag -- ≈ 1 sec

b. Emergency power source start -- <15 sec

c. Pump motor startup (from the time the pump motor line circuit breaker closes until the pump attains full speed) -- ≈ 10 sec

d. Injection valve opening time -- <10 sec

e. Borated water storage tank outlet valves -- <10 sec

Total (only b and c are additive) ≈ 25 sec

6.1.3.3 Special Features

The core flooding nozzles (Figure 3-47) will be specially designed to insure that they will safely take the differential temperatures imposed by the accident condition. Special attention also will be given to the ability of the injection lines to absorb the expansion resulting from the recirculating water temperature.

For most of their routing, the emergency injection lines will be outside the reactor and steam generator shielding, and hence protected from missiles originating within these areas. The portions of the injection lines located between the primary reactor shield and the reactor vessel wall are not subject to missile damage because there are no credible sources of
missiles in that area. To afford further missile protection, a high- 6
pressure injection line connects to each reactor coolant inlet pipe and the two core flooding nozzles are located on opposite sides of the reactor vessel.

All water used for emergency injection fluid will be maintained at a minimum concentration of 2,270 ppm of boron. The temperature, pressure, and level of these tanks will be displayed in the control room, and alarms will sound when any condition is outside the normal limits. The water will be periodically sampled and analyzed to insure proper boron concentration.

6.1.3.4 Check Valve Leakage - Core Flooding System

The action that would be taken in the case of check valve leakage would 1
be a function of the magnitude of the leakage.

Limited check valve leakage will have no adverse effect on reactor operation. The valves will be specified to meet the tightness requirements of MSS-SP-61. For these valves, this amounts to a maximum permissible leakage of 140 cc/hr per valve. Two valves in series are provided in each core flooding line; hence, leakage should be below this value.

Leakage across these check valves can have three effects: (a) it can cause a temperature increase in the line and core flooding tank, (b) it can cause a level and resultant pressure increase in the tank, and (c) it can cause dilution of the borated water in the core flooding tank.

Leakage at the rate mentioned above causes insignificant changes in any of these parameters. A leakage of 140 cc/hr causes level increase in the tank of less than 1 in./mo. The associated temperature and pressure increase is correspondingly low. 1

If it were assumed that the leakage rate is 100 times greater than specified, then there would still be no significant effect on reactor operation since the level change would be approximately 2 in./day. A 2-in. level change will result in a pressure increase of approximately 10 psi. With redundant temperature, pressure, and level indicators and alarms available to monitor the core flooding tank conditions, the most significant effect on reactor operations is expected to be a more frequent sampling of tank boric acid concentration.

To insure that no temperature increase will occur in the tank, even at higher leakage rates, the portion of the line between the two check valves and the line to the tanks will be left uninsulated to promote convective losses to the building atmosphere.

In summary, reactor operation may continue with no adverse effects coincident with check valve leakage. Maximum permissible limits on core flooding tank parameters (level, temperature, and boron concentration) will be established to insure compliance with the core protection criteria and final safety analyses.

All active components, as listed in Table 6-3, of the emergency injection systems will be tested periodically to demonstrate system readiness. In addition, normally operating components will be inspected for leaks from pump seals, valve packing, flanged joints, and safety valves.

Table 6-2

Single Failure Analysis-Emergency Injection

Component	Malfunction	Comments and Consequences
A. High Pressure Injection		
1. Pneumatic valve at makeup tank outlet.	Valve remains open.	When the tank is empty, tank pressure would be less than the high-pressure injection pump suction pressure (with borated water storage tank on the line), thus preventing the release of hydrogen from the tank to the pump suction line.
2. Pneumatic suction valve for makeup pumps from borated water storage tank.	Fails to open.	Similar valve in other makeup pump string will deliver required flow. 6
3. Makeup pump.	Out of maintenance.	Two pumps will still be available. Only one pump is required for engineered safeguards. 6
4. Makeup pump.	Fails (stops).	Other makeup pump delivers required flow. 6
5. Makeup pump isolation valve.	Left inadvertently closed.	See Item A-4 above. Valves will normally be left open since the check valve in each pump discharge will prevent backflow. Operating procedures will call for pump isolation valves to be closed only for maintenance.

Table 6-2 (Cont'd)

Component	Malfunction	Comments and Consequences	
6. Makeup pump discharge check valve.	Sticks closed.	This is considered incredible since the pump discharge pressure of 2,700 psi at no flow would tend to open even a very tightly stuck check disc.	
7. Pressurizer level control valve.	Fails to close.	No consequences.	6
8. Seal injection control valve.	Fails to close.	Injection flow through this line would be small compared to the flow through the injection lines due to the high flow resistance of the reactor coolant pump seals.	
9. Pneumatic valve in high-pressure injection line.	Fails to open.	Flow from one pump will go through the alternate line. Other pump will operate as normal.	6
10. Check valve in injection line (inside reactor building).	Sticks closed.	See comment on Item A-6.	
11. Injection line inside reactor building.	Rupture.	Flow rate indicators in the four injection lines would indicate the gross difference in flow rates. Check valve in the injection line would prevent additional loss of coolant from the reactor. The line is protected from missiles by reactor coolant system shielding.	6

Table 6-2 (Cont'd)

Component	Malfunction	Comments and Consequences
12. Pneumatic valve from decay heat coolers.	Inadvertently left open.	No significant consequences. A small percentage of LP injection flow will be bypassed to HP suction.
13. Pneumatic valve from decay heat coolers.	Fails to open.	This valve, if required to be opened, need not be opened until about 39 min. after the accident. This provides time for manual opening.
14. Double manual valves connecting pump lines.	Inadvertently left open.	Not credible that both valves will inadvertently be left open.

6

Table 6-2 (Cont'd)

Component	Malfunction	Comments and Consequences
B. Core Flooding System		
1. Flooding line check valve.	Sticks closed.	This is considered incredible based on the valve size and opening pressure applied.
C. Decay Heat Removal System		
1. Check valve at reactor vessel.	Sticks closed.	This is considered incredible since these valves will be used periodically during decay heat removal, and the opening force will be approximately 5,000 pounds.
2. Air-operated injection valve.	Fails to open.	Second injection line will deliver required flow.
3. Safety valve.	Stuck open.	Loss of injection flow is small since valve is small.
4. Decay heat cooler.	Isolation valve left closed.	Other heat exchanger will take required injection flow and remove required heat. Valves will be closed only for maintenance of heat exchanger.

