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7. INSTRUMENTATION AND CONTROL

Complete supervision of both the nuclear and turbine-generator sections of the plant is accomplished by the instrumentation and control systems from the control room. The instrumentation and control systems are designed to permit periodic on-line test to demonstrate the operability of the reactor protection system.

7.1 GENERAL DESIGN CRITERIA

Criteria applying in common to all Instrumentation and Control Systems are given in the following listing. Thereafter, criteria which are specific to one of the instrumentation and control systems are discussed in the appropriate portion of the description of that system.

7.1.1 INSTRUMENTATION AND CONTROL SYSTEMS CRITERIA

Instrumentation and Control Systems

Criterion: Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables. (GDC 12)

Instrumentation and controls essential to avoid undue risk to the health and safety of the public are provided to monitor and maintain neutron flux, primary coolant pressure, flow rate, temperature, and control rod positions within prescribed operating ranges.

The non-nuclear regulating process and containment instrumentation measures temperatures, pressure, flow, and levels in the Reactor Coolant System, Steam Systems, Containment and ot[illegible] Auxiliary Systems. Process variables required on a continuous basis for the startup, power operation, and shutdown of the plant are controlled and indicated or recorded from the control room access - to which is supervised. The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems and processes over the full operating range of the plant.

7.1.2 RELATED CRITERIA

The following are criteria which are related to all instrumentation and control systems but are more specific to other plant features or systems, therefore are discussed in other sections, as listed.

Name		Discussion
Suppression of Power Oscillations	(GDC #7)	Section 3
Reactor Core Design	(GDC #6)	Section 3
Quality Standards	(GDC #1)	Section 4
Performance Standards	(GDC #5)	Section 4
Fire Protection	(GDC #3)	Sections 5 and 9
Missile Protection	(GDC #40)	Sections 4, 5 and 6
Emergency Power	(GDC #39)	Section 8

7.2 PROTECTIVE SYSTEMS

The protective systems consist of both the reactor protection system and the engineered safety features. Equipment supplying signals to any of these protective systems is considered a part of that protective systems.

7.2.1 DESIGN BASES

Control Room

Criterion: The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the control room under any credible post-accident condition or as an alternative, access to other areas of the facility as necessary to shut down and maintain safe control of the facility without excessive radiation exposures of personnel. (GDC 11)

The plant is equipped with a control room which contains those controls and instrumentation necessary for operation of the reactor and turbine generator under normal and accident conditions.

The control room is continuously occupied by the qualified operating personnel under all operating and MCA accident conditions.

Sufficient shielding, distance, and containment integrity are provided to assure that control room personnel shall not be subjected to doses under postulated accident conditions during occupancy of, ingress to and egress from the control room which, in the aggregate would exceed suggested limits in 10 CRF 100. The control room ventilation consists of a system having a large percentage of recirculated air. The fresh air intake can be cloased to stop the intake of airborne activity if monitors indicate that such action is appropriate.

Core Protection Systems

Criterion: Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits. (GDC 14)

The basic reactor tripping philosophy is to define a region of power and coolant temperature conditions allowed by the primary tripping functions, the overpower high ΔT trip, the over-temperature high ΔT trip and the nuclear overpower trip. The allowable operating region within these trip settings is provided to prevent any combination of power, temperature and pressure which would result in DNB with all reactor coolant pumps in operation. Additional tripping functions such as a high pressurizer pressure trip, low pressurizer pressure trip, high pressurizer water level trip, loss of flow trip, steam and feedwater flow mismatch trip, steam generator low-low water level trip, turbine trip, safety injection trip, nuclear source and intermediate range trips, and manual trip are provided to back up the primary tripping functions for specific accident condition and mechanical failures.

A dropped rod signal blocks automatic rod withdrawal and also provides a turbine load cutback if above a given power level. The dropped rod is indicated from individual rod position indicators or by a rapid flux decrease on any of the power range nuclear channels.

Overpower, over-temperature and T_{AVG} deviation rod stops prevent abnormal power conditions which could result from excessive control rod withdrawal initiated by a malfunction of the reactor control system or by operator violation of administrative procedures.

Engineered Safety Features Protection Systems

Criterion: Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features. (GDC 15)

Instrumentation and controls provided for the protective systems are designed to trip the reactor, when necessary, to prevent or limit fission product release from the core and to limit energy release; to signal containment isolation; and to control the operation of engineered safety features equipment.

The engineered safety features systems are actuated by the engineered safety features actuation channels. Each coincidence network energizes an engineered safety features actuation device that operates the associated engineered safety features equipment, motor starters and valve operators. The channels are designed to combine redundant sensors, and independent channel circuitry, coincident trip logic and different parameter measurements so that a safe and reliable system is provided in which a single failure will not defeat the channel function. The action initiating sensors, bistables and logic is shown in the figures included in the detailed Engineered Safety Features Instrumentation Description given in the System Design section. The Engineered Safety Features Instrumentation System actuates (depending on the severity of the condition) the Safety Injection System, the Containment Isolation System, the Containment Air Recirculation System and the Containment Spray System.

The passive accumulators of the Safety Injection System do not require signal or power sources to perform their function. The actuation of the active portion of the Safety Injection System is described on page 7.2.16.

The containment air recirculation coolers are normally in use during plant operation and would therefore not normally require an initiating signal. These units are, however, in the automatic sequence which actuates the engineered safety features upon receiving the necessary actuating signals indicating an accident condition.

Containment spray is actuated by coincident and redundant high containment pressure signals.

The Containment Isolation System provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the outside environment in the event of a loss-of-coolant accident. The actuation of the containment isolation is by coincident and redundant containment high pressure signals.

Protection Systems Reliability

Criterion: Protection systems shall be designed for high functional reliability and in-service testability necessary to avoid undue risk to the health and safety of the public. (GDC 19)

The reactor uses a higher speed version of the Westinghouse magnetic-type control rod drive mechanisms used in the San Onofre, and Connecticut Yankee plants. Upon a loss of power to the coils, the rod cluster control assemblies with full length absorber rods are released and fall by gravity into the core.

The reactor internals, fuel assemblies, RCC assemblies and drive system components are designed as Seismic Class I equipment. The RCC assemblies are fully guided through the fuel assembly and for the maximum travel of the control rod into the guide tube. Furthermore, the RCC assemblies are never fully withdrawn from their guide thimbles in the fuel assembly. Due to this and the flexibility designed into the RCC assemblies, abnormal loadings and misalignments can be sustained without impairing operation of the RCC assemblies.

The Rod Cluster Control (RCC) assembly guide system is locked together with pins throughout its length to ensure against misalignments which might impair control rod movement under normal operating conditions and credible accident conditions. An analogous system has successfully undergone 4132 hours of testing in the Westinghouse Reactor Evaluation Channel during which about 27,200 feet of step-driven travel and 1461 trips were accomplished with test misalignments in excess of the maximum possible misalignment that may be experienced when installed in the plant.

All reactor trip protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power.

Removal of one trip circuit is accomplished by placing that circuit in a half-tripped mode; i.e., a two-out-of-three circuit becomes a one-out-of-two circuit. Testing does not trip the system unless a trip condition exists in a concurrent channel.

Reliability and independence is obtained by redundancy within each tripping function. In a two-out-of-three circuit, for example, the three channels are equipped with separate primary sensors. Each channel is continuously fed from its own independent electrical source. Failure to de-energize a channel when required would be a mode of malfunction that would affect only that channel. The trip signal furnished by the two remaining channels would be unimpaired in this event.

Protection Systems Redundancy and Independence

Criterion: Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection function to be served. (GDC 20).

The reactor protection systems are designed so that the most probable modes of failure in each protection channel result in a signal calling for the protective trip. Each protection system design combines redundant sensors and channel independence with coincident trip philosophy so that a safe and reliable system is provided in which a single failure will not defeat the channel function, cause a spurious plant trip, or violate reactor protection criteria.

Channel independence is carried throughout the system extending from the sensor to the relay actuating the protective function. The protective and control functions when combined are combined only at the sensor. Both of these functions are fully isolated in the remaining part of the channel, control being derived from the primary protection signal path through an isolation amplifier. As such, a failure in the control circuitry does not affect the protection channel. This approach is used for pressurizer pressure and water level channels, steam generator water level, T_{avg} and ΔT channels, steam flow-feedwater flow and nuclear source, power range channels.

The engineered safety features equipment is actuated by one or the other of the engineered safety features actuation channels. Each coincidence network actuates an engineered safety actuation device that operates the associated engineered safety features equipment, motor starters and valve operators. As an example, the control circuit of a safety injection pump is typical of the control circuit for a large pump operated from switchgear. The actuation relay, energized by the Engineered Safety Features Instrumentation System has normally open contacts. These contacts energize the circuit breaker closing coil to start the pump when the control relay is energized. The Engineered Safety Features Instrumentation System actuates (depending on the severity of the condition) the Safety Injection System, the Containment Isolation System, Containment Air Recirculation System and Containment Spray System.

In the Reactor Protection System, two reactor trip breakers are provided to interrupt power to the full length rod drive mechanisms. The breaker main contacts are connected in series (with power supply) so that opening either breaker interrupts power to all full length rod mechanisms, permitting them to fall by gravity into the core. In the event of a loss of rod control power reactor trip breaker is de-energized and trips to an open mode.

Further detail on redundancy is provided through the detail descriptions of the respective systems covered by the various sections in this chapter. In summary, reactor protection is designed to meet all presently defined reactor protection criteria and is in accordance with the IEEE "Standard for Nuclear Plant Protection Systems 279". Redundancy and independence are more than achieved by protection channel designs which combine more than one sensor and parameter measurement with coincident trip circuitry (e. g. pressure coincident with level and interlocked with flow or nuclear flux).

Required continuous electrical supply is discussed in Section 8.

Protection Against Multiple Disability for Protection Systems

Criterion: The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function or shall be tolerable on some other basis. (GDC 23)

The components of the protection system are designed and laid out so that the mechanical and thermal environment accompanying any emergency situation in which the components are required to function does not interfere with that function.

Separation of redundant analog protection channels originates at the process sensors and continues back through the field wiring and containment penetrations to the analog protection racks. Physical separation is used to the maximum practical extent to achieve separation of redundant transmitters. Separation of field wiring is achieved using separate wireways, cable trays, conduit runs and containment penetrations for each redundant channel. Redundant analog equipment is separated by locating redundant components in different protection racks. Each channel is energized from a separate a-c instrument bus.

Demonstration of Functional Operability of Protection Systems

Criterion: Means shall be included for suitable testing of the active components of protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred. (GDC 25)

The signal conditioning equipment of each protection channel in service at power is capable of being tested and tripped independently by simulated analog input signals to verify its operation. This includes checking through to the trip breakers which necessarily involves the trip logic. Thus, the operability of each trip channel can be determined conveniently and without ambiguity.

Testing of the diesel-generator starting may be performed from the diesel-generator control board. The generator breaker is not closed automatically after starting during this testing. The generator may be manually sychronized to the 480 volt bus for loading. Complete testing of the starting of diesel generators can be accomplished by tripping the associated 480 volt undervoltage relays. The ability of the units to start within the prescribed time and to carry load can be periodically checked. (The electrical system is discussed in more detail in Section 8.2.3).

Protection Systems Failure Analysis Design

Criterion: The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced. (GDC 26).

Each reactor trip circuit is designed so that trip occurs when the circuit is de-energized; therefore, loss of channel power causes the system to go into its trip mode. In a two-out-of-three circuit, the three channels are equipped with separate primary sensors and each channel is energized from an independent electrical bus. Failure to de-energize when required is a mode of malfunction that affects only one channel. The trip signal furnished by the two remaining channels is unimpaired in this event.

Reactor trip is implemented by interrupting power to the magnetic latch mechanisms on all drive allowing the full length rod clusters to insert by gravity. The protection system is thus inherently safe in the event of a loss of power.

The engineered safety features actuation circuits are designed on the same "de-energize to operate" principle as the reactor trip circuits with the exception of the containment spray actuation circuit which is energized to operate in order to avoid spray operation on inadvertent power failure.

Automatic starting of all emergency diesel-generators is initiated by undervoltage relays on any 480 volt bus or by the safety injection signal. Engine cranking is accomplished by a stored energy system supplied solely for the associated diesel-generator. The undervoltage relay scheme is designed so that loss of 480 volt power does not prevent the relay scheme from functioning properly.

Redundancy of Reactivity Control

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

One of the two reactivity control systems employs rod cluster control assemblies to regulate the position of Ag-In-Cd neutron absorbers within the reactor core. The other reactivity control system employs the Chemical and Volume Control System to regulate the concentration of boric acid solution neutron absorber in the Reactor Coolant System.

Reactivity Control Systems Malfunction

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

Reactor shutdown with rods is completely independent of the normal control functions since the trip breakers completely interrupt the power to the full length rod mechanisms regardless of existing control signals. Effects of continuous withdrawal of a rod control assembly and of deboration are described in Sections 7.3.1 and 7.3.2, and Sections 9 and 14.

Principles of Design

Redundancy and Independence

The Protective systems are redundant and independent for all vital inputs and functions. Each channel is functionally independent of every other channel and

13 receives power from an independent source. Isolation of redundant protection channels is described in further detail in Section 7.2.1, Isolation of Redundant Protection Channels.

Manual Actuation

Means are provided for manual initiation of protective system action. Failures in the automatic system do not prevent the manual actuation of protective functions. Manual actuation requires the operation of a minimum of equipment.

Channel Bypass or Removal from Operation

The system is designed to permit any one channel to be maintained, and when required, tested or calibrated during power operation without system trip. During such operation the active parts of the system continue to meet the single failure criterion. Since the channel under test is either tripped or superimposed test signals are used which do not negate the process signal.

EXCEPTION: "one-out-of-two' systems are permitted to violate the single failure criterion during channel bypass provided that acceptable reliability of operation can be otherwise demonstrated and bypass time interval is short.

Capability for Test and Calibration

The bistable portions of the protective system (e.g., relays, bistables, etc.) provide trip signals only after signals from analog portions of the system reach preset values. Capability is provided for calibrating and testing the performance of the bistable portion of protective channels and various combinations of the logic networks during reactor operation.

The analog portion of a protective channel provides analog signals of reactor or plant parameters. The following means are provided to permit checking the analog portion of a protective channel during reactor operation:

a. Varying the monitored variable

b. Introducing and varying a substitute transmitter signal.

c. Cross checking between identical channels or between channels which bear a known relationship to each other and which have readouts available.

The design permits the administrative control of the means for manually bypassing channels or protective functions.

The design permits the administrative control of access to all trip settings, module calibration adjustments, test points, and signal injection points.

Information Readout and Indication of By-Pass

The protective systems are designed to provide the operator with accurate, complete, and timely information pertinent to their own stats and to plant safety.

Indication is provided in the control room if some part of the system has been administratively bypassed or taken out of service.

Trips are indicated and identified down to the channel level.

Vital Protective Functions and Functional Requirements

The Reactor Protective System monitors parameters related to safe operation and trips the reactor to protect the reactor core against fuel rod cladding damage caused by departure from nucleate boiling (DNB), and to protect against reactor coolant system damage caused by high system pressure. The engineered safety features instrumentation system monitors parameters to detect failure of the Reactor Coolant System, and initiates containment isolation and engineered safety features operation to contain radioactive fission products.

This section covers those protective system provided to:

a. Trip the reactor to prevent or limit fission product release from the core and to limit energy release.

b. Isolate containment and activate the Isolation Valve Seal Water System when necessary.

c. Control the operation of engineered safety features provided to mitigate the effects of accidents.

The core protective systems in conjunction with inherent plant characteristics are designed to prevent anticipated abnormal conditions from causing fuel damage exceeding limits established in Section 3, or reactor coolant system damage exceeding effects established in Section 4.

Completion of Protective Action

Where operating requirements necessitate automatic or manual bypass of a protective function, the design is such that the bypass is removed automatically whenever permissive conditions are not met. Devices used to achieve automatic removal of the bypass of a protective function are part of the protective system and are designed in accordance with the criteria of this section.

The protective systems are so designed that, once initiated, a protective action goes to completion. Return to normal operation requires administrative action by the operator.

Multiple Trip Settings

Where it is necessary to change to a more restrictive trip setting to provide adequate protection for a particular mode of operation or set of operating conditions, the design provides positive means of assuring that the more restrictive trip setting is used. The devices used to prevent improper use of less restrictive trip settings are considered a part of the protective system and are designed in accordance with the other provisions of these criteria.

Interlocks and Administrative Procedures

Interlocks and administrative procedures required to limit the consequences of fault conditions other than those specified as limits for the protective function comply with the protective system criteria.

Protective Actions

The Reactor Protective System automatically trips the reactor to protect the reactor core under the following conditions:

a) The reactor power, as measured by neutron flux, reaches a preset limit.

b) The temperature rise across the core as determined from loop ΔT reaches a limit; both from a variable ΔT set point (analog adjusted by neutron flux distribution) or a variable ΔT set point (function of T_{avg} and pressurizer pressure, also adjusted by neutron flux distribution).

c) The pressurizer pressure reaches an established minimum limit.

d) Loss of reactor coolant flow as sensed by low flow, loss of pump power or pump breakers opening.

The Reactor Protective System automatically trips the reactor when the pressurizer pressure or level reaches an established maximum limit.

Interlocking functions of the Reactor Protective System inhibit control rod withdrawal on the occurrence of a specified parameter reaching a value lower than the value at which reactor trip is initiated.

For anticipated abnormal conditions, protective syst[illegible] in conjunction with inherent plant characteristics and engineered safety features are designed to ensure that limits for energy release to the containment and for radiation exposure (as in 10 CFR 100) are not exceeded.

7.2.2 SYSTEM DESIGN

Reactor Protective System Description

Figure 7.2-2 is a block diagram of the Reactor Protective System.

Figure 7.2-3 illustrates the core limits and shows the maximum trip points which are used for the protection system. The solid lines indicate a typical locus of DNBR = 1.30 at four pressures, and the dashed lines indicate maximum permissible trip points for the overtemperature high ΔT reactor trip. Actual set points are lower to allow for measurement and instrumentation errors. The cross-hatched region represents the region of permissible operation (in which the DNBR exceeds 1.30) for a system pressure of 2200 psia.

The maximum and minimum pressures shown (2400 psia and 1700 psia) represent the set points for the high pressure and low pressure reactor trips.

Adequate margins exist between the worst steady state operating point (including all temperature, calorimetric, and pressure errors) and required trip points to preclude a spurious plant trip during design transients.

Engineered Safety Features Instrumentation Description

Figures 7.2-4, -5 and -6 show the action initiating sensors, bistables and logic for the engineered safety features instrumentation.

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Indication

All transmitted signals (flow, pressure, temperature, etc.) which can cause a reactor trip are either indicated or recorded for every channel.

The same channel isolation and separation criteria as describe for the reactor protection circuits are applied to the engineered safety features actuation circuits.

Protective Actions

The engineered safety features actuation system automatically performs the following vital functions:

a) Start operation of the Safety Injection System upon coincidence of low pressurizer pressure and low pressurizer level signals, or high containment pressure signals (Hi level; approximately 10% of containment design pressure), or on coincidence of high differential pressure between any two steam generators, or on coincidence of high steam flow in any two steam lines (automatically blocked when T_{avg} and steam pressure are above certain limits).

b) Operate the containment isolation valves in non-essential process lines upon detection of high containment pressure signals (Phase A containment isolation - Hi level). The Isolation Valve Seal Water System is actuated upon automatic actuation of the Safety Injection System.

c) Start the Containment Spray System and operate the remaining containment isolation valves upon detection of a containment pressure signal higher than required in item (b) above (Phase B containment isolation; Hi-Hi level, approximately 25 psig).

d) Start operation of the Containment Air Recirculation and Filtration System is not operating after operation of the required Safety Injection System is completed.

Steam Line Isolation

Any of the following signals will close all steam line isolation valves:

1. Coincidence of high steam flow in any two steam lines (automatically blocked when T_{avg} and steam pressure are above certain limits).

2 High containment pressure signals (2/3 + 2/3 high-high pressure).

3. Steam line isolation valves can also be closed one at a time by manual action.

Feedwater Line Isolation

Any safety injection signal will isolate the main feedwater lines by closing all control valves (main and bypass valves), tripping the main feedwater pumps and closing the pump discharge valves.

System Safety Features

Separation of Redundant Protection Channels

The reactor protection system is designed on a channelized basis to achieve separation between redundant protection channels. The channelized design, as applied to the analog as well as the logic portions of the protection system, is illustrated by Figure 7.2-1 and is discussed below. Although shown for four channel redundancy, the design is applicable to two and three channel redundancy.

Separation of redundant analog channels originates at the process sensors and continues through the field wiring and containment penetrations to the analog protection racks. Physical separation is used to the maximum practical extent to achieve separation of redundant transmitters. Separation of field wiring is achieved using separate wireways, cable trays, conduit runs and containment penetrations for each redundant channel. Analog equipment is separated by locating redundant components in different protection racks. Each channel is energized from a separate a-c power feed.

The reactor trip bistables are mounted in the protection racks and are the final operational component in an analog protection channel. Each bistable drives two logic relays ("C" & "D"). The contacts from the "C" relays are interconnected to form the required actuation logic for Trip Breaker No. 1 through d-c power feed No. 1. The transition from channel identity to logic identity is made at the logic relay coil/relay contact interface. As such, there is both electrical and physical separation between the analog and the logic portions of the protection system. The above logic network is duplicated for Trip Breaker No. 2 using d-c power feed No. 2 and the contacts from the "D" relays. Therefore, the two redundant reactor trip logic channels will be physically separated and electrically isolated from one another. Overall, the Protection System is comprised of identifiable channels which are physically, electrically and functionally separated and isolated from one another.

Physical Separation

The physical arrangement of all elements associated with the protective system reduces the probability of a single physical event impairing the vital functions of the system.

System equipment is distributed between instrument cabinets so as to reduce the probability of damage to the total systems by some single event.

Wiring between vital elements of the system outside of equipment housing is routed and protected so as to maintain the true redundancy of the systems with respect to physical hazards.

Loss of Power

A loss of power in the Reactor Protective System causes the affected channel to trip. All bistables operate in a normally energized state and go to a de-energized state to initiate action. Loss of power thus automatically forces the bistables into the tripped state.

Availability of power to the engineered safety features instrumentation is continuously indicated. The loss of instrument power to the sensors, instruments, logic or actuating devices in the engineered safety features instrumentation, starts the engineered safety features equipment associated with the affected channels, except for containment spray which requires instrument power for actuation.

Reactor Trip Signal Testing

Provisions on non-nuclear instrumentation are made to manually place the output of the bistable in a tripped condition for "at power" testing of all portions of each trip circuit including the reactor trip breakers. Administrative procedure requires that the final element in a trip channel (required during power operation) is placed in the trip mode before that channel is taken out of service for repair or testing so that the single failure criterion is met by the remaining channels. In the source and intermediate ranges where the trip logic is one-out-of-two for each range, bypasses are provided for this testing procedure.

Nuclear power range channels are tested by superimposing a test signal on the normal sensor signal so that the reactor trip protection is not bypassed. Based upon coincident log (214) this will not trip the reactor; however, a trip will occur if a reactor trip is required.

Provision is made for the insertion of test signals in each analog loop. Verification of the test signal is made by station instuments at test points specifically provided for this purpose. This enables testing and calibration of meters and bistables. Transmitters and sensors are checked against each other and against precision read-out equipment during normal power operation.

Analog Channel Testing

The basic elements comprising an analog protection channel are shown in Figure 7.2-7, and consist of a transmitter, power supply, bistable, bistable trip switch and proving lamp, test signal injection switch, test signal injection jack, test point, logic test switch, and test operate switch.

Each protection rack will include a test panel containing those switches, test jacks and related equipment needed to test the channels contained in the rack. A hinged cover encloses the test panel. Opening the cover or placing the test-operate switch in the "TEST" position will initiate an alarm. These alarms are arranged on a rack basis to preclude entry to more than one redundant protection rack (or channel) at any time. The test panel cover is designed such that it cannot be closed (and the alarm cleared) unless the test signal plugs (described below) are removed. Closing the test panel cover will mechanically return the test switches to the "OPERATE" position.

Administrative procedures will require that the bistable in the channel under test be placed in the tripped mode prior to test. This places a proving lamp across the bistable output so that the bistable trip point can be checked during channel calibration. The bistable trip switches must be manually reset after completion of a test. Closing the test panel cover will not restore these switches to the untripped mode.

Administrative controls prevent the nuclear instrumentation source range and intermediate range protection channels from being disabled during periodic testing. Power range overpower protection cannot be disabled since this function is not affected by the testing of circuits. Administrative controls also prevent the power range dropped-rod protection from being disabled by testing. In addition, the rod position system would provide indication and associated corrective actions for a dropped rod condition.

Actual channel calibration will consist of injecting a test signal from an external calibration signal source into signal injection jack. Where applicable, the channel power supply will serve as a power source for the calibration source and permit verifying the output load capacity of the power supply. Test points are located in the analog channel and provide an independent means of measuring the calibration signal level.

Logic Channel Testing

The general design features of the logic system are described below. The trip logic channels for a typical two-out-of-three and a two-out-of-four trip function are shown in Figure 7.2-8. The analog portions of these channels are shown in Figure 7.2-9. Each bistable drives two relays ("A" & "B" for level and "C" & "D" for pressure). Contacts from the "A" and "C" relays are arranged in a 2/3 and 2/4 trip matrix for Trip Breaker 1. The above configuration is duplicated for Trip Breaker 2 using contacts from the "B" and "D" relays. A series configuration is used for the trip breakers since they are actuated (opened) by undervoltag coils. This approach is consistent with a de-energize-to-trip preferred failure mode. The planned logic system testing includes exercising the reactor trip breakers to demonstrate system integrity. By-pass breakers are provided for this purpose. During normal operation, these by-pass breakers are open. Administrative control will be used to minimize the amount of time these breakers are closed. Closure of the breaker is controlled from its respective logic test panel in the control room. An interlock is provided that trips both by-pass breakers open if a second by-pass breaker is closed. The status of the breaker is indicated in the control room by indicating lights.

As shown in Figure 7.2-8, the trip signal from the logic network is simultaneously applied to the main trip breaker associated with the specific logic chain as well as the by-pass breaker associated with the alternate trip breaker. Should a valid trip signal occur while AB-1 is by-passing TB-1, TB-2 will be opened through its associated logic train. The trip signal applied to TB-2 is simultaneously applied to AB-1 thereby opening the by-pass around TB-1. TB-1 would either have been opened manually as part of the test or would be opened through its associated logic train which would be operational or tripped during a test.

An auxiliary relay is located in parallel with the undervoltage coils of the trip breakers. This relay is tied to an event recorder which is used to indicate transmission of a trip signal through the logic network during testing. Lights are also provided to indicate the status of the individual logic relays.

The following procedure illustrates the method used for testing Trip Breaker No. 1 and its associated logic network.

a. With the by-pass breaker (AB-1) racked-out, manually set and trip AB-1 to verify operation.

b. Rack-in and set AB-1. Trip TB-1.

c. Sequentially de-energize the trip relays (A1, A2, A3) for each logic combination (1-2, 1-3, 2-3). Verify that the logic network de-energizes the UV coil on TB-1 for each logic combination. Since the event recorder monitors the signal applied to the UV coil, operation of the UV coil can be determined from the event recorder.

d. Repeat "C" for every logic combination in each matrix.

e. Reset TB-1.

f. Trip TB-1 to validate prior test results as evidenced by the event recorder.

g. Reset TB-1. Trip and rack-out AB-1.

In order to minimize the possibility of operational errors from either the standpoint of tripping the reactor inadvertently or only partially checking all logic combinations, each logic network includes a logic channel test panel. This panel includes those switches, indicators and recorders needed to perform the logic system test. The front panel arrangement is shown in Figure 7.2-10. The test switches used to de-energize the trip bistable relays operate through inter-posing relays as shown in Figure 7.2-7 and 7.2-9. This approach avoids violating the separation philosophy used in the analog channel design. Thus, although test switches for redundant channels are conveniently grouped on a single panel to facilitate testing, physical and electrical isolation of redundant protection channels are maintained by the inclusion of the interposing relay which is actuated by the logic test switches.

Primary Power Source

The primary source of control power for the Reactor Protective System is the vital instrument buses described in Section 8. The source of power for the measuring elements and the actuation circuits in the engineered safety features instrumentation is also from those buses.

Protective Actions

Reactor Trip Description

Rapid reactivity shutdown is provided by the insertion of RCC assemblies by gravity fall to compensate for fast reactivity effects, e.g., Doppler and moderator temperature effects. Duplicate series-connected circuit breakers supply all power to the control rod drive mechanisms. The full length control rods must be energized to remain withdrawn from the core. The full length RCC assemblies are dropped upon the loss of power to the control rods. The trip breakers are opened by the under-voltage coils on both breakers (normally energized) which become de-energized by any of several trip signals.

The electrical state of the devices providing signals to the circuit breaker undervoltage trip coils is such as to cause these coils to trip the breaker in the event of reactor trip or power loss.

Certain reactor trip channels are automatically bypassed at low power where they are not required for safety. Nuclear source range and intermediate range trips which are specifically provided for protection at low power or subcritical operation are bypassed by operator manual action after receiving a permissive signal from the next higher range of instrumentation to establish operational status for permit low power operation.

During power operation, a sufficiently rapid shutdown capability in the form of RCC assemblies is administratively maintained through the control rod insertion limit monitors. Administrative control requires that all shutdown rods be in the fully withdrawn position during power operation.

A resume of reactor trips, means of actuation and the coincident circuit requirements is given in Table 7.2-1. The permissive circuits, referred to (e.g. P-7) are listed in Table 7.2-2.

Manual Trip

The manual actuating devices are independent of the automatic trip circuitry, and are not subject to failures which make the automatic circuitry inoperable. Either of two manual trip devices located in the control room will initiate a reactor trip.

High Nuclear Flux (Power Range) Trip

This circuit trips the reactor when two of the four power range channels read above the trip set-point. There are two independent trip settings, a high and a low setting. The high trip setting provides protection during normal power operation. The low setting, which provides protection during startup, can be manually bypassed when two out of the four power range channels read above approximately 10% power (P-10). Three out of the four channels below 10% automatically reinstates the trip protection. The high setting is always active.

High Nuclear Flux (Intermediate Range) Trip

This circuit trips the reactor when one out of the two intermediate range channels reads above the trip set-point. This trip, which provides protection during reactor startup, can be manually bypassed if two out of four power range channels are above approximately 10% (P-10). Three out of four channels below this value automatically reinstates the trip protection. The intermediate channels (including detectors) are separate from the power range channels.

High Nuclear Flux (Source Range) Trip

This circuit trips the reactor when one of the two source range channel count levels (neutraflux) reads above the level trip set-point. The trip, which provides protection during reactor startup, can be manually bypassed when one of two intermediate range channels reads above the P-6 set-point value and is atuomatically reinstated when both intermediate range channels decrease below this value (P-6). This trip is also bypassed by two out of four high power range signals (P-10). It can also be reinstated below P-10 by an administrative action requiring coincident manual actuation.

The trip point is set between the source range cutoff power level and the maximum source range power level.

Overtemperature ΔT Trip

The purpose of this trip is to protect the core against DNB. This circuit trips the reactor on coincidence of two out of the four signals, with two channels (two sets of temperature measurements) per loop. The set point

For this reactor trip is continuously calculated for each channel by solving equations of this form:

$$\Delta T_{\text{set point}}' = \Delta To \left[K_1 - K_2 (T_{avg} - T') + K_3 (P - P') - f(\Delta I)\right]$$

Where:

T_{avg} = average coolant temperature (two measurements in each loop), (°F)

P = pressurizer pressure, four independent measurements (psia)

K_1 = set point bias (°F)

K_2, K_3 = constants based on the effect of temperature and pressure on the DNB limits

$f(\Delta I)$ = same reset function described above in conjunction the overpower ΔT trip

P' = nominal pressure at rated power

Overpower ΔT Trip

The purpose of this trip is to protect against excessive power (fuel rod rating protection). This circuit trips the reactor on coincidence of two out of the four signals, with two channels (two sets of temperature measurements) per loop.

The set point for this reactor trip is continuously calculated for each channel by solving equations of the form:

$$\Delta T_{\text{set point}} = \Delta To \left[K_4 - K_5 \frac{dT_{avg}}{dt} - K_6 (T_{avg} - T') - f(\Delta I)\right]$$

Where: T' = Indicated T_{avg} at nominal conditions at rated power, °F

K_4 = Set point bias

K_5 = Constant

K_6 = Constant

$f(\Delta I)$ = even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup test.

Low Pressurizer Pressure Trip

The purpose of this circuit is to protect against excessive core steam voids which could lead to DNB. The circuit trips the reactor on coincidence of two out of the four low pressurizer pressure signals. This trip is blocked when any three of the four power range channels and two of two turbine first stage pressure channels reads below approximately 10% power (P-7).

High Pressurizer Pressure Trip

The purpose of this circuit is to limit the range of required protection from the overtemperature ΔT trip and to protect against Reactor Coolant System over-pressure. This circuit trips the reactor on coincidence of two out of the three high pressurizer pressure signals.

High Pressurizer Water Level Trip

This trip is provided as a backup to the high pressurizer pressure trip. The coincident of two out of the three high pressurizer water level signals trips the reactor. This trip is bypassed when any three of the four power range channels and two of the two turbine first stage pressure channels read below approximately 10% power (P-7).

Low Reactor Coolant Flow Trip

This trip protects the core from DNB following a loss of coolant flow accident. The means of sensing a loss of coolant flow accident are as follows:

a. Measured low flow in the reactor coolant loop

The low flow trip signal is actuated by the coincidence of 2/3 signals for any reactor coolant loop. The loss of flow in any two loops causes a reactor trip in the power range above approximately 10% (P-7). Above 70% power, the loss of flow in any loop causes a reactor trip (P-8). The instrument used for flow measurement is an elbow tap and is discussed in Section 4.

b. Reactor Coolant Pump circuit breaker open

1. Undervoltage on any two of the four reactor coolant pump buses will cause a reactor trip above approximately 10% power (P-7). Undervoltage on any single bus will cause a reactor trip above approximately 70% power (P-8)

2. Underfrequency on any two of our four reactor coolant pump buses will trip all four reactor coolant pumps and cause a reactor trip above approximately 10% power (P-7).

c. Undervoltage on any two of the four reactor coolant pump buses causes a direct reactor trip above approximately 10% power (P-7).

Below the permissive power set point P-8 and above the permissive set point P-7, a loss of flow in two loops would cause a reactor trip. This permits an orderly plant shutdown under administrative control following a single loop loss of flow during low power operation. Since the plant will not be maintained in operation above permissive power setting P-7 without three loops in service, independent accidents simultaneous with a single loop loss of flow at low power are not considered in the Protection System design.

Safety Injection System (SIS) Actuation Trip

A reactor trip occurs when the safety injection system is actuated. The means of actuating the SIS trips are:

a. Low pressurizer pressure in coincidence with low pressurizer water level (1/3 pairs - a pair is defined as a pressure signal and its assigned level signal).

b. High containment pressure (2/3, Hi level) set at approximately 10% of containment design pressure.

c. High differential pressure between any two steam generators (2/3).

d. High steam flow (2/4).

e. Manual.

Turbine Generator Trip

A turbine trip is sensed by two out of three signals from auto stop oil pressure. A turbine trip is accomplished by a direct reactor trip and a controlled short term release of steam to the condenser which removes sensible heat from the reactor coolant system while avoiding steam generator safety valve actuation.

In addition to speed and load control, the turbine control system automatically trips the turbine generator under any of the following conditions:

a. Turbine overspeed
b. Excessive shaft vibration
c. Generator electrical faults
d. Low condenser vacuum

e. Thrust bearing failure

f. Low bearing oil pressure

g. Low control oil pressure

h. Low feedwater flow

i. Reactor trip

j. Manual trip

Further details are discussed in Section 10.

Steam/Feedwater Flow Mismatch Trip

This trip protects the reactor from a sudden loss of heat sink The trip is actuated by a steam/feedwater flow mismatch (1/2) in coincidence with low water level (1/2) in any steam generator.

Low-Low Steam Generator Water Level Trip

The purpose of this trip is to protect the steam generators for the case of a sustained steam/feedwater flow mismatch. The trip is actuated on two out of the three low-low water level signals in any steam generator. A diagram of the steam generator level control and protection system is shown in Figure 7.2-13.

Rod Stops

A list of rod stops is listed in Table 7.2-3. Some of these have been previously noted under permissive circuits, but are listed again for completeness.

Rod Drop Protection

Two independent systems are provided to sense a dropped rod, (1) a rod bottom position detection system and (2) a system which senses sudden reduction in out-of-core neutron flux. Both protection systems initiate protective action in the form of blocking automatic rod withdrawal and also turbine load cutback if above a given power level. This action compensates for possible adverse core power distributions and permits an orderly retrieval of the dropped RCC.

The primary protection for the dropped RCC accident is the rod bottom signal derived for each rod from its individual position indication system. With the position indication system, initiation of protection is independent of rod location or reactivity worth.

Backup protection is provided by use of the out-of-core power range nuclear detectors and is particularly effective for larger nuclear flux reductions occurring in the region of the core adjacent to the detectors..

The rod drop detection circuit from nuclear flux consists basically of a comparison of each ion chamber signal with the same signal taken through a first order lag network. Since a dropped RCC assembly will rapidly depress the local neutron flux, the decrease in flux will be detected by one or more of these four sensors.

Such a sudden decrease in ion chamber current will be seen as a difference signal. A negative signal output greater than a preset value (approximately 10 per cent) from any one of the four power range channels will actuate the rod drop protection.

Figure 7.4-2 indicates schematically the dropped rod detection circuit and the Nuclear Protection System in general. The potential consequences of any dropped RCC without protective action are limited to localized fuel failure, and the integrity of the Reactor Coolant System is maintaine

Alarms

Any of the following conditions actuate an alarm:

a. Reactor trip (first-out annunciator)
b. Trip of any reactor trip channel
c. Actuation of any permissive circuit or override
d. Significant deviation of any major control variable (pressure, T_{avg}, pressurizer water level, and steam generator water level)

Control Group Rod Insertion Limits

The lower insertion limit system is used in an administrative control procedure with the objective to maintain an RCCA shutdown margin.

The control group rod insertion limits, Z_{LL}, are calculated as a linear function of reactor power and reactor coolant average temeprature. The equation is:

$$Z_{LL} = A\ (\Delta T)_{avg} + B\ \overline{(T_{avg})} + C$$

where A, B are preset manually adjustable gains and C is a preset manually adjustable bias. The $(\Delta T)_{avg}$ and $\overline{(T_{avg})}$ are the average of the individual temperature differences and the coolant average temperatures respectively measured from the reactor coolant hot leg and the cold leg.

One insertion limit monitor with two alarm set points is provided for each control bank. A description of control and shutdown rod groups is provided in section 7.3 under System Design. The "Low" alarm alerts the operator of an approach to a reduced shutdown reactivity situation requiring boron addition by following normal procedures with the Chemical and Volume Control System (Section 9). Actuation of "Low-Low" alarm requires the operator to take immediate action to add boron to the system by any one of several alternate methods.

7.2.3 SYSTEM EVALUATION

Reactor Protection System and DNB

The following is a description of how the reactor protection system prevents DNB.

The plant variables affecting the DNB ratio are:

- Thermal Power
- Coolant flow
- Coolant temperature
- Coolant pressure
- Core power distribution (hot channel factors)

Figure 7.2-11 illustrates the core limits for which DNBR for the hottest rod is 1.30 and shows the overpower and overtemperature ΔT reactor trips locus as a function of T_{avg} and pressure.

Variations in both flow and power are monitored by the overpower and overtemperature ΔT trips since a decrease in flow would have the same effect on the measured loop ΔT signal as an increase in power. It is the nature of the DNB limits that a reduction in flow of 10% would require a reduction in power of only about 5% to maintain the same DNBR, all other variables remaining constant. Thus, the permissible ΔT increases somewhat at a reduced flow. The trip setpoints are therefore set for a maximum flow. A reduction in flow increases the margin between the trip point and the actual core limit. Periodic measurements using the in-core instrumentation system are used to verify that the actual core power distributions are within design limits.

Reactor trips for a fixed high pressurizer pressure and for a fixed low pressurizer pressure are provided to limit the pressure range over which core protection depends on the variable overpower and overtemperature high ΔT trips.

Reactor trips on nuclear overpower and low reactor coolant flow are provided for direct, immediate protection against rapid changes in these variables. However, for all cases in which the calculated DNBR approaches 1.30, a reactor trip on overpower and/or overtemperature ΔT would be actuated.

The Reactor Protection System actuates a reactor trip for a set of conditions for which the calculated DNBR for the worst fuel rod approaches 1.30. Because of the statistical nature of the W-3 correlation and the statistical makeup of a portion of the hot channel factors there exists a finite probability of a few rods being in DNB for a calculated ratio of 1.30 for the worst fuel rod. (Reference is made to Section 3).

Two points are made in this regard however:

1. For the anticipated abnormal conditions it is highly unlikely that the exact combination of conditions (reactor coolant pressure, temperature and core power, instrumentation inaccuracies etc.) exists to even approach a DNBR of 1.30 before a reactor trip. The accident condition most likely to approach a DNBR of 1.30 for the calculated worst fuel rod is the simultaneous loss of power to the reactor coolant pumps. In any event the DNBR at the worst fuel rod is near 1.30 for only a few seconds.

2. The hottest rods are not adjacent to one another but are located near the spare RCC thimbles and neighboring rods have higher DNB ratios.

The ΔT trip functions are based on the differences between measurements of the hot leg and cold leg temperatures, which are proportional to core power.

The ΔT trip functions are provided with a nuclear flux feedback to reflect a measure of axial power distribution. This will assist in preventing an adverse distribution which could lead to exceeding allowable core conditions.

The reactor is equipped with part length control rods which are manually controlled from the control room and which provide flexibility in the shaping of core axial power distribution, thus improving the capability of maintaining desired core conditions while providing the means to damp out potential axial xenon oscillations.

Overpower Protection

In addition to the high power range nuclear flux trips, a overpower ΔT trip is provided (2 out of 4 logic) to limit the maximum overpower. This trip is modified by an axial power distribution function of the following form:

$$\Delta T_{set\ point} = \Delta To \left[K_4 - K_5 \frac{dT_{avg}}{dt} - K_6 \left(T_{avg} - T' \right) - f(\Delta I) \right]$$

where:

ΔTo = Indicated ΔT at rated power T

T_{avg} = Average temperature, °F

T' = Indicated T_{avg} at nominal conditions at rated power, °F

K_4 = Set point bias

K_5 = Constant

K_6 = Constant

$f(\Delta I)$ = even function of the indicated difference between top and bottom detectors of the power range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup test.

Four long ion chamber pairs are provided and each one independently feeds a separate overpower ΔT trip channel. Thus, a single failure neither defeats the function nor causes a spurious trip. The reset function is only in the direction of decreasing the trip setpoint; it cannot increase the setpoint. As shown above, if the difference between the top and bottom detectors exceeds a preset limit indicative of excess power generation in either half of the core, a proportional signal is transmitted to the overpower ΔT trip to reduce its setpoint.

In addition, a rod stop function and turbine runback function is provided in the form:

$$\Delta T_{rod\ stop} = \Delta T_{trip} - B_p$$

B_p = set point bias (°F) with a programmed turbine runback until $\Delta T_{trip} < \Delta T_{rod\ stop}$

This function serves to maintain essentially a constant margin to trip and give the operator the opportunity to adjust the part length rods to reshape the flux before a reactor trip occurs.

Overtemperature Protection

A second ΔT trip (2 out of 4 logic) provides an overtemperature trip which is a function of coolant average temperature and pressurizer pressure derived as follows:

$$\Delta T_{trip} = \Delta T_o \left[K_1 - K_2 (T_{avg} - T') + K_3 (P - P') - f(\Delta I) \right]$$

where

T_{avg} = reactor coolant average temperature, two measurements in each loop

T' = indicated T_{avg} at nominal condition at rated power

P = pressurizer pressure, four independent measurements

P' = nominal pressure at rated power

K_1 = set point bias

K_2, K_3 = constants based on the effect of temperature and pressure on the DNB limits

$f(\Delta I)$ = same reset function described above in conjunction with the overpower ΔT trip

A similar rod stop and turbine runback function is provided in the form:

$$\Delta T_{rod\ stop} = \Delta T_{trip} - B_T$$

B_T = set point bias (°F)

In summary, the operator manually adjusts part length rods on an infrequent basis using long ion chamber information displayed on the control board to maintain the difference between top and bottom detectors within a desired range. In the event the difference exceeds the desired range, automatic feedback signals are provided to reduce the overpower-overtemperature trip setpoints, block rod withdrawal and reduce load to maintain appropriate operating margins to the trip setpoint.

Interaction of Control and Protection

The design basis for the control and protection system permits the use of a detector for both protection and control functions. Where this is done, all equipment common to both the protection and control circuits are classified as part of the protection ystem. Isolation amplifiers prevent a control system failure from affecting the protection system. In addition, where failure of a protection system component can cause a process excursion which requires protective action, the protection system can withstand another, independent failure without loss of function. Generally, this is accomplished with two-out-of-four trip logic. Also, wherever practical, provisions are included in the protection system to prevent a plant outage because of single failure of a sensor.

Specific Control and Protection Interactions

Nuclear Flux

Four power-range nuclear flux channels are provided for overpower protection. Isolated outputs from all four channels are averaged for automatic control rod regulation of power. If any channel fails in such a way as to produce a low output, that channel is incapable of proper overpower protection. In principle, the same failure would cause rod withdrawal and overpower. Two-out-of-four overpower trip logic will ensure an overpower trip if needed even with an independent failure in another channel.

In addition, the control system will respond only to rapid changes in indicated nuclear flux; slow changes or drifts are overriden by the temperature control signals. Also, a rapid decrease of any nuclear flux signal will block automatic rod withdrawal as part of the rod drop protection circuitry. Finally, an overpower signal from any nuclear channel will block automatic rod withdrawal. The set point for this rod stop is below the reactor trip set point.

Coolant Temperature

Four T_{avg} channels are used for overtemperature-overpower protection. (See Figure 7.2-12 for single channel). Isolated output signals from all four channels are also averaged for automatic control rod regulation of power and temperature. In principle, a spuriously low temperature signal from one sensor would partially defeat this protection function and also cause rod withdrawal and overtemperature.

In addition, channel deviation alarms in the control system will block automatic rod motion (insertion or withdrawal) if any temperature channel deviates significantly from the others. Automatic rod withdrawal blocks will also occur if any one of four nuclear channels indicates an overpower condition or if any one of four temperature channels indicates an over-temperature condition. Two-out-of-four trip logic is used to ensure that an overtemperature trip will occur if needed even with an independent failure in another channel. Finally, as shown in Section 14.1, the combination of trips on nuclear overpower, high pressurizer water level, and high pressurizer pressure also serve to limit an excursion for any rate of reactivity insertion.

Pressurizer Pressure

Four pressure channels are used for high and low pressure protection and for overpower-overtemperature protection. Isolated output signals from these channels also are used for pressure control and compensation signals for rod control. These are discussed separately below:

(1) Control of rod motion: one of the pressure channels is used for rod control with a low pressure signal acting to withdraw rods. The discussion for coolant temperature is applicable, i.e., two-out-of-four logic for overpower-overtemperature protection as the primary protection, with backup from multiple rod stops and "backup" trip circuits. In addition, the pressure compensation signal is limited in the control system such that failure of the pressure signal cannot cause more than about a 10°F change in T_{avg}. This change can be accommodated at full power without a DNBR less than 1.30. Finally, the pressurizer safety valves are adequately sized to prevent system overpressure.

(2) Pressure Control: Spray, power-operated relief valves, and heaters are controlled by isolated output signals from the pressure protection channels.

a) Low Pressure

A spurious high pressure signal from one channel can cause low pressure by spurious actuation of spray and/or a relief valve. Additional redundancy is provided in the protection system to ensure underpressure protection, i.e., two-out-of-four low pressure reactor trip logic and one-out-of-three logic for safety injection. (Safety injection is actuated on one-out-of-three coincident low pressure and low level.)

In addition, interlocks are provided in the pressure control system such that a relief valve will close if either of two independent pressure channels indicates low pressure. Spray reduces pressure at a lower rate, and some time is available for operator action (about three minutes at maximum spray rate before a low pressure trip is reached.)

b) High Pressure

The pressurizer heaters are incapable of overpressurizing the reactor coolant system. Maximum steam generation rate with heaters is about 15,000 lbs/hr., compared with a total capacity of 1,224,000 lbs/hr. for the two safety valves and a total capacity of 358,000 lbs/hr. for the two power-operated relief valves. Therefore, overpressure protection is not required for a pressure control failure. Two-out-of-three high pressure trip logic is therefore used.

In addition, either of the two relief valves can easily maintain pressure below the high pressure trip point. The two relief valves are controlled by independent pressure channels, one of which is independent of the pressure channel used for heater control. Finally, the rate of pressure rise achievable with heaters is slow, and ample time and pressure alarms are available for operator action.

Pressurizer Level

Three pressurizer level channels are used for high level reactor trip (2/3) and low level safety injection (1/3 logic level coincident with pressure). Isolated output signals from these channels are used for volume control, increasing or decreasing water level. A level control failure could fill or empty the pressurizer at a slow rate (on the order of half an hour or more).

(a) High Level

A reactor trip on pressurizer high level is provided to prevent rapid thermal expansions of reactor coolant fluid from filling the pressurizer: the rapid change from high rates of steam relief to water relief can be damaging to the safety valves and the relief piping and pressure relief tank. However, a level control failure cannot actuate the safety valves because the high pressure reactor trip is set below the safety valve set pressures. Therefore, a control failure does not require protection system action.

In addition, ample time and alarms are available for operator action.

(b) Low Level

For control failures which tend to empty the pressurizer, one-out-of-three logic for safety injection actuation on low level coincident with low pressure ensures that the protection system can withstand an independent failure in another channel.

In addition, a signal of low level from either of two independent level control channels will isolate letdown, thus preventing the loss of coolant. Also, ample time and alarms exist for operator action.

Steam Generator Water Level; Feedwater Flow

Before describing control and protection interaction for these channels, it is beneficial to review the protection system basis for this instrumentation.

The basic function of the reactor protection circuits associated with low steam generator water level and low feedwater flow is to preserve the steam generator heat sink for removal of long term residual heat. Should a complete loss of feedwater occur with no protective action, the steam generators would boil dry and cause an overtemperature-overpressure excursion in the reactor coolant. Reactor trips on temperature, pressure, and pressurizer water level will trip the plant before there is any damage to the core or reactor coolant system. However, residual heat after trip would cause thermal expansion and discharge of the reactor coolant to containment through the pressurizer relief valves. Redundant emergency feedwater pumps are provided to prevent this. Reactor trips act before the steam generators are dry to reduce the required capacity and starting time requirements of these pumps and to minimize the thermal transient on the reactor coolant system and steam generators. Independent trip circuits are provided for each steam generator for the following reasons:

1. Should severe mechanical damage occur to the feedwater line to one steam generator, it is difficult to ensure the functional integrity of level and flow instrumentation for that unit. For instance, a major pipe break between the feedwater flow element and the steam generator would cause high flow through the flow element. The rapid depressurization of the steam generator would drastically affect the relation between downcomer water level and steam generator water inventory.

2. It is desirable to minimize thermal transient on a steam generator for credible loss of feedwater accidents.

   It should be noted that controller malfunctions caused by a protection system failure affect only one steam generator. Also, they do not impair the capability of the main feedwater system under either manual control or automatic T_{avg} control. Hence, these failures are far from being the worst case with respect to decay heat removal with the steam generators.

(1) Feedwater Flow

A spurious high signal from the feedwater flow channel being used for control would cause a reduction in feedwater flow and prevent that channel from tripping. A reactor trip on low-low water level, independent of indicated feedwater flow, will ensure a reactor trip if needed.

In addition, the three-element feedwater controller incorporates reset on level, such that with expected gains, a rapid increase in the flow signal would cause only a 12 inch decrease in level before the controller re-opened the feedwater valve. A slow increase in the feedwater signal would have no effect at all.

(2) Steam Flow

A spurious low steam flow signal would have the same effect as a high feedwater signal, discussed above.

(3) Level

A spurious high water level signal from the protection channel used for control will tend to close the feedwater valve. This level channel is independent of the level and flow channels used for reactor trip on low flow coincident with low level.

a) A rapid increase in the level signal will completely stop feedwater flow and actuate a reactor trip on low feedwater flow coincident with low level.

b) A slow drift in the level signal may not actuate a low feedwater signal. Since the level decrease is slow, the operator has time to respond to low level alarms. Since only one steam generator is affected, automatic protection is not mandatory and reactor trip on two-out-of-three low-low level is acceptable.

TABLE 7.2-1

LIST OF REACTOR TRIPS & CAUSES OF ACTUATION OF: ENGINEERED SAFETY FEATURES, CONTAINMENT AND STEAM LINE ISOLATION & AUXILIARY FEEDWATER

REACTOR TRIP	COINCIDENCE CIRCUITRY AND INTERLOCKS	COMMENTS
1. Manual	1/2, no interlocks	
2. Overpower nuclear flux	2/4	High and low settings; manual block and automatic reset of low setting by P-10 Permissive 10, Table 7.2-2
3. Overtemperature ΔT	2/4, no interlocks	
4. (ower ΔT	2/4, no interlocks	
5. Low pressurizer pressure	2/4, blocked by P-7	
6. High pressurizer pressure (fixed set point)	2/3, no interlocks	
7. High pressurizer water level	2/3, blocked by P-7	
8a Low reactor coolant flow	2/3, per loop blocked by P-7, P-8	
8b Reactor coolant pump breaker	1/1, per loop, blocked by P-7, P-8	
8c Undervoltage on reactor coolant pump bus	2/4 blocked by P-7	Reactor coolant pump breaker is tripped on underfrequency
9. Safety injection signal (Actuation)	1/3, (pairs of low pressurizer pressure and low pressurizer water level, manual block permitted by 2/3 low pressurizer pressure): or 2/3 high containment pressure (Hi level): or 2/3 high differential pressure between any two steam generators: or manual 1/2: or 2/4 high steam flow (Blocked by 3/4 high T_{avg} and 3/4 high steam pressure).	

TABLE 7.2-1 (Continued)

REACTOR TRIP	COINCIDENCE CIRCUITRY AND INTERLOCKS	COMMENTS
10. Turbine-generator (Low auto stop oil pressure signal)	2/3, blocked by P-7	
11. Low feedwater flow	1/2 coincidence low steam generator water level and low feedwater flow, any loop	
12. Low-low steam generator water level	2/3, per loop	
13. High intermediate range nuclear flux	1/2, manual block permitted by P-10	Manual block and automatic reset
14. High source range nuclear flux	1/2, manual block permitted by P-6, also blocked by P-10	" " " " "
CONTAINMENT ISOLATION ACTUATION		
15. Safety Injection Signal (Phase A)	See Item 9	Actuates all non-essential service containment isolation trip valves and actuates Isolation Valve Seal Water System
16. Containment pressure (Phase B)	Coincidence of two 2/3 containment pressure (Hi-Hi pressure, same signal which actuates containment spray), or manual 2/2	Actuates all essential service containment isolation trip valves
17. High containment activity	High activity signal, from air particulate detector or radiogas detector. (1/2)	This additional signal closes containment purge supply, exhaust ducts and pressure relief duct only.
ENGINEERED SAFETY FEATURES ACTUATION		
18. Safety injection signal (S)	See Item 9	
19. Containment spray signal (P)	Coincidence of two 2/3 containment pressure (Hi-Hi pressure); or manual 2/2	

TABLE 7.2-1 (Continued)

REACTOR TRIP	COINCIDENCE CIRCUITRY AND INTERLOCKS	COMMENTS
20. Spray additive valves	Coincidence of two 2/3 containment pressure (Hi-Hi pressure, same signal which actuates containment spray)	
ENGINEERED SAFETY FEATURES ACTUATION		
21. Containment air recirculation cooling and filtration signal	Safety injection signal initiates starting of all fans in accordance with the Safety Injection Starting Sequence, 2/3 high containment pressure or manual 1/2	
22. Isolation valve seal water signal	Safety injection signal	
STEAM LINE ISOLATION ACTUATION		
23. Steam flow	High steam flow in 2/4 lines plus (a) low T_{avg} in 2/4 lines or (b) low steam line pressure in 2/4 lines.	
24. Containment pressure	Coincidence of two 2/3 containment pressure (Hi-Hi pressure) (Note: Bistables are energized-to-operate)	
25. Manual	1/1 per steam line	
AUXILIARY FEEDWATER ACTUATION		
26. Turbine driven pump	Coincidence of 2/3 low level in two steam generators; or loss of voltage on 2/2 480 volt buses 5A and 6A and unit trip; or manual 1/2	
27. Motor driven pumps	2/3 low level in any steam generator; or trip of 1/2 main feedwater pump turbines; or safety injection signal; or manual 1/2; or loss of 2/2 480 volt buses 5A and 6A and unit trip	

TABLE 7.2-1 (Continued)

REACTOR TRIP	COINCIDENCE CIRCUITRY AND INTERLOCKS	COMMENTS
MAIN FEEDWATER ISOLATION		
28. Close main feedwater control valves trip main feedwater pumps	Any Safety Injection Signal (see Item 9)	

TABLE 7.2-2

INTERLOCK AND
PERMISSIVE CIRCUITS

Number	Function	Input for Blocking
1	Prevent rod withdrawal on overpower	1/4 high nuclear flux (power range) or 1/2 high nuclear flux (intermediate range or 1/4 overtemperature ΔT or 1/4 overpower ΔT.
2	Auto-rod withdrawal stop at low powers	Low MWe load signal
3	Auto-rod withdrawal stop on rod drop	1/4 rapid decrease of nuclear flux (power range) or 1/1 rod bottom indication
4*		
5	Steam dump interlock	Rapid decrease of MWe load signal
6	Manual block of source range level trip	1/2 high intermediate range flux allows manual block, 2/2 low intermediate range defeats block
7	Permissive power (block various trips required only at power)	3/4 low nuclear flux signals (power range) and 2/2 low turbine first stage pressure signals
8	Block single primary loop loss of flow trip	3/4 low nuclear flux (power range)
9*		
10	Manual block of low trip (power range) and intermediate range trips	2/4 high nuclear flux allows manual block, 3/4 low nuclear flux (power range) defeats manual block

* not applicable to this plant

TABLE 7.2-3

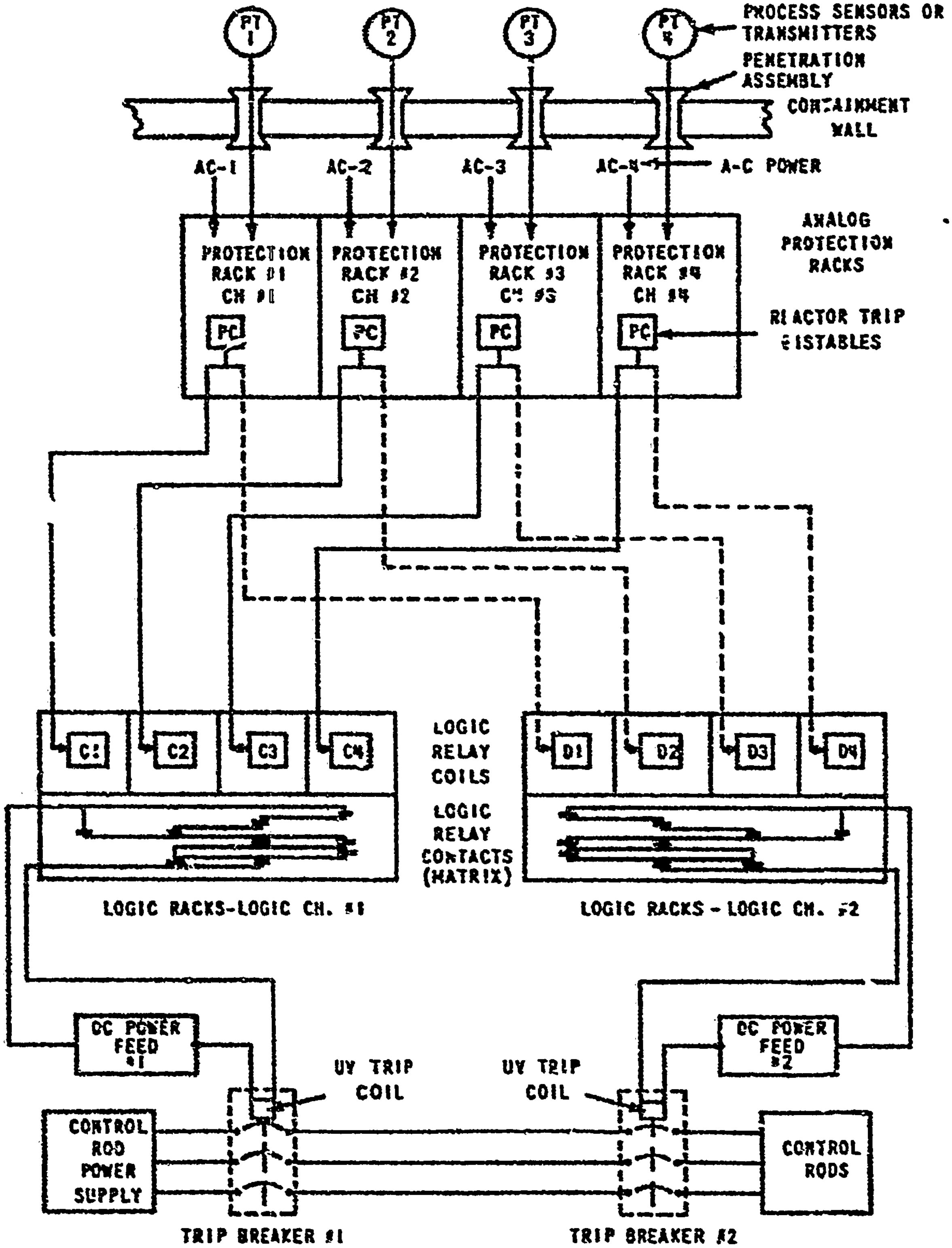
ROD STOPS

Rod Stop	Actuation Signal	Rod Motion to be blocked
1. Rod Drop	1/4 rapid power range nuclear flux decrease or any rod bottom signal	Automatic Withdrawal
2. Nuclear Overpower	1/4 high power range nuclear flux or 1/2 high intermediate range nuclear flux	Automatic and Manual Withdrawal
3. High ΔT	1/4 overpower ΔT or 1/4 over-temperature ΔT	Automatic and Manual Withdrawal

Actuation of rod stop (item 1) initiates a turbine load reduction above a given power level.

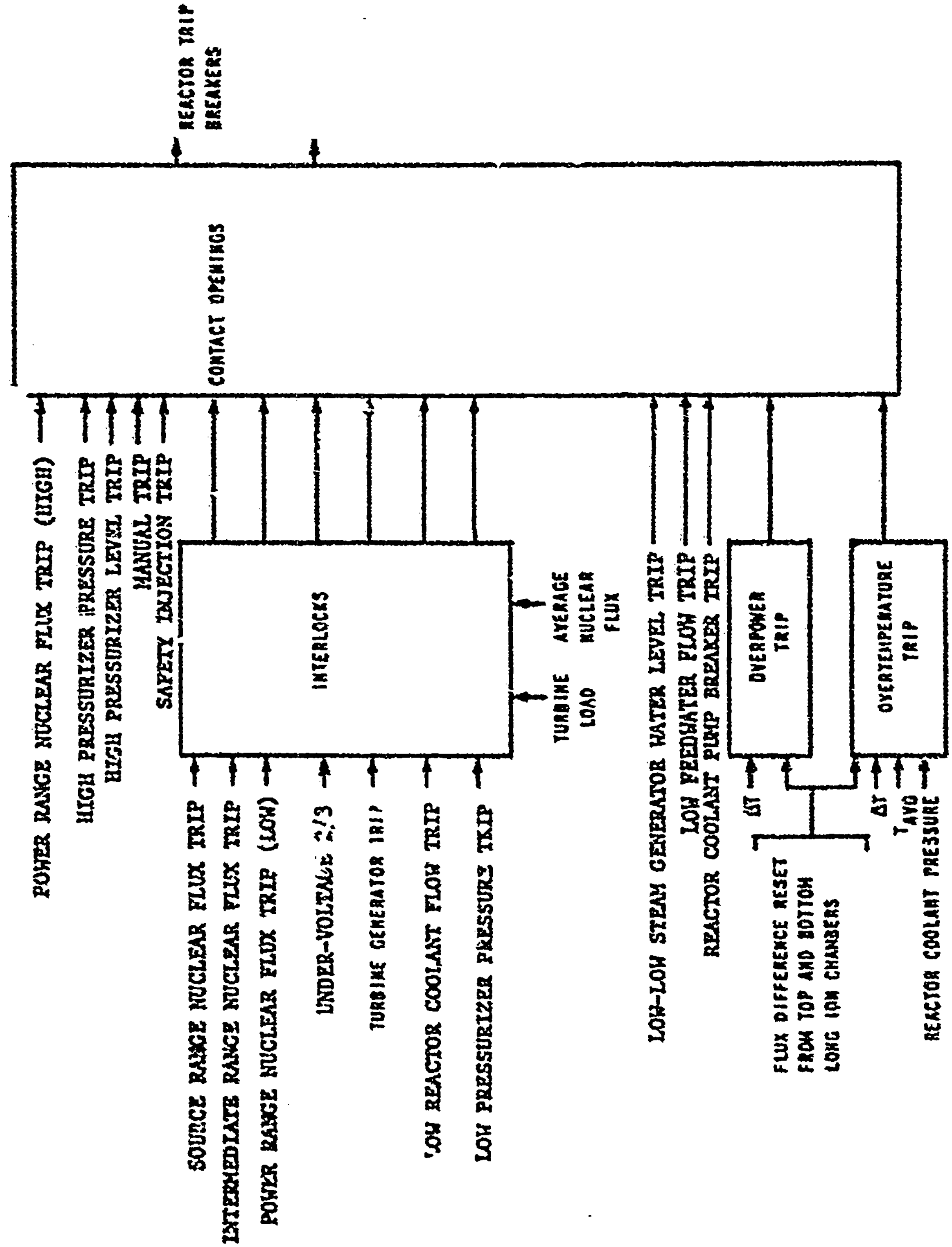
Actuation of rod stop (item 3) initiates a load cutback at any power level.

Low Power	Low first stage pressure turbine load signals	Automatic Withdrawal
5. T_{avg} Deviation	1/4 T_{avg} deviation from average T_{avg}	Automatic Withdrawal and Insertion

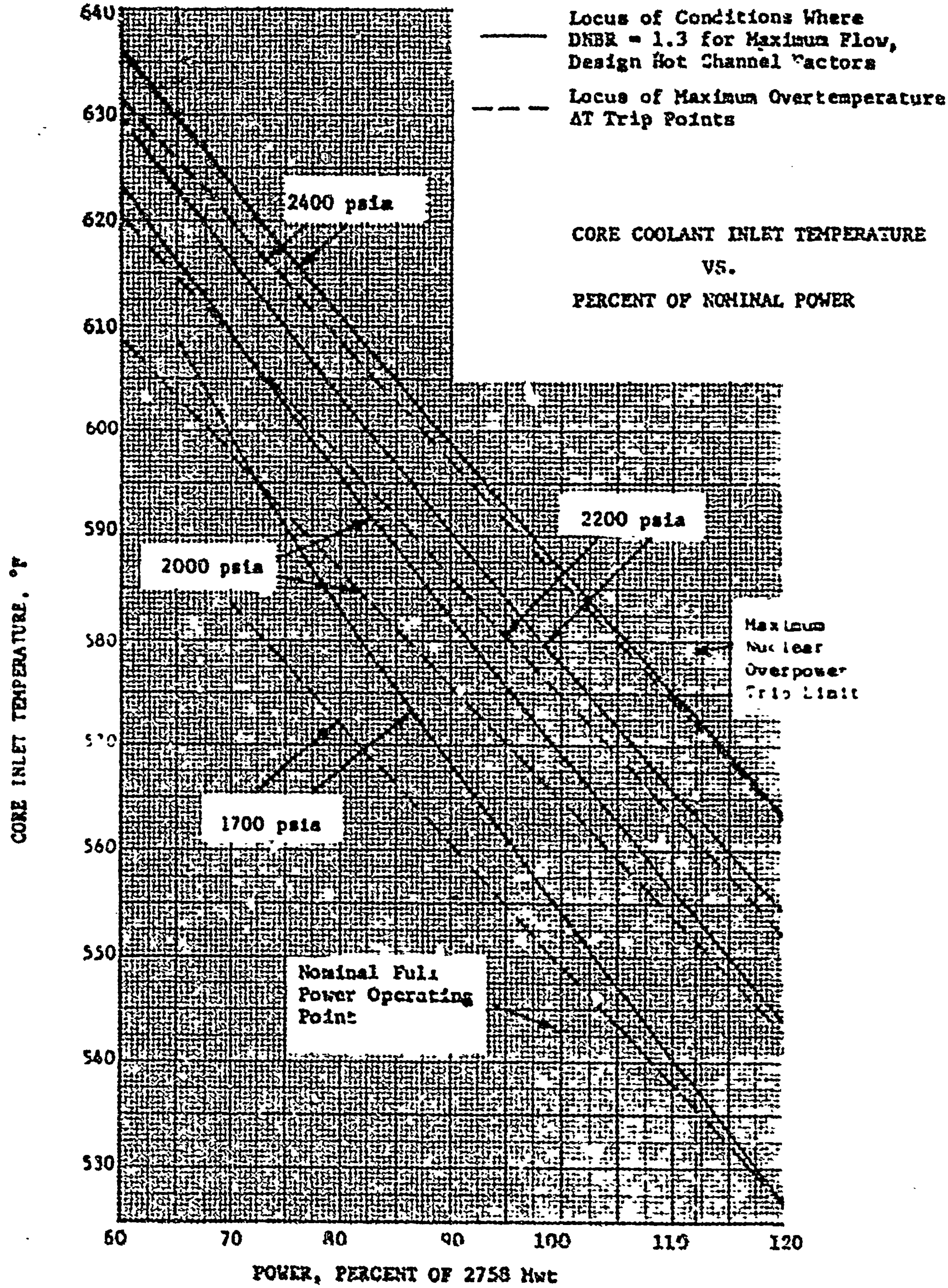


DESIGN PHILOSOPHY TO ACHIEVE ISOLATION BETWEEN CHANNELS

FIG. 7.2-1

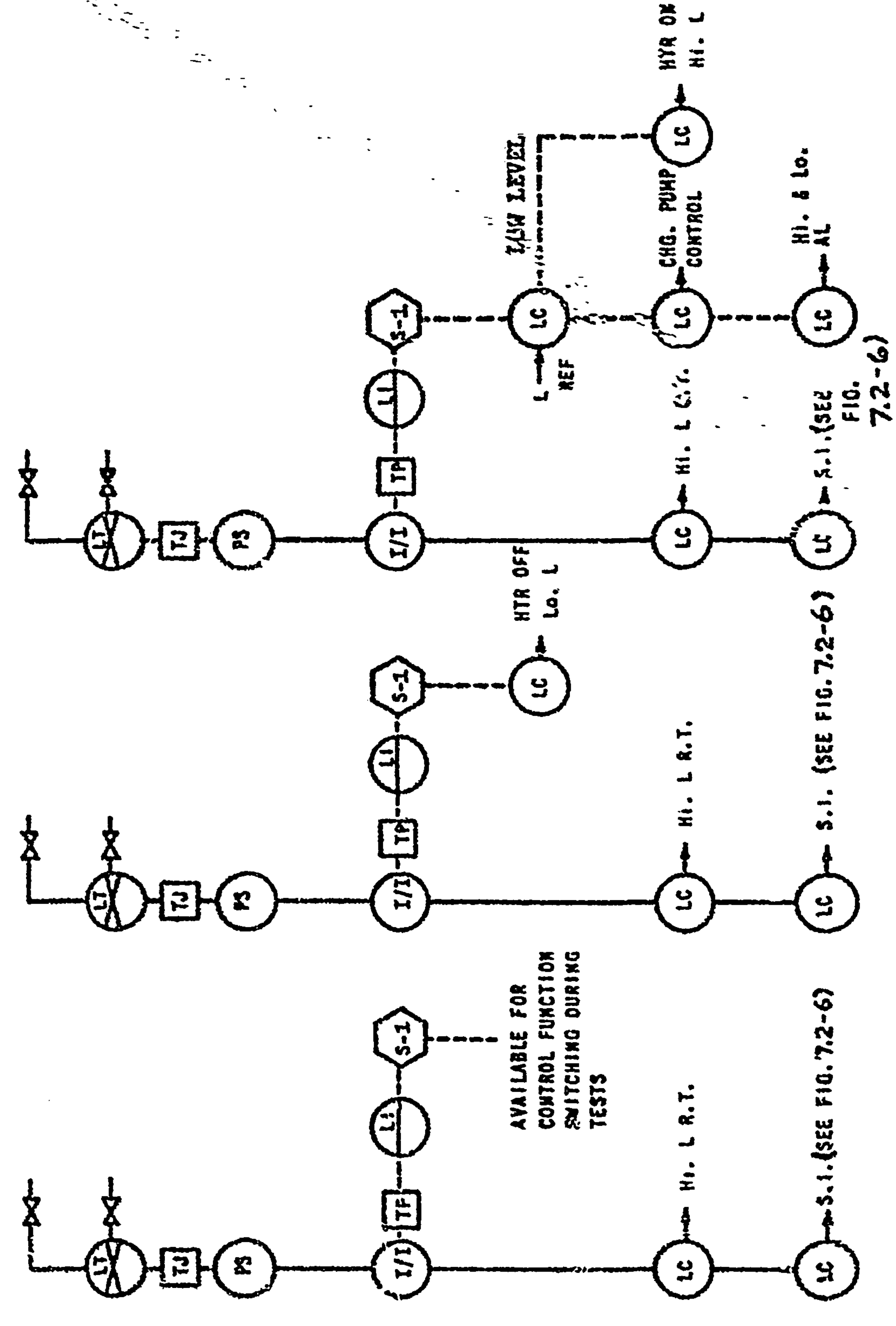


REACTOR PROTECTION SYSTEMS
FIG. 7.2-2

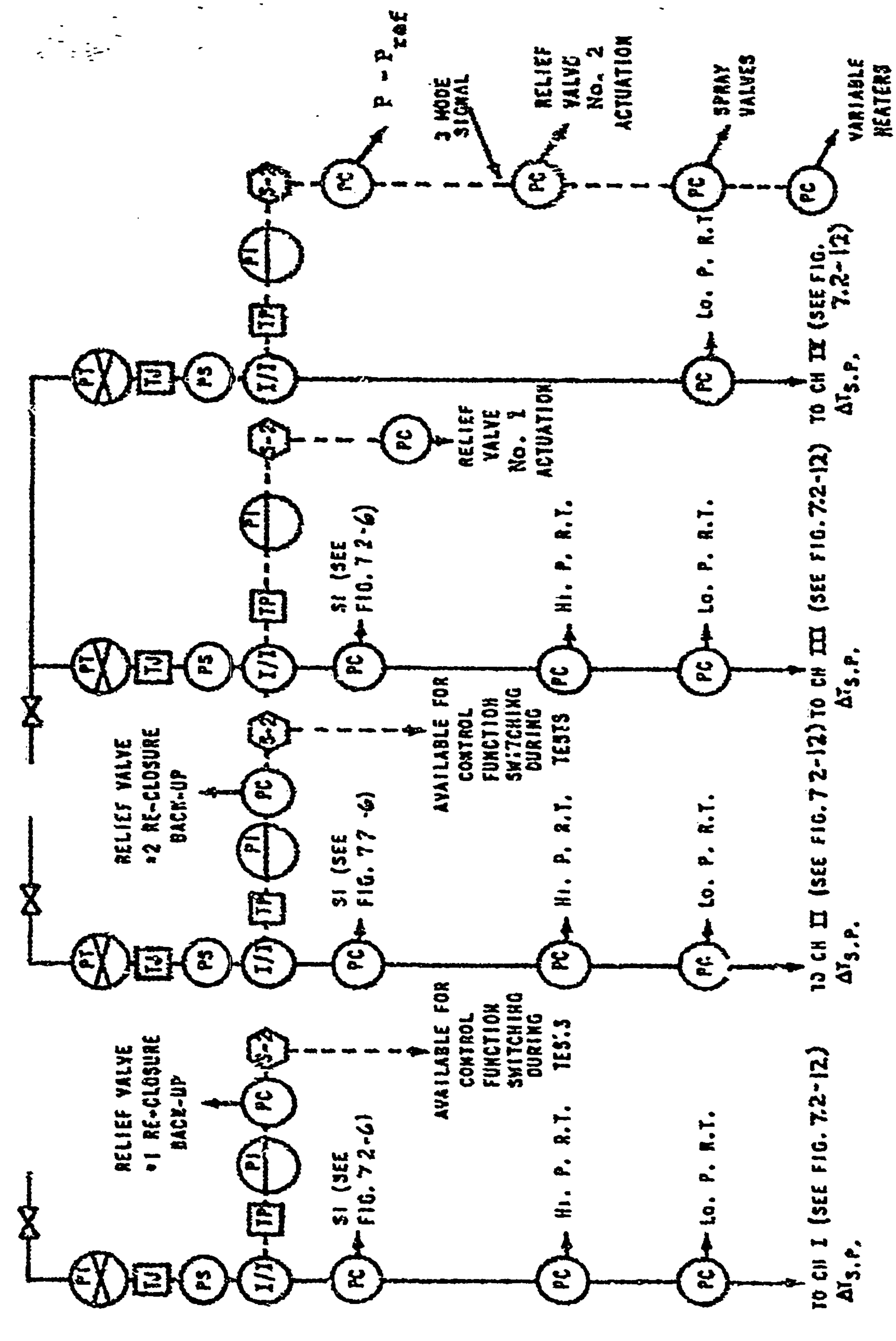


Locus of Conditions Where DNBR = 1.3 for Maximum Flow, Design Hot Channel Factors
Locus of Maximum Overtemperature ΔT Trip Points
CORE COOLANT INLET TEMPERATURE
VS.
PERCENT OF NOMINAL POWER
2400 psia
2200 psia
2000 psia
1700 psia
Maximum Nuclear Overpower Trip Limit
Nominal Full Power Operating Point
CORE INLET TEMPERATURE, °F
640
630
620
610
600
590
580
570
560
550
540
530
60
70
80
90
100
110
120
POWER, PERCENT OF 2758 Mwt