Table 6-2 (Cont'd)

Component	Malfunction	Comments and Consequences	
5. Decay heat cooler.	Massive rupture.	Not credible. During normal decay heat removal operation, heat exchanger will be exposed to higher pressure and approximately the same temperature as the postaccident temperature and pressure.	
6. Decay heat cooler.	Out for maintenance.	Remaining heat exchanger will take required injection flow.	
7. Decay heat pump isolation valve.	Left closed.	Remaining pump will deliver required injection flow.	6
8. Decay heat pump discharge check valve.	Sticks closed.	See comment on Item C-1 above.	
9. Decay heat pump.	Fails to start.	Remaining pump will deliver required injection flow.	6
10. Stop-check valve at borated water storage tank outlet.	Sticks closed.	Alternate line will permit required flow.	6
11. Air-operated valve permitting suction from reactor building sump.	Fails to open.	Two lines and valves are provided, but need not be actuated until 39 min after start of accident which provides time for manual operation.	

Table 6-2 (Cont'd)

	Component	Malfunction	Comments and Consequences	
12.	Reactor building sump outlet pipe.	Becomes clogged.	This is considered incredible because of the dual sump line arrangement, the size of the lines, and the sump design. The two recirculation lines take suction from the different portions of t[illegible] sump. A grating will be provided over the sump, and additional heavy duty strainers will be provided for suction lines.	
13.	Dual manual valves connecting suction headers.	Inadvertently left open.	Not credible that valves will inadvertently be left open.	6
14.	Air-operated valve permitting suction from reactor building sump.	Inadvertently and prematurely opened after LOCA.	See answer to Question 17.6, Amendment #4, Supplement 3.	6

Table 6-3

Emergency Injection Equipment Performance Testing

Makeup Pumps	One pump is operating continuously. The other two pumps will be periodically tested. 6
High Pressure Injection Line Valves	The remotely operated stop valve in each line will be opened partially one at a time. The flow devices will indicate flow through the lines.
Makeup Pump Suction Valve	The makeup tank water level will be raised to equalize the pressure exerted by the storage tank and the borated water storage tank. The valves will then be opened individually and closed. 6
Decay Heat Pumps	In addition to use for shutdown cooling, these pumps will be tested singly by opening the borated water storage tank outlet valves and the bypasses in the borated water storage tank fill line. This will allow water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.
Borated Water Storage Tank Outlet Valves	The operational readiness of these valves will be established in completing the pump operational test discussed above. During this test, each of the valves will be tested separately for flow.
Low Pressure Injection Valves	With pumps shut down and borated water storage tank outlet valves closed, these valves will be opened and reclosed by operator action.
Valve for Suction From Sump	With pumps shut down and borated water storage tank outlet valves closed, these valves will be opened and reclosed by operator action.
Valves in Core Flooding Injection Lines	Valves can be operated during each shutdown to determine performance. Isolation valves will be closed to contain water in core flooding tanks during shutdown.

6.2 REACTOR BUILDING ATMOSPHERE COOLING AND WASHING

6.2.1 DESIGN BASES

Emergency building atmosphere cooling and washing is provided to limit postaccident building pressures to design values and reduce the postaccident level of fission products in the building atmosphere.

Reactor building air recirculation and cooling units, backed up by reactor building sprays, are used for emergency atmosphere cooling. Chemical additives contained in the building sprays are used to reduce postaccident fission product concentrations in the building atmosphere.

6.2.2 DESCRIPTION

The schematic flow diagram of the emergency reactor building atmosphere cooling and washing and associated instrumentation is given in Figure 6-5.

Emergency and normal cooling are performed with the same basic units. Each 1
unit contains an emergency cooling coil, a normal cooling coil, and a two-speed fan. For emergency cooling, all units will operate under postaccident conditions with the heat being rejected to river water. Each of these units 6
can remove 80 x 10^6 Btu/hr under peak reactor building temperature conditions. Figure 6-6 shows the heat exchange characteristics versus building ambient conditions for these units. The design data for the cooling units are shown in Table 6-4.

Table 6-4

Reactor Building Cooling Unit Performance and Equipment Data 1

(capacities are on a per unit basis)

	Duty	
	Emergency	Normal
No. Installed	3	3
No. Required	3	2
Type Coil	Finned Tube	Finned Tube
Peak Heat Load, Btu/hr	80 x 10^6	1.25 x 10^6
Fan Capacity, cfm	54,000	108,000
Reactor Building Atmosphere Inlet Conditions		
Temperature, F	281	110
Steam Partial Pressure, psia	50	--
Air Partial Pressure, psia	20	--
Total Pressure, psig	55	Atmospheric
Cooling Water Flow, gpm	1,780	250
Cooling Water Inlet Temperature, F	85	85 6
Cooling Water Outlet Temperature, F	175	95

Simultaneously with the air recirculation cooling, reactor building sprays are supplied with water by two pumps which take suction on the borated water storage tank until this coolant source is exhausted. The sodium thiosulfate chemical additive required for the reactor building sprays is supplied from a storage tank connected by dual lines containing check valves to the suction of the spray and decay heat removal pumps. Sufficient sodium thiosulfate is injected into the borated water to create a 1 wt % concentration in the reactor building water inventory. 1

After the supply from the borated water storage tank is exhausted, the spray pumps take suction from the reactor building sump recirculation lines. This continued spraying serves to reduce the reactor building atmosphere to the temperature of the reactor building sump.

Design data for the reactor building spray system components are given in Table 6-5, and the flow characteristics of the reactor building spray pumps are given in Figure 6-7. Design data for components of the reactor building cooling and decay heat removal systems used in this phase of engineered safeguards operation are given in Section 9 and supplemented by Figures 6-3, 6-4, 6-6, and 6-7 of this Section.

Table 6-5

Reactor Building Spray System Performance and Equipment Data
(capacities are on a per unit basis)

Reactor Building Spray Pumps		
Number	2	
Flow, gpm	1,500	
Developed Head at Rated Flow, ft	430	1
Motor Horsepower, hp	250	
Material	SS	
Design Pressure, psi	300	
Design Temperature, F	300	
Sodium Thiosulfate Tank		
Number	1	
Volume, ft^3	1,500	
Material	SS	
Design Pressure, psi	50	1 6
Design Temperature, F	150	
Sodium Thiosulfate Concentration, wt %	30	
Spray Header		
Number	2	
Spray Nozzles per Header	375	1

6.2.3 DESIGN EVALUATION

The function of cooling the reactor building atmosphere is fulfilled by either of the two methods described above, and redundancy of equipment within both methods will provide for protection of building integrity. The reactor building sprays through duplication, basic washing concept, and chemical additive will serve to reduce fission product levels in the building atmosphere.