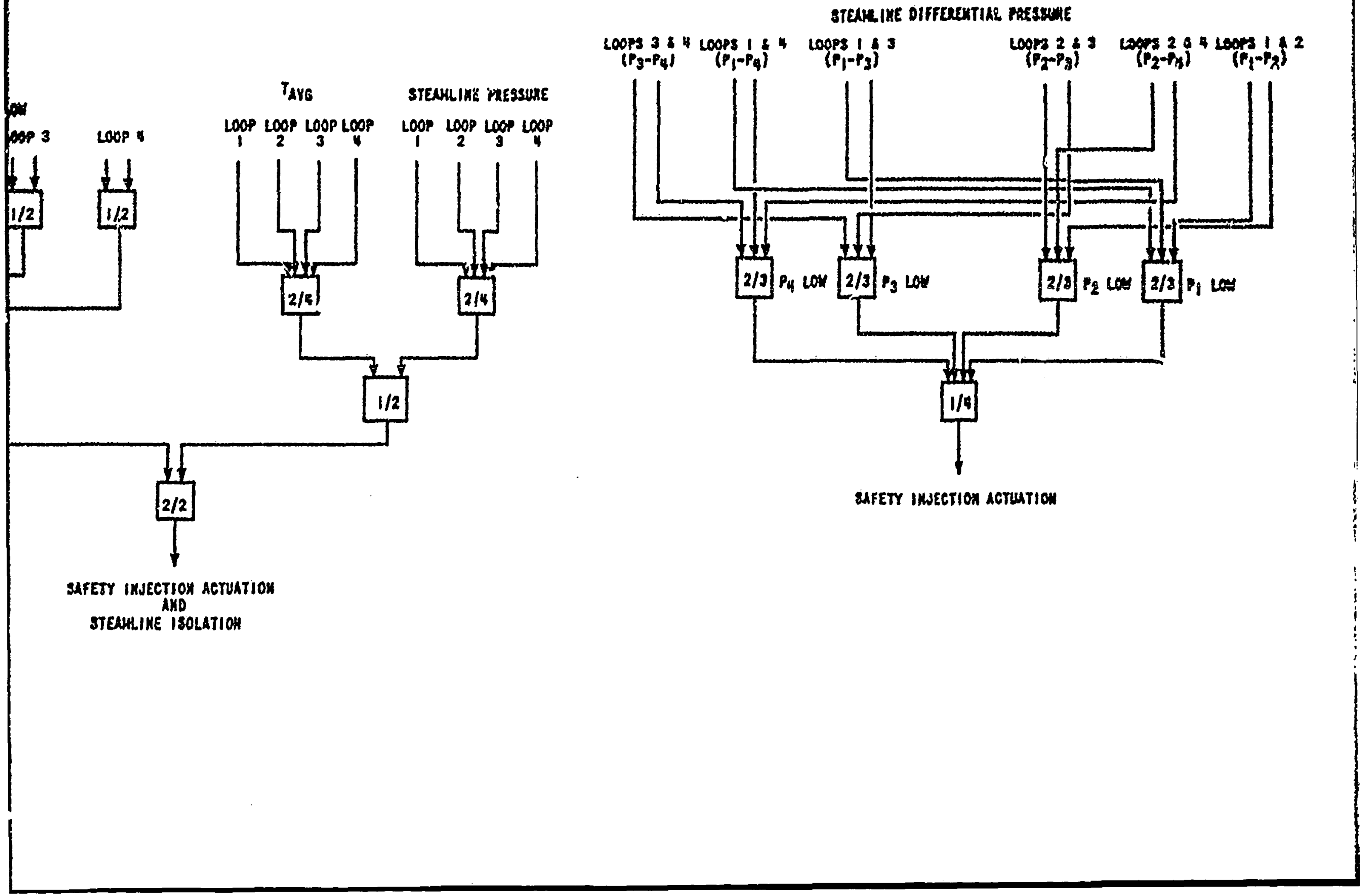
FIGURE 7.2-3

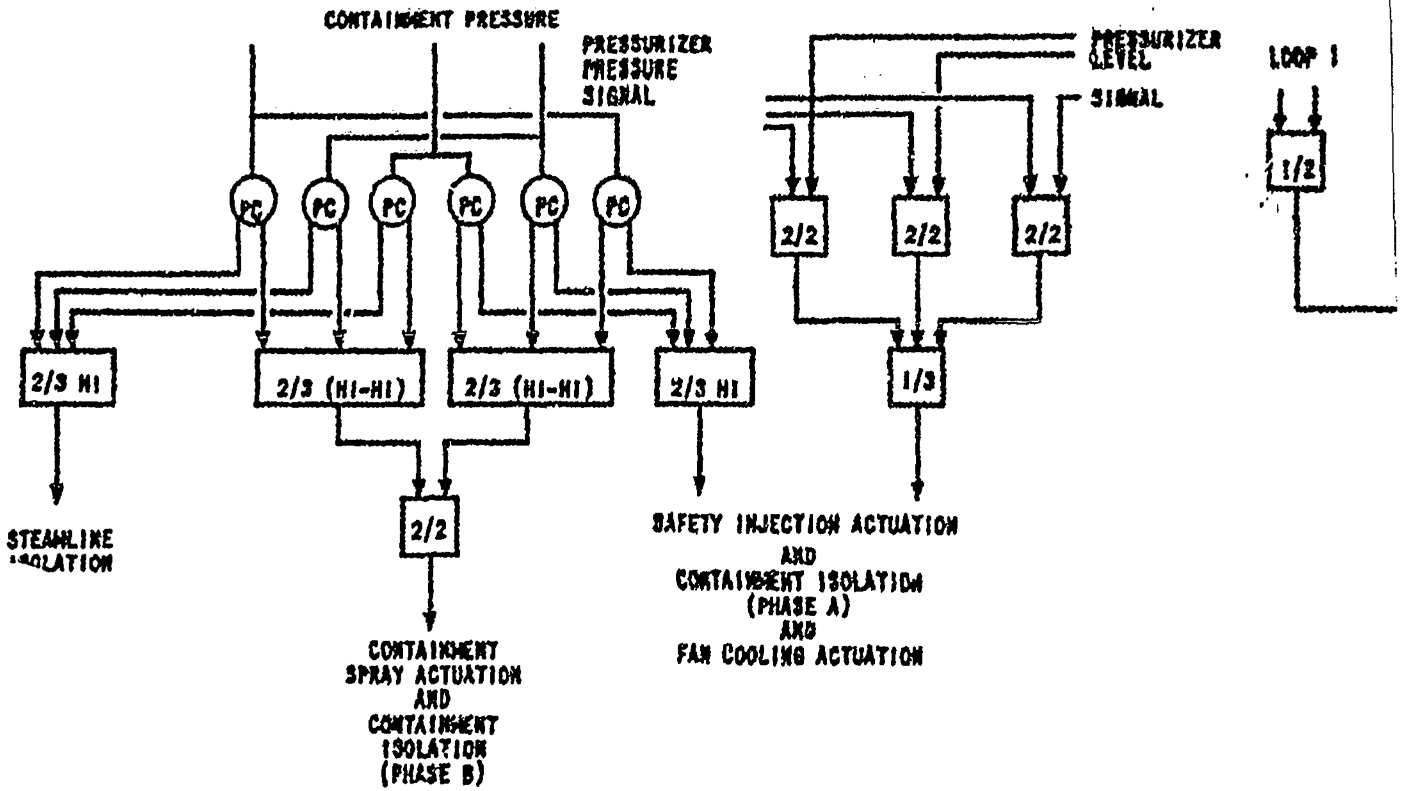


PRESSURIZER LEVEL CONTROL AND PROTECTION SYSTEM
FIG. 7.2-4

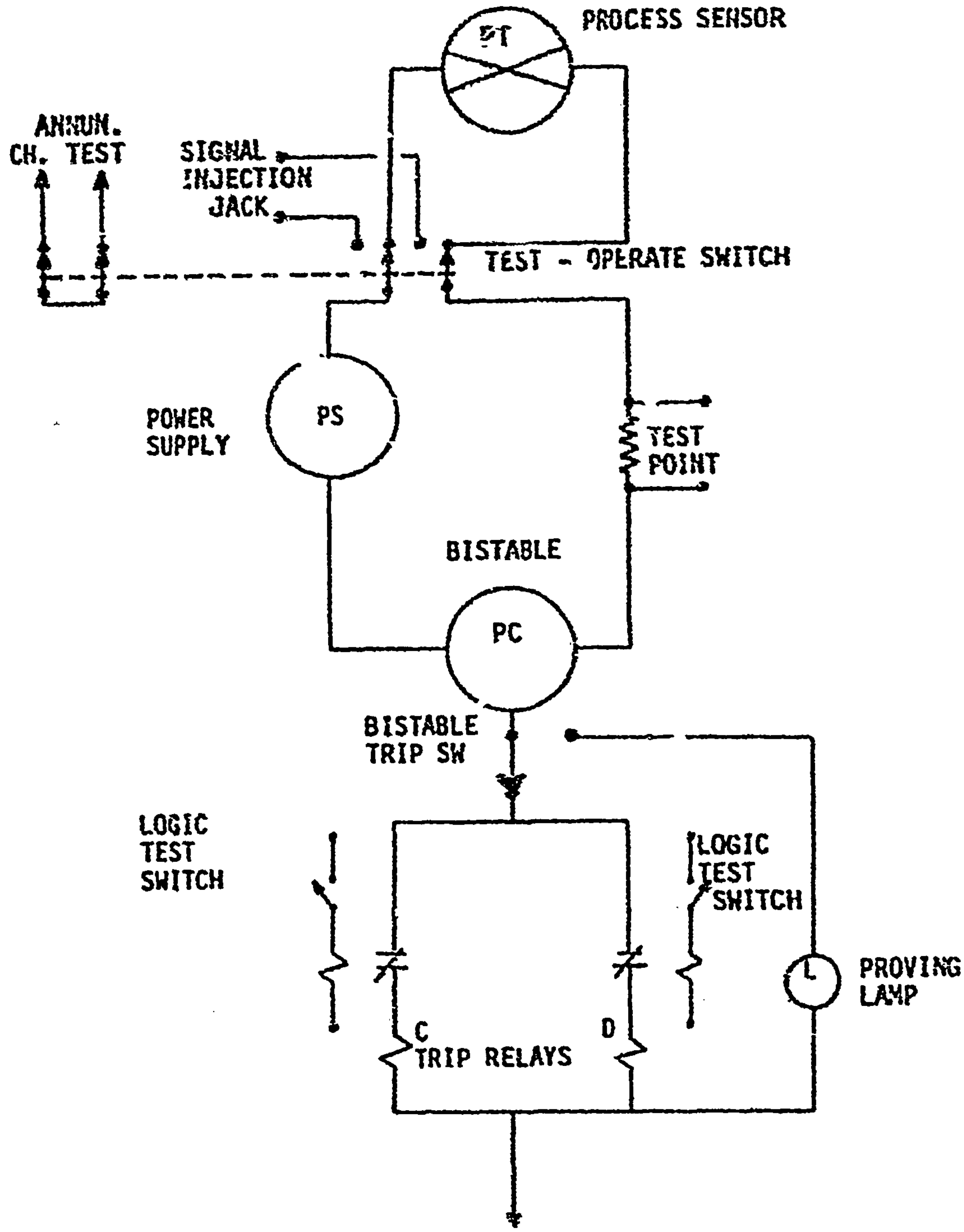


PRESSURIZER PRESSURE CONTROL AND PROTECTION SYSTEM
FIG. 7.2-5

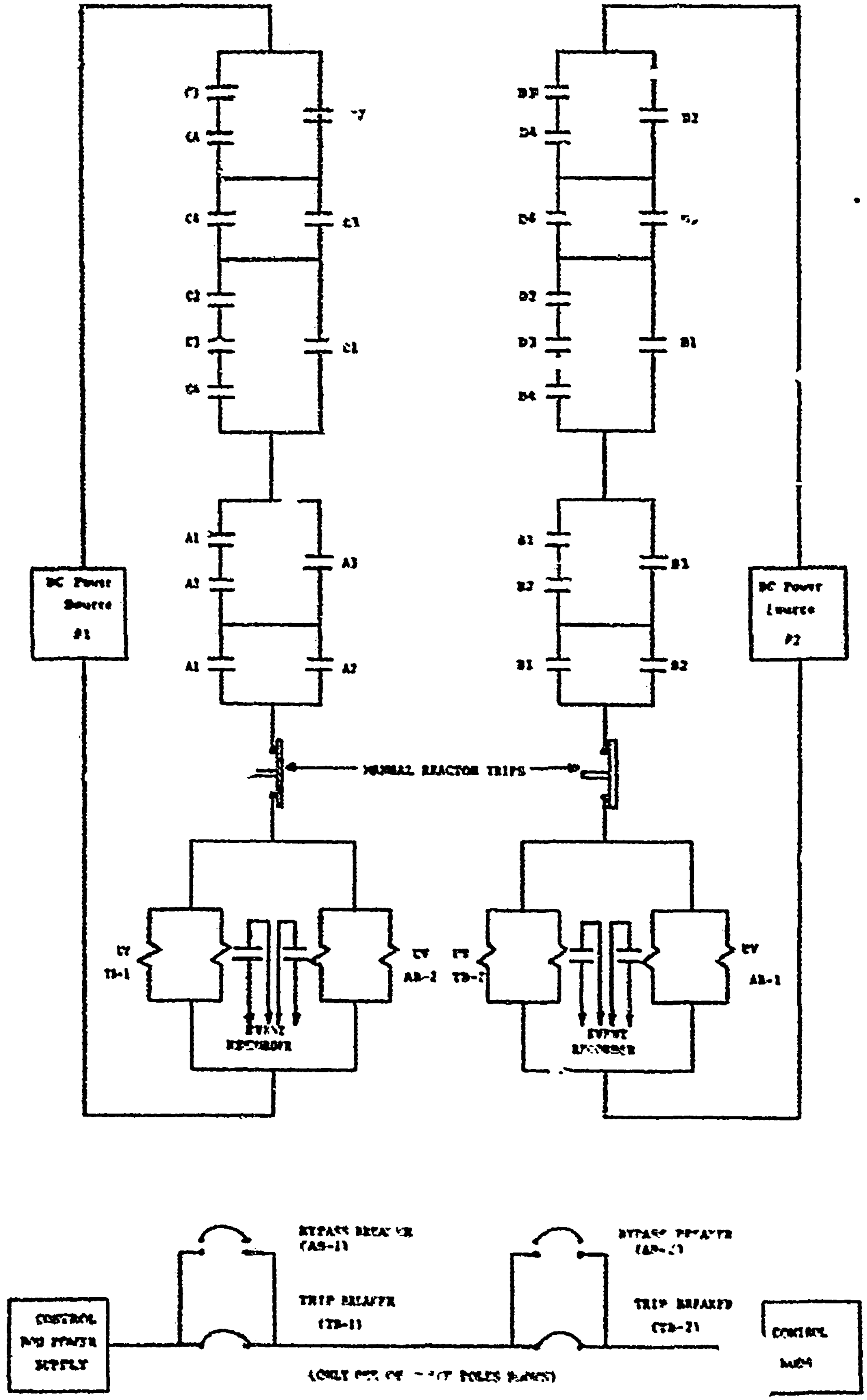

OW
OOP 3
LOOP 4
1/2
1/2
T AVG
LOOP 1
LOOP 2
LOOP 3
LOOP 4
STEAMLINE PRESSURE
LOOP 1
LOOP 2
LOOP 3
LOOP 4
2/4
2/4
1/2
2/2
SAFETY INJECTION ACTUATION
AND
STEAMLINE ISOLATION
STEAMLINE DIFFERENTIAL PRESSURE
LOOPS 3 & 4 (P3-P4)
LOOPS 1 & 4 (P1-P4)
LOOPS 1 & 3 (P1-P3)
LOOPS 2 & 3 (P2-P3)
LOOPS 2 & 4 (P2-P4)
LOOPS 1 & 2 (P1-P2)
2/3 P4 LOW
2/3 P3 LOW
2/3 P2 LOW
2/3 P1 LOW
1/4
SAFETY INJECTION ACTUATION



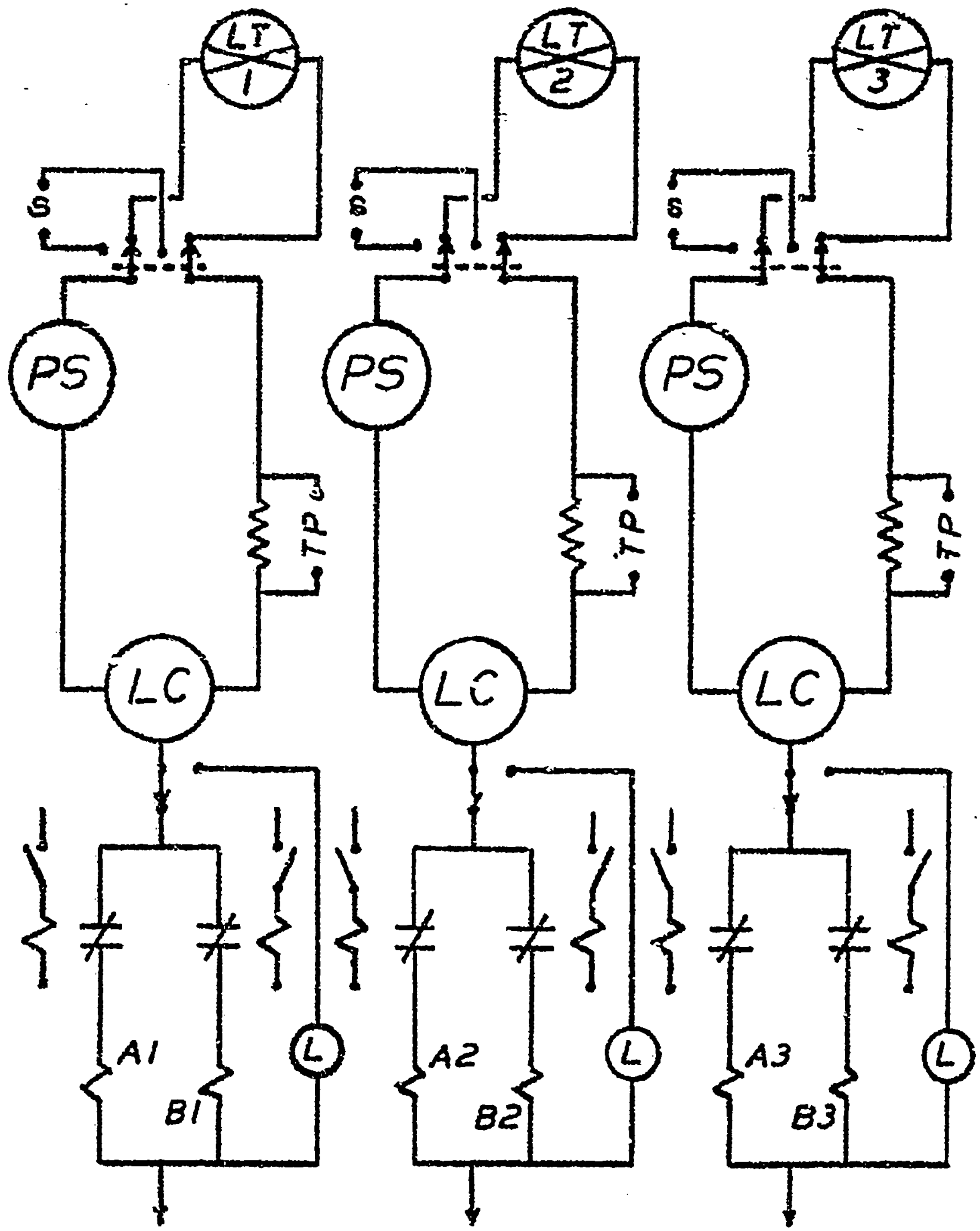
CONTAINMENT PRESSURE
PRESSURIZER PRESSURE SIGNAL
PRESSURIZER LEVEL
SIGNAL
LOOP 1
1/2
PC
PC
PC
PC
PC
PC
2/2
2/2
2/2
2/3 HI
2/3 (HI-HI)
2/3 (HI-HI)
2/3 HI
1/3
2/2
STEAMLINE ISOLATION
CONTAINMENT SPRAY ACTUATION AND CONTAINMENT ISOLATION (PHASE B)
SAFETY INJECTION ACTUATION AND CONTAINMENT ISOLATION (PHASE A) AND FAN COOLING ACTUATION



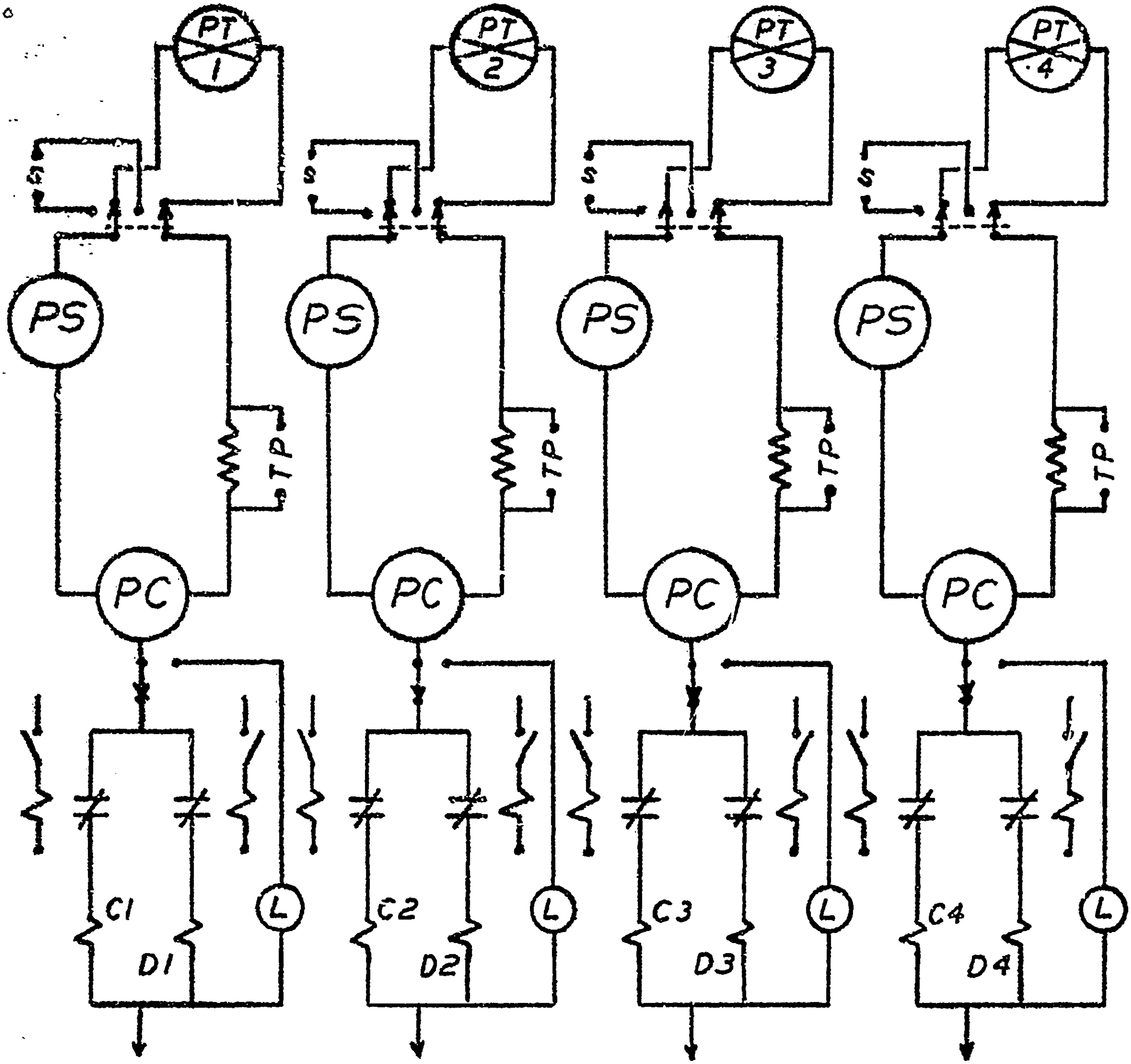
TYPICAL ANALOG CHANNEL TESTING ARRANGEMENT
FIG. 7.2-7



TRIP LOGIC CHANNELS
FIG. 7.2-8



LT 1
LT 2
LT 3
S
S
S
PS
PS
PS
TP
TP
TP
LC
LC
LC
A1
B1
A2
B2
A3
B3
L
L
L

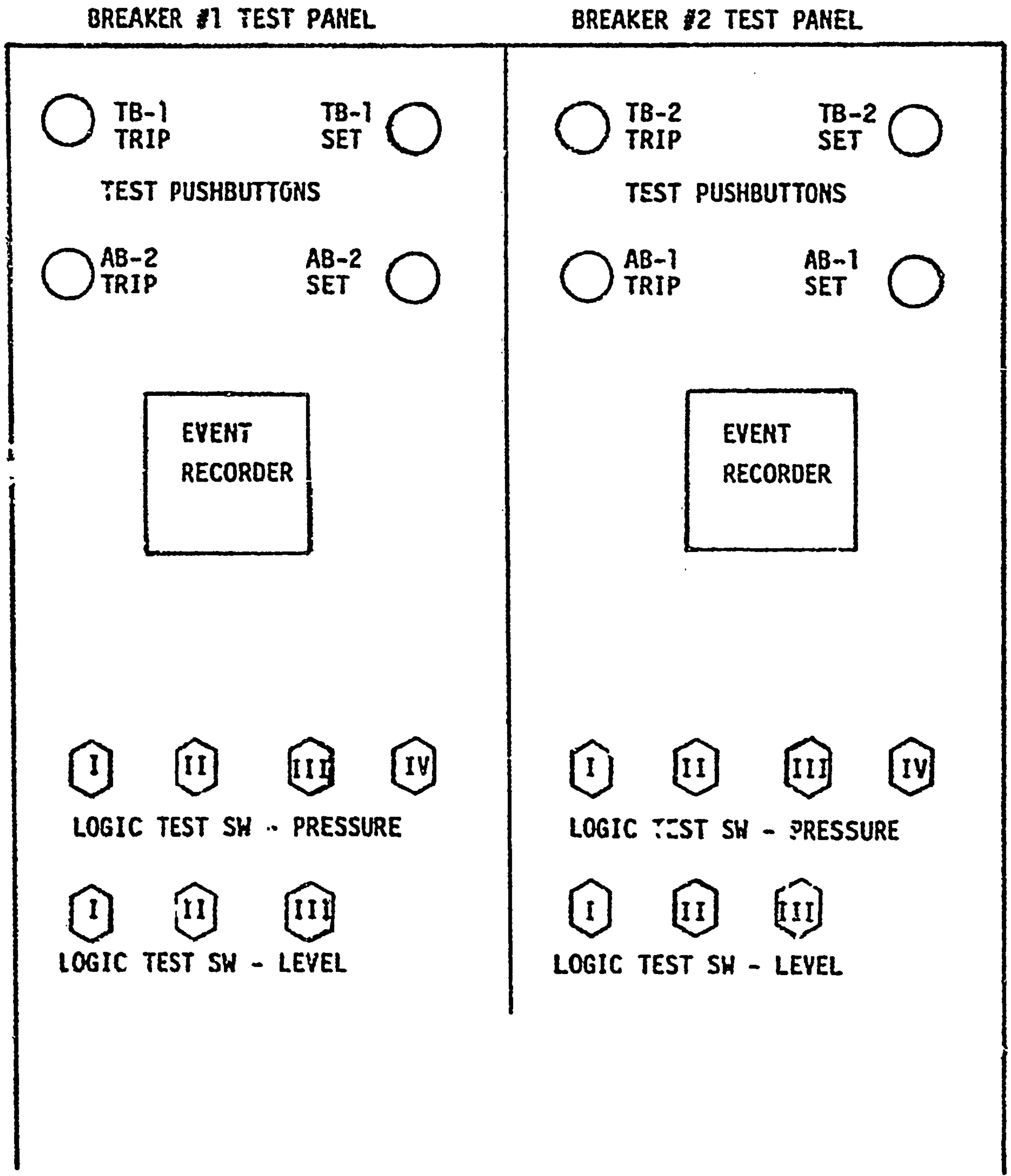


S - SIGNAL INJECTION

TP-TEST POINT

NOTE- REDUNDANT CHANNELS
ARE ISOLATED

ANALOG CHANNELS
FIG. 7.2-



LOGIC CHANNEL TEST PANELS
FIG. 7.2-10

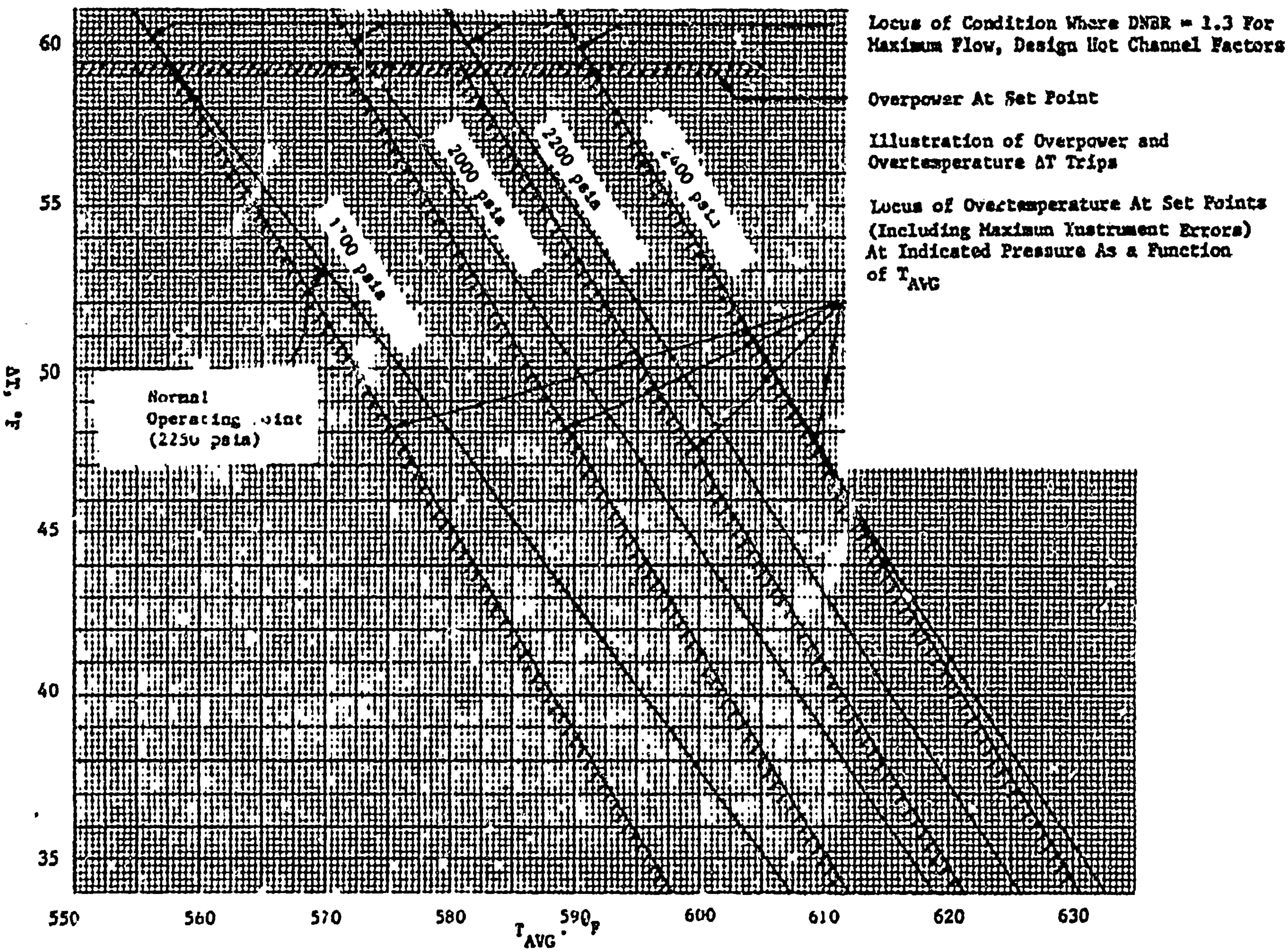
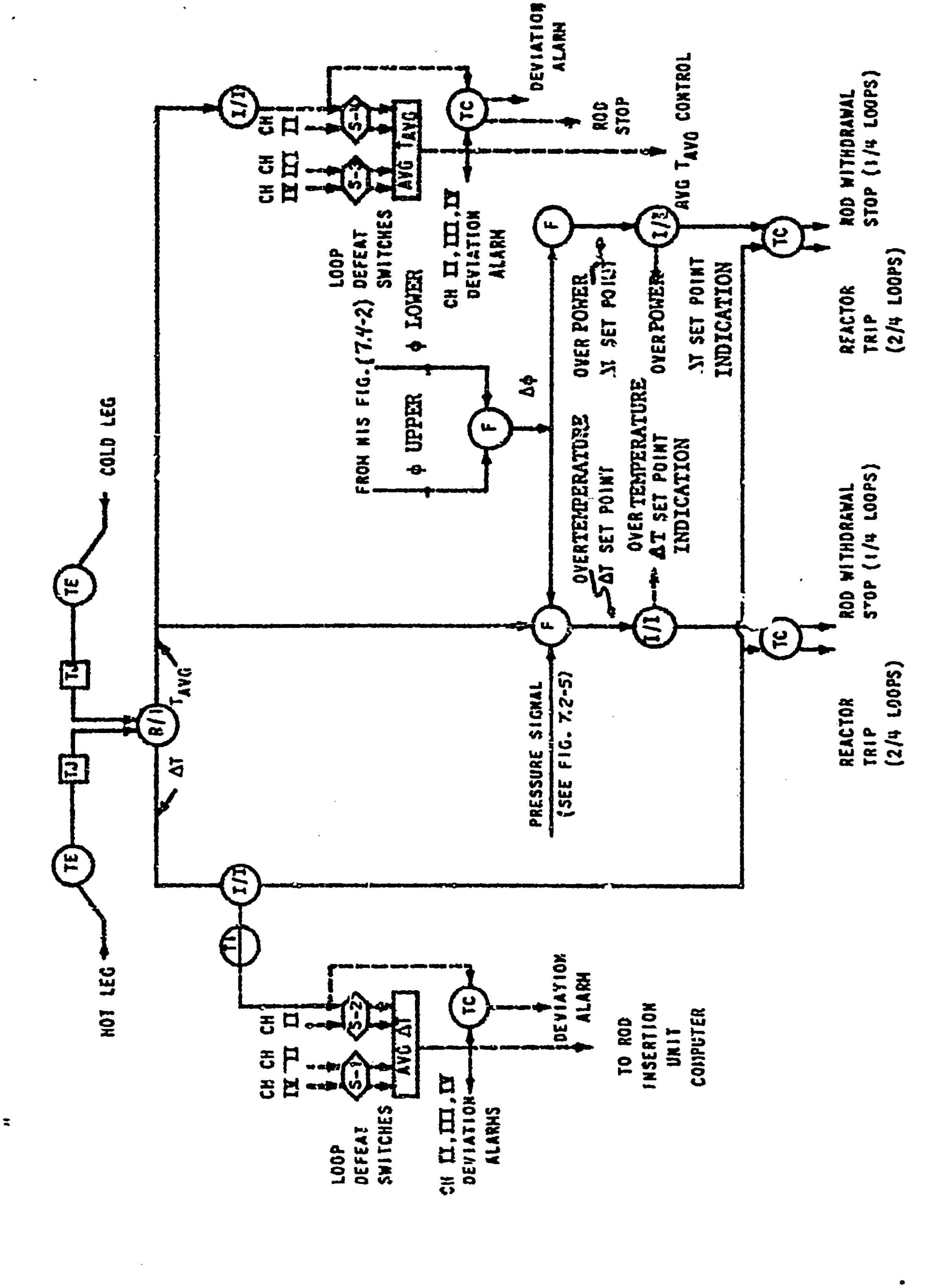

Locus of Condition Where DNBR = 1.3 For Maximum Flow, Design Hot Channel Factors
Overpower At Set Point
Illustration of Overpower and Overtemperature ΔT Trips
Locus of Overtemperature At Set Points (Including Maximum Instrument Errors) At Indicated Pressure As a Function of T_{AVG}
1700 psia
2000 psia
2200 psia
2400 psia
Normal Operating Point (2250 psia)
ΔT, °F
60
55
50
45
40
35
550
560
570
580
590
600
610
620
630
T_{AVG}, °F

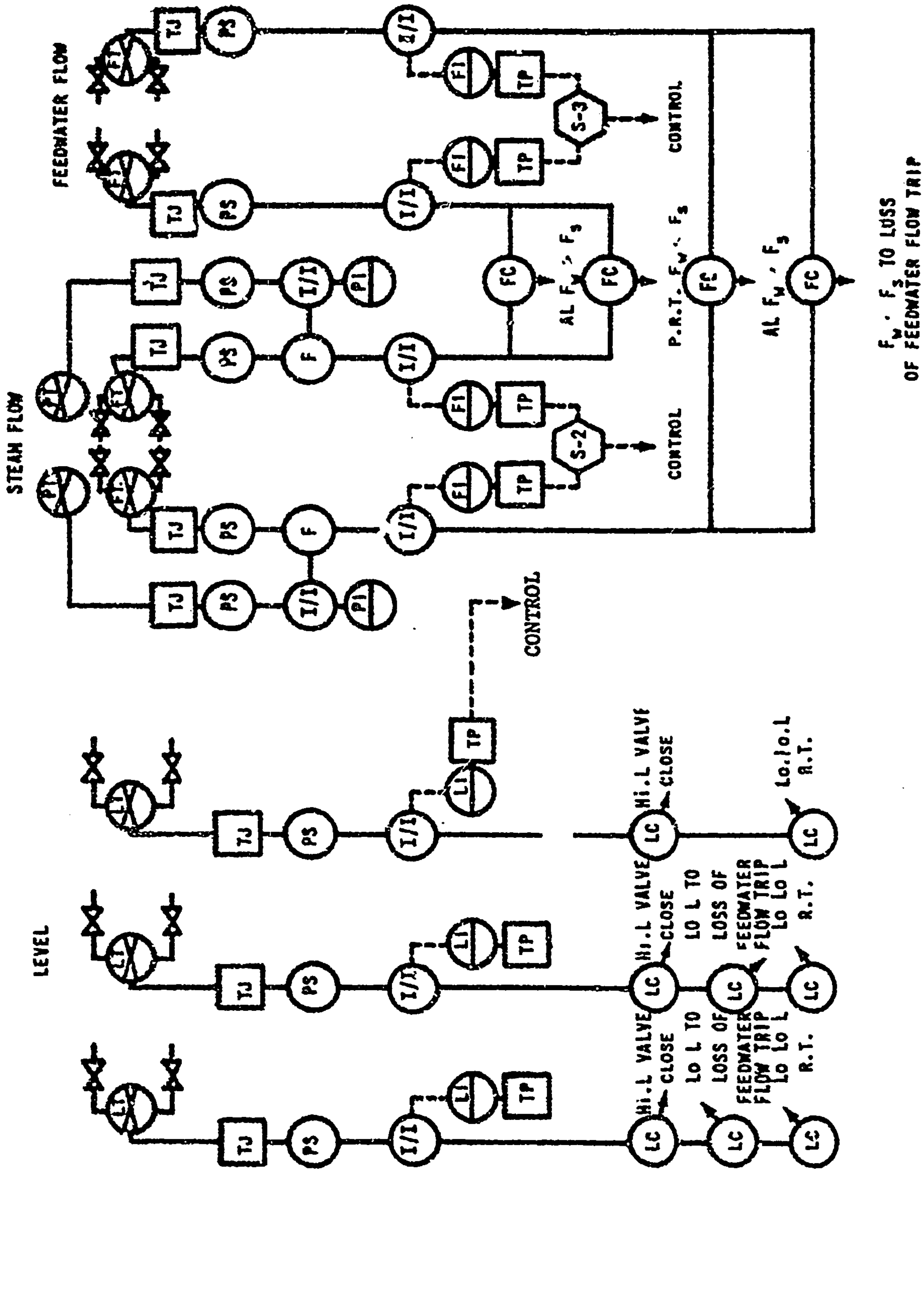
FIGURE 7.2-11



$T_{AVG}/\Delta T$ CONTROL AND PROTECTION SYSTEM

(SINGLE CHANNEL)

FIG. 7.2-12



STEAM GENERATOR LEVEL CONTROL AND PROTECTION SYSTEM

FIG. 7.2-13

ANALOG SYSTEM SYMBOLS

Al	-	Alarm
Buf	-	Buffer
f	-	Special Function (such as a pressure compensation unit or lead/log compensation)
FC	-	Flow controller (off-on unless output signal is shown)
FI	-	Flow Indicator
FT	-	Flow Transmitter
Hi LRT	-	High Level Reactor Trip
Hi PRT	-	High Pressure Reactor Trip
I/I	-	Isolation Current Repeater
ISOL	-	Isolation (other than I/I)
LC	-	Level controller (off-on unless output signal is shown)
LI	-	Level Indicator
L-Low	-	Low Level
Lo L	-	Low Level
Lo LRT	-	Low Level Reactor Trip
Lo PRT	-	Low Pressure Reactor Trip
L_{ref}	-	Programmed Reference Level
LT	-	Level Transmitter
NC	-	Nuclear Flux Controller
NE	-	Nuclear Detector
NI	-	Nuclear Flux Indicator
NQ	-	Nuclear Power Supply
PC	-	Pressure controller (off-on unless output signal is shown)
PI	-	Pressure Indicator
P_{ref}	-	Programmed Reference Pressure
PS	-	Power Supply
PI	-	Pressure Transmitter
R/I	-	Resistance to Current Connector
S	-	Control channel transfer switch (used to maintain auto channel during test of the protection channel)
SI	-	Safety Injection
T	-	Built-in Test Point
TE	-	Temperature Element
TJ	-	Test Signal Insertion Jack
TP	-	Test Point
$\phi_{U,L}$	-	Out of core upper or lower ion chamber flux signals

FIG. 7.2-14

7.3 REGULATING SYSTEMS

7.3.1 DESIGN BASIS

The reactor control system is designed to limit nuclear plant transients for prescribed design load perturbations, under automatic control, within prescribed limits to preclude the possibility of a reactor trip in the course of these transients.

Overall reactivity control is achieved by the combination of chemical shim and 53 control rod clusters of which 37 are in control group and 16 are in shutdown group. Long-term regulation of core reactivity is accomplished by adjusting the concentration of boric acid in the reactor coolant. Short-term reactivity control for power changes or reactor trip is accomplished by movement of control rod clusters.

The primary function of the Reactor Control System is to provide automatic control of the rod clusters during power operation of the reactor. The system uses input signals including neutron flux; coolant temperature and pressure; and plant turbine load. The Chemical and Volume Control System (Section 9) serves as a secondary reactor control system by the addition and removal of varying amounts of boric acid solution.

There is no provision for a direct continuous visual display of primary coolant boron concentration. When the reactor is critical, the best indication of reactivity status in the core is the position of the control group in relation to plant power and average coolant temperature. There is a direct, predictable, and reproducible relationship between control rod position and power and it is this relationship which establishes the lower insertion limit calculated by the rod insertion limit monitor. There are two alarm setpoints to alert the operator to take corrective action in the event a control bank approaches or reaches its lower limit.

Any unexpected change in the position of the control group when under automatic control or a change in coolant temperature when under manual control provides a direct and immediate indication of a change in the reactivity status of the reactor. In addition, periodic samples of coolant boron concentration are taken. The variation in concentration during core life provides a further check on the reactivity status of the reactor including core depletion.

The Reactor Control System is designed to enable the reactor to follow load changes automatically when the plant output is above 15% of nominal power. Control rod positioning may be performed automatically when plant output is above this value, and manually at any time.

The system enables the nuclear plant to accept a generation step load increase of 10% and a ramp increase of 5% per minute within the load range of 15% to 100% without reactor trip subject to possible xenon limitations. Similar step and ramp load reductions are possible within the range of 100% to 15% of nominal power.

The operator is able to select any single bank of rods (shutdown or control) for manual operation. Using a single switch, he may not select more than one bank from these two groups. He may also select automatic reactor control, in which case, the control banks can be moved only in their normal sequence with some overlap as one bank reaches its full withdrawal position and the
15 next bank begins to withdraw. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn simultaneously.

The control system is capable of restoring coolant average temperature to within the programmed temperature deadband, following a scheduled or transient change in load.

The reactor plant can be placed under automatic control in the power range between (15) per cent of load and full load for the following design transient

a) ± 10% step change in load without turbine by-pass
b) ± 5% per minute loading and unloading.
c) -50% step change in load from approximately 100% load with steam dump.

A programmed pressurizer water level as a function of load is provided in conjunction with the programmed coolant average temperature to minimize the requirements of the Chemical and Volume Control and Waste Disposal System resulting from coolant density changes during loading and unloading from full power to zero power.

Following a reactor and turbine trip, sensible heat stored in the reactor coolant is removed without actuation of steam generator safety valves by means of controlled steam bypass to the condenser and by injection of feedwater to the steam generators. Reactor coolant system temperature is reduced to the no load condition. This no load coolant temperature is maintained by steam bypass to the condensers to remove residual heat.

The control system is designed to operate as a stable system over the full range of automatic control throughout core life without requiring operator adjustment of set points other than normal calibration procedures.

7.3.2 SYSTEM DESIGN

A block diagram of the Reactor Control System is shown in Figure 7.3-1.

Rod Control

There are 61 total RCC assemblies of which 53 are full length and 8 are part length rods. The full length rods are divided into (1) a shutdown group comprising two shutdown banks of 8 rod clusters each and two shutdown banks of 4 rod clusters each, and (2) a control group comprising 4 control banks containing 8, 4, 8 and 9 rod clusters.

Figure 3.2-1 shows the location of the full and part length rods in the core. The four banks of the control group are the only rods that can be manipulated under automatic control. The banks are divided into subgroups to obtain smaller incremental reactivity changes. All RCC assemblies in a subgroup are electrically paralleled to step simultaneously. Position indication for each RCC assembly type is the same. There are two types of drive mechanism for the RCC assemblies, those for the control and shutdown groups and those for the part length rod group (Section 3.2.3).

Control Group Rod Control

The automatic rod control system maintains a group programmed reactor coolant average temperature with adjustments of control group rod position for equilibrium plant conditions. The reactor control system is capable of restoring programmed average temperature following a scheduled or transient change in load. The coolant average temperature increases linearly from zero power to the full power conditions.

The control system will also compensate initially for reactivity changes caused by fuel depletion and/or xenon transients. Final compensation for these two effects is periodically made with adjustments of boron concentration. The control system then readjusts the control group rod in response to changes in coolant average temperature resulting from changes in boron concentration.

The coolant average temperatures are measured from the hot leg and the cold leg in each reactor coolant loop. The average of the four measured average temperatures is the main control signal. This signal is sent to the control group rod programmer through a proportional plus rate compensation unit. The control group rod programmer commands the direction and speed of control group rod motion. A compensated pressurizer pressure signal, and a power-load mismatch signal are also employed as control signals to improve the plant performance. The power-load mismatch channel takes the difference between nuclear power (average of all four power range channels) and a signal of turbine load (first stage turbine pressure), and passes it through a high-pass filter such that only a rapid change in flux or power causes rod motion. The pressure compensation and the power-load mismatch compensation serve to speed up system response and to reduce transient peaks.

The rod control group is divided into four banks comprising 8, 4, 8 and 9 RCCs respectively, to follow load changes over the full range of power operation. Each rod control bank is driven by a sequencing, variable speed rod drive control unit. The rods in each control bank are divided into two subgroups; the subgroups are moved sequentially one step at a time. The sequence of motion is reversible; that is, a withdrawal sequence is the reverse of the insertion sequence. The variable speed sequential rod control affords the ability to insert a small amount of reactivity at low speed to accomplish fine control of reactor coolant average temperature about a small temperature deadband. Any reactor trip signal causes the rods to insert by gravity into the core.

Manual control is provided to manually move a control bank in or out at a preselected fixed speed.

Proper sequencing of the RCCA is assured first, by fixed programming equipment in the Rod Control System, and second, through administrative control of the reactor plant operator. Startup of the plant is accomplished by first manually withdrawing the shutdown rods to the full out position. This action requires that the operator select the SHUTDOWN BANK position on a control board mounted selector switch and then position the IN-HOLD-OUT level (which is spring return to the HOLD position) to the OUT position.

RCCA are then withdrawn under manual control of the operator by first selecting the MANUAL position on the control board mounted selector switch and then positioning the IN-HOLD-OUT lever to the OUT position. In the MANUAL selector switch position, the rods are withdrawn (or inserted) in a predetermined programmed sequence by the automatic programming equipment.

When the reactor power reaches approximately 15% off, the operator may select the AUTOMATIC position, where the IN-HOLD-OUT lever is out of service, and rod motion is controlled by the Reactor Control and Protection Systems. A permissive interlock limits automatic control to reactor power levels above 15%. In the AUTOMATIC position, the rods are again withdrawn (or inserted) in a predetermined programmed sequence by the automatic programming equipment.

Programming is set so that as the first bank out (control bank C-2) reaches a preset position near the top of the core, the second bank out (control bank C-3) begins to move out simultaneously with the first bank. When control bank C-2 reaches the top of the core, it stops, and control bank C-3 continues until it reaches a preset position near the top of the core where control bank C-4 motion begins. This withdrawal sequence continues until the plant reaches the desired power level. The programmed insertion sequence is the opposite of the withdrawal sequence, i.e., the last control bank out is the first control bank in.

With the simplicity of the rod program, the minimal amount of operator selection, and two separate direct position indications available to the operator, there is very little possibility that rearrangement of the control rod sequencing could be made.

Shutdown Rod Group Control

The shutdown group of control rods together with the control group are capable of shutting the reactor down. They are used in conjunction with the adjustment of chemical shim and the control group to provide shutdown margin of at least one per cent following reactor trip with the most reactive control rod in the fully withdrawn position for all normal operating conditions. The shutdown banks are manually controlled during normal operation and are moved at a constant speed with staggered stepping of the subgroups within the banks. Any reactor trip signal causes them to insert by gravity into the core. They are fully withdrawn during power operation and are withdrawn first during startup. Criticality is always approached with the control group after withdrawal of the shutdown banks. Four shutdown banks with a total of 24 clusters are provided.

Part Length Rod Control

Eight part length rods are provided in the reactor in addition to the normal control rods. The function of these rods, which have neutron absorber material in only the bottom one quarter of the length (three feet), is to shape the axial power distribution and thus stabilize axial xenon oscillations. In addition, they are beneficial in flattening the axial power distribution and thus in reducing hot channel factors.

Superseded by Suppl. 6 2/70

of at least one per cent following reactor trip with the most reactive control rod in the fully withdrawn position for all normal operating conditions. The shutdown banks are manually controlled during normal operation and are moved at a constant speed with staggered stepping of the subgroups within the banks. Any reactor trip signal causes them to insert by gravity into the core. They are fully withdrawn during power operation and are withdrawn first during startup. Criticality is always approached with the control group after withdrawal of the shutdown banks. Two shutdown banks of two subgroups each are provided.

X-Y Group Control

The X-Y RCC assemblies are divided into two banks of four (4) assemblies each. X-Y rod motion is manually controlled by the operator. These assemblies may be moved during motion of either shutdown or control banks. Eight (8) selector switches are provided on the control board for the X-Y rods. During startup, one bank of four (4) assemblies is withdrawn to the top of the core and then the second bank of four (4) assemblies is withdrawn to the top of the core. For correction of flux oscillations or tilts, one (1) or a maximum of two (2) assemblies are manually inserted. An alarm is provided to indicate that more than two (2) assemblies have left the top position.

Part Length Rod Control

Eight part length rods are provided in the reactor in addition to the normal control rods. The function of these rods, which have neutron absorber material in only the bottom one quarter of the length (three feet), is to shape the axial power distribution and thus stabilize axial xenon oscillations. In addition, they are beneficial in flattening the axial power distribution and thus in reducing hot channel factors.

The part length rods are operated only by manual control by the operator from the control console. They are moved together as a bank to make the upper and lower ion chamber readings approach a prescribed relationship within a prescribed allowable region of travel.

The part length rods do not trip since power is required to change their position.

Interlocks

The rod control group is used for automatic control and is interlocked with measurements of turbine-generator load and reactor power to prevent automatic control rod withdrawal below 15% of nominal power. The manual and automatic controls are further interlocked with measurements of nuclear flux, ΔT and rod drop indication to prevent approach to an overpower condition.

Rod Drive Performance

The control banks are driven by a sequencing, variable speed rod drive programmer. In the control bank of RCC assemblies, control subgroups (each containing a small number of RCC assemblies) are moved sequentially in a cycle such that all subgroups are maintained within one step of each other.

The sequence of motion is reversible, that is, withdrawal sequence is the reverse of the insertion sequence. The sequencing speed is proportional to the control signal from the Reactor Control System. This provides control group speed control proportional to the demand signal from the control system.

A solid state control system provides power to the rod drive mechanism coils from the output of two paralleled motor generator sets. Two reactor trip breakers are placed in series with the output of the M-G sets. To permit on-line testing, a bypass breaker is provided across each of the two breakers.

RCCA Position Indication

Two separate systems are provided to sense and display control rod position as described below:

a) Analog System - An analog signal is produced for each individual rod by a linear position transmitter.

An electrial coil stack is located above the stepping mechanisms of the control rod magnetic jacks, external to the pressure housing, but concentric with the rod travel. When the associated control rod is at the bottom of the core, the magnetic coupling between the primary and secondary coil winding of the detector is small and there is a small voltage induced in the secondary. As the control rod is raised by the magnetic jacks, the relatively high permeability of the lift rod causes an increase in magnetic coupling. Thus, an analog signal proportional to rod position is obtained.

Direct, continuous readout of every control rod is presented to the operator on individual indicators.

A deviation monitor alarm is actuated if an individual rod position deviates from its group position by a preselected distance.

Lights are provided for rod bottom positions for each rod. The lights are operated by bistable devices in the analog system.

b) Digital System - The digital system counts pulses generated in the rod drive control system. One counter is associated with each subgroup of control and shutdown rods, eight are provided for the X-Y rods and one is provided for the part length rods. Readout of the digital system is in the form of electromechanical add-subtract counters reading the number of steps or rod withdrawal with one display for each subgroup. These readouts are mounted on the control panel.

The digital and analog systems are separate systems· each serves as backup for the other. Operating procedures require the reactor operator to compare the digital and analog readings upon recognition of any apparent malfunction. Therefore, a single failure in rod position indication does not in itself lead the operator to take erroneous action in the operation of the reactor.

Full Length Rod Drive Power Supply

The full length control rod drive power supply concept using a single scram bus system has been successfully employed on all Westinghouse PWR plants. Potential fault conditions with a single scram bus system are discussed in this section. The unique characteristics of the latch type mechanisms with its relatively large power requirements makes this system with the redundant series trip breakers particularly desirable.

The solid state rod control system is operated from two parallel connected 400 KVA generators which provide 260 volt line to line, three phase, four wire power to the rod control circuits through two series connected reactor trip breakers. This AC power is distributed from the trip breakers to a line-up of identical solid state power cabinets using a single overhead run of enclosed bus duct which is bolted to and therefore comprises part of the power cabinet arrangement. Alternating current from the motor-generator sets is converted to a profiled direct current by the power cabinet and is then distributed to the mechanism coils. Each complete rod control system includes a single 70 volt DC power supply which is used for holding the mechanisms in position during maintenance of normal power supply.

This 70 volt supply, which receives its input from the AC power source downstream of the reactor trip breakers, is distributed to each power cabinet and permits holding mechanisms in groups of four by manually positioning switches located in the power cabinets. The 30 ampere output capacity limits the holding capability to eight rods.

Reactor Trip

Current to the mechanisms is interrupted by opening either of the reactor trip breakers. The 70 volt DC maintenance supply will also be interrupted since this supply receives its input power through the reactor trip breakers.

Trip Breaker Arrangement

The trip breakers are arranged in the reactor trip switchgear in individual metal enclosed compartments. The 1000 ampere bus work, making up the connections between scram breakers will be separated by metal barriers to prevent the possibility that any conducting object could short circuit, or bypass, scram breaker contacts.

Maintenance Holding Supply

The 70 VDC holding supply and associated switches have been provided to avoid the need for bringing a separate DC power source to the rod control system during maintenance on the power cabinet circuits. This source is adequate for holding a maximum of eight mechanisms and satisfies all maintenance holding requirements.

Control System Construction

The rod control system equipment is assembled in enclosed steel cabinets. Three phase power is distributed to the equipment through a steel enclosed bus duct, bolted to the cabinets. DC power connections to the individual mechanisms are routed to the reactor head area from the solid state cabinets through insulated cables, enclosed junction boxes, enclosed reactor containment penetrations, and sealed connectors. In view of this type of construction, any <u>accidental</u> connection of either an AC or DC power source, either internal or external to the cabinets, is not considered credible.

AC Power Connections

The three phase four wire supply voltage required to energize the equipment is 260 volts line to line, 58.3 Hz, 438 KVA capacity, zig-zag connected. It is unlikely that any power supply, and in particular one as unusual as this four wire power source could be accidently connected, in phase, in the required configuration. Also it should be noted that this requires multiple connections, not single connections. The closest outside sources available in the plants are 480 volt auxiliary power sources and 208 volt lighting sources.

Connections of either a 480 or 208 volt, 60 Hz source to the single AC bus supplying the rod control system causes currents to flow between the sources due to an out of phase condition. These currents flow until the generator accelerates to a speed synchronous with the 60 Hz outside source, a time sufficient to trip the generator breakers. The out-of-phase currents for an unlimited capacity outside source, an outside source with a capacity equivalent to the normal generator KVA, and for either one or two M-G sets in service are tabulated below:

Out of Phase Currents (Amperes)

		One M-G Set in Service	Two M-G Sets in Service
480V	Unlimited Capacity	25,000	50,000
	438 KVA Capacity	12,000	25,000
208V	Unlimited Capacity	16,000	32,000
	438 KVA Capacity	8,000	16,000

All of the foregoing currents are sufficiently high to trip out the generator breakers on either overcurrent or reverse current. This trip-out is detectable by annunciation in the control room. If the outside power source trips, the connection is of no concern.

Each solid state power cabinet is tied to the main AC bus through three fused disconnect switches; one for the stationary gripper coil circuits, one for the movable gripper coil circuits, and one for the lift coil circuits. Reference voltages to operate the control circuits for all three coil circuits must be in phase with the supply to all coil circuits for proper operation of the system. If the outside power source were brought into an individual cabinet, nine (9) normal source connections would have to be disconnected and the outside source would have to be tied in phase to the proper nine (9) points plus one (1) neutral point to allow movement of the rods. This is not considered credible.

Connection of a single phase AC source (i.e. one line to neutral) is also considered improbable. This would again require a high capacity source which would have to be connected in-phase with the non-synchronous M-G set supply. Again more than one connection is needed to achieve this condition. Each power cabinet contains three alarm circuits (stationary, movable, and lift) that would annunciate the condition to the operator. In addition, calculations show that a single phase source of 208 volts, 260 volts, or 480 volts will not supply enough current to hold the rods. Therefore, a jumper across two trip circuit breaker contacts in series which results in a single phase remaining closed would not provide sufficient current to hold-up the rods.

The normal source generators are connected in a zig-zag winding configuration to eliminate the effects of direct current saturation of the machines resulting from the direct currents that flow in the half wave bridge rectifier circuits. If this connection were not used, the generator core would saturate and loss of generating action would occur. This condition would also occur in a transformer. An outside source not having the zig-zag configuration would have to have a large capacity (>400 KVA) to avoid the loss of transformer action from saturation.

Most of the components in the equipment are applied with a 100% safety factor. Therefore, the possibility exists that the system will operate at 450 volts with a source of sufficient capacity. The system will definitely operate at 208 volts with a source of sufficient capacity.

The connection of an outside source of AC power to one rod control system would first require a need for this source. No such need exists since two power sources (M-G sets) are already provided to supply the system. If the source were connected in spite of the need, extreme measures would have to be taken by the intruder to complete the connection. The outside source would have to be a large capacity (400 KVA) one. The currents that flow would require the routing of large conductors or bus bars, not the usual clip leads. Then, the disassembly of switchgear or enclosed bus duct would be required to expose the single AC bus. Large bolted cable or bus bar terminations would have to be completed. A total of four conductors would have to be connected in phase with a non-synchronous source. To expect that a connection could be completed with the equipment either energized or de-energized in view of the obstacles which would prevent such a connection is incredible. However, even if the connection were completed, the outside source connection would be detectable by the operator through the tripping of the generator breakers.

DC Power Connections

An external DC source could, if connected inside the power cabinet, hold the rods in position. This would require a minimum supply voltage of 50 volts. Since the holding current for each mechanism coil is 4 amperes, the DC current capacity would have to be approximately 180 amperes to hold all rods. Achieving this situation would require several acts - bringing in a power source which is not required for any type of operation in the rod control system, preferentially connecting it into the system at the correct points, and actuating specific holding switches so as to interconnect all rods. Closure of twelve switches in four separate cabinets would be required to hold all rods. One switch could hold as many as four rods.

The application of a DC voltage to an individual rod external to the power cabinet would affect only a single rod - connection with other rods in the group being prevented by the blocking diodes in the power circuits.

Should an external DC source be connected to the system, the system is provided with features to permit its detection.

Each solid state power cabinet contains circuitry which compares the actual currents in the stationary and movable gripper coils with the reference signals from the step sequencing unit (slave cycler). In taking a single step, the current to the stationary gripper coil will be profiled from the holding value to the maximum, to zero and return to holding level. Correspondingly, the movable gripper coil must change from zero to maximum and return to zero. The presence of an external DC source on either the stationary or movable coils would prevent the related currents from returning to zero.

This situation would be instantaneously annunciated by way of the comparison circuit. Therefore, any rod motion would actuate an alarm indicating the presence of an external DC source. In addition, an external DC source would prevent rods from stepping. Thus, an external source could be detected by the rod position indication system indicating failure of the rod(s) to move.

Connection of an external DC power source to the output lines of the 70 volt DC power supply can be detected by opening the three phase primary input of the supply and checking the output with a built-in voltmeter.

Evaluation Summary

In view of the preceding discussion, the postulated connection of an external power source (either AC or DC) or occurrence of short circuits that would prevent dropping of the rods is not considered credible. Specifically:

a. The need for an outside power source has been eliminated by incorporating build-in holding sources as part of the rod control system and by providing two M-G sets.

b. The equipment is contained within enclosed steel cabinets precluding the possibility of an accidental connection of either AC or DC power in the cabinets.

c. AC power distribution is accomplished using steel enclosed bus duct. The high capacity (400 KVA) AC power source is unique and not readily available. Multiple connections are required.

d. DC power is distributed to the individual mechanisms through insulated cables and enclosed electrical connections precluding the accidental connection of an outside DC source external to the cabinets. The high capacity DC source required to hold rods is not readily available in the rod control system, would require multiple connections, and would require deliberate positioning of switches within the enclosed cabinets.

e. Provisions are made in the system to permit detection of an external DC source which could preclude a rod release.

The total capacity of the system including the overload capability of each motor generator set is such that a single set out of service does not cause limitations in rod motion during normal plant operation. In order to minimize reactor trip as a result of a unit malfunction, the power system is normally operated with both units in service.

Turbine By-Pass

A turbine by-pass system is provided to accommodate a reactor trip with turbine trip, 50% loss of load without reactor and turbine trip. The turbine by-pass system removes steam to reduce the transient imposed upon the reactor coolant system so that the control rods can reduce the reactor power to a new equilibrium value without allowing overtemperature, overpressure conditions in the reactor coolant system.

A turbine by-pass is actuated by the coincidence of compensated coolant average temperature higher than the programmed value by a preset value and electrical load decrease greater than a preset value. All the turbine by-pass valves stroke to full open immediately upon receiving the by-pass signal. The by-pass valves are modulated by the compensated coolant average temperature signal after they are full open. The turbine by-pass reduces proportionally as the control rods act to reduce the coolant average temperature. The artifical load is therefore removed as the coolant average temperature is restored to its programmed equilibrium value.

The turbine by-pass steam capacity is 40 percent of full load steam flow at full load steam pressure. The bypass flows to the main condenser.

Part Length Power Supply

As shown in Figure 7.3-2, the prime mover in each part length mechanism is a six phase reluctance type motor. By proper sequencing of the motor windings, the armature can be rotated incrementally and the rotational motion converted to linear rod travel through the roller nut assembly.

The movement of the rods is derived through the part length rod power supply which utilizes a silicon controlled rectifier (SCR) solid state switch. As shown in Figure 3.1-2, the three phase AC power is first stepped down and transformer into six phase power which is then fed through the SCR power switches to the drive motors. The direction and speed of the drive motors are determined by the pulse sequencing pattern and repetition rate generated by the programmer. The IN-HOLD-OUT rod motion switch, mounted on the control console, selects the direction of rotation of the programmer motor and the stepping rate (rod speed) is fixed by the line frequency applied to the hysteresis type programmer motor. The sequencing pattern is generated by a coded disc/photocell circuit geared to the programmer. The programmer output is sensed and amplified by the gating circuits which trigger the proper SCR's to achieve the required stepping sequence of the drive motor coils.

The control system includes a set of disconnect switches which de-energize the individual rod drive motors and permit realignment of the rod if this becomes necessary.

Control of the part length rods is a manual operation. There are no inputs to the part length control system from the reactor protection system or other rod control system.

The power supply contains various alarm, inhibit and interlock functions. Specifically, the momentary existance of any of the following conditions shall prevent initiation of part length rod operation or, if the rods are in motion, shall interrupt the sequencing operation and stop rod motion:

a. Loss of AC power to the control system and/or the mechanisms

b. Faulty programmer operation to include incorrect rod motion rotation signals from the programmer caused by failure of the direction relays or a single short in the rod motion control switch

c. Faulty SCR gating

Restoration of system operation following an inhibit signal requires correction of the fault and operation of a local reset.

The implementation of the detailed design of the system reflects the following functional requirements:

a. The control system shall be capable of continuously operating the part length rods at a speed of 15 inches/minute ± 10%.

b. The system shall be designed to operate all part length rods simultaneously.

c. On loss of power during rod motion, all rods shall stop and the relative position of rods in the bank shall change by less than 3/8 inch.

d. Control or electrical failure shall not cause reversal of the direction of any rod or the complete bank during motion.

e. Control or electrical failure shall not cause withdrawal or insertion of any rod or the complete bank when in the hold condition.

f. Control or electrical failure shall not cause the rod speed to exceed the maximum controlled speed specified when driving in either the IN or OUT direction.

g. No control or electrical failure shall not cause a rod to drop.

Part Length Position Indication

Thy rod position indication system is essentially identical to that used for the full length rods, i.e., the actual position of each rod as well as the bank demand signal is continuously displayed on the operating console.

Actual Position Indication

This system derives the position signal directly from measurements of the drive rod position utilizing a linear variable differential transformer (LVDT) as a detector. The drive shaft varies the amount of coupling between the primary and secondary windings of the coils which generates an analog signal proportional to rod position. The LVDT signal is conditioned and displayed on individual meters mounted on the operating console.

Demand Position Indication

The bank demand position signal is derived from the programmer and is displayed on an add-subtract pulse counter mounted in the control console.

Rod Deviation

As stated above, both the demand and actual rod position signals are monitored by a rod deviation monitoring system which provides an alarm whenever the individual rod position signal deviates from the bank demand signal by a preset limit.

Operational Information

The part length rods are used to correct for changes in axial power distribution in the core. Control information is derived through measurements of the out-of-core neutron flux distributions using the power range instrumentation in the nuclear instrumentation system.

Feedwater Control

Each steam generator is equipped with a three-element feedwater controller which maintains a programmed water level as a function of load on the secondary side of steam generator. The three-element feedwater controller continuously compares actual feedwater flow with steam flow compensated by steam pressure with a water level set point to regulate the feedwater valve opening. The individual steam generators are operated in parallel, both on the feedwater and on the steam side.

Continued delivery of feedwater to the steam generators is required as a sink for the heat stored and generated in the coolant following a reactor trip and turbine trip. A reactor trip signal provides an override signal to the feedwater control system. Upon receiving the override signal, all feedwater valves open fully to insure the full supply of feedwater following a reactor trip and turbine trip. Another override signal closes the feedwater valves when the coolant average temperature is below a preset temperature value or when the respective steam generator level rises to a preset value. Manual override of the feedwater control systems is also provided.

Pressure Control

The reactor coolant system pressure is maintained at constant value by using heaters in the water region and spray in the steam region of the pressurizer. Electrical immersion heaters are located near the bottom of the pressurizer. A portion of the heater groups are proportional heaters and are used for small pressure variation control and to compensate for heat losses and the smaller continuous spray. The remaining (backup) heaters are turned on either when the pressurizer pressure controller signal is below a preset value or when the pressurizer level exceeds the programmed level setpoint by a preset amount.

Spray valves are located at the top of the pressurizer. Spray is initiated when the pressure controller signal is above a preset set point. Spray rate increases proportionally with increasing pressure until it reaches the maximum spray capacity. Steam condensed by spray reduces the pressurizer pressure. A small continuous spray is normally maintained to reduce thermal stresses and thermal shock when the spray valves open and help maintain uniform water chemistry and temperature in the pressurizer.

Two power relief valves limit system pressure to 2350 psia for large load reduction transients.

One relief valve is operated on the pressurizer pressure controller signal, the other one is operated on the actual pressure signal. A separate interlock is provided for each so that if a second pressure channel indicates abnormally low, at the time the relief valve operation is called for by the control channel, the valve activation is blocked. The logic for each is thus basically two out of two. However, during normal operation at normal pressure, the interlock is not actuated and only the operating signals are required to actuate the valves. Actuation of the interlock also sounds the low pressure alarm.

Three spring-loaded safety valves limit system pressure to 2750 psia following a complete loss of load without direct reactor trip or turbine by-pass.

7.3.3 SYSTEM DESIGN EVALUATION

Plant Stability

The Control System is designed to limit the amplitude and the frequency of continuous oscillation of coolant average temperature about the control system set point within acceptable values. Continuous oscillation can be induced by the introduction of a feedback control loop with an effective loop gain which is either too large or too small with respect to the process transient response, i.e., instability induced by the control system itself. Because stability is more difficult to maintain, at low power under automatic control, no provision is made to provide automatic control below 15 per cent of full power.

The control system is designed to operate as a stable system over the full range of automatic control throughout core life.

Step Load Changes Without Turbine Bypass

A typical reactor power control requirement is to restore equilibrium conditions, without a plant trip, following a plus or minus 10 per cent change in step load demand, over the 15 to 100 per cent power range for automatic control. The design must necessarily be based on conservative conditions and a greater transient capability is expected for actual operating conditions. A load demand greater than full power is prohibited by the turbine control load limit devices.

The function of the control system is to minimize the reactor coolant average temperature deviation during the transient within an acceptable value and to restore average temperature to the programmed set point within an acceptable time. Excessive pressurizer pressure variations are prevented by using spray and heaters in the pressurizer.

The margin to over-temperature ΔT reactor trip is of primary concern for the step load changes. This margin is influenced by nuclear flux, pressurizer pressure, and reactor coolant average temperature and temperature rise across the core.

Loading and Unloading

Ramp loading and unloading is provided over the 15 to 100 per cent power range under automatic control. The function of the control system is to maintain the coolant average temperature and the secondary steam pressure as functions of turbine-generator load within acceptable deviation from the programmed values. The minimum control rod speed provides a sufficient reactivity rate to compensate the reactivity changes resulting from the moderator temperature coefficient and the power coefficient.

The coolant average temperature is increasing during loading and there is a continuous insurge to the pressurizer resulting from coolant expansion. The sprays limit the resulting pressure increase. Conversely as the coolant average temperature is decreasing during unloading, there is a continuous outsurge from the pressurizer resulting from coolant contraction. The heaters limit the resulting system pressure decrease. The pressurizer level is programmed such that the water level has an acceptable margin above the low level heater cutout set point during the loading and unloading transients.

The primary concern for the loading is to limit the overshoot in coolant average temperature to provide sufficient margin to the over-temperature ΔT trip.

The automatic load controls are designed to safely adjust the unit generation to match load requirements within the limits of the unit capability and licensed rating.

Loss of Load With Turbine By-Pass

The reactor control system is designed to accept 50% loss of load. No reactor trip or turbine trip will be actuated. The automatic turbine by-pass system is able to accommodate this abnormal load rejection and to reduce the transient imposed upon the reactor coolant system. The reactor power is reduced at a rate consistent with the capability of the rod control system. The reducing of the reactor power is automatic down to 15 per cent of full power. Manual control is used when the power is below this value. The by-pass is removed as fast as the control rods are capable of inserting negative reactivity.

The pressurizer relief valves might be actuated for the most adverse conditions, e.g., the most negative Doppler coefficient, and the minimum incremental rod worth. The relief capacity of the power operated relief valves is sized large enough to limit the system pressure to prevent actuation of high pressure reactor trip for the most adverse conditions.

Turbine - Generator Trip With Reactor Trip

Turbine-generator unit trip is accompanied by reactor trip. With a secondary system design pressure of 1100 psia, the plant is operated with a programmed average temperature as a function of load, with the full load average temperature significantly greater than the saturation temperature corresponding to the steam generator safety valve set point. This, together with the fact that the thermal capacity in the reactor coolant system is greater than that of the secondary system, requires a heat sink to remove heat stored in the reactor coolant to prevent actuation of steam generator safety valves for turbine and reactor trip from full power.

This heat sink is provided by the combination of controlled release of steam to the condenser and by makeup of cold feedwater to the steam generators. The turbine by-pass system is controlled from the reactor coolant average temperature signal whose reference set point is reset upon trip to the no load value. Turbine by-pass actuation must be rapid to prevent steam generator safety valve actuation. With the by-pass valves open the coolant average temperature starts to reduce quickly to the no load set point. A direct feedback of reactor coolant average temperature acts to proportionally close the valves to minimize the total amount of steam by-passed.