For the first 30-40 min following the maximum blowdown loss-of-coolant accident, i.e., during the time that the reactor building spray pumps take their suction from the borated water storage tank, this system provides more than 100 per cent of the heat removal capacity of the reactor building cooling system.

The reactor building spray system design is based on the spray water being raised to the temperature of the reactor building in falling through the steam-air mixture within the building. Detailed evaluation of system performance is presented in Section 14. Each of the following equipment arrangements will provide sufficient heat removal capability to maintain the postaccident reactor building pressure below the design value:

a. Reactor building spray system.

b. All emergency units in the reactor building cooling system. | 1

c. Two emergency cooling units and the reactor building spray system at one-half capacity.

The reactor building spray system shares the suction lines from the borated water storage tank and the tank itself with the high and low pressure injection safeguards. | 6

6.2.3.1 Failure Analysis

A single failure analysis has been made on all active components of the systems used to show that the failure of any single active component will not prevent fulfilling of the design functions. This analysis is shown in Table 6-6. Assumptions inherent in this analysis are the same as those presented in 6.1.3 in regard to valve functioning, failure types, etc. Results of full and partial performance of these safeguards are presented in Section 14 under analysis of postaccident conditions.

Table 6-6

Single Failure Analysis-Reactor Building Atmosphere Cooling and Washing

Component	Malfunction	Comments and Consequences
1. Reactor building spray nozzles.	Clogged.	Large number of nozzles (375 on each of two headers) renders clogging of significant number of nozzles as incredible. 1
2. Reactor building spray header.	Rupture.	This is considered incredible due to low operating pressure differential.
3. Check valve in spray header line.	Sticks closed.	This is considered incredible due to large opening force available at pump shutoff head.
4. Air-operated valve in spray header line.	Fails to open.	Second header delivers 50 per cent flow.
5. Spray pump isolation valve.	Left closed.	Flow and cooling capacity reduced to 50 per cent of design. In combination with emergency coolers, 150 per cent of total design requirement is still provided.
6. Reactor building spray pump.	Fails to start.	Flow and cooling capacity reduced to 50 per cent of design. In combination with emergency coolers, 150 per cent of total design requirement is still provided.
7. Normal and emergency cooling unit fan.	Stops.	Emergency cooling by the other operating units with supplemental cooling by the sprays.

Table 6-6 (Cont'd)

	Component	Malfunction	Comments and Consequences
8.	Normal and emergency cooling unit.	Rupture of emergency cooling coil.	The tubes are designed for 200 psi and 300 F which exceeds maximum operating conditions. Tubes are protected against credible missiles. Hence, rupture is not considered credible.
9.	Normal and emergency cooling unit.	Rupture of casing and/or ducts.	Consideration will be given during detailed design to the dynamic forces resulting from the pressure buildup during a postaccident situation. The units will also be inspectable and protected against credible missiles. Cooling with these units will be supplemented by the sprays.
10.	Normal and emergency cooling units.	Rupture of system piping.	Rupture is not considered credible since all piping is Schedule 40, permitting an allowable working pressure of at least 500 psi at 650 F for all sizes. Piping is inspectable and protected from missiles. Maximum actual internal pressure will be less than 200 psi at temperatures below 300 F.
11.	Air-operated valve at inlet penetration.	Sticks closed.	Flow will be periodically established through the line to check the operational capability f system. Such tests will show if valve is malfunctioning.
12.	Air-operated valve at outlet penetration.	Fails to open.	Comments for Item 11 apply.

Table 6-6 (Cont'd)

Component	Malfunction	Comments and Consequences
13. Power operated valve at sodium thiosulfate storage tank outlet.	Fails to open.	Alternate check valve will permit flow required for sprays.

6

6.2.3.2 Reactor Building Cooling Response 1

Air recirculation established during normal operation through two of the three building ventilation units continues under accident conditions. In addition the third unit will be placed in operation for the accident condition. Alternate cooling coils in these ventilation units, supplied with emergency cooling water, are placed into service after reactor building pressure increases to 4 psi. Cooling continues utilizing these coils until the building pressure reaches near-atmospheric, and the decay heat removal system is placed into emergency service, recirculating and cooling fluid from the reactor building sump.

The reactor building spray system will likewise be activated by a single parameter signal. Two of three signals signifying high reactor building pressure will start both of the reactor building spray pumps, open the reactor building spray inlet valves, and open the suction valves from the borated water storage tank. The system components may also be actuated by operator action from the control room for performance testing.

The total time to delivery of reactor building sprays is approximately 1 min 3
after building pressure reaches 10 psi.

6.2.3.3 Special Features

The casing design for the ventilation units will be of a conventional nature unless additional analysis shows the possibility of pressure wave collapse. In that event, quick, inward-opening hinged doors, or other protective features, will be incorporated into the design to maintain postaccident operability. The ventilation units are located outside the concrete shield for the reactor vessel, steam generators, and reactor coolant pumps at an elevation above the water level in the bottom of the reactor building at postaccident conditions. In this location, the systems in the reactor building are protected from credible missiles and from flooding during postaccident operations. Also, this location provides shielding so that the design radiation dose level is 25 mrem/hr and allows for maintenance and repair, and inspections to be performed during power operation.

The spray headers of the reactor building spray system are located outside and above the reactor and steam generator concrete shield. During operation, a shield also provides missile protection for the area immediately above the reactor vessel. The spray headers are therefore protected from missiles originating within the shield. The spray pumps are located outside the reactor building and are thus available for operative checks during Station operation.

6.2.4 TESTS AND INSPECTIONS

Active components of the ventilation units will normally be in service. Valving on the emergency coils can be periodically cycled, thus placing the coils into service periodically during operation.

The active components of the reactor building spray system will be tested on a regular schedule as follows:

Reactor Building Spray Pumps	These pumps will be tested singly by opening the valves in the test line and the borated water storage tank outlet valves. Each pump in turn will be started by Station operator action and checked for flow establishment to each of the spray headers. Flow will also be tested through each of the borated water storage tank outlet valves by operating these valves.
Borated Water Storage Tank Outlet Valves	These valves will be tested in performing the pump test listed above.
Reactor Building Spray Injection Valves	With the pumps shut down and the borated water storage tank outlet valves closed, these valves will each be opened and closed by operator action.
Reactor Building Spray Nozzles	Under the conditions specified for the previous test, and with the reactor building spray valves closed, low pressure air will be blown through the test connections. 3

6.3 ENGINEERED SAFEGUARDS LEAKAGE AND RADIATION CONSIDERATIONS

6.3.1 INTRODUCTION

The use of normally operating equipment for engineered safeguards functions and the location of some of this equipment outside the reactor building require that consideration be given to direct radiation levels after fission products have accumulated in these systems with leakage from these systems. Although the engineered safeguards equipment is designed for control room operation following an accident, long-term postaccident operation could necessitate manual operation of certain valves.

The shielding for components of the Engineered Safeguards is designed to provide protection for personnel to perform all operations necessary for mitigation of the accident within the limits of 10 CFR 100 in the event of an MHA.

6.3.2 SUMMARY OF POSTACCIDENT RECIRCULATION AND LEAKAGE CONSIDERATIONS

Following a loss-of-coolant accident and exhaustion of the borated water storage tank, reactor building sump recirculation to the reactor vessel and the reactor building sprays is initiated.

While the reactor auxiliary systems involved in the recirculation complex are closed to the auxiliary building atmosphere, leakage is possible through component flanges, seals, instrumentation, and valves.

The leakage sources considered are:

a. Valves.

(1) Disc leakage when valve is on recirculation complex boundary.

(2) Stem leakage.

(3) Bonnet flange leakage.

b. Flanges.

c. Pump stuffing boxes.

While leakage rates have been assumed for these sources, maintenance and periodic testing of these systems will preclude all but a small percentage of the assumed amounts. With the exception of the boundary valve discs, all of the potential leakage paths may be examined during periodic tests or normal operation. The boundary valve disc leakage is retained in the other closed systems and therefore will not be released to the auxiliary building.

While valve stem leakage has been assumed for all valves, the manual valves in the recirculation complex are backseating.

6.3.3 LEAKAGE ASSUMPTIONS

Source	Quantities
a. Valves - Process	
(1) Disc leakage	10 cc/hr/in. of nominal disc diameter.
(2) Stem leakage	1 drop/min
(3) Bonnet flange	10 drops/min
b. Valves - Instrumentation	
Bonnet flange and stem	1 drop/min
c. Flanges	10 drops/min
d. Pump Stuffing Boxes	50 drops/min

For the analysis, it was assumed that the water leaving the reactor building was less than 200 F when recirculation occurs. 3

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(DELETED) 3

6.3.4 DESIGN BASIS LEAKAGE

The design basis leakage quantities derived from these assumptions for post-accident sump recirculation are tabulated in Table 6-7.

6.3.5 LEAKAGE ANALYSIS CONCLUSIONS

It may be concluded from this analysis (in conjunction with the discussion and analysis in 14.2.2.4.5) that leakage from Engineered Safeguards equipment outside the reactor building does not pose a public safety problem.

Table 6-7
Leakage Quantities to Auxiliary Building Atmosphere

Leakage Source	No. of Sources	Per Source, drops/min	Total cc/hr	
a. Pump Seals				
Decay heat pumps	2	50	300	6
Spray pumps	2	50	300	
b. Flanges[a]	114	10	3,320	
c. Process Valves	35	1	105	
d. Instrumentation Valves	25	1	75	
e. Valve Seats at Boundaries	11	(b)	750	
Total			4,850	6

(a) Assumes process and boundary valves, and process components are flanged.

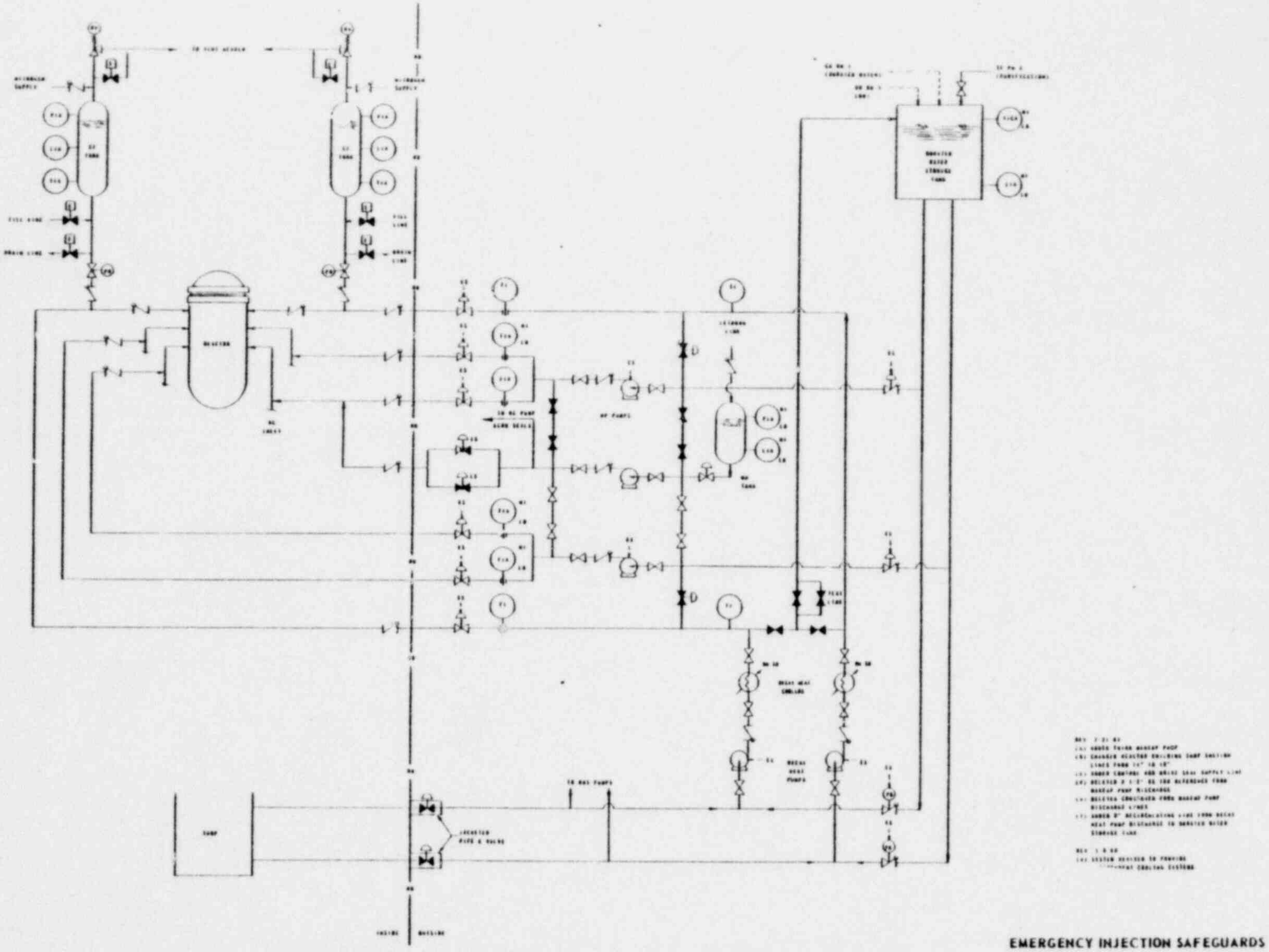
(b) Assumes 10 cc/hr/in. of nominal disc diameter.