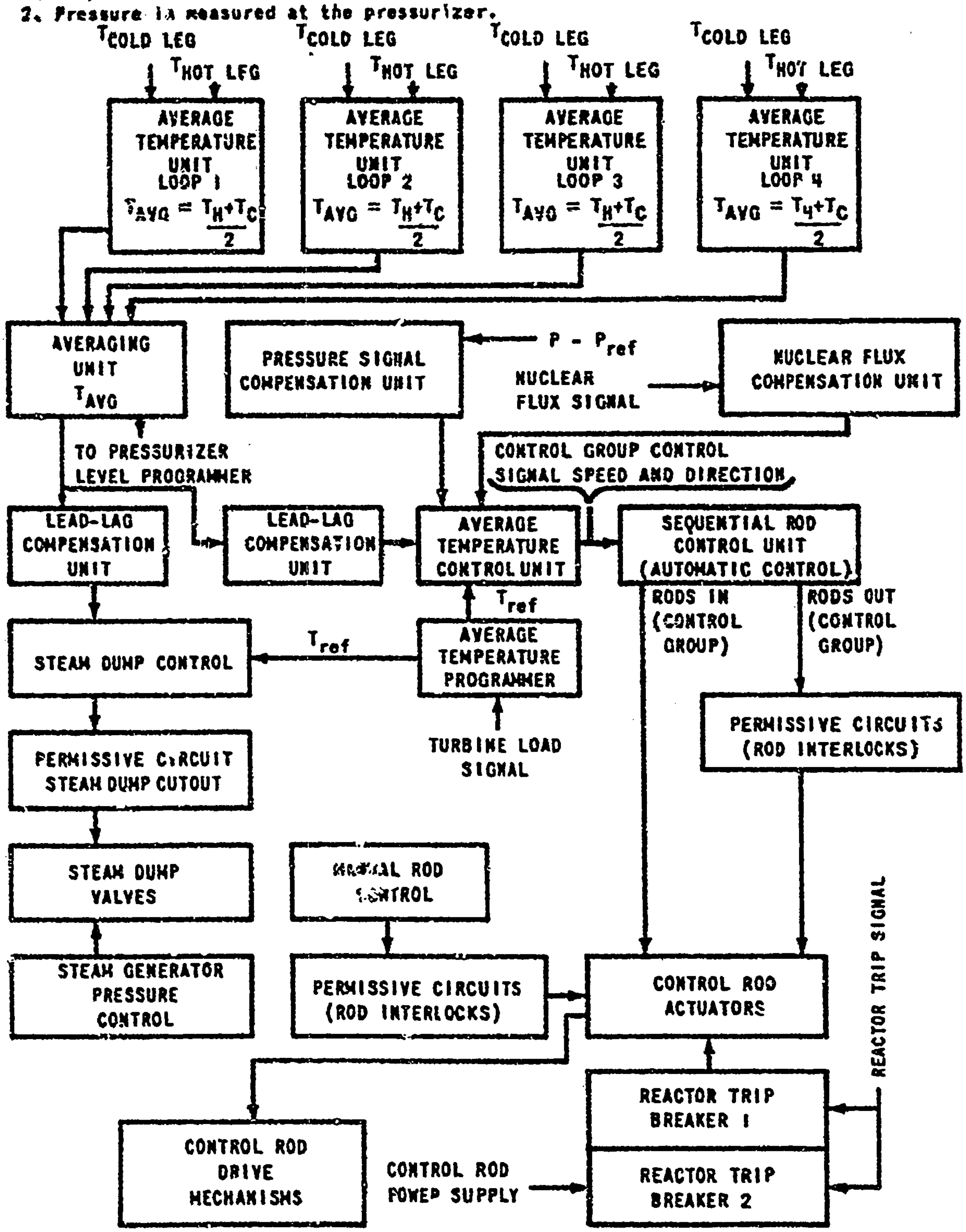
Following turbine trip, the steam voids in the steam generators will collapse and the fully opened feedwater valves will provide sufficient feedwater flow to restore water level in the downcomer. The feedwater flow is cut off when the reactor coolant average temperature decreases below a preset temperature value or when the steam generator water level reaches a preset high set point.

Additional feedwater makeup is then controlled manually to restore and maintain steam generator level while maintaining the reactor coolant at the no load temperature. Residual heat removal (manually selected) is maintained by the steam generator pressure controller which controls the amount of turbine by-pass to the condensers. This controller operates the same bypass valves to the condensers which are controlled by coolant average temperature during the initial transient following turbine and reactor trip.

The pressurizer pressure and level fall very fast during the transient resulting from the coolant contraction. The pressurizer water level is programmed so that the level has an acceptable margin above the low level safety injection set point following turbine and reactor trip to prevent accidental safety injection actuation. If heaters become uncovered following the trip, the Chemical and Volume Control System will provide full charging flow to restore water level in the pressurizer. Heaters are then turned on to heat up pressurizer water and restore pressurizer pressure to normal.

The turbine by-pass and feedwater control systems are designed to prevent the coolant average temperature falling below the programmed no load temperature following the trip to ensure adequate reactivity shutdown margin.

NOTES:
1. Temperatures are measured at steam generator's inlet and outlet.
2. Pressure is measured at the pressurizer.



SIMPLIFIED BLOCK DIAGRAM OF REACTOR CONTROL SYSTEMS
FIG. 7.3-1

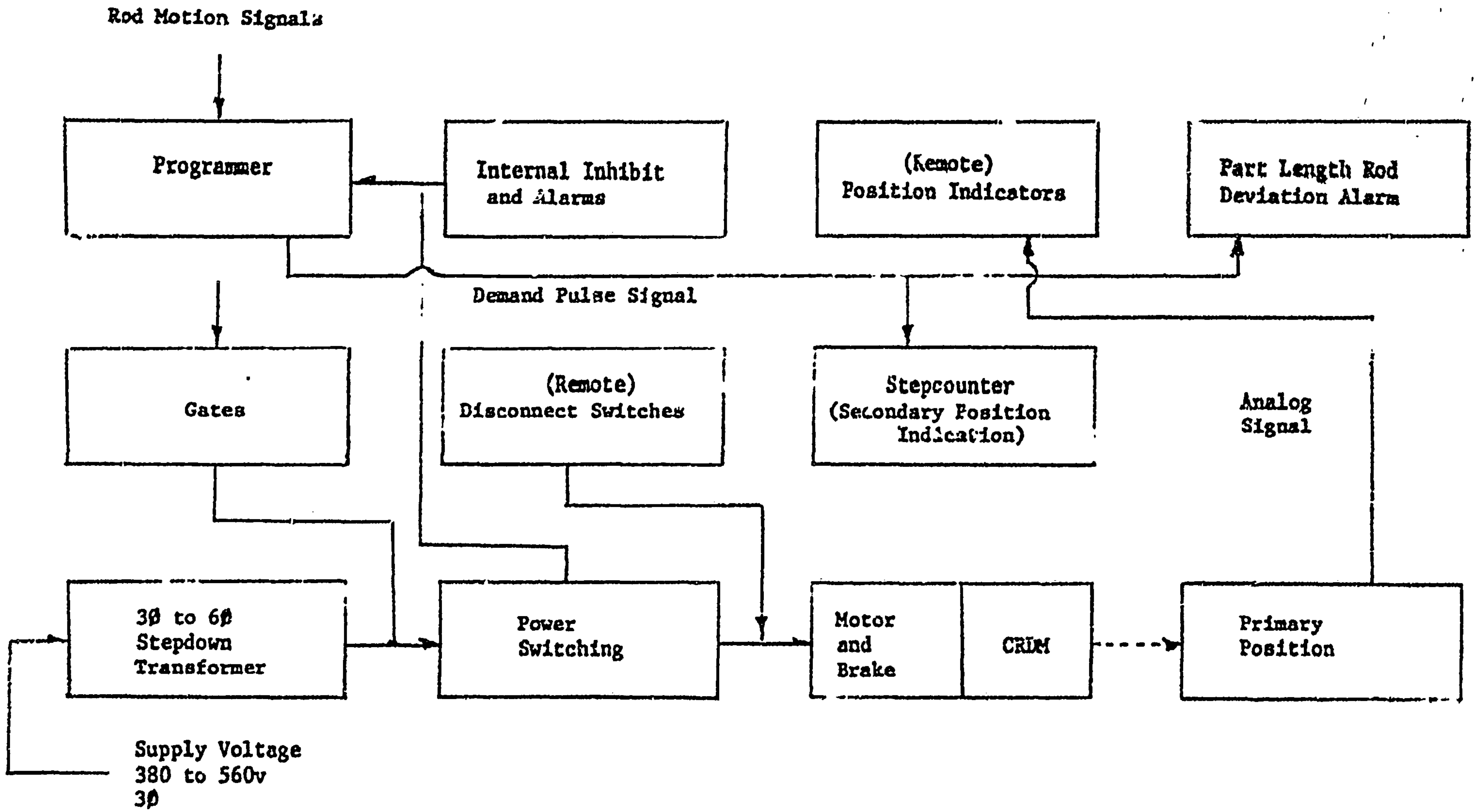


FIGURE 7.3-2

PART LENGTH CRDM CONTROL AND POSITION INDICATION

7.4 NUCLEAR INSTRUMENTATION

7.4.1 DESIGN BASES

Fission Process Monitors and Controls

Criterion: Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in reactivity of the core. (GDC 13)

The Nuclear Instrumentation System is provided to monitor the reactor power from source range through the intermediate range and power range up to 120 percent full power. The system provides indication, control, and alarm signals for reactor operation and protection.

The operational status of the reactor is monitored from the control room. When the reactor is subcritical (i.e., during cold or hot shutdown, refueling and approach to criticality) the relative reactivity status (neutron source multiplication) is continuously monitored and indicated by proportional counters located in instrument wells in the primary shield adjacent to the reactor vessel. Two source-detector channels are provided for supplying information on multiplication while the reactor is subcritical. A reactor trip is actuated from either channel if the neutron flux level becomes excessive. This system is checked prior to operations in which criticality may be approached. This is accomplished by the use of an in-core source to provide a meaningful count rate even at the refueling shutdown condition. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate, is slow enough to give ample time to start corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical.

When the reactor is critical, means for showing the relative reactivity status of the reactor is provided by control bank positions displayed in the control room. The position of the control banks is directly related to the reactivity status of the reactor when at power and any unexpected change in the position of the control banks under automatic control or change in the coolant temperature under manual control provides a direct and immediate indication of a change in the reactivity status of the reactor. Periodic samples of the coolant boron concentration are taken. The variation in concentration during core life provides a further check on the reactivity status of the reactor including core depletion.

High nuclear flux protection is provided both in the power and intermediate ranges by reactor trips actuated from either range if the neutron flux level exceeds trip set-points. When the reactor is critical, the best indications of the reactivity status in the core (in relation to the power level and average coolant temperature) is the control room display of the rod control group position.

7.4.2 SYSTEM DESIGN

The three instrumentation ranges are provided with overlap between adjacent ranges so that continuous readings will be available during transition from one range to another. The sensitivity of the neutron detector is illustrated on Figure 7.4-1.

Detectors

The system consists of a total of six detector assemblies located in instrument wells around the reactor as shown in the (plan view) lower right hand corner of Figure 7.4-2. The six assemblies provide the following instrumentation:

a) Power Range

This range consists of four independent long uncompensated ionization chamber assemblies. Each assembly is made up of two sensitive lengths. One sensitive length covers the upper half of the core, and the other length covers the lower half of the core.

The arrangement provides in effect a total of eight separate ionization chambers approximately one-half the core height. The eight uncompensated (guard-ring) ionization chambers sense thermal neutrons in the range from 2.5×10^3 to 2.5×10^{10} neutrons per sq cm per sec.

Each has a nominal sensitivity of 1.7×10^{-13} amperes per neutron per sq cm per sec. The four long ionization chamber assemblies are located in vertical instrument wells adjacent to the four "corners" of the core. The assembly is manually positioned in the assembly holders and is electrically isolated from the holder by means of insulated stand-off rings.

b) Start-Up Range (Intermediate and Source)

There are two separate assemblies. Each assembly covers two ranges. Each assembly contains one compensated ionization detector (intermediate range) and one proportional counter (source range).

The source range neutron detectors are boron -10 lined proportional counters with a nominal sensitivity of 10 counts per neutron per sq cm. The detectors sense thermal neutrons in the range from 10^{-1} to 5×10^4 neutrons per sq cm per sec, to produce a pulse rate between 10^0 and 5×10^5 counts per second. The range of the source range channel is 10^0 to 10^6 counts per second.

The neutron detectors are positioned in detector assembly containers by means of a linear, high density moderator insulator. The detector and insulator units are packaged in a housing which is inserted into the guide thimbles.

The detector assembly is electrically isolated from the guide thimble by means of insulated stand-off rings.

The intermediate range neutron detectors are compensated ionization chambers that sense thermal neutrons in the range from 2.5×10^2 to 2.5×10^{10} neutrons per sq cm per sec and have a nominal sensitivity of 4×10^{-14} amperes per neutron per sq cm per second. They produce a corresponding direct current of 10^{-11} to 10^{-3} amp. These detectors are located in the same detector assemblies as the proportional counters for the source range channels.

The electronic equipment for each of the source, intermediate and power range channels is contained in a draw-out panel mounted adjacent to the main control board.

Power Range Channels

There are three sets of power range measurements. Each set utilizes four individual currents as follows:

a) Four currents directly from the lower sections of the long ionization chambers.

b) Four currents directly from the upper sections.

c) Four total currents of (a) and of (b), equivalent to the average of each section.

For each of the four currents in (a) and (b), the current measurement is indicated directly by a microammeter and isolated signals are available for control console indication and recording. An analog signal proportional to individual currents are transmitted through buffer amplifiers to the overtemperature and overpower ΔT channels and provide automatic reset of the trip point for these protection functions. The total current, equivalent to the average, is then applied through a linear amplifier to the bistable trip circuits. The amplifiers are equipped with gain and bias controls for adjustment to the actual output corresponding to 100 per cent rated reactor power.

Each of the four amplifiers also provides amplified isolated signals to the main control board for indication and for use in the Reactor Control System. Each set of bistable trip outputs is operated as a 2-out-of-4 coincidence to initiate a reactor trip. Bistable trip outputs are provided at low and high power set points depending on the operating power. To provide more protection during start-up operation the low range power bistable is used. This trip is manually blocked after a permissive condition is obtained by 2 of 4 power range channels. The high power trip bistable is always active.

The four amplifier signals corresponding to (c) are supplied to circuits which compare a referenced channel output with the corresponding signal from the other channels. Alarms are provided to present deviations which might be indicative of quadrant flux asymmetries.

The overpower trip will be set so that, for operating limit reactor conditions concurrent with the maximum instrumentation and bistable set point error, the maximum reactor overpower condition will be limited to 112 per cent. This limit is accomplished by the use of solid state instrumentation and long ionization chambers which permit an integration of the flux external to the core over the total length of the core, thereby reducing the influence of axial flux distribution changes due to control rod motion.

The ion chamber current of each detector is measured by sensitive meters with an accuracy of 0.5 per cent. A shunt assembly and switch in parallel with each meter allows selection of one of four meter ranges. The available ranges are 0-100, 0-500, 0-1,000 and 0-5,000 microamperes. The shunt assemblies are designed in such a manner that they will not disconnect the detector current to the summing assembly upon meter failure or during switching. An isolation amplifier provides an analog signal proportional to ion chamber current for recording, data logging and delta flux indication. A test calibration unit provides necessary switches and signals for checking and calibrating the power range channels.

The linear amplifier accepts the output currents from each of the two chamber sections and derives a nuclear power signal proportional to the summed direct currents. This unit amplifies the currents and converts the normal current signal to a voltage signal suitable for operation of associated components such as bistables and isolation amplifiers.

Multiple power supplies furnish necessary positive and negative voltages for the individual channels and detector power.

Mounted on the front panel of each power range channel drawer are the ion chamber current meters, shunt selector switches with appropriate positions and the nuclear power indicator (0 to 120 per cent full power).

The isolated nuclear power signals are available for recording by the nuclear instrumentation system recorder. An isolated nuclear power signal is available for recording overpower conditions up to 200% full power.

Alarm signals for dropped-rod - rod stop, overpower - rod stop, overpower reactor trip, and channel test are annunciated on the main control board. Control signals sent to the reactor control and protection system include dropped-rod - rod stop, overpower - rod stop, overpower - reactor trip, and permissive circuit signals. These are described in Section 7.2.

Intermediate Range Channels

There are two intermediate range channels, which utilize two compensated ionization chambers. Direct current from the ion chambers is transmitted through triaxial cables to transistor logarithmic current amplifiers in the nuclear instrumentation equipment.

The logarithmic amplifier derives a signal proportional to the logarithm of the current as received from the output of the compensated ion chamber. The output of the logarithmic amplifier provides an input to the level bistables for reactor protection purposes and source range cutoff. The bistable trip units are similar to those in the other ranges. The trip outputs can be manually blocked after receiving a permissive signal from the power range channels. On decreasing power the intermediate range trips for reactor protection are automatically inserted when the power range permissive signal is not present.

Low voltage power supplies contained in each drawer furnish the necessary positive and negative voltages for the channel electronic equipment. Two medium voltage power supplies, one in each channel, furnish compensating voltage to the two compensated ion chambers. The high voltage for the compensated ion chambers is supplied by separate power supplies also located in the intermediate range drawers.

On the front panel of the intermediate range channel cabinet and on the control board are mounted a neutron (log N) flux level indicator calibrated in terms of ion chamber current (10^{-11} to 10^{-3} amp).

Isolated neutron flux level signals are available for recording and start-up rate computation. The start-up rate for each channel is indicated at the main control board in terms of decades per minute over the range of -.5 to 5.0 DPM.

Channel test, intermediate channel above source range cutoff point, intermediate range trips not armed, block rod withdrawal, and reactor trip signals are alarmed on the main control board annunciator. The latter signal is sent to the reactor protection system.

Source Range Channels

There are two source range channels utilizing B_{10} lined proportional counters. Neutron flux, as measured in the primary shield area, produces current pulses in the detectors. These preamplified pulses are applied to transistor amplifiers and discriminators located in the control room. Triaxial cable is used for all interconnections from the detector assemblies to the instrumentation in the control room. The preamplifiers are located outside the reactor containment.

These channels indicate the source range neutron flux and start-up rate and provide high flux level reactor trip and alarm signals to the Reactor Control and Protection System. The reactor trip signal is manually blocked when a permissive signal from the intermediate range is available. They are also used at shutdown to provide audible alarms in the reactor containment and control room of any inadvertent increase in reactivity. An audible count rate signal is used during initial phases of startup and is audible in both the reactor containment and control room.

Amplifiers are used to obtain a high level signal prior to elimination of noise and gamma pulses by the discriminator. The discriminator output is shaped for use by the log integrator.

The log integrator derives an analog signal, proportional to the logarithm of the number of pulses per unit time, as received from the output of the previous unit. This unit performs log integration of the pulse rate to determine the count rate, a linear amplifier amplifies the log integrator output for indication, recording, control, and rate computation through isolation amplifiers.

Each source range contains two bistable trip units. Both units trip on high flux level but one is used during shutdown to alarm reactivity changes and the other provides overpower protection during shutdown and startup. The shutdown alarm unit is blocked manually prior to startup or can serve as a start-up alarm. When the input to either unit is below its set point, the bistable is in its normal position and assumes a "fully-on" status. When an input from the log amplifier reaches or exceeds the set point, the unit reverses its condition and goes "fully-off". The output of the reactor trip unit controls relays in the reactor protection system.

Power supplies furnish the positive and negative voltages for the transistor circuits and alarm lights and the adjustable high voltage for the neutron detector.

A test calibration unit can insert selected test or calibration signals into the preamplifier channel input, or the log amplifier input. A set of precalibrated level signals are provided to perform channel tests and calibrations. An alarm is registered on the main control board annuclator whenever a channel is being tested or calibrated. A trip bypass switch is also provided to prevent a reactor trip during channel test under certain reactor conditions.

The neutron detector high-voltage cutoff assembly receives a trip signal when a one-of-two matrix controlled by intermediate range channel flux level bistables and manual block condition are present and disconnects the voltage the source range channel high voltage power supply to prevent operation of the boron -10 lined counter outside its design range. High voltage and reactor trip circuits are reactivated automatically when two of the intermediate range signals are below the permissive trip setting.

Mounted on the front panel of the source range channel is a neutron flux level indicator calibrated in terms of count rate level (1 to 10^6 cps). Mounted on the control board is a neutron count rate level indicator (1 to 10^6 cps). Isolated neutron flux signals are available for recording by the nuclear instrumentation system recorder and startup rate computation. The startup rate for each channel is indicated at the main control board in terms of decades per minute over the range of -0.5 to +5.0 DPM. The isolation network for these signals prevents any electrical malfunction in the external circuitry from affecting the signal being supplied to the flux level bistables. The signals for channel test, high neutron flux at shutdown, and source reactor trip are alarmed on the main control board annunciator. In addition, there are annunciators for the following source range conditions: "manual block of hi-flux level at shutdown", "source range level trip not armed," and "loss of high voltage", and individual "source trip bypass".

Auxiliary Equipment

Comparator Channel

The comparator channel compares the four nuclear power signals of the power range channels with one another. A local alarm on the channel is actuated when any two channels deviate from one another by a preset adjustable amount. During full power operation, the comparator serves to sense and annunciate channel failures and/or deviations.

Dropped Rod Protection

As backup to the primary protection for the dropped RCC accident, the rod bottom signal, there is provided an independent detection means using the out-of-core power range nuclear channels. The dropped-rod sensing unit contains a difference amplifier, which compares the instantaneous nuclear power signal with an adjustable power lag signal and responds with a trip signal to the bistable amplifier when the difference exceeds a preset adjustable amount. Above a given power level the signal blocks automatic rod withdrawal and initiates protective action in the form of a turbine load cutback.

Audio Count Rate Channel

The audio count rate channel provides audible source range information during refueling operations in both the control room and the reactor containment. In addition, this channel signal is fed to a scaler-timer assembly which produces a visual display of the count rate for an adjustable sampling period.

Recorders

One large, two-pen strip chart recorder is mounted on the main control board for recording the complete range of the source and intermediate channels. It is also possible to record any two power range channels as linear signals. Variable chart speeds will be provided. It also includes controls for changing the span and zero during intermediate range operation.

Switching of inputs to the recorders does not cause any spurious signals, that would initiate false alarms or reactor trips.

Four two-pen recorders are provided, one for each power range, to record the flux level from each of the eight sections comprising the four long ion chambers.

Power Supply

The nuclear instrumentation system is powered by four independent vital bus circuits (see Section 8).

7.4.3 SYSTEM EVALUATION

Loss of Power

The nuclear instrumentation draws its primary power from vital instrument buses and is discussed in Section 8.

Loss of nuclear instrumentation power would result in the initiation of all reactor trips associated with the channel power failure. In addition, all trips which were blocked prior to loss would be unblocked and initiated.

Reliability and Redundancy

The requirements established for the reactor protective system apply to the nuclear instrumentation. All channel functions are independent of every other channel.

Safety Factors

The relation of the power range channels to the Reactor Protective System has been described in Section 7.2. To maintain the desired accuracy in trip action, the total error from drift in the power range channels will be held to ± 1 per cent at full power. Routine tests and recalibration will ensure that this degree of deviation is not exceeded. Bistable trip set points of the power range channels will also be held to an accuracy of ± 1 per cent of full power. The accuracy and stability of the equipment will be verified by vendor tests.

Overpower Trip Set Point

The overpower trip set point for the Indian Point Unit No. 2 Reactor is 106.5%. This trip point was selected to provide adequate assurance that spurious reactor trips will not occur in normal operation. Table 7.4-1 gives a comparison of the factors which determine the trip set point for the Indian Point Unit No. 2 reactor and those used in other designs.

As shown by the information in Table 7.4-1, the difference in set point is due to reduction in errors due to RCC motion from +4% to $\pm 1.5\%$ and a reduction in draft and set point errors from $\pm 4\%$ to $\pm 2\%$.

The reduction in errors due to RCC motion is made possible by the use of long ion chambers instead of short ion chambers at the top and bottom of the core. These long ion chambers, which extend over almost the entire core length, give a more accurate indication of flux changes than the combination of detectors at the top and bottom of the core.

The reduction of drift and set point errors is due to the use of more accurate, solid state electronic devices. Application of the reduced trip set point and improved performance of these control devices results in a maximum calculated overpower of 112% as compared with 118% obtained with previous designs. Table 7.4-2 gives a comparison of factors to which this difference is attributed.

It is seen that the reduction in overpower is due to (1) a reduction from 3.0 to 2.0 in the allowed calorimetric error, based on experience with conventional stations, and (2) the effects of reduced errors due to rod motion, drift, and set point reproducibility as described earlier. The net result is an improvement in plant capability with the same assurance against both spurious trips and excessive power as was afforded by previous designs.

TABLE 7.4-1

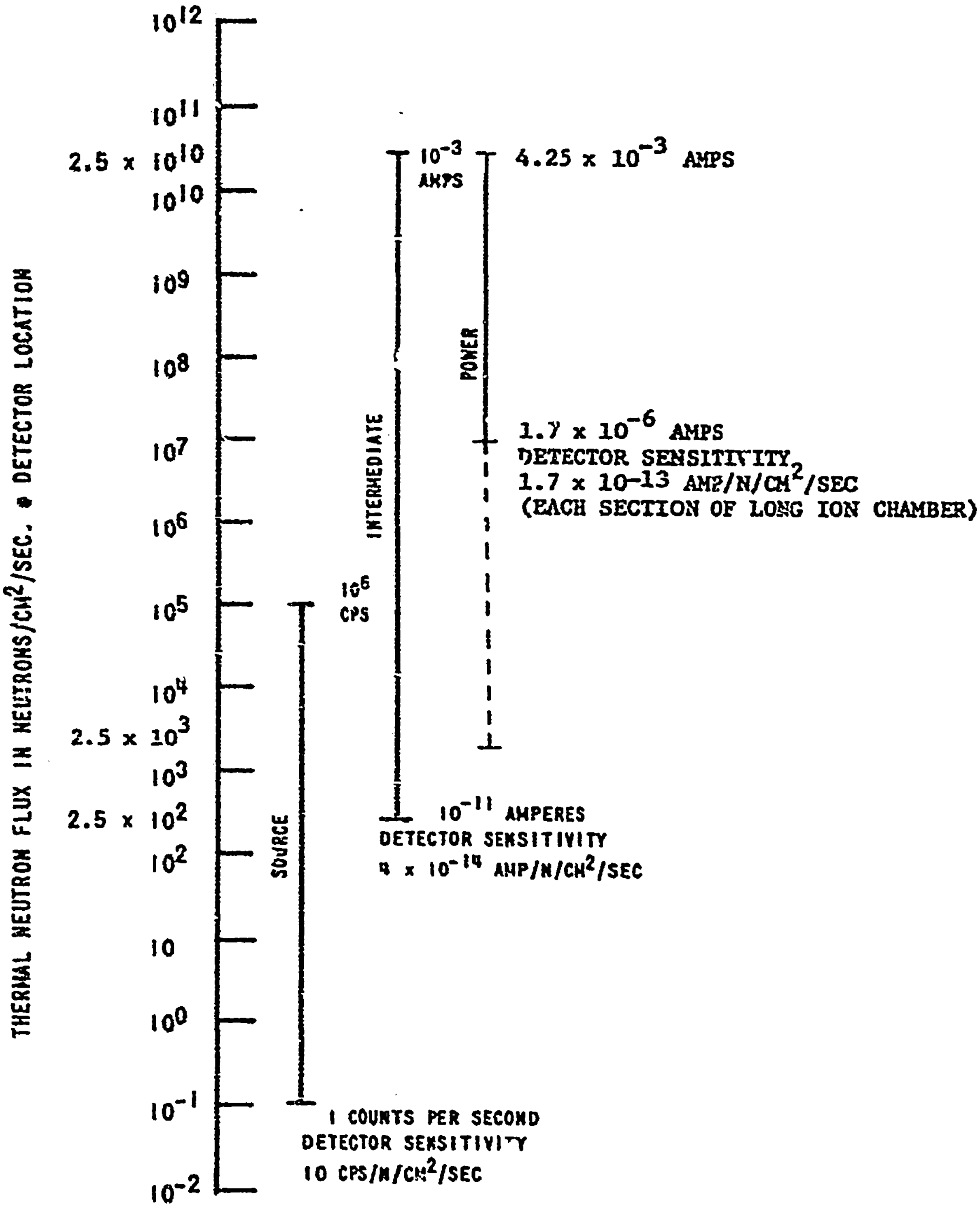
OVERPOWER TRIP SET POINT

Set Point Factor	Indian Point Unit No. 2 (per cent)	Previous Designs (per cent)
Normal Power	100.0	100.0
Overshoot	3.0	3.0
Errors in Ion Chamber Output Due to RCC Motion for Design Transients	1.5	4.0
Errors Due to Drift and Set Point Reproducibility	2.0	4.0
Set Point	106.5	111.0

TABLE 7.4-2

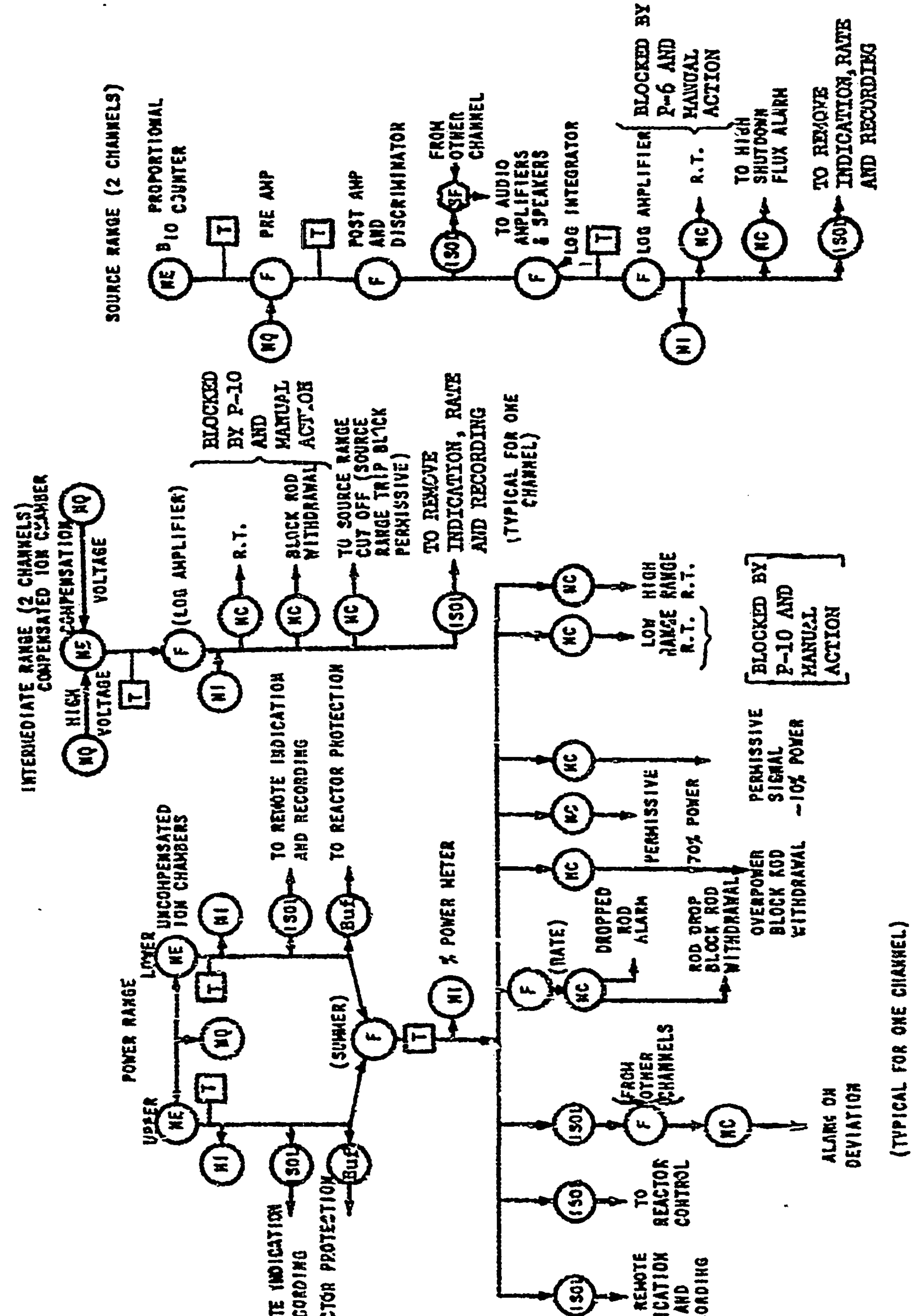
MAXIMUM OVERPOWER LEVEL

Overpower Factor	Indian Point Unit No. 2 (per cent)	Previous Designs (per cent)
Normal Power	100.0	100.0
Calorimetric Error	2.0	3.0
Overshoot	3.0	3.0
Errors in Ion Chamber Output Due to RCC Motion for Design Transient	3.0	4.0
Errors Due to Drift and Set Point Reproducibility	4.0	8.0
Maximum Overpower	112.0	118.0

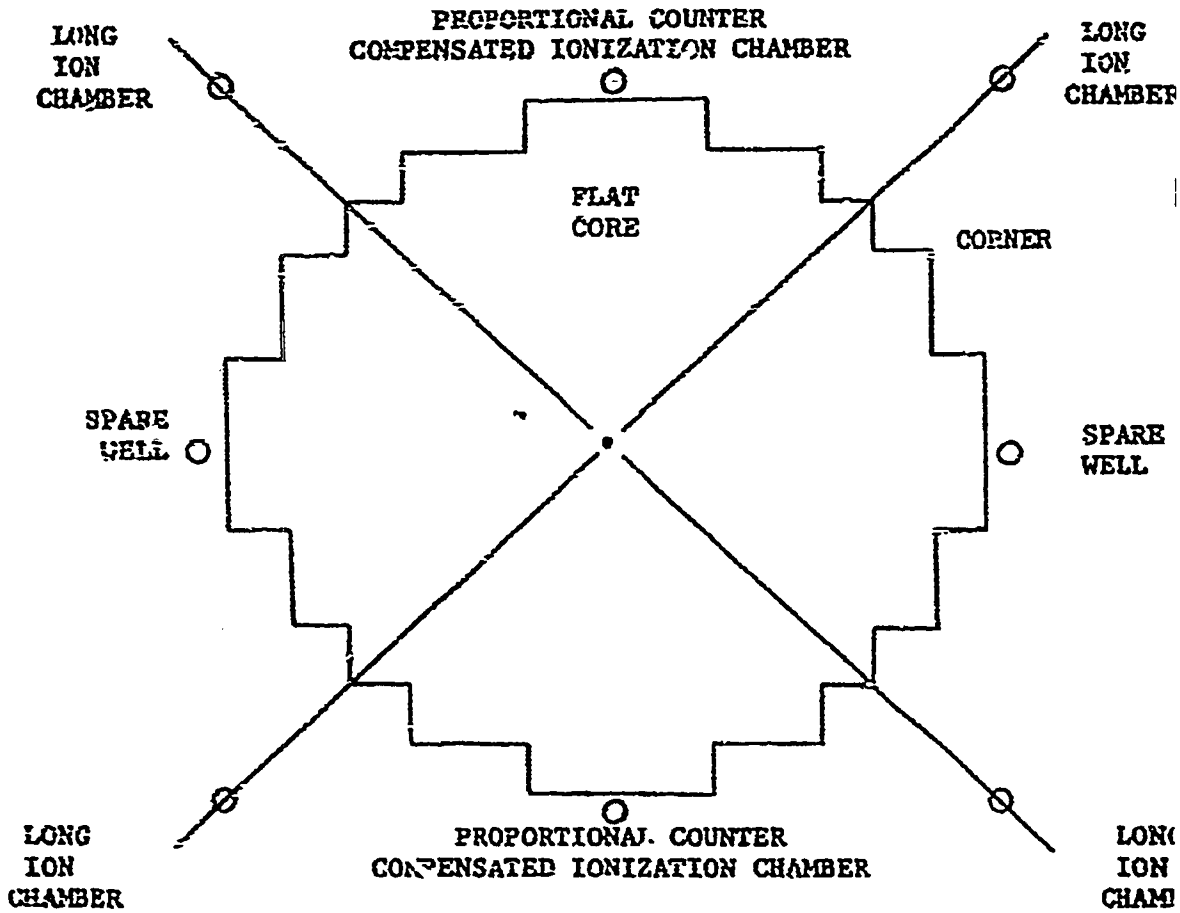


NEUTRON DETECTORS AND RANGE OF OPERATION

FIG. 7.4-1



NUCLEAR PROTECTION SYSTE[M]
FIG. 7.4-



PLAN VIEW INDICATING DETECTOR LOCATION RELATIVE TO COR

FIGURE 7.4-

7.5 PROCESS INSTRUMENTATION

7.5.1 DESIGN BASES

The non-nuclear process instrumentation measures temperatures, pressures, flows, and levels in the reactor coolant system, steam system, reactor containment and auxiliary systems. Process variables required on a continuous basis for the startup, operation, and shutdown of the unit are indicated, recorded and controlled from the control room. The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems and processes over the full operating range of the plant.

Certain controls which require a minimum of operator attention, or are only in use intermittently, are located on local control panels near the equipment to be controlled. Monitoring of the alarms of such control systems are provided in the control room.

7.5.2 SYSTEM DESIGN

Much of the process instrumentation provided in the plant has been described in the Reactor Control and Protection and Nuclear Instrumentation System. The most important instrumentation used to monitor and control the plant have been described in the above systems descriptions. The remaining portion of the process instrumentation is generally shown on the respective systems process flow diagrams.

Condensate pots and wet legs are used to prevent process temperatures from actually reaching the transmitters.

Engineered Safety Features

The following instrumentation ensures coverage of the effective operation of the engineered safety features:

Containment Pressure

The containment pressure is transmitted to the main control board for post accident monitoring. Six transmitters are installed outside the containment to prevent potential missile damage. The pressure is indicated (all six channels) on the main control board.

The six channels, monitoring containment pressure, reflect the effectiveness of engineered safety features.

Containment Sump Level

Redundant (two) containment sump level indicators in each sump show that water has been delivered to the containment following a loss of coolant accident and, subsequently show that sufficient water has been collected by the pump to permit recirculation to the reactor and/or to the spray headers. These transmitters are mounted inside the containment and are designed to withstand accident conditions. Each board indication is a series of 5 lights each energized from the associated instrument as a pre-set level is exceeded. Reference is made to Section 6 for further description of the two sumps serving the internal and external recirculation loops.

Refueling Water Storage Tank Level

Two redundant channels indicate that safety injection and containment spray systems have removed water from the storage tank. One is located at the tank and the other is transmitted from the tank to a control board indicator.

Safety Injection Pumps Discharge Pressure

These channels show that the safety injection pumps are operating. The transmitters are outside the containment.

Pump Energization

All pump motor power feed breakers indicate that they have closed by energizing indicating lights on the control board.

Valve Position

All engineered safety features valves have position indication on the control board to show proper positioning of the valves. Air-operated and solenoid-operated valves are selected so as to move in a preferred direction on the loss of air or power. Motor-operated valves remain in the position at time of loss of power to the motor. Spray flow is also indicated on the control board.

Residual Heat Exchangers

Combined exit flow is indicated and combined inlet and combined exit temperatures are recorded on the control board to monitor operation of the residual heat exchangers.

Air Coolers

The service water discharge flow and temperature of each of the five air recirculation coolers are indicated locally. A control room alarm is actuated if the flow is low. The transmitters are outside the reactor containment. In addition, the exit flow is monitored for radiation and alarmed in the control room if high radiation should occur. This is a common monitor and the faulty cooler can be located by manually blocking the flow to each unit in turn with locally operated valves.

Alarms

Visual and audible alarms are provided to call attention to abnormal conditions. The alarms are of the individual acknowledgement type; that is, the operator must recognize and silence the audible alarm for each alarm point. For most control systems, the sensing device and circuits for the alarms are independent, or isolated from, the control devices.

In addition to the above, the following local instrumentation is available.

a. Containment spray test lines total flow
b. Safety injection test line pressure and flow

7.5.3 SYSTEM EVALUATION

Redundant instrumentation has been provided for all inputs to the protective systems and vital control circuits.

Where wide process variable ranges and precise control are required, both wide range and narrow range instrumentation is provided.

Instrumentation components are selected from standard commercially available products.

All electrical and electronic instrumentation required for safe and reliable operation is supplied from four redundant instrumentation busses.