6.3.6 SYSTEM INTEGRITY

In addition to the proper selection of materials, fabrication quality control, and design, additional measures have been taken to further increase the integrity of the engineered safeguards systems, i.e. High Pressure Injection, Decay Heat Removal System, Core Flooding System, and the Reactor Building Spray System. Provisions are included for the physical separation of each engineered safeguard system and also the physical separation of redundant components within any system. Figure 1-3 shows the physical separation provided for these systems and their redundant components. The entire train of the systems is included within the confines of the separation barriers and ensures that the failure of any system or redundant component will have no detrimental effect on any other system or component. 5

The drain lines from the reactor building sump from the point where they leave the building up to and including their first isolation valve are enclosed within a barrier. Leak detection equipment is also provided within the individual cubicles which will permit isolation of that particular portion of the system which may have failed.

The combination of double barriers, physical separability, and leak detection methods precludes the loss of function of any safeguard system.



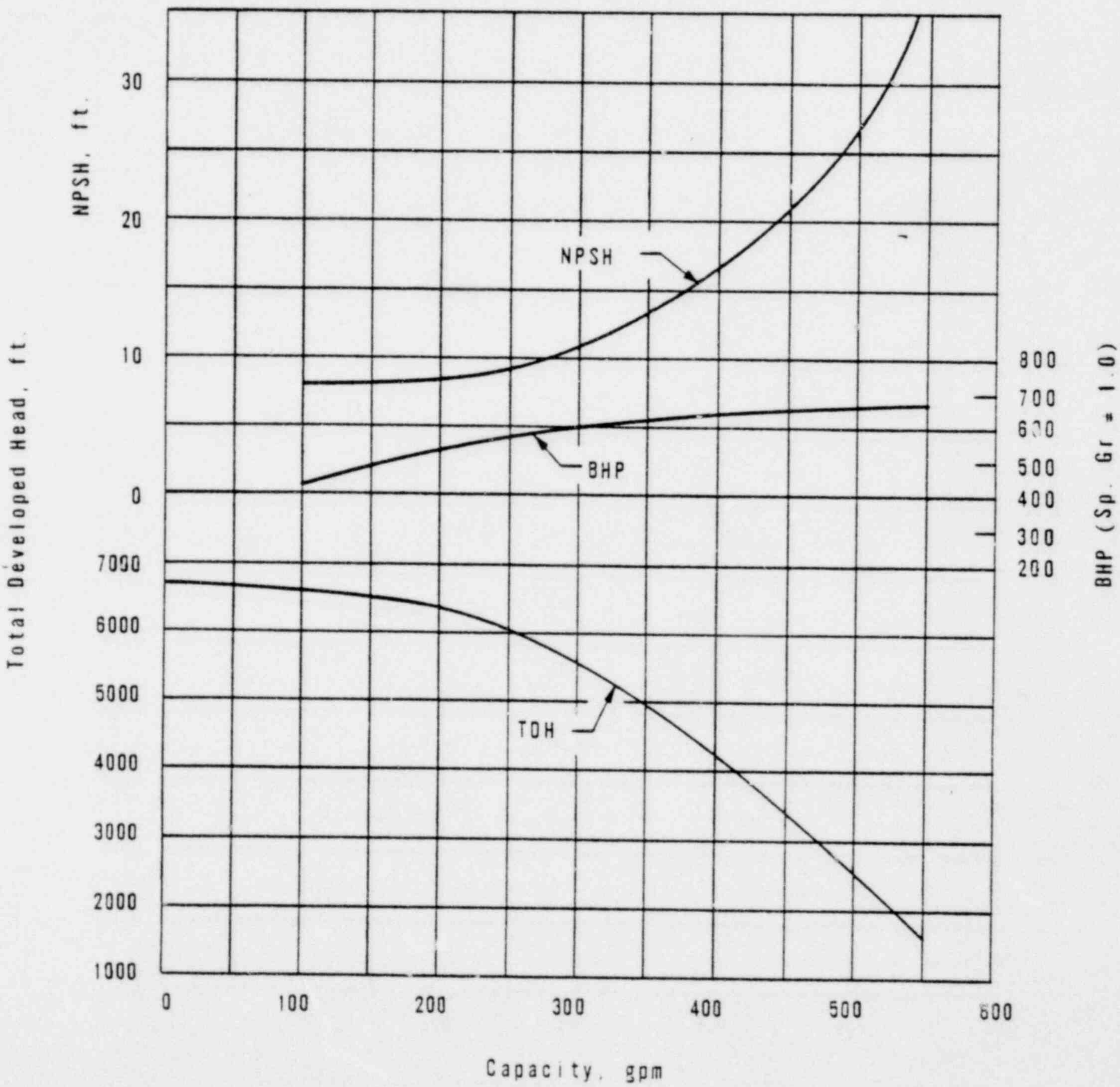
EMERGENCY INJECTION SAFEGUARDS

Met-Ed FIGURE 6-1

THREE MILE ISLAND NUCLEAR STATION

AMEND. 6 (1-8-68)

690 1000



MAKE-UP PUMP CHARACTERISTICS

FIGURE 6-2

THREE MILE ISLAND NUCLEAR STATION

AMEND. 6 (1-6-68)

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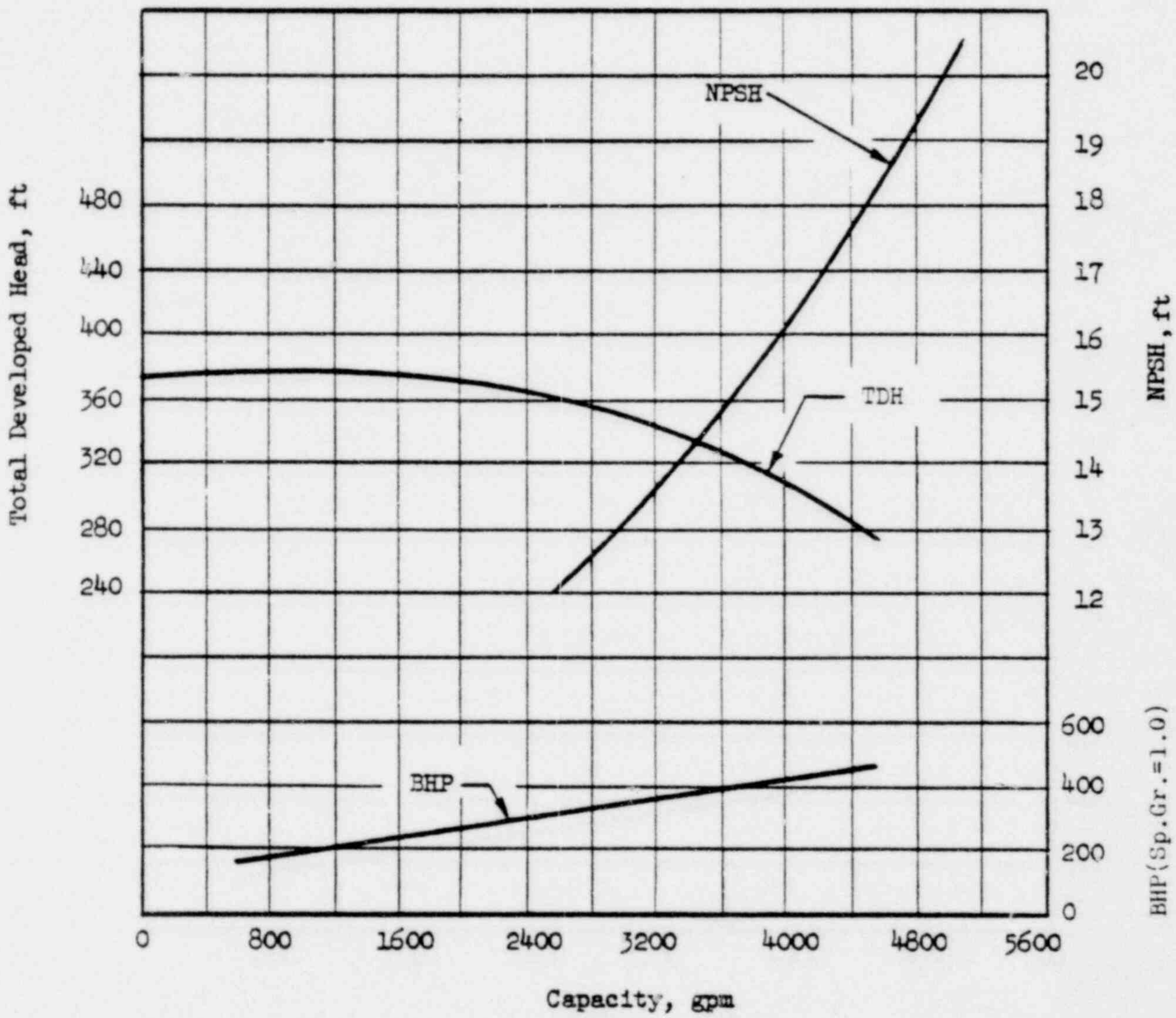