7.6 IN-CORE INSTRUMENTATION

7.6.1 DESIGN BASIS

The in-core instrumentation is designed to yield information on the neutron flux distribution and fuel assembly outlet temperatures at selected core locations. Using the information obtained from the in-core instrumentation system, it is possible to confirm the reactor core design parameters and calculated hot channel factors. The system provides means for acquiring data and performs no operational plant control.

7.6.2 SYSTEM DESIGN

The in-core instrumentation system consists of thermocouples, positioned to measure fuel assembly coolant outlet temperature at preselected locations; and flux thimbles, which run the length of selected fuel assemblies to measure the neutron flux distribution within the reactor core.

The experimental data obtained from the in-core temperature and flux distribution instrumentation system, in conjunction with previously determined analytical information, can be used to determine the fission power distribution in the core at any time throughout core life. This method is more accurate than using calculational techniques alone. Once the fission power distribution has been established, the maximum power output is primarily determined by thermal power distribution and the thermal and hydraulic limitations determine the maximum core capability.

The in-core instrumentation provides information which may be used to calculate the coolant enthalpy distribution, the fuel burnup distribution, and an estimate of the coolant flow distribution.

Both radial and azimuthal symmetry of power may be evaluated by combining the detector and thermocouple information from the one quadrant with similar data obtained from the other three quadrants.

Thermocouples

Chromel-alumel thermocouples are threaded into guide tubes that penetrate the reactor vessel head through seal assemblies, and terminate at the exit flow end of the fuel assemblies. The thermocouples are provided with two primary seals, a conseal and swage type seal from conduit to head. The thermocouples are enclosed in stainless steel sheaths within the above tubes to allow replacement if necessary. Thermocouple readings are recorded in the control room. The support of the thermocouple guide tubes in the upper core support assembly is described in Section 3.

Movable Miniature Neutron Flux Detectors

Mechanical Configuration

Six fission chamber detectors (employing U_3O_8 which is 90 percent enriched in U_{235}) can be remotely positioned in retractable guide thimbles to provide flux mapping of the core. Maximum chamber dimensions are 0.188-inch in diameter and 2.10 inches in length. The stainless steel detector shell is welded to the leading end of the helical wrap drive cable and the stainless steel sheathed coaxial cable. Each detector is designed to have a minimum thermal neutron sensitivity of 1.5×10^{-17} amps/nv and a maximum gamma sensitivity of 3×10^{-14} amps/R/hr. Operating thermal neutron flux range for these probes is 1×10^{11} to 8×10^{13} nv. Other miniature detectors, such as gamma ionization chambers and boron-lined neutron detectors, can also be used in the system. The basic system for the insertion of these detectors will be shown in Figure 7.6-1 and 7.6-2. Retractable thimbles into which the miniature detectors are driven are pushed into the reactor core through conduits which extend from the bottom of the reactor core through conduits which extend from the bottom of the reactor vessel down through the concrete shield area and then up to a thimble seal zone.

The thimbles will be closed at the leading ends, are dry inside, and serve as the pressure barrier between the reactor water pressure and the atmosphere. Mechanical seals between the retractable thimbles and the conduits are provided at the seal line.

During reactor operation, the retractable thimbles are stationary. They are extracted downward from the core during refueling to avoid interference within the core. A space above the seal line is provided for the retraction operation.

The drive system for the insertion of the miniature detectors consists basically of six drive assemblies, six path group selector assemblies and six rotary selector assemblies, as shown in Figures 7.6-1A and 7.6-1B. The drive system pushes hollow helical-wrap drive cables into the core with the miniature detectors attached to the leading ends of the cables and small diameter sheathed coaxial cables threaded through the hollow centers back to the ends of the drive cables. Each drive assembly generally consists of a gear motor which pushes a helical-wrap drive cable and detector through a selective thimble path by means of a special drive box and includes a storage device that accommodates the total drive cable length. Further information on mechanical design and support is described in Section 3.

Control and Readout Description

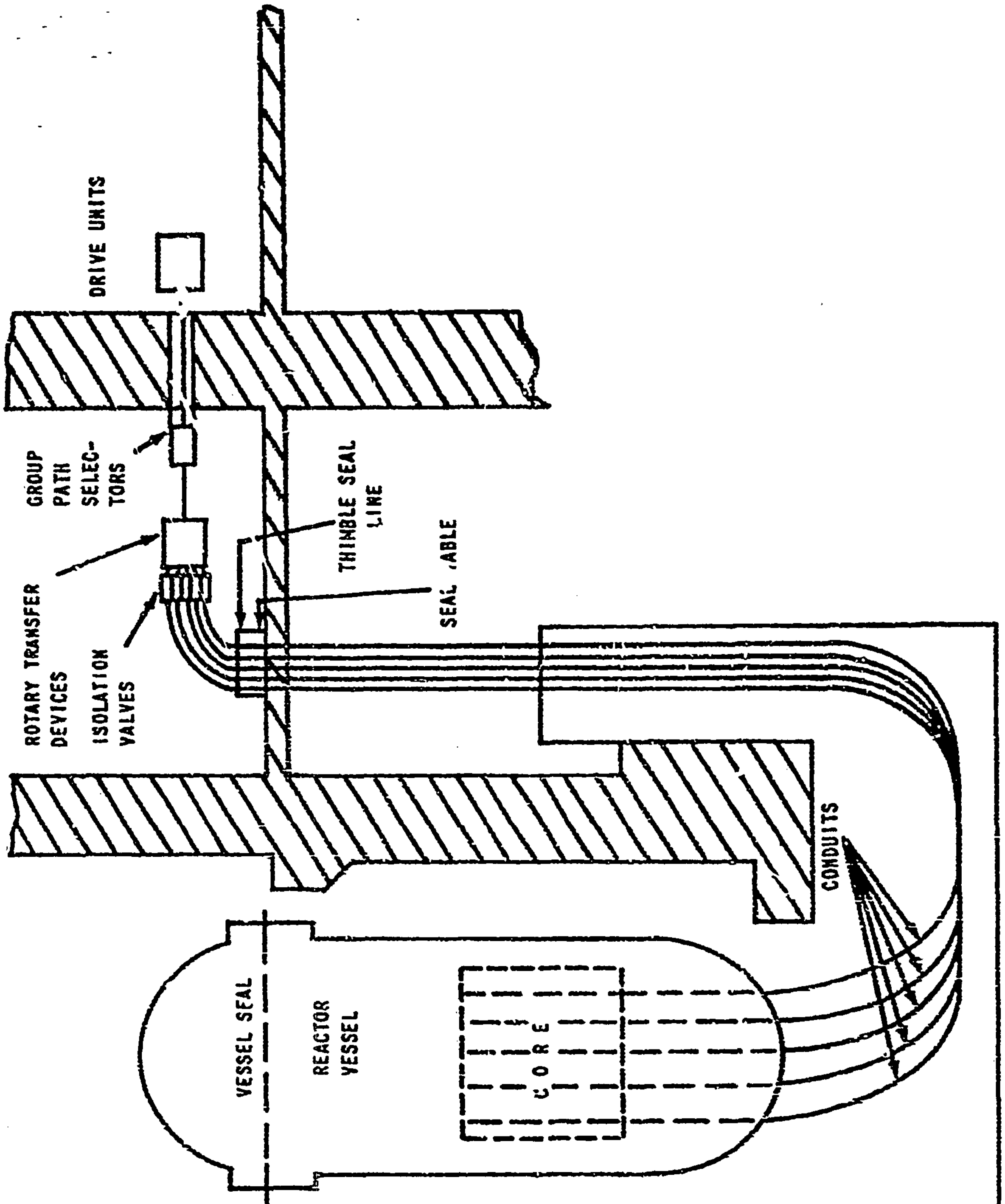
The control and readout system provides means for inserting the miniature neutrons detectors into the reactor core and withdrawing the detectors at a selected speed while plotting a level of induced radioactivity versus detector position. The control system consists of two sections, one physically mounted with the drive units, and the other contained in the control room. Limit switches in each drive conduit provide means for pre-recording detector and cable positioning in preparation for a flux mapping operation. Each gear box drives an encoder data positional data plotting. One group path selector is provided for each drive unit to route the detector into one of the flux thimble groups. A rotary transfer assembly is a transfer device that is used to route a detector into any one of up to ten selectable paths. Ten manually operated isolation valves allow free passage of the detector and drive wire when open, and prevents steam leakage from the core in case of a thimble rupture, when closed. A path common to each group of flux thimbles is provided to permit cross calibration of the detectors.

The control room contains the necessary equipment for control, position indication, and flux recording. Panels are provided to indicate the core position of the detectors, and for plotting the flux level versus the detector position. Additional panels are provided for such features as drive motor controls, core path selector switches, plotting and gain controls. A "flux-mapping" consists, briefly, of selecting (by panel switches) flux thimbles in given fuel assemblies at various core quadrant locations. The detectors are driven or inserted to the top of the core and stopped automatically. An x-y plot (position vs. flux level) is initiated with the slow withdrawal of the detectors through the core from top to a point below the bottom. In a similar manner other core locations are selected and plotted.

Each detector provides axial flux distribution data along the center of a fuel assembly. Various radial positions of detectors are then compared to obtain a flux map for a region of the core.

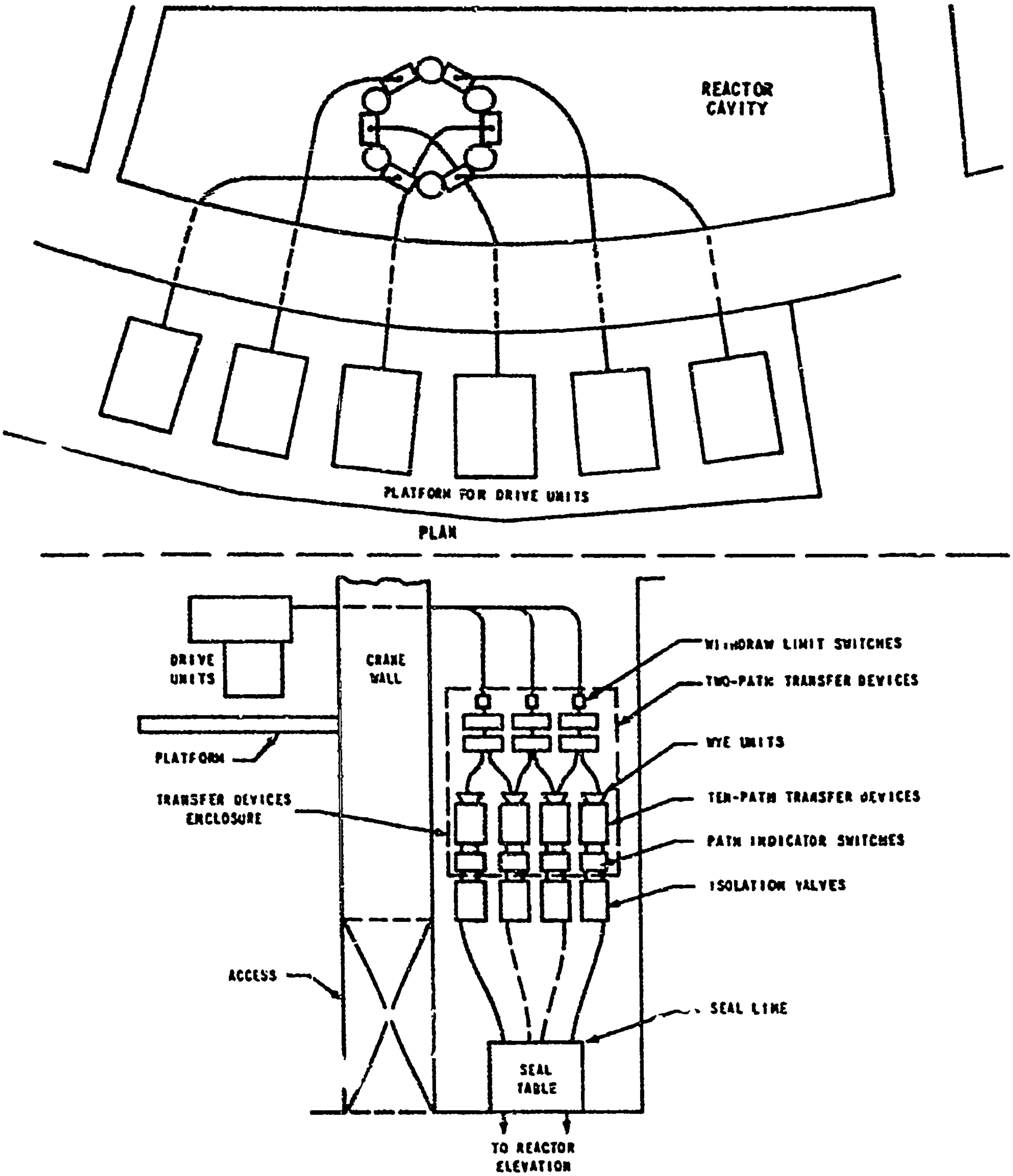
7.6.3 SYSTEM EVALUATION

The thimbles are distributed nearly uniformly over the core with about the same number of thimbles in each quadrant. The number and location of thimbles have been chosen to permit measurement of local to average peaking factors to an accuracy of ± 10% (95% confidence). Measured nuclear peaking factors will be increased by 10% to allow for possible instrument error. The DNB ratio calculated with the measured hot channel factor will be compared to the DNB ratio calculated from the design nuclear hot channel factors. If the measured power peaking is larger than expected, reduced power capability will be indicated.



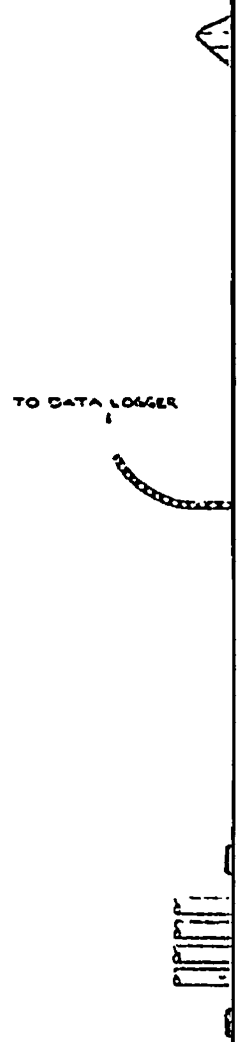
Typical Arrangement of Movable Miniature Neutron Flux Detector System
(Elevation View)

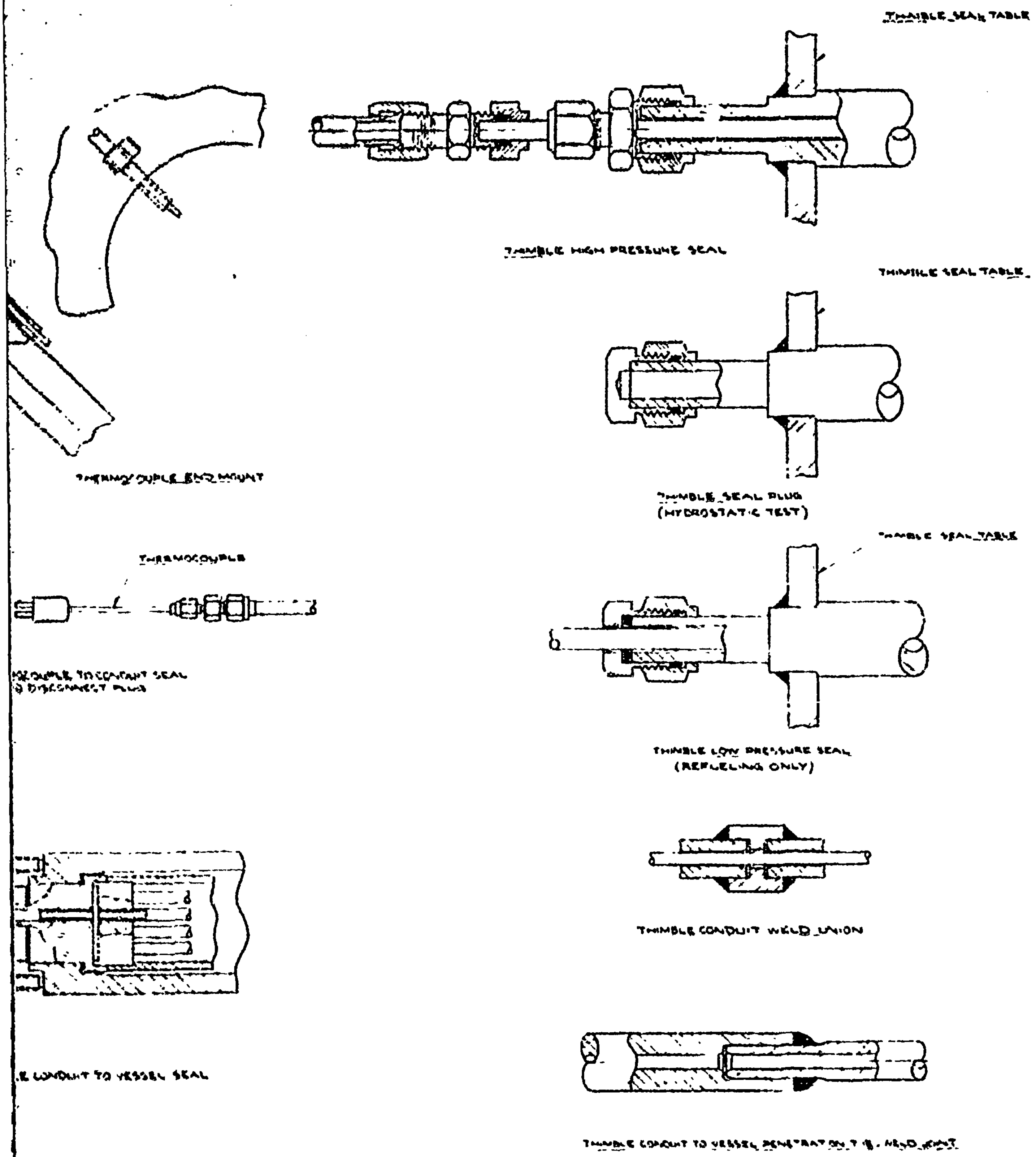
Figure 7.6-1A



ARRANGEMENT OF IN-CORE FLUX DETECTOR

FIG. 7.6-1B


TO DATA LOGGER



IN-CORE INSTRUMENTATION - DETAILS
FIG. 7.6-2

7.7 OPERATING CONTROL STATIONS

7.7.1 STATIONS LAYOUT

The principal criteria of control station design and layout is that all controls, instrumentation displays and alarms required for the safe operation and shutdown of the plant are readily available to the operators in the control room.

During other than normal operating conditions, other operators will be available to assist the control operator. Figure 7.7-1 shows the control room layout for the unit. The control board is divided into relative areas to show the location of control components and information display pertaining to various sub-systems.

7.7.2 INFORMATION DISPLAY AND RECORDING

Alarms and annunciators in the control room provide the operators warning of abnormal plant conditions which might lead to damage of components, fuel or other unsafe conditions. Other displays and recorders are provided for indication of routine plant operating conditions and for the maintenance of records.

Consideration is given to the fact that certain systems normally require more attention from the operator. The control system therefore is centrally located on the three section board.

On the left section of the control board, individual indicators present a direct, continuous readout of every control rod position. Fault detectors in the rod drive control system are used to alert the operator should an abnormal condition exist for any individual or group of control rods. Displayed in this same area are limit lights for each control rod group and all nuclear instrumentation information required to start up and operate the reactor. Control rods are manipulated from the left section.

Variables associated with operation of the secondary side of the station are displayed and controlled from the control board. These variables include steam pressure and temperature, feedwater flow and temperature, electrical load, and other signals involved in the plant control system. The control board also contains provisions for indication and control of the reactor coolant system. Redundant indication is incorporated in the system design since pressure and temperature variables of the Reactor Coolant System are used to initiate safety features. Control and display equipment for station auxiliary systems are also located here.

The Engineered Safety Features Systems are controlled and monitored from a vertical panel to the left of the control board. Valve position indicating lights are provided as a means of verifying the proper operation of the control and isolation valves following initiation of the engineered safety features. Control switches located on this panel allow manual operation or test of individual units. Also located on this section are the control switches, indicating lights, and meters for fans and pumps required for emergency conditions. Also mounted on this section are auxiliary electrical system controls required for manual switching between the various power sources described in Section 8.2.2.

Controls and indications for all ventilation systems, the containment isolation valves, and the Isolation Valve Seal Water System are located on a vertical panel. Radiation monitoring information is indicated immediately behind and to the left of the main control board.

Audible Reactor Building alarms are initiated from the radiation monitoring system and from the source range nuclear instrumentation. Audible alarms will be sounded in appropriate areas throughout the station if high radiation conditions are present.

7.7.3 EMERGENCY SHUTDOWN CONTROL

The Control Room, its equipment and furnishings have been designed so that the likelihood of fire or other conditions which could render the control room inaccessible even for a short time is extremely small.

A criterion of the station design and layout is that all controls, instrumentation displays and alarms required for the safe operation and shutdown of the plant are readily available to the operators in the control room.

It is design policy that the functional capacity of the control room shall be maintained at all times inclusive of accident conditions, such as an MCA or a fire; the following features are incorporated in the design to ensure that this criterion is met.

Structural and finish materials for the control room and the cable spreading room below were selected on the basis of fire resistant characteristics. Structural floors are concrete reinforced. Interior partiti are metal paneling joints. The control room ceiling covering is fire retardant egg crate diffusers. Door frames and doors are metallic. Wood trim is not used.

The control room is equipped with portable fire extinguishers sized and located in accordance with National Fire Code and National Fire Protection Association specifications. Extinguishers carry the Underwriter's Laboratory label of approval and are electrical shock resistant.

The cable spreading room has a fire and smoke detection system and flame sensing devices which can actuate alarms.

The control room ventilation consists of a system having a large percentage of recirculated air. The fresh air intake can be closed to control the intake of airborne activity if monitors indicate that such action is appropriate.

Control cables used throughout [illegible] allation have been selected on the basis of flame testing described i[illegible] ection 8 and have superior flame retardant capability. Each conductor has a flame retardant glass braid over the insulation. In addition electrical circuits are limited in the control room to those associated with lighting, instrumentation and control. Lighting circuits operate on 120 volts; instrumentation and control circuits operated at either 120 volt ac, 125 volt dc or at millivolt level. All 120 and 125 volt circuits are protected against both overload and short circuits by either fuses or circuit breakers. The power levels on the millivolt circuits are so low that it is inconceivable that short circuits in these could become a fire hazard.

All control and indication is transmitted into the control room assuring that no combustible process fluids are carried into the r

Cables that penetrate the control room floor pass through sealing devices to minimize fume and flame transmission from possible fire sources external to the control room.

All internal wiring in switchboards and instrument racks is type SIS cross linked polyethylene which has excellent resistance to propagation of flame. As a result of the design criterion discussed above the amount of combustible material in the control room is of such small quantity that a fire of the magnitude that would require evacuation of the control room is not credible.

As a further measure to assure safety, provisions have been made so that plant operators can shutdown and maintain the plant in a safe condition by means of controls located outside the control room. During such a period of control room inaccessibility the reactor will be tripped and the plant maintained in a hot shutdown condition. If the period extends for a long time the Reactor Coolant System can be borated to maintain shutdown as Xenon decays.

Local controls are located so that the stations to be manned and the times when attention is needed are within the capability of the plant operating staff. The plant intercom system will provide communication among the personnel so that the operation can be coordinated.

The functions for which local control provisions have been made are listed below along with the type of control and its location in the plant. Transfer to these local controls is annunciated in the control room.

Reactor Trip

If the control room should be evacuated suddenly without any action by the operators, the reactor can be tripped by either of the following.

1. Open rod control breakers in the control building.

2. Actuate the manual turbine trip at the control standard in the turbine building.

Following evacuation of the control room the following systems and equipment are provided to maintain the plant in a safe shutdown condition from outside the control room:

(a) Residual heat removal

(b) Reactivity control. i.e., boron injection to compensate for fission product decay

(c) Pressurizer pressure and level control

(d) Electrical system as required to supply the above systems

(e) Other equipment, as described

(a) Residual Heat Removal

Following a normal plant shutdown an automatic steam dump control syst bypasses steam to the condenser and maintains the reactor coolant temperature at its no load value. This implies the continued operation of the steam dump system, condensate circuit, condenser cooling water, feed pumps and steam generator instrumentation. Failure to maintain water supply to the steam generators would result in steam generator dry out after some 400 seconds and loss of the secondary system for decay heat removal. Redundancy and full protection where necessary is built into the system to ensure the continued operation of the steam generator units. If the automatic steam dump control system is not available independently controlled relief valves on each steam generator maintain the steam pressure. These relief valves are further backed up by coded safety valves on each steam generator. Numerous calculations, verified by start-up tests on the Connecticut-Yankee and San Onofre Power Plants have shown that with the steam generator safety valves operating alone the reactor coolant system maintains itself close to the nominal no load condition. The steam relief facility is adequately protected by redundancy and local protection. For decay heat removal it is only necessary to maintain the control on one steam generator.

For the continued use of the steam generators for decay heat removal, it is necessary to provide a source of water, a means of delivering that water and, finally, instrumentation for pressure and level indication.

The normal source of water supply is the secondary feed circuit; this implies satisfactory operation of the condenser, air ejector, condenser cooling circuit etc. In addition to the normal feed circuit the plant may fall back on:

(1) The condensate storage tanks
(2) The city water storage tank
(3) The city water supply.

Feed water may be supplied to the steam generators by the electrical feed pumps or by the steam driven feed pump, these pumps and associated valves have local controls.

(b) Reactivity Control

Following a normal plant shutdown to hot shutdown condition soluble poison is added to the primary system to maintain subcriticality. For boron addition the chemical and volume control system is used. Routine boration requires the use of:

> Charging pumps and volume control tank with associated piping. Boric Acid transfer pumps with tanks and associated piping. Let down station, non-regenerative heat exchanger and associated equipment Component cooling and service systems. Periodic operation of one M. C. pump for pressurizer homogenization; the auxiliary spray/heaters could be used if necessary. Compressed air for valve operation - manual could be adopted if necessary.

It is worthy of note that with the reactor held at hot shutdown conditions boration of the plant is not required immediately after shutdown. The Xenon transient does not decay to the equilibrium level until some 10 to 15 hours after shutdown and a further period would elapse before the 1% reactivity shutdown margin provided by the full length control rods had been cancelled. This delay would provide useful time for emergency measures.

(c) Pressurizer Pressure and Level Control

Following a reactor trip the primary temperature will automatically reduce to the no load temperature condition as dictated by the steam generator temperature conditions. This reduction in the primary water temperature reduces the primary water volume and if continued pressure control is to be maintained primary water makeup is required.

The pressurizer level is controlled in normal circumstances by the chemical and volume control system. This requirement implies the charging pump duty referred to for boration plus a guaranteed borated water supply. The facility for boration is provided as discribed above; it is only necessary to supply water for makeup. Water may readily be obtained from normal sources; i.e., the volume control tank.

Startup of Other Equipment

Although not directly related to plant safety, the air temperature inside containment should be kept below 120°F. For this reason the containment air recirculation fan coolers should be continuted in operation. If they have stopped, at least one should be restarted within five minutes with the others started later as required. Similarly the nuclear service water pumps will be checked and at least one of them restarted if none are already operating. The fan coolers and the service water pump controls are located in the switchgear room.

Electrical Systems

Offsite or onsite emergency power should be available to supply the above systems and equipment for the hot shutdown condition.

Indication and Controls Provided Outside the Control Room

The specific indication and controls provided outside the control room for the above capability are summarized as follows:

Indication

1) Level Indication For the Individual Steam Generators.
   One set visable from the auxiliary feed pumps
   One set visable from the main feed control valves

2) Pressure Indication For the Individual Steam Generators.
Visable from the auxiliary feed pumps.

3) Pressurizer Level and Pressure Indicators.
One set visable from the auxiliary feed pumps
One set visable from the charging pump local control point
All instruments at the auxiliary feed pumps are grouped on a local gauge board.

Controls

Local stop/start push button motor controls with a selector swich are provided at each of the following motors. The selector switch will transfer control of the switch gear from the control room to local at the motor. Placing the local selector switch in the local operating position will give an annunciator alarm in the control room and will turn out the motor control position lights on the control room panel.

1) Auxiliary Motor Driven Feedwater Pumps.
2) Charging Pumps.
3) Boric Acid Transfer Pumps.

Remote stop/start push button motor controls with a selector switch are provided for each of the following motors. These controls are grouped at one point in the switch gear room convenient for operation. The selector switch will transfer control of the switch gear from the control room to the remote point. Placing the selector switch to local operation will give an annunciator alarm in the control room and will turn out the motor control position lights on the control room panel.

1) Service Water Pumps.
2) Containment Air Recirculation Fans.
3) Control Room Air Handling Unit Including Control for the Air Inlet Dampe

Alternate Motor control points are not required for the following:

1) Component Cooling Water Pumps. (Automatically restarted on a blackout once the diesel generators are operating.)

2) Instrument Air Compressors and Cooling Pumps. (These will start automatically on low pressures in the air and water services, once the diesel automatically energizes the bus and the motor control centers are manually energized. The control point is local to the compressors.)

Speed Control

Speed control is provided locally for:

1) The Auxiliary Turbine Driven Feed Pump
2) The Charging Pump

<u>Valve Control</u>

Local Valve Control is Provided At The:

1) Main Feed Regulators.

2) Auxiliary Feed Control Valves. (These valves are located local to the auxiliary feed pumps.)

3) Atmospheric Dump. (Auto control normally at hot shutdown.)

4) All other valves requiring operation during hot standby can be locally operated at the valve.

5) Letdown orifices isolation valves locally to the charging pumps. Local stop and start buttons with selector switch and position lamp.

Pressurizer Heater Control

Stop and start buttons with selector switch and position .amp locally to the charging pumps for one 485KW backup heater group.

Lighting

Emergency lighting is provided in all operating areas as defined by the foregoing.

Communications

The communication network provides communications between the area of the auxiliary feed pumps and the charging pumps, boric acid transfer pumps, diesel generators, and the outside exchange without requiring the control room.

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Figure 7.7-1, Titled "Control Room Arrangement"

The above was **redacted** by NRC staff as **sensitive information** to allow release of this document to the public.

7.8 MAXIMUM SAFETY SYSTEM SETTINGS AND MINIMUM CONDITIONS FOR OPERATION

Table 7.2-1 lists the reactor protection and engineered safety features actuation systems and Table 7.2-2 lists the associated interlocks. Maximum permissible settings for safe operation for these functions are given in Technical Specifications, Section 15.

7.9 SURVEILLANCE REQUIREMENTS

The purpose of this section is to identify those items which are expected to be the subject of the Technical Specifications. Table 7.9-1 enumerates the instrumentation list of channels designated for such periodic testing. The type of test action (channel calibration or test function) to be taken during operation of the plant and the minimum frequency for testing (each refueling shift, or month) for the indicated channels will be supplied as part of the Technical Specifications.

As indicated, the instrumentation channels which are covered include for example, nuclear, reactor coolant temperature and flow, pressurizer pressure and level and auxiliary process channels or components necessary to assure that facility operation is maintained within the safe limits.

Table 7.9-2 enumerates further for system and/or system component. The frequency for the checks is supplied in the Technical Specifications.

TABLE 7.9-1

PERIODIC TESTS, CALIBRATIONS AND CHECKS OF INSTRUMENT CHANNELS

	Channel Description	Action	Minimum Frequency
1.	Nuclear Power	*	*
2.	Intermediate Range		
3.	Source Range		
4.	Reactor Coolant Temperature		
5.	Reactor Coolant Flow		
6.	Pressurizer Level		
7.	Pressurizer Pressure		
8.	Low Pressure Trip Set-Point		
9.	Rod Position Indicator		
10.	Rod position Counters		
11.	Steam Generator Level		
12.	Steam Generator Flow Mismatch		
13.	Charging Flow		
14.	Residual Heat Pump Flow		
15.	Boric Acid Tank Level		
16.	Refueling Water Storage Tank Level		
17.	Volume Control Tank Level		
18.	Reactor Containment Pressure		
19.	Radiation Monitoring System	*	*
20.	Boric Acid Control		
21.	Containment Sump Level		
22.	Valve Temperature Interlocks		
23.	Pump-Valve Interlock		
24.	Boron Injection Tanks		
25.	Engineering Safety Channels		

Code

Minimum Frequency	Action
D - Daily	Cal - Calibrate
W - Weekly	T - test
B/W - Biweekly	Ch. - Check
M - Monthly	
S - Each Shift	
R. - Each Refueling	

*This information to be supplied as part of Technical Specifications

TABLE 7.9-2

PERIODIC TESTS AND CHECKS OF EQUIPMENT AND SYSTEMS

		Check	Frequency
1.	Reactor Coolant Samples	Radio-chemical Analysis	*
2.	Reactor Coolant Boron	Boron concentration	
3.	Refueling Water Storage Tank Water Sample	Boron concentration	
4.	Control Rods	Check rod drop times of all rods	
5.	Control Rod	Partial movement of all rods	
6.	Pressurizer Safety Valves	Check set point	
7.	Main Steam Safety Valves	Check set point	
8.	Main Steam Isolation Valves	Check functioning	
9.	Reactor Containment Isolation Trip Valves	Check functioning	
9.a.	Isolation Valve Seal Water System	Check functioning	
10.	Refueling System Interlocks	Check functioning	

*This information will be supplied as part of Technical Specifications

Chap 8

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8 ELECTRICAL SYSTEMS

8.1 DESIGN BASES

The main generator will feed electrical power at 22 kv through an isolated phase bus to two half-sized main power transformers. The bulk of the power required for station auxiliaries during normal operation will be supplied by an auxiliary transformer connected to the isolated phase bus. This practice has been proven highly satisfactory for fossil-fueled units. Deviat[illegible] from past practices are reflected in the provisions for stand-by or emergency power which have been included to further ensure the continuity of electrical power for critical loads.

The function of the Auxiliary Electrical System is to provide reliable power to those auxiliaries required during any normal or emergency mode of plant operation.

The design of the system is such that sufficient independence or isolation between the various sources of electrical power is provided in order to guard against concurrent loss of all auxiliary power.

8.1.1 PRINCIPAL DESIGN CRITERIA

Performance Standards

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

All electrical systems and components vital to plant safety, including the emergency diesel generators are designed as Class I and designed so that their integrity is not impaired by the maximum potential earthquake, wind storms, floods or disturbances on the external electrical system. Power, control and instrument cabling, motors and other electrical equipment required for operation of the engineered safety features are suitably protected against the effects of either a nuclear system accident or of severe external environment phenomena in order to assure a high degree of confidence in the operability of such components in the event that their use is required.

Emergency Power

Criterion: An emergency power source shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning of the engineered safety features and protection systems required to avoid undue risk to the health and safety of the public. This power source shall provide this capacity assuming a failure of a single component. (GDC 39 and GDC 24)

Independent alternate power systems are provided with adequate capacity and testability to supply the required engineered safety features and protection systems.

The plant is supplied with normal, standby and emergency power sources as follows:

1. The normal source of auxiliary power during plant operation is the generator. Power is supplied via the unit auxiliary transformer which is connected to the main leads of the generator.

2. Standby power required during plant startup, shutdown and after reactor trip is supplied from the Consolidated Edison Co. 138 kv system by overhead line from a substation approximately 3/4 mile from the plant to the station auxiliary transformer. In addition, a 21 MW gas turbine is available as an emergency blackout startup power supply. The capacity of the gas turbine generator requires that the station load be reduced to a minimum for startup.

3. Three diesel generators are each connected to their respective engineered safety features buses to supply emergency shutdown power in the event of loss of all other a.c. auxiliary power. There are no automatic ties between these buses.

4. Emergency power supply for vital instruments and control and supplies for emergency lighting is from the two 125 volt dc station batteries.

The diesel-generaotrs are located adjacent to the primary auxiliary building and each are connected to three (3) separate 480 volt auxiliary system buses. Each diesel will be started automatically on a safety injection signal or upon the occurrence of undervoltage on any 480 volt bus. Any two diesels have adequate capacity to supply the engineered safety features for the hypothetical accident concurrent with loss of outside power. This capacity is adequate to provide a safe and orderly plant shutdown in the event of loss of outside electrical power.

8.2 ELECTRICAL SYSTEM DESIGN

8.2.1 NETWORK INTERCONNECTIONS

The external transmission system provides two basic functions for the nuclear generating station; namely, it provides auxiliary power as required for startup and normal shutdown and transmits the output of the station.

Electrical energy generated at 22 kv is raised to 345 kv by the two main generator transformers and delivered to Buchanan Switching Station via a 345 kv, 2,000 amp, 25,000 MVA synchronizing circuit breaker. The system side of this generator circuit breaker is connected to Millwood Substation by an overhead transmission circuit. The overhead line is connected through circuit breakers to the Millwood bus. Millwood Substation has circuits to Pleasant Valley Substation, which is the interconnection point between Consolidated Edison Company and the Niagara Mohawk, and Connecticut Light and Power systems. Millwood 345 kv Substation also has connections to Sprain Brook Substation, which in turn has connections to southern high-voltage switching stations.

The standby power supply is furnished from the Buchanan 138 kv Substation, which has connections to Millwood Substation, Indian Point Unit #1 Generation and to Lovett Station of the Orange & Rockland system (see Figure 8.2-1). In addition, a 21 MW gas turbine generator located on Indian Point Plant property is connected to the 13 kv buses and to the 6.9kv buses via an auto transformer.

Single-Line Diagram

A single-line diagram, showing the connections of the main generator to the power system grid, and standby power source is shown in Figure 8.2-1.

Reliability Assurance

In the initial arrangement three external sources of stand-by power are available to Indian Point #2. They are the 13 kv tie from Buchanan, the 138 kv tie from Buchanan, and the gas turbine generator. Loss of any two of these sources will not affect the third.

The 138 kv supply from the Buchanan bus with its connections to the Consolidated Edison Company system and Orange & Rockland Company, provides a dependable source of station auxiliary power. Future 138 kv ties are designed to provide additional auxiliary power sources for Indian Point #2 from Indian Point #1 and from a second 138 kv feeder to Buchanan Substation. A further guarantee of reliable auxiliary power supply independent of system connections is provided by the gas turbine generator installed at the plant site.

8.2.2 STATION DISTRIBUTION SYSTEM

The Auxiliary Electrical System is designed to provide a simple arrangement of buses requiring the minimum of switching to restore power to a bus in the event that the normal supply to that bus is lost.

Single Line Diagrams

The basic components of the station electrical system are shown on the Main Electrical One Line Diagrams, Figures 8.2-2 through 8.2-6, which include the 6900 volt, the 480 volt, the 120 volt a.c. instrument, and the 125 volt d.c. bus systems.