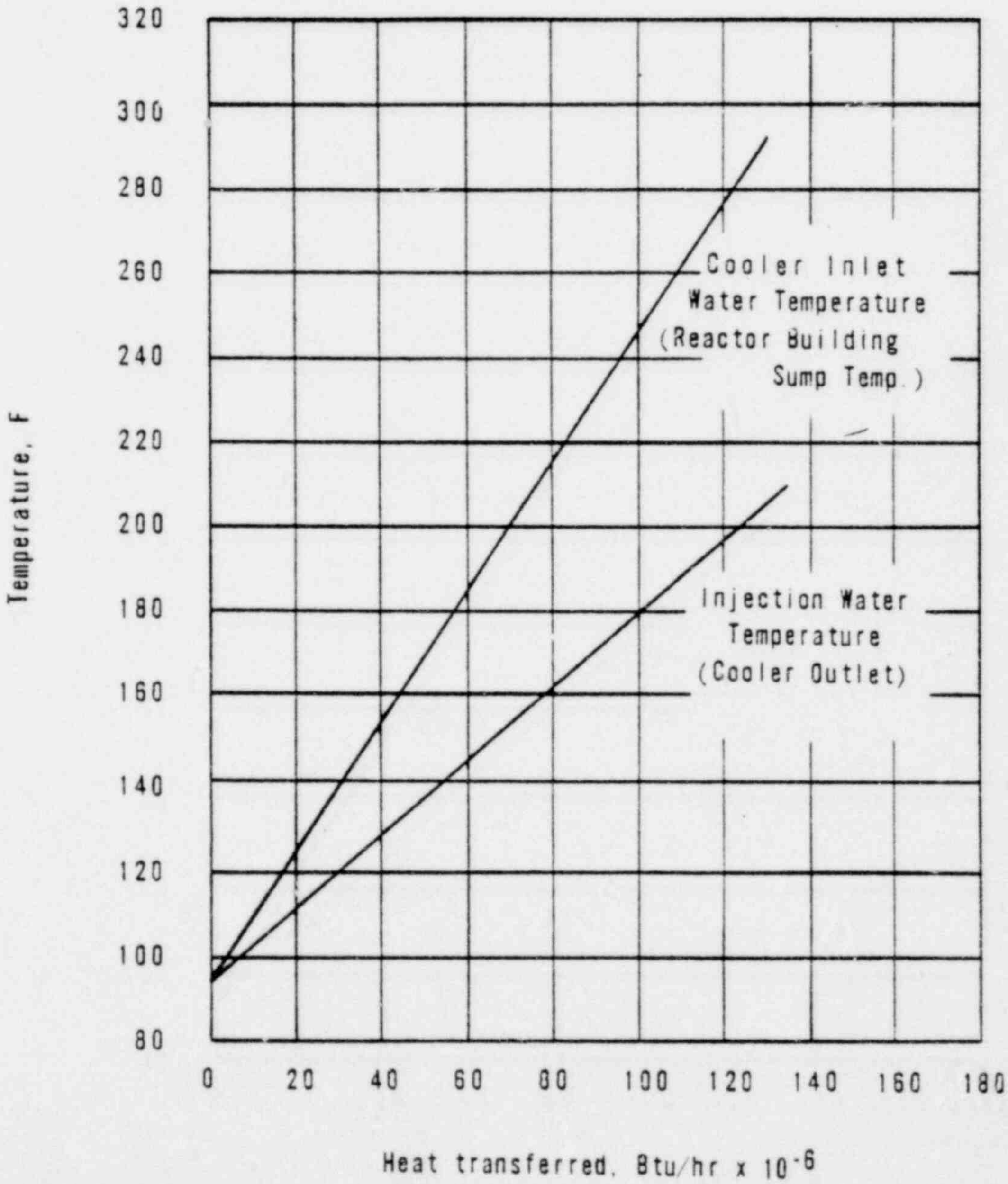
DECAY HEAT REMOVAL PUMP CHARACTERISTICS

Met Ed FIGURE 6-3

THREE MILE ISLAND NUCLEAR STATION

Eme.gency Design Conditions

Injection Water Flow Per Cooler - 3000 gpm
Decay Heat Service Water Flow Per Cooler - 3000 gpm
Decay Heat Service Water Inlet Temperature - 95 F



DECAY HEAT REMOVAL COOLER CHARACTERISTICS

FIGURE 6-4

THREE MILE ISLAND NUCLEAR STATION

AMEND. 6 (1-8-68)

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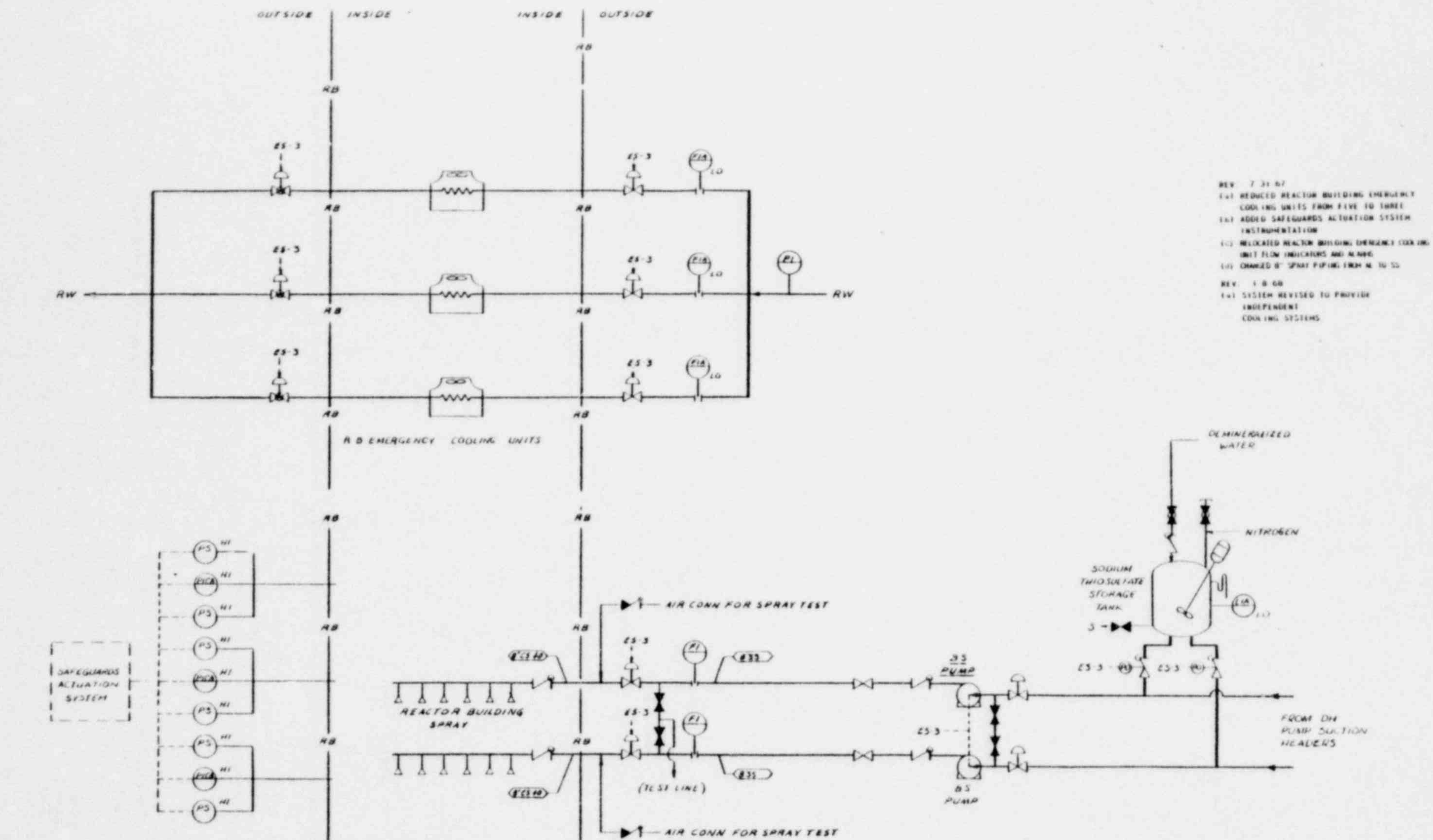

NOTE:
FOR LEGEND & NOMENCLATURE SEE FIG 9-1

REACTOR BUILDING ATMOSPHERE COOLING & WASHING SAFEGUARDS

Met Ed FIGURE 6-5

THREE MILE ISLAND NUCLEAR STATION

AMEND. 6 (1-8-68)

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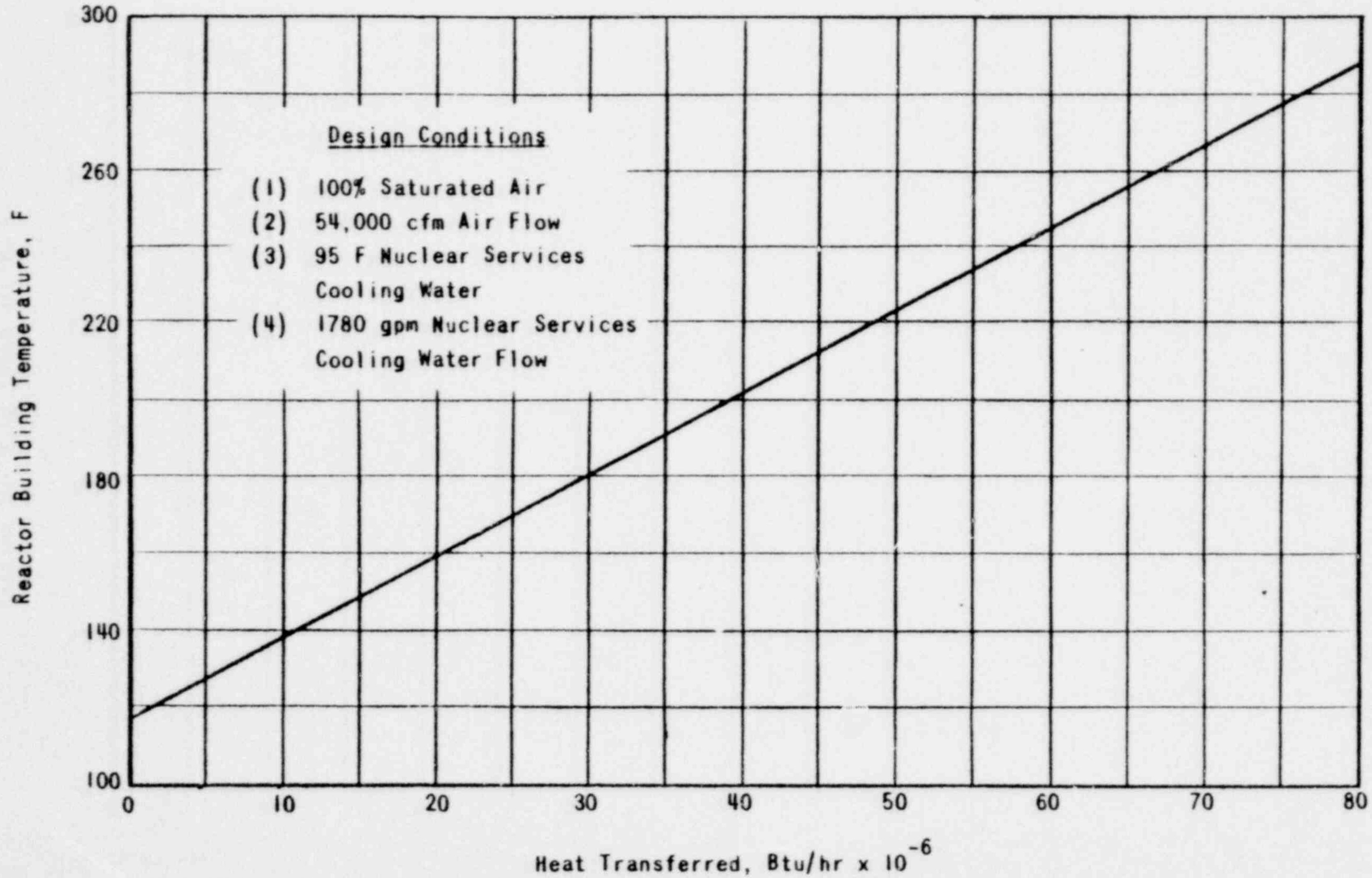
REV: 7-21-67
REVISED FOR 80 BTU/HR x 10^6



REACTOR BUILDING EMERGENCY COOLER CHARACTERISTICS

FIGURE 6-6

THREE MILE ISLAND NUCLEAR STATION

AMEND. 1 (7-21-

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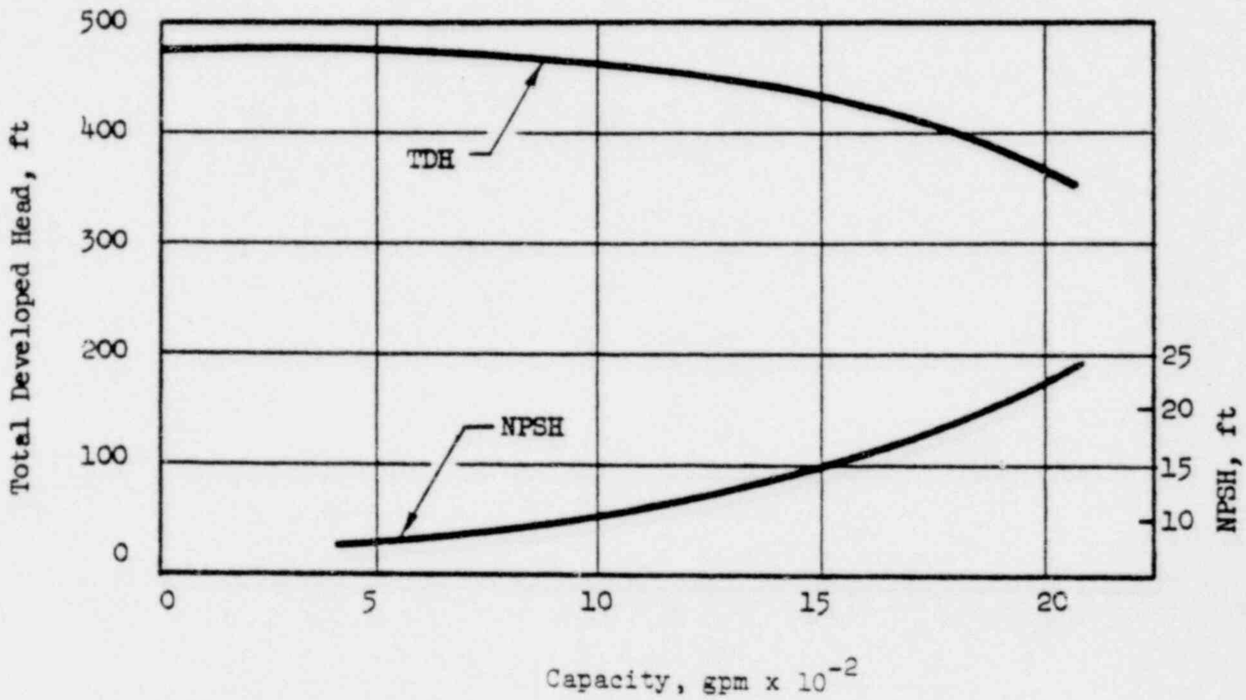

REACTOR BUILDING SPRAY PUMP CHARACTERISTICS

Met Ed

FIGURE 6-7

THREE MILE ISLAND NUCLEAR STATION

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