Unit Auxiliary, Station Auxiliary and Station Service Transformers

The plant's generator serves as the main source of auxiliary electrical power during "on-the-line" operation of the plant. Power to the auxiliaries is supplied via a 22-6.9 kv two winding Unit Auxiliary Transformer that is connected to the main leads from the generator. Power to the 480 volt buses is fed throug' four 6900/480 volt station service transformers of non-explosive, fire-resistant, air-insulated, dry type. Solid insulation in the transformers consists of inorganic materials such as porcelain, mica, glass or asbestos in combination with a sufficient quantity of a high temperature binder, to impart the necessary mechanical strength to the insulation structure. This insulation is defined by ASA standards as Group III material.

These transformers are designed and constructed in accordance with the applicable standards of ASA, IEEE and NEMA. During engineered safeguards loading and operation, these transformers will not be loaded beyond their rating. Manufacturer shop tests of the transformers are conducted in accor with the latest revision of American Standard Test Code C 57.12.90. This series of tests consists of the following:

1. Resistance measurements of all windings
2. Ratio tests
3. Polarity and phase relation tests
4. No-load losses
5. Exciting current
6. Impedence and load loss
7. Temperature test
8. Applied potential tests
9. Induced potential tests.

Auxiliary power required during plant start-up, shutdown and after reactor trip is supplied from the 138 kv switchyard. After reactor or turbine generator trip, the auxiliaries on the four 6900 volt buses supplied by the Unit Auxiliary Transformer are transferred by a dead fast transfer scheme using stored energy breakers to the Station Auxiliary Transformer. The 138 kv system is the normal supply for the auxiliary load associated with plant engineered safeguards.

6900 Volt System

The 6900 volt system is divided into six buses. Two buses, numbers 5 and 6, are connected to the 138 kv system via bus main breakers and the Station Auxiliary Transformer. Buses 1, 2, 3, and 4 are connected to the generator leads via bus main breakers and the Unit Auxiliary Transformer. Buses1 and 2 can be tied to bus 5 and buses 3 and 4 can be tied to bus 6 via bus tie breakers. Buses 2, 3, 5 and 6 each serve one 6900-480 volt station service transformer.

480 Volt System

The 480 volt system is divided into four buses. The 480 volt buses are supplied from the 6900 volt buses as follows: 2A from 2; 3A from 3; 5A from 5; and 6A from 6.

The required safeguards equipment circuits are dispersed among the 480 volt buses. The normal source of power for buses 5A and 6A is the 138 kv system (via station auxiliary transformer, and 6900 volt buses 5A and 6A), and no transfer is required in the event of an incident.

One emergency diesel-generator set is connected to bus 5A, one to 6A and the other to bus 2A. Each set will be automatically started upon under-voltage on any one of the 480 volt buses. 12

The power for the safeguards valve motors will be supplied from two motor control centers which in turn will be supplied from the 480 volt system. Each motor control center bus will be fed through a circuit breaker on the 480 volt system. These circuit breakers will be on different 480 volt buses and the bus that supplies each breaker will be supplied by an emergency diesel generator.

Each of the four 480 volt switchgear buses which supply power to the safeguards equipment receives DC · ntrol power from both battery sources. An automatic transfer device on each bus seeks whichever DC source is energized with battery source 21 being the preferred source for buses 5A and 3A and battery source 22 being the preferred source for buses 6A and 2A.

125 Volt D.C. System

The 125 volt d-c system is divided into two buses with one battery and battery charger (supplied from the 480 volt system) serving each. The battery chargers supply the normal d-c loads as well as maintaining proper charges on the batteries. A bus tie between the main distribution panels allows either battery or battery charger to be removed for maintenance.

120 Volt A.C. System

The 120 volt a-c instrument supply is split into four buses. Two of the buses are fed by inverters which are in turn supplied from separate 125 volt d-c buses. The other two buses are supplied by constant voltage transformers connected to separate 480 volt buses. In the event an inverter or a constant voltage transformer is taken out of service, a backup supply from the lighting panel is available to feed the associated bus.

Evaluation of Layout and Load Distribution

The physical locations of electrical distribution system equipment is such as to minimize vulnerability of vital circuits to physical damage as a result of accidents.

Station and unit auxiliary transformers, and the main transformer are located outdoors physically separated from each other.

Lightning arresters are used where applicable for lightning protection. All oil filled transformers are covered by automatic spray systems to extinguish oil fires quickly and prevent the spread of fire. Transformers are spaced to minimize their exposure to fire, water and mechanical damage.

The 6900 volt switchgear and 480 volt load centers are located in areas which minimize their exposure to mechanical, fire and water damage. This equipment will be properly coordinated electrically to permit safe operation of the equipment under normal and short circuit conditions.

The 480 volt motor control centers are located in the areas of electrical load concentration. Those associated with the turbine-generator auxiliary system in general are located below the turbine-generator operating floor level. Those associated with the nuclear steam supply system are located in the Primary Auxiliary Building.

Nonsegregated, metal-enclosed 6900 volt buses are ed for all major bus runs where large blocks of current are to be carried. The routing of this metal-enclosed bus is such as to minimize its exposure to mechanical, fire and water damage.

The application and routing of control, instrumentation and power cables are such as to minimize their vulnerability to damage from any source. All cables are designed using conservative margins with respect to their current carrying capacities, insulation properties and mechanical construction. Power cable insulation in the Reactor Building has fire resistant sheathing, selected to minimize the harmful effects of radiation, heat and humidity.

Appropriate instrumentation cables are shielded to minimize induced voltage and magnetic interference. Wire and cables related to engineered safeguard and reactor protective systems are routed and installed to maintain the integrity of their respective redundant channels and protect them from physical damage.

Cable loading of trays and consequently heat dissipation of cable throughout the plant has been carefully studied and controlled to ensure no overloading. The criteria for electrical loading has been developed using IPCEA standard P-46-426, manufacturer recommendations and good engineering practic

Derating factors for cables in trays without maintained spacing are taken from Table VIII of the IPCEA publication. Derating factors for the maximum ambient temperature existing in any area of the plant are also taken from th IPCEA publication. These factors are applied against ampacities selected from appropriate tables in other portions of the standard.

For physical loading of trays, the following criteria was followed: 6 kv power, one horizontal row of cables was allowed in a tray; 480 volt power, two horizontal rows of cables were allowed in a tray; for control and instrumentation 70 per cent of the cross-sectio . a tray was the maximum fill. A computer program monitored the loading and prevented routing anything greater than this amount to the trays.

In general, for instrumentation cables, four basic channels are routed through the plant. These channels include cables for systems 65 volts and less. Cables assigned to these four channels will remain in their respective channels throughout the run.

Certain other cables are run in with the four instrument channels such as thermocouple cable, public address system cabling and instrument power supplies.

Control cables are separated into two basic channels as required for redundant circuits. These groups of cables are set up for systems above 65 volts and less than 500 volts and include multiconductor control cable or other cable as required. Cables assigned to these two channels for separation will be in their respective channels, and so designated from the beginning of the cable to the final termination. These cables include the following:

a) Motor Operated Valves - Two (2) channels for the redundant valves.

b) Solenoid Valves - Two (2) channels where required for redundant valves and safeguards. Otherwise not separated.

c) Detector Drives are run in any channel as convenient.

d) Motor Controls except safeguards are run in any channel as convenient.

e) Small power cables are run in any channel as convenient.

f) Safeguard Control cables are run in two (2) channels as required.

g) Safeguard Power Cables are separated into sufficient channels to provide for minimum functions. For example, three channels are provided for containment fan cooler motors.

The physical channeling is accomplished by either separate trays or trays with metal barriers and in some cases by separate conduit.

In general, redundant circuits are separated horizontally rather than vertically. Where physical conditions prevented this, horizontal fire barriers separate power trays from instrument trays.

To assure that only fire retardant cables are used throughout the plant a careful study of cable insulation systems was undertaken early in plant design.

Insulation systems that appeared to have superior flame retardant capability were selected and manufacturers were invited to submit cable samples for testing. An extensive flame testing program took place which included ASTM vertical flame testing and Consolidated Edison Company vertical flame and bonfire tests. A report summarizing the testing was prepared by Consolidated Edison Company. These flame tests were used as one of the means of qualifying cables and the specifications were written on the basis of results from the tests.

The following tests were made to determine the flame resistant qualities of the covering and insulations of various types of cables for Indian Point #2.

1) Standard Vertical Flame Test - made in accordance with ASTM-D-470-59T. "Tests For Rubber and Thermal-plastic Insulated Wire and Cable".

2) Five-Minute Vertical Flame Test - made with cable held in vertical position and 1750°F flame applied for 5 minutes.

3) Bon-Fire Test - consisting of exposing, for 5 minutes, bundles of three or six cables to flame produced by igniting transformer oil in 12 inch pail. The cable was supported horizontally over the center of the pail the lowest cable 3 inches above the top of the pail. The time to ignite the cable and the time the cable continued to flame after the fire was extinguished were noted.

On the basis of these tests, the cables were selected for the reactor containment vessel penetration.

To prevent the propagation of flame, firestops have been provided where
cable trays or openings for cables pass through walls and floors, and
13 under switchgear and other equipment. The firestop consist of transite
sheet barriers or packed fiberglass barriers coated on either side with
Flame Mastic 71A.

Smoke detectors of the ionization type are provided in the cable tunnel and other unattended areas where large concentrations of cables are installed, for fire detection and alarm.

In areas where missile protection could not be provided (such as near the reactor coolant system) redundant instrument impulse lines and cables were run by separate routes. These lines were kept as far apart as physically possible, or were protected by heavy (1/4") metal plates interposed where inherent missile protection could not be provided by spacing.

8.2.3 EMERGENCY POWER

Sources Description

The three sources of off-site emergency power are 1) the 138 - 6.9 kv
15 Station Auxiliary Transformer, 2) 13.8 kv tie, and 3) gas turbine. If
these sources should fail, the on-site sources of emergency power are
three emergency diesel-generator sets. The emergency power facilities
consist of three diesel engine-generator sets, each consisting of an Alco
model 16-251-E engine coupled to a Westinghouse 2188 kva, 0.8 power factor,
900 RPM, 3 phase, 60 cycle 480 volt generator. The units have a capability
of 2000 kw for 2000 hours and 1750 kw continuous.

Any two units, as a backup to the normal standby AC power supply are capable of sequentially starting and supplying the power requirement of one complete set of safeguards equipment. The units are located in a Class I structure located near the Primary Auxiliary Building.

Each emergency diesel is automatically started by two redundant air motors, each unit having a complete 53 cu. ft. air storage tank and compressor system powered from a 480 volt motor. The piping and the electrical services are arranged so that manual transfer between units is possible. Each air receiver has sufficient storage for 4 starts. The diesel will consume, however, only enough air for one automatic start during any particular power failure. This is due to the engine control system which is designed to shutdown and lock-out any engine which did not start during the initial try. The emergency units are capable of being started and sequence load begun within 10 seconds after the initial signal. The starting system is completely redundant for each diesel generator. The units have the capability of being fully loaded within 30 seconds after the initial starting signal. The starting system is completely redundant for each diesel generator.

To ensure rapid start the units are equipped with water jacket and lube coil heating and pre-lube pump for circulation of lube oil when the unit is not running. The units are located in heated rooms.

An audible and visual alarm system will be located in the main control room and will alarm off-normal conditions of jacket water temperature, lube oil temperature, fuel oil level, and starting air pressure.

The abnormal conditions that can shut down the diesel generator during an accident are:

a) overcranking
b) low oil pressure
c) overspeed

A shutdown of the diesel generator is indicated in the control room by an audible alarm on the control board and by the generator bus voltmeter.

The diesel generator is designed to start and come up to speed within ten seconds after initiation of the starting signal. Failure of the engine to start within the timing period of the overcrank time (20 seconds) indicates a malfunction. Shutdown conserves the starting air supply so that the engine can be subsequently started after the malfunction is corrected.

Low oil pressure indicated by two out of three oil pressure switches shuts down the diesel generator, since the engine cannot run without proper lubrication. Shutdown permits corrective action to be taken before the engine is damaged, and the diesel generator can then be returned to normal operation.

An overspeed condition causes improper generator output and therefore, the diesel generator should be shut down for corrective action to be taken to restore the generator output to normal.

The units will use diesel oil spec No. 2. A two hour storage tank is located at each of the units. Three fuel oil storage tanks will be on site having a capacity of 7,700 gallons each. The transfer pumps stop automatically at a level of 12" in the tanks. Thus approximately 7,200 gallons of fuel oil per tank is available. The minimum allowable available fuel oil is 19,000 gallons. This quantity will be sufficient to run two diesels with a recirculation load for 30 hours. Transfer of oil from the underground storage tank to automatically maintain level in each unit tank is accomplished by a motor driven pump for each unit tank. Any oil transfer pump is capable of serving any one or more emergency generator unit tanks through manual valving.

Additional [illegible]ties of diesel oil are available locally. Under normal conditio[illegible], 25,000 gallons can be delivered on a one or two day notice. Additio[illegible]e are also maintained in the New Rochelle - Mount Vernon area ([illegible]iles from the plant) and are available for use during emergenc[illegible]bject to extreme cold weather conditions (increased domestic heating usage) and available transportation.

Diesel-Generator Separation

The 2350 HP diesel engine-generators are located in a sheet metal, steel framed building immediately South of the primary auxiliary building. The engine generators are arranged on 13'-0" centers, parallel to each other with approximately 10'-0" of clean space between engine components. The engine foundations are surrounded with a 6" high concrete curb containing sufficient volume to hold all of the lube oil or fuel released from a single engine in the event of an inadvertent spill or line break. Individual fire detection and automatic protection spray systems are provided over each engine and fuel oil day tank. The detection system which annunciates in the main control room is designed to sense heat in order to quickly actuate the detection and spray devices. In addition, manual fire fighting equipment is located in each personnel access door.

A control panel is located on the West end of the building which contains relays and metering equipment for all three diesel-engine generators. The panels are compartmentalized with controls for each engine separated from each other. With the compartmentalized panel design, spread of fire to other electrical components is minimized. A reinforced concrete wall will be placed between the diesel generators and the control panel.

Each diesel generator has its own small fuel storage (day) tank that feeds the fuel oil pump on the engine. All day tanks are automatically filled during engine operation from three separate underground storage tanks outside the diesel generator building. Each storage tank has its own supply pump mounted in a manhole opening in the top of the tank above oil level. It is therefore unlikely that a fire associated with any one of the small fuel oil storage (day) tanks would prevent oil from being supplied to the remaining two diesels.

Loading Description

Each unit is to be started on the occurrence of either of the following incidents:

1. Initiation of safety injection operation.

2. Undervoltage on any of the 480 volt buses.

On occurrence of undervoltage the engines run at idle and can be connected to deenergized buses by the operator from the control room if desired. If there is coincident or subsequent requirement for engineered safeguards, automatic sequencing is initiated as follows:

1. All 480 volt breakers, except those feeding the valve motor control centers numbers 26A and 26B, are tripped and all automatically operated non-Safeguards' feeders are locked out. All engineered safeguards motors are operated from the 480 volt buses.

2. Connect the diesel generators to their respective buses.

3. Magnitude of loads for the diesel generators is given in Table 8.2-1.

The loads on the three diesel generators will be sequenced as follows:

		Bus 5A	Bus 2A-3A	Bus 6A
a.	Auxiliary Component Cooling Pumps	1		1
b.	Safety Injection Pumps	1	1	1
c.	Residual Heat Removal Pumps		1	1
d.	Spray Pumps (if start signal present)	1		1
e.	Nuclear Service Water Pumps	1	1	1
f.	Containment Air Re-circulation Cooling Fans	2	2	1
g.	Auxiliary Feedwater Pumps		1	1
h.	Spray Pumps (action taken in case high-high containment pressure did not exist at step (d).)	1		1

The recirculation phase is manually initiated by control switches on the supervisory panel in the main control room. As the sequence switches are operated the bus loads are modified to give those shown in Table 8.2-1. The loads are given for a post-blowdown containment pressure transient.

To verify that the emergency power system will respond within the required time limit and when required, the following tests shall be performed periodically.

a. Manually initiated demonstration of the ability of the diesel generators to start, and deliver power up to name plate rating, when operating in parallel with other power sources. Normal plant operation will not be affected. The duration of the test shall be at least 2 hours.

b. Demonstration of the readiness of the system and the control systems of vital equipment to automatically start or restore to operation particular vital equipment by simulating a loss of all normal AC station service power supplies. This test can be conducted during each refueling interval.

The starting of the diesel-generator sets can be tested from the Diesel Building. The ability of the units to start within the prescribed time and to carry intended loads are checked periodically.

Batteries and Battery Chargers

Each of the two 60 cell, lead acid station batteries has been sized to carry its expected shutdown loads following a plant trip and a loss of all AC power for a period of 2 hours without battery terminal voltage falling below 105 volts. Major loads with their approximate operating times on each battery are listed in Table 8.2-2.

15 Each of the two battery chargers has been sized to re-charge either of the above partially discharged batteries within 15 hours while carrying its normal load.

Reliability Assurance

The electrical system equipment is arranged so that no single contingency can inactivate enough safeguards equipment to jeopardize the plant safety. The 480-volt equipment is arranged on 4 buses. The 6900-volt equipment is supplied from 6 buses.

The plant auxiliary equipment is arranged electrically so that multiple items receive their power from the two different sources. The charging pumps are supplied from the 480-volt buses Nos. 3A, 5A and 6A. The six service water pumps and the five containment fans are divided among the four 480-volt buses. Valves are supplied from motor control centers, Nos. 26A and 26B, which are supplied from buses 5A and 6A.

The outside source of power is adequate to run all normal operating equipment. The 138 - 6.9 kv station transformer can supply all the auxiliary loads.

The bus arrangements specified for operation ensure that power is available to an adequate number of safeguards auxiliaries.

Two diesel generators have capacity enough to start and run a fully loaded set of engineered safeguards equipment. These safeguards can adequately cool the core for any loss-of-coolant incident, and they also maintain the containment pressure within the design value.

One battery charger shall be in service so that the batteries will always be at full charge in anticipation of loss-of-ac power incident. This ensures that adequate dc power will be available for starting the emergency generators and other emergency uses.

TABLE 8.2-1

LOAD SCHEDULE FOR DIESEL GENERATORS
FOR A
LOSS OF COOLANT ACCIDENT

A. Engineered Safety Features Loads

Component	Quantity	Rated hp, Each	Injection Phase = 15 min. Quantity Required	kw	Recirculation Phase Quantity Required	kw
Safety Injection Pump	3	400	2	665	1	332
Residual Heat Removal Pump	2	400	1	332	0*	0
Nuclear Service Water Pump	3	350	2	580	2	580
Conventional Service Water Pump	3	350	0	0	1	290
Containment Fan	5	350	3	870	3	870**
Recirculation Pump	2	350	0	0	1*	290
Auxiliary Component Cooling Pump	2	3	2***	6	0	0
Auxiliary Feedwater Pump	2	400	1	332	1	332
Component Cooling Pump	3	250	0	0	1	207
Containment Spray Pump	2	400	1	332	0	0
Valve MCCS	2	25	2	42	2	42
Battery Charger	2	30	0	0	1	25
Auxiliary Building Fans	2	30	0	0	1	25
Total ESF Loads:		3338		3159		2993

TABLE 8.2-1 (Cont'd)

B. Non-Engineered Safety Feature Loads

Component	Quantity	Rated hp, Each	Injection Phase = 15 min. Quantity Required	kw	Recirculation Phase Quantity Required	kw
Seal Oil Pump	1	30	0	0	1	25
Turbine Building MCC			0	0		42
Turning Gear	1	50	0	0	1	42
Turning Gear Oil Pump	1	75	0	0	1	62
Total Non-ESF Loads:		155				171

C. Load Capacity of Each Diesel Generator

Magnitude kw	Length of Time Diesel can Support this Magnitude
2000	2000 hours
1750	Continuous

System load capacity can be carried with a minimum of 2 diesel generators.

Footnotes

* The effect of recirculation through a residual heat removal pump instead of through a recirculation pump is to increase the load by 42 kw.

** Load for containment design pressure is given; actual pressure will be less.

*** Both auxiliary component cooling pumps operate, but only one is required.

TABLE 8.2-2

MAJOR BATTERY LOADS

Battery 21	
Inverter 21	2 Hours
Emergency Lighting & Control Power	2 Hours
Turbine Generator Emergency Oil Pump (60 HP)	1 Hour
Battery 22	
Boiler Feed Pump Emergency Oil Pump	15 Min.
Air Side Seal Oil Back-up Pump (25 HP)	1 Hour
Inverter 22	2 Hours
Emergency Lighting & Control Power	2 Hours

INSTRUMENT BUS-22			
CIRCUIT NO.	DESCRIPTION	BREAKER RATING	REF. DWG.
1	BUS VOLTMETER	15A	SHEET 9
2	RCPS	15A	
3	FEEDWATER CONTROL	15A	
4	S/G INSTRUMENTATION	15A	
5	NUCLEAR INST. SYS.	15A	
6	REACT. COOLANT SYS. INST.	15A	
7	CVCS INSTRUMENTS	15A	
8	MTR 21A/B & C EXT STM SOL VLV	15A	
9	MTR 20A EXT STM SOL VLV	15A	
10	MTR 22C EXT STM SOL VLV	15A	
11	MTR 23B EXT STM SOL VLV	15A	
12	FLASH EVAPORATOR	15A	
13	MTR 21A & 22A MTR DRN SOL VLV	15A	
14	MTR 21B & 22B MTR DRN SOL VLV	15A	
15	MTR 21C & 21C MTR DRN SOL VLV	15A	
16			
17			
18			
19			
20			
21			
22			
23			
24			

INSTRUMENT BUS-24			
CIRCUIT NO.	DESCRIPTION	BREAKER RATING	REF. DWG.
1	BUS VOLTMETER	15A	SHEET 9
2	RCPS	15A	
3	FEEDWATER CONTROL	15A	
4	S/G INSTRUMENTATION	15A	
5	NUCLEAR INST. SYS.	15A	
6	REACT. COOLANT SYS. INST.	15A	
7	CVCS INSTRUMENTS	15A	
8	MTR 25A&C EXT STM SOL VLV	15A	
9	MTR 25B EXT STM SOL VLV	15A	
10	MTR 23A EXT STM SOL VLV	15A	
11	MTR 23C EXT STM SOL VLV	15A	
12	MTR 22A & 24A MTR DRN SOL VLV	15A	
13	MTR 23B & 24B MTR DRN SOL VLV	15A	
14	MTR 23C & 24C MTR DRN SOL VLV	15A	
15	RAD MONIT CAB	15A	CAB 31
16			
17	RCPT, INSTR, & RELAYS	15A	
18			
19			
20			
21			
22			
23			
24			

INSTRUMENT BUS-21			
CIRCUIT NO.	DESCRIPTION	BREAKER RATING	REF. DWG.
1	BUS VOLTMETER	15A	SHEET 9
2	RCPS	15A	
3	FEEDWATER CONTROL	15A	
4	S/G INSTRUMENTATION	15A	
5	NUCLEAR INST. SYS.	15A	
6	REACT. COOLANT SYS. INST.	15A	
7	RAD. MONIT. CAB 30	15A	
8	AUX. COOLANT SYS. INST	15A	
9	ROD POSITION [illegible]	15A	
10	SUMP LEVEL IND.	15A	DWG [illegible] SK 2??
11			
12			
13			
14			
15			
16			
17	SERVICE WATER SCREENS BY-PASS VALVE	15A	
18	PRESSURE RELIEF FAN DUCT CONT SOLENOID VALVES	15A	
19			
20			
21			
22			
23			
24			

INSTRUMENT BUS-23			
CIRCUIT NO.	DESCRIPTION	BREAKER RATING	REF. DWG.
1	BUS VOLTMETER	15A	SHEET 9
2	RCPS	15A	
3	FEEDWATER CONTROL	15A	
4	S/G INSTRUMENTATION	15A	
5	NUCLEAR INST. SYS.	15A	
6	REACT. COOLANT SYS. INST.	15A	
7	RAD. MONIT. CAB. 30	15A	
8	AUX. COOLANT SYS. INST.	15A	
9	MISC. TRANSMITTERS	15A	
10	SUMP LEVEL IND.		DWG [illegible] SK 2??
11			
12			
13			
14			
15			
16			
17	[illegible] EMERGENCY [illegible]	15A	
18	P.A. SYSTEM	15A	
19			
20			
21			
22			
23			
24			

Table 8.2-2. Instrument Bus Circuits

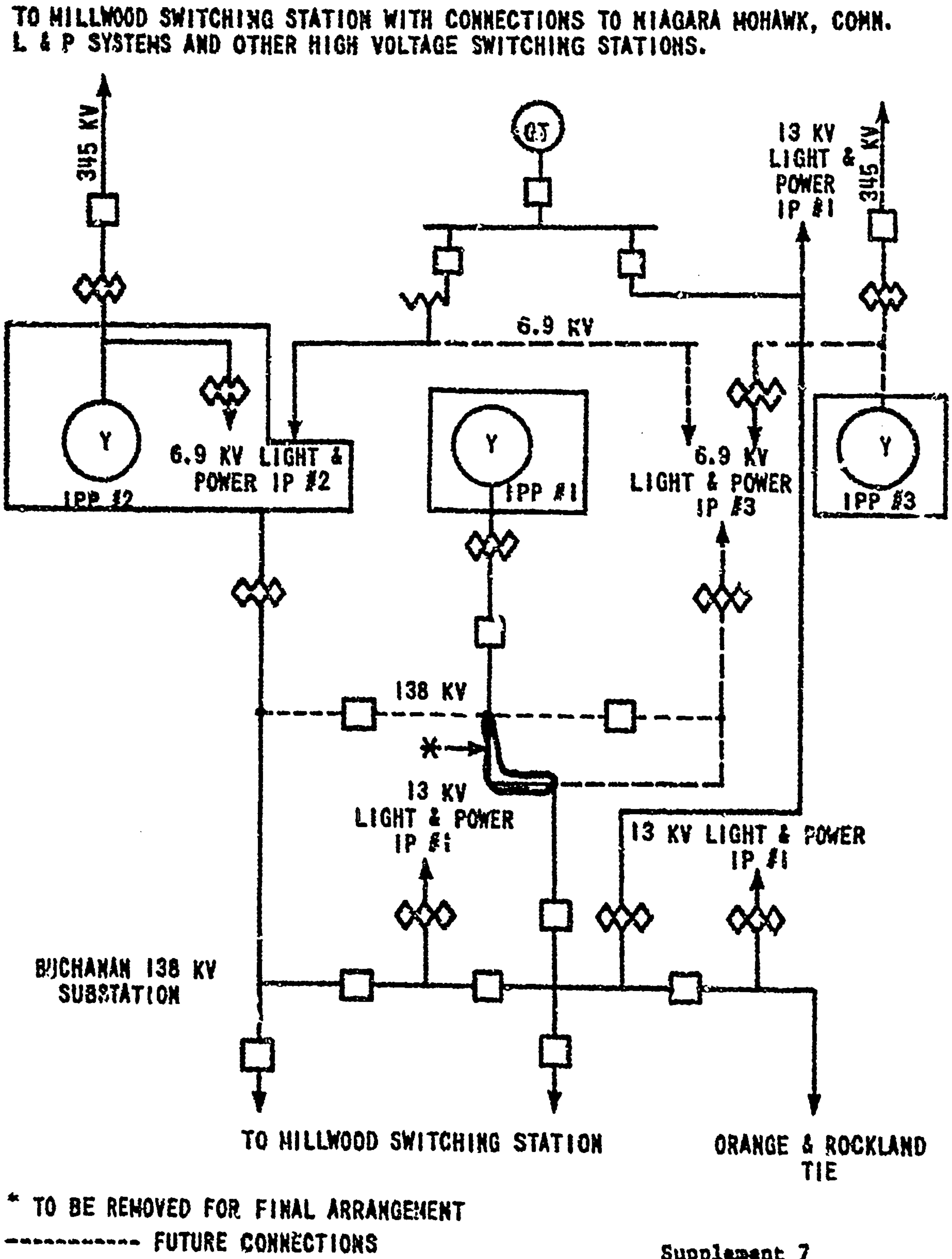

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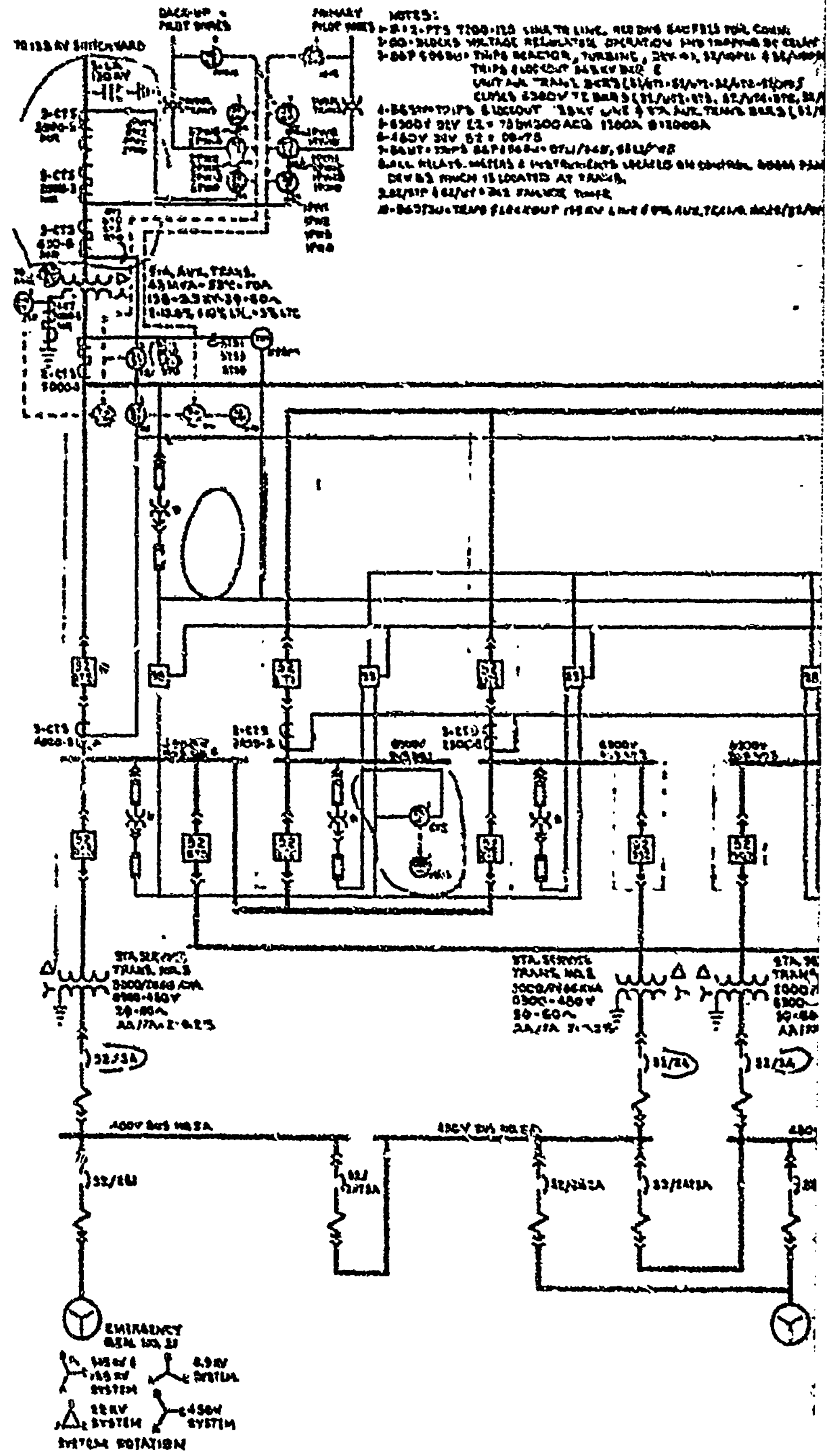
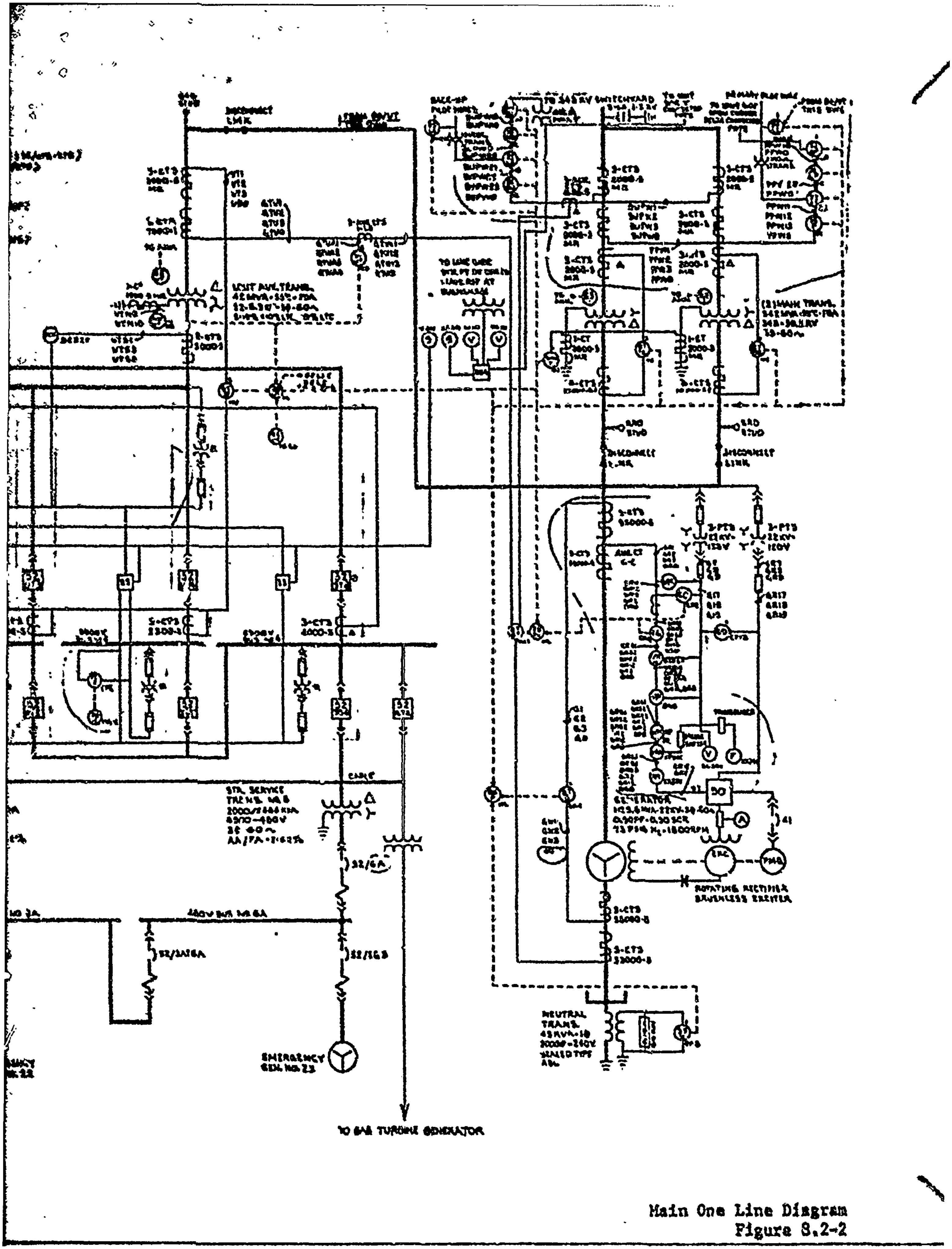
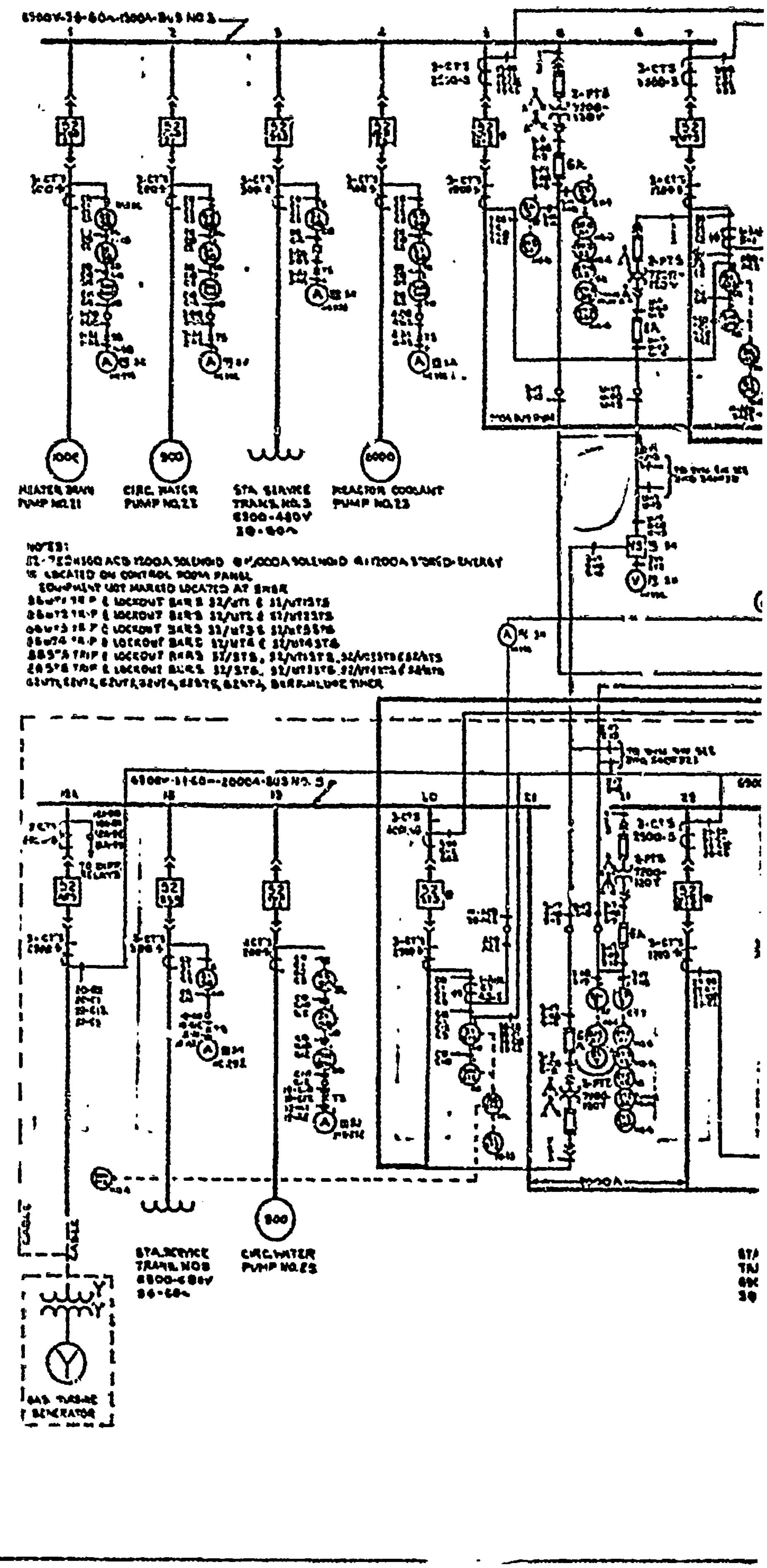
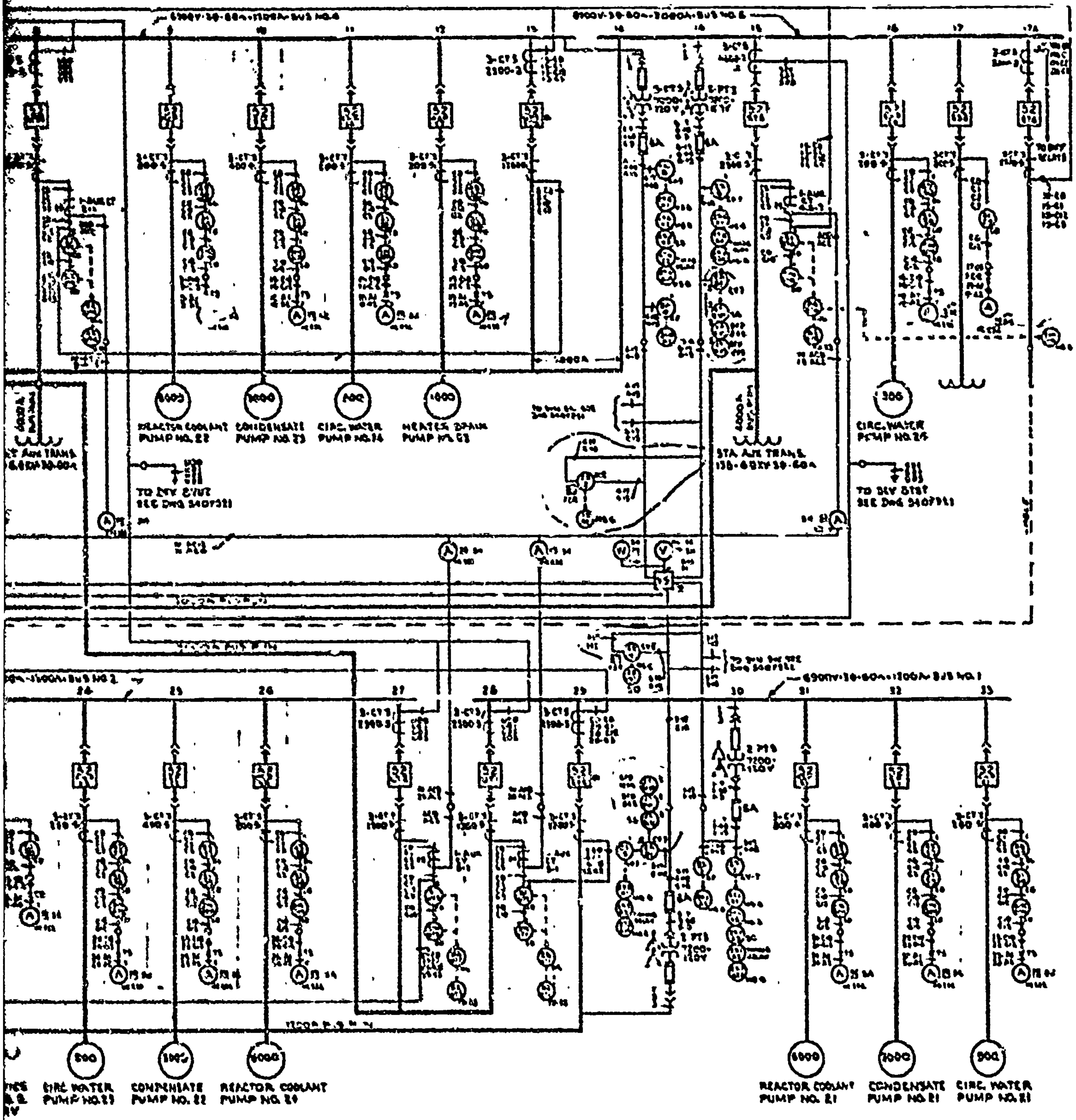
Figure 8.2-1. Transmission Connections


NOTES:
SYSTEM ROTATION

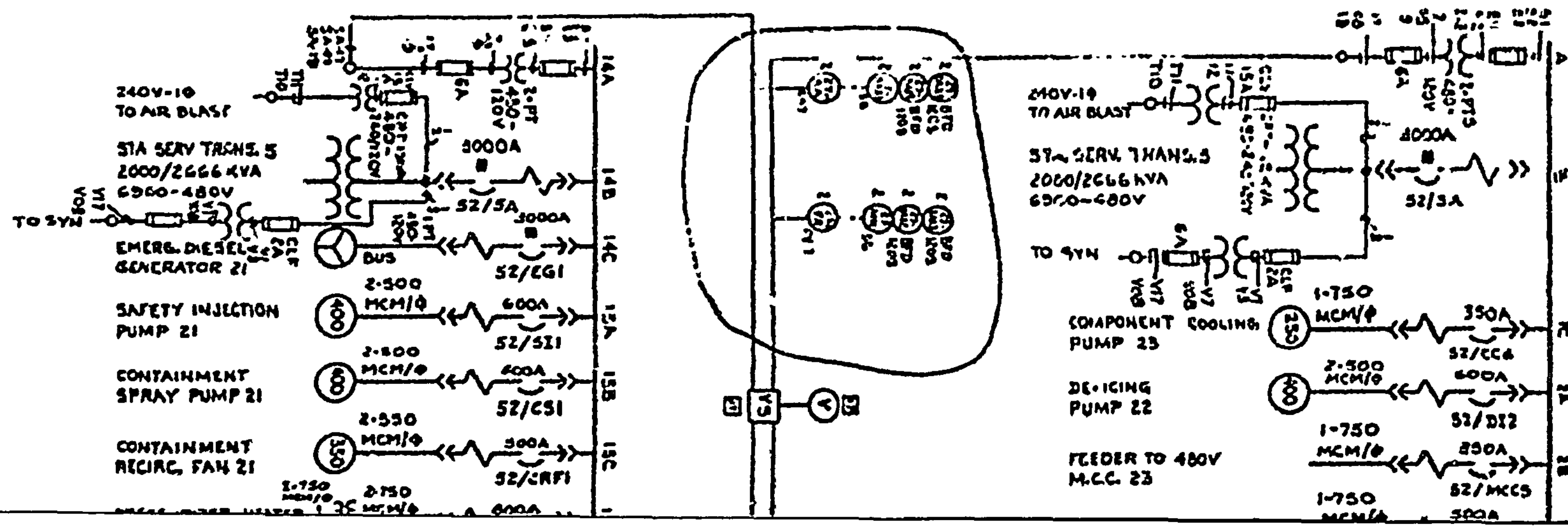


Main One Line Diagram
Figure 8.2-2

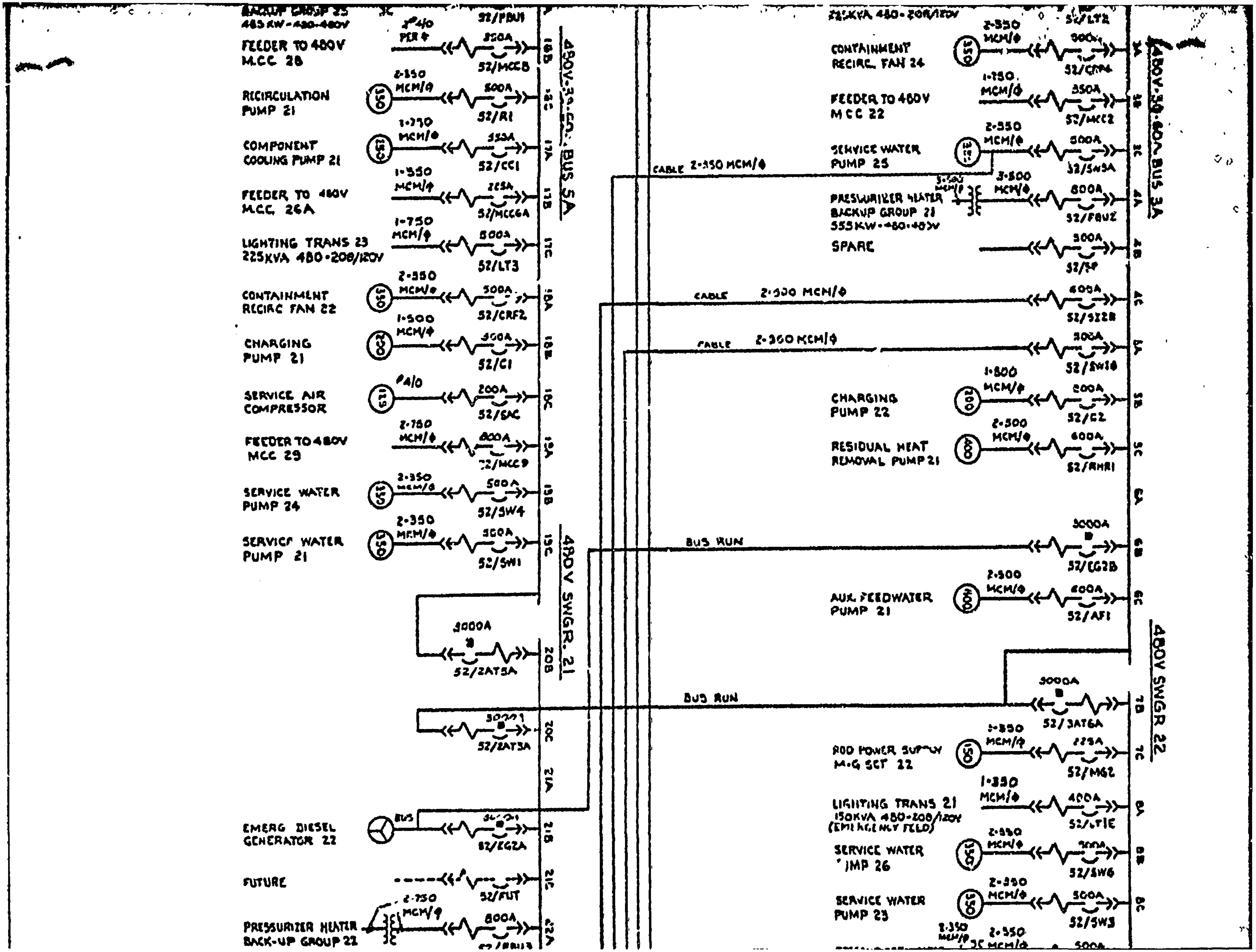

HEATER DRAIN PUMP NO.21
CIRC. WATER PUMP NO.22
STA. SERVICE TRANS. NO.3 6900-480V 3Ø-60~
REACTOR COOLANT PUMP NO.23
IS LOCATED ON CONTROL ROOM PANEL
EQUIPMENT NOT MARKED LOCATED AT SWGR
STA. SERVICE TRANS. NO.5
CIRC. WATER PUMP NO.25
GAS TURBINE GENERATOR
CABLE



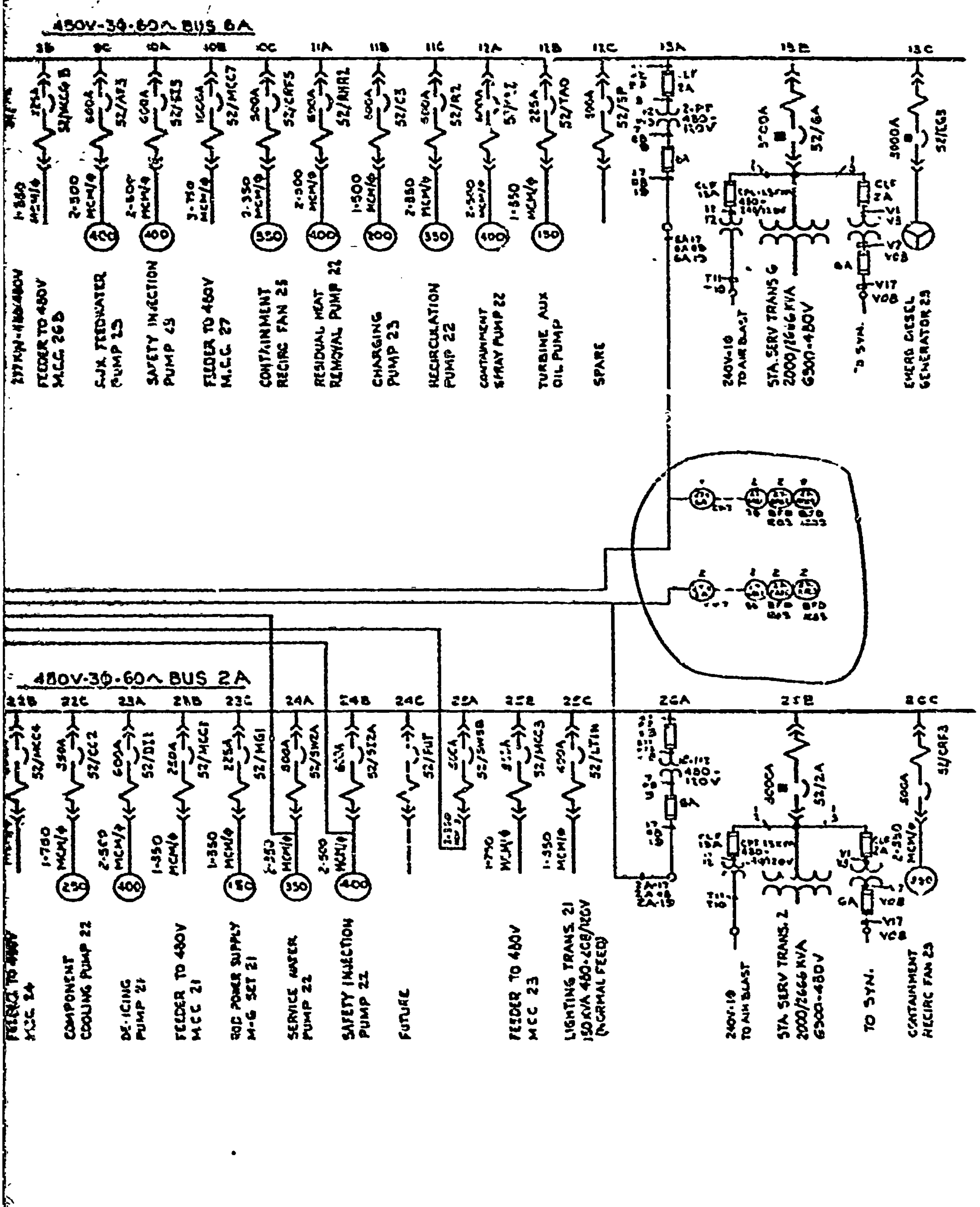
6900 Volt One Line Diagram
Figure 8.2-3



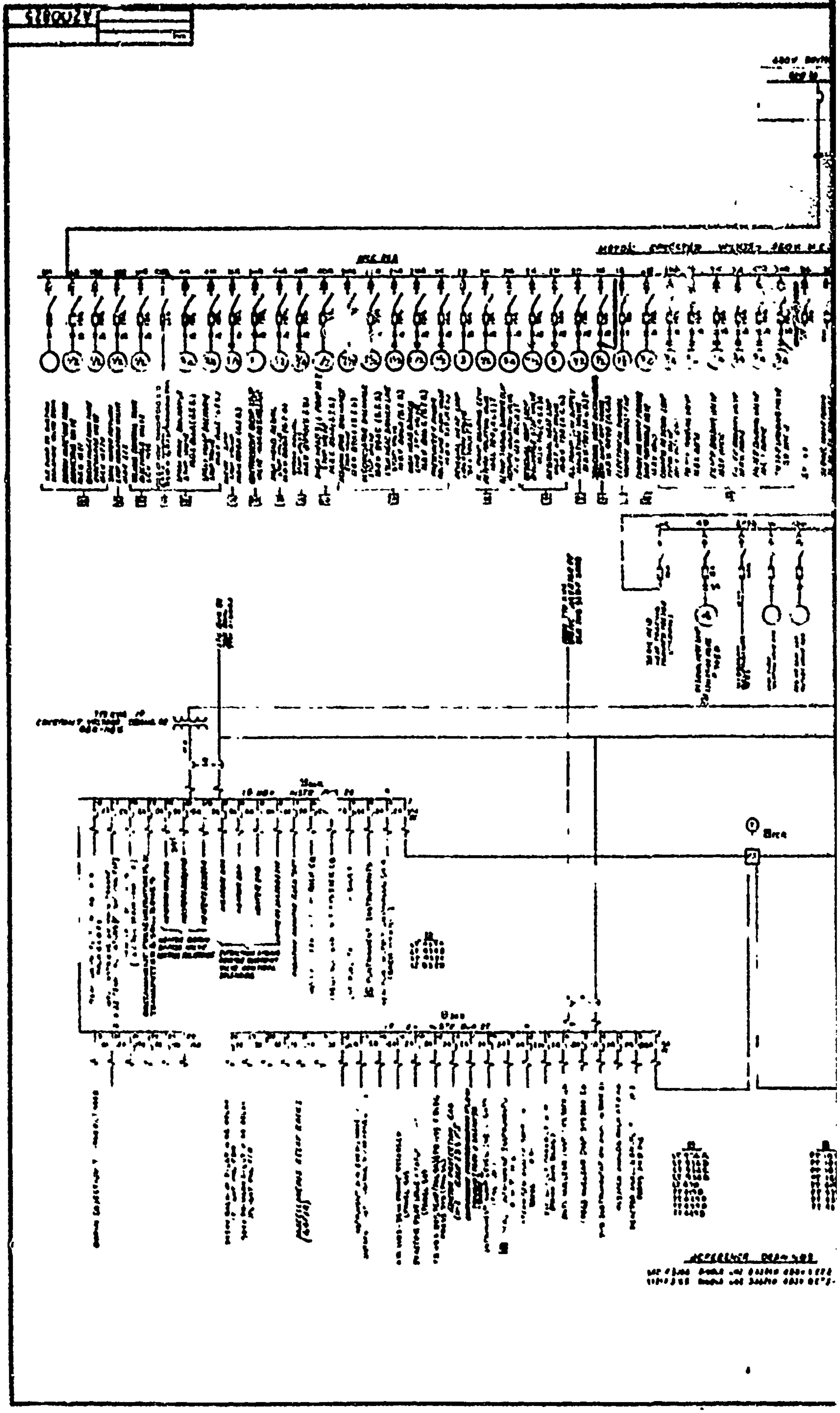
NOTES:
1. TYPE DB-50 ACB   ■: DB-75 ACB
2. ⊡ LOCATED ON CONTROL ROOM PANEL



FEEDER TO 480V M.C.C. 28
52/MCC8
RECIRCULATION PUMP 21
52/R1
COMPONENT COOLING PUMP 21
52/CC1
FEEDER TO 480V M.C.C. 26A
52/MCC6A
LIGHTING TRANS 23 225KVA 480-208/120V
52/LT3
CONTAINMENT RECIRC FAN 22
52/CRF2
CHARGING PUMP 21
52/C1
SERVICE AIR COMPRESSOR
52/SAC
FEEDER TO 480V MCC 29
SERVICE WATER PUMP 24
52/SW4
SERVICE WATER PUMP 21
52/SW1
480V SWGR. 21
3000A
52/2ATSA
52/2AT5A
EMERG DIESEL GENERATOR 22
52/EG2A
FUTURE
52/FUT
PRESSURIZER HEATER BACK-UP GROUP 22
CABLE 2-350 MCM/Φ
CABLE 2-500 MCM/Φ
BUS RUN
CONTAINMENT RECIRC. FAN 24
FEEDER TO 480V MCC 22
52/MCC2
SERVICE WATER PUMP 25
52/SW5A
PRESSURIZER HEATER BACKUP GROUP 21 555KW-480-480V
52/FBU2
SPARE
52/SP
CHARGING PUMP 22
52/C2
RESIDUAL HEAT REMOVAL PUMP 21
52/RHR1
52/EG2B
AUX. FEEDWATER PUMP 21
52/AF1
480V SWGR 22
52/3AT6A
ROD POWER SUPPLY M-G SET 22
52/MG2
LIGHTING TRANS 21 150KVA 480-208/120V (EMERGENCY FEED)
SERVICE WATER PUMP 26
52/SW6
SERVICE WATER PUMP 23
52/SW3



480 Volt One Line Diagram
Figure 8.2-4



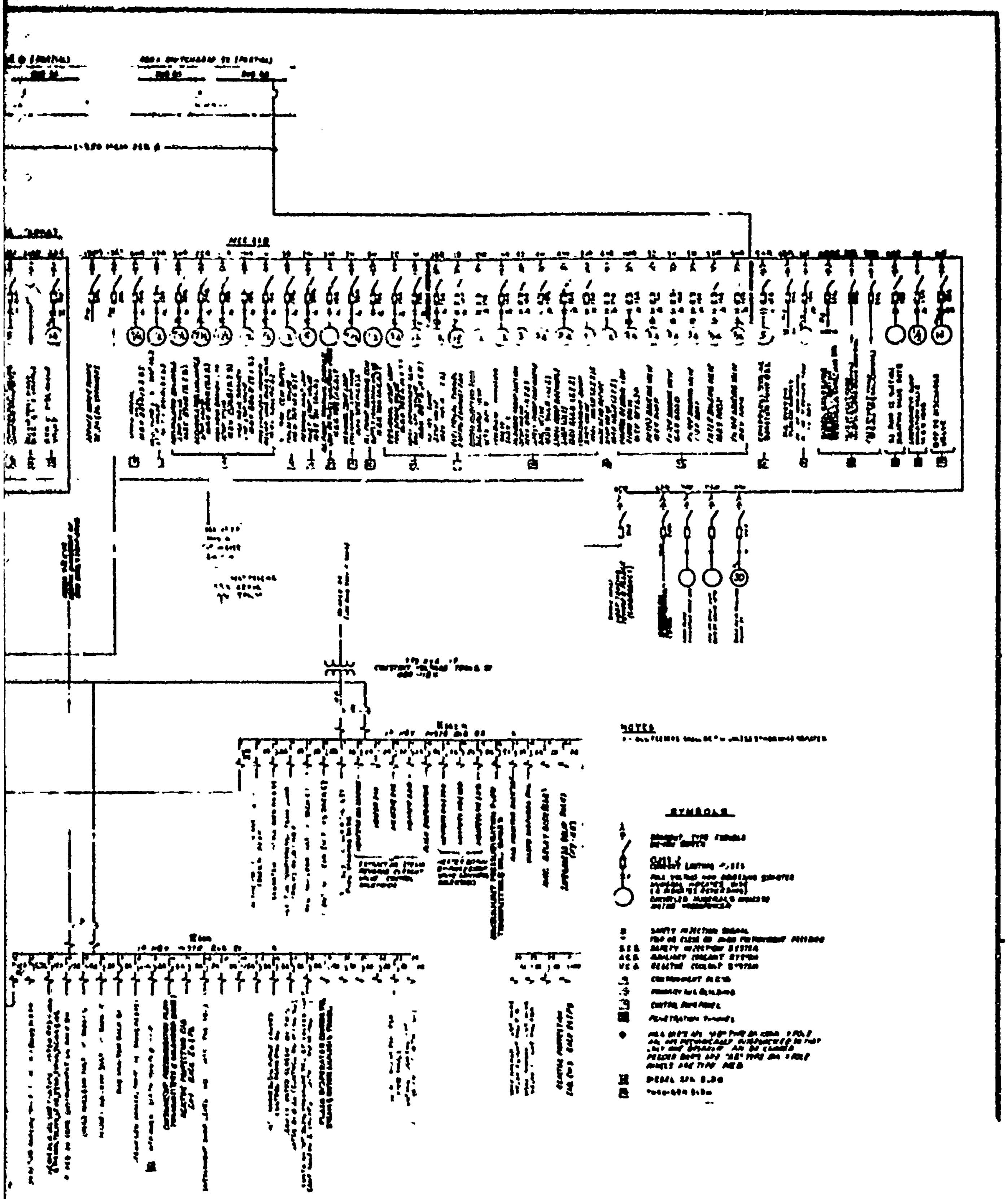


FIGURE 8.2-5
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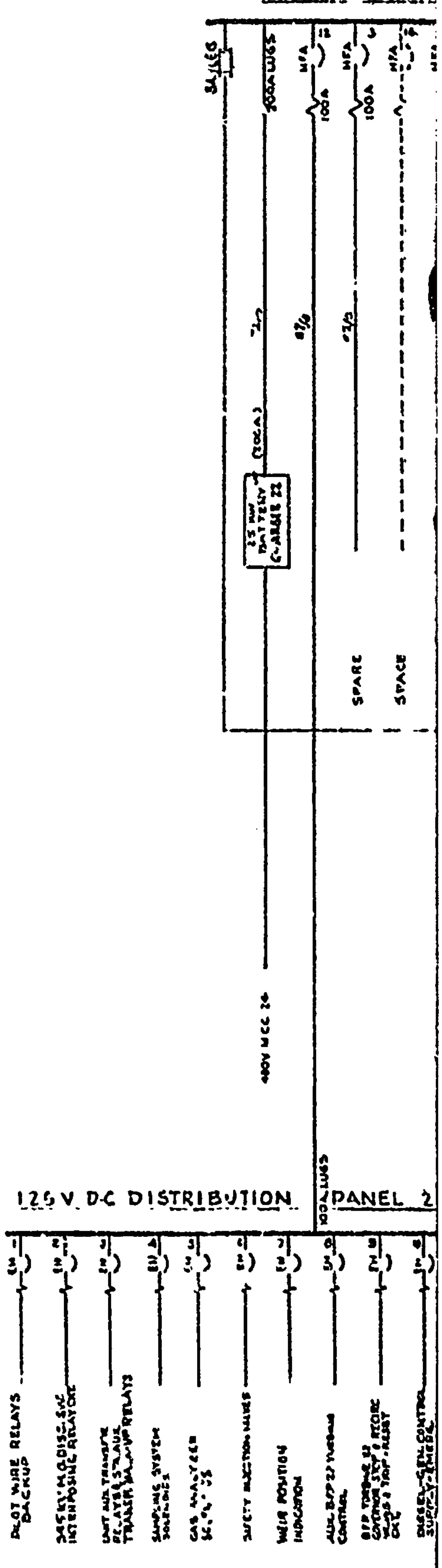
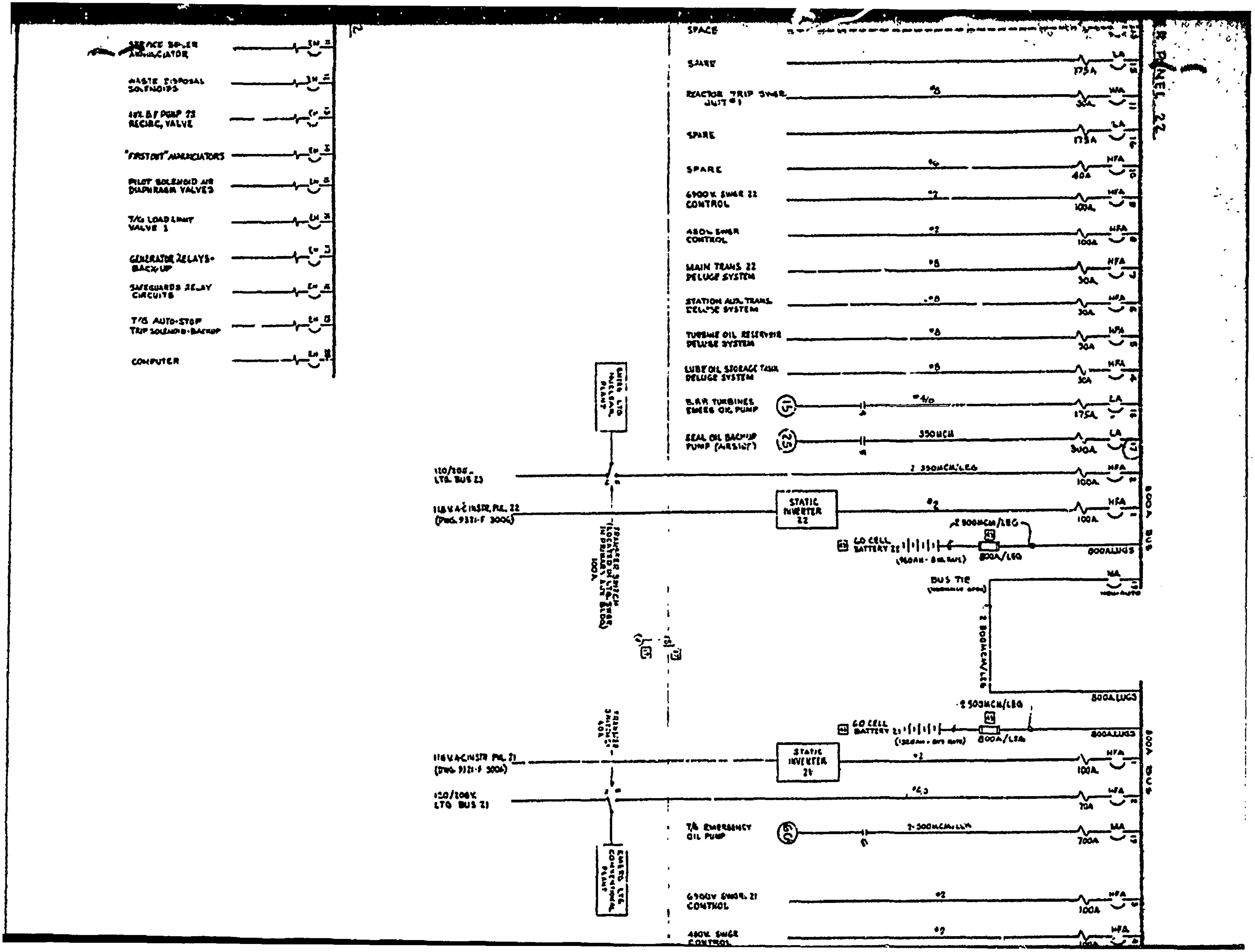
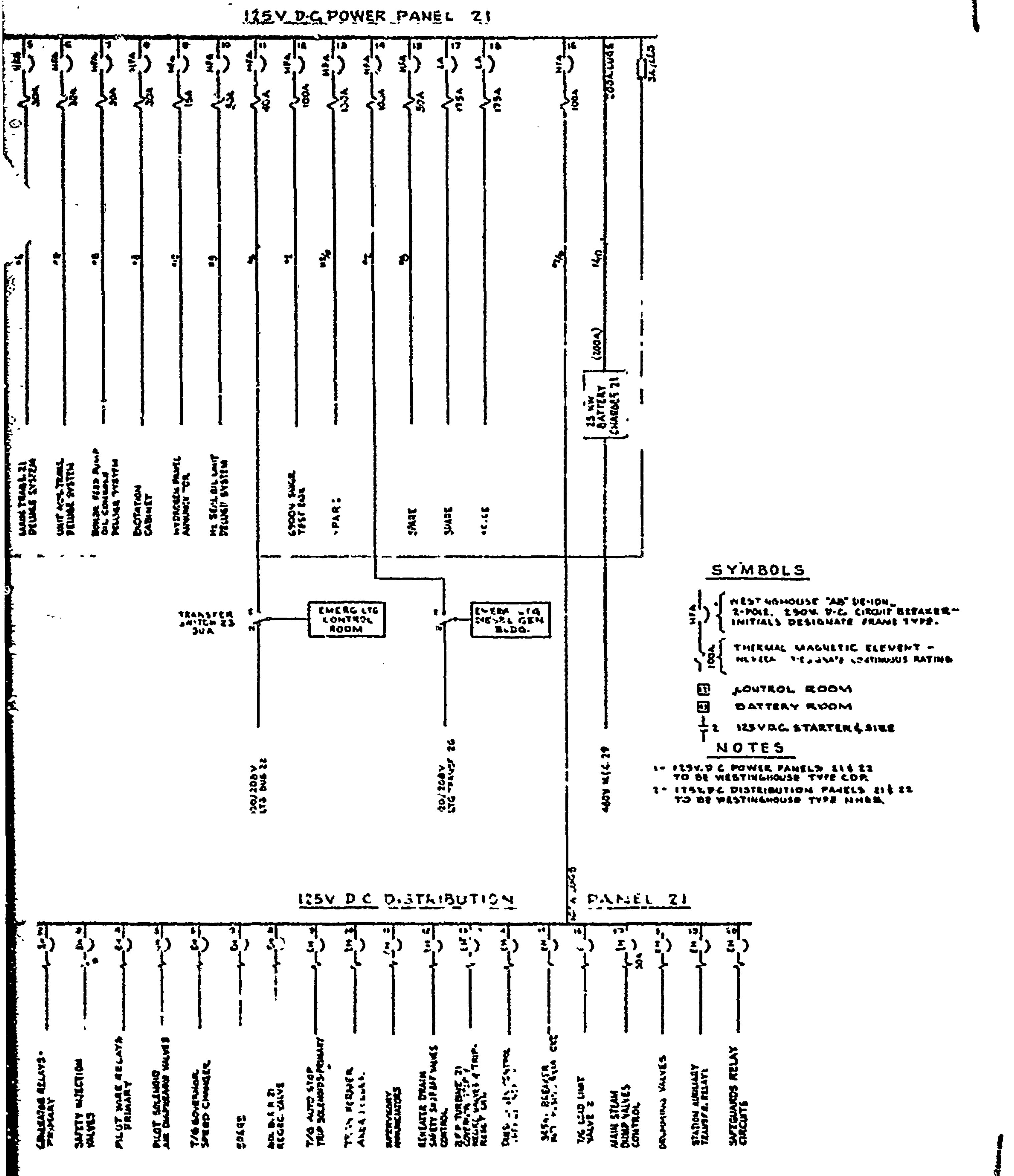


125 V D-C POW
SPARE
SPACE
125 V D-C DISTRIBUTION
PANEL 2



ER PANEL 22
800A BUS
800A BUS
SPACE
SPARE
REACTOR TRIP SWGR. UNIT #1
SPARE
SPARE
6900V SWGR 22 CONTROL
480V SWGR CONTROL
MAIN TRANS 22 DELUGE SYSTEM
STATION AUX. TRANS. DELUGE SYSTEM
TURBINE OIL RESERVOIR DELUGE SYSTEM
LUBE OIL STORAGE TANK DELUGE SYSTEM
SEAL OIL BACKUP PUMP (ASSIST)
STATIC INVERTER 22
60 CELL BATTERY 22
800A LUGS
BUS TIE
2 500MCM/LEG
60 CELL BATTERY 21
STATIC INVERTER 21
T/G EMERGENCY OIL PUMP
6900V SWGR. 21 CONTROL
480V SWGR CONTROL
EMERG LTG NUCLEAR PLANT
TRANSFER SWITCH (LOCATED IN LTG. SWGR. IN PRIMARY AUX BLDG) 100A
TRANSFER SWITCH 40A
EMERG LTG CONVENTIONAL PLANT
120/208V LTG. BUS 22
118 V A-C INSTR. PNL. 22
118 V A-C INSTR PNL. 21
120/208V LTG BUS 21
WASTE DISPOSAL SOLENOIDS
"FIRSTOUT" ANNUNCIATORS
PILOT SOLENOID AIR DIAPHRAGM VALVES
T/G LOAD LIMIT VALVE 1
GENERATOR RELAYS-BACK-UP
SAFEGUARDS RELAY CIRCUITS
T/G AUTO-STOP TRIP SOLENOID-BACKUP
COMPUTER



125 Volt D.C. One Line Diagram
Figure 8.2-6

8.3 MINIMUM OPERATING CONDITIONS

The reactor shall not be critical for other than "just critical" testing without:

START

1) The 138 kv - 6.9 kv station auxiliary transformer in service.

2) 2 diesel generators available. If one diesel generator is out of service, the remaining two will be started by manual actuation and run on a standby basis.

3) 3 of 4 480-volt buses energized by manual actuation.

4) 6.9 kv buses 1 thru 4 are energized via station auxiliary.

5) 1 of 2 battery chargers in service.

RUN

1) 22 kv 6.9 kv unit auxiliary transformer in service.

8.4 TESTS AND INSPECTIONS

The tests specified are designed to demonstrate that the dies generators will provide power for operation of equipment. They also assure that the emergency generator system controls and the control systems for safeguards equipment will function automatically in the event of a loss of all normal 480 volt AC station service power.

The testing frequency specified will be often enough to identify and correct deficiencies in systems under test before they can result in a system failure. The fuel supply and starting circuits and controls are continuously monitored and any faults are alarm indicated. An abnormal condition in these systems would be signaled without having to place the diesel generators themselves on test.

To verify that the emergency power system will respond within the required time limit and properly when required, the following tests shall be performed periodically:

a. Manually initiated demonstration of the ability of the diesel generators to start, and deliver power up to name plate rating, when operating in parallel with other power sources. Normal plant operation will not be affected. The duration of the test shall be at least 2 hours.

b. Demonstration of the readiness of the system and the control systems of vital equipment to automatically start or restore to operation particular vital equipment by initiating an actual loss of all normal AC station service power supplies. This test will be conducted during each refueling interval.

The starting of the diesel-generator sets can be tested from the Diesel Building. The ability of the units to start within the prescribed time and to carry intended loads are checked periodically.

Chap 9

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BY QUESTION RESPONSES IN
VOLUMES 5 AND 6

Section	Title	Question
9.2	Temperature detectors, alarm and control systems, electrical power requirements for heat tracing used in the boric acid piping and tanks. Evaluate single failure in the controller/alarm units, as well as loss of one source of power.	9.1
	Instrumentation and/or methods used to monitor concentration and level in the concentrated boric acid storage tanks.	9.2
	Instrumentation used to verify injection of concentrated boric acid flow into the primary system.	9.3
	Provisions for analyzing primary coolant for boron concentration, frequency of sampling and analysis for boron content, on-site capability	9.4
9.5	Spent fuel pool - thermal stresses, provisions to limit cracking and prevent leakage	5.12
	Prevention of handling of hearing objects over fuel racks	9.5
	Provisions to prevent dropping spent fuel element cask into spent fuel storage pool. Analysis of dropping cask into spent fuel storage pool - loss of water and loss of cooling function.	9.6
9.6	Fire protection system - seismic design classification, single failure, diagram of k_{eff}, failure of Class II positions which could damage Class I structures or equipment or prevent fire protection	9.7

Section	Title	Question
9.9	Dose design criteria and analysis for the control room using assumptions stated in Question 14.1	14.2

9. AUXILIARY AND EMERGENCY SYSTEMS

The Auxiliary and Emergency Systems are supporting systems required to insure the safe operation or servicing of the Reactor Coolant System (detailed in Section 4).

In some cases, the dependable operation of several systems is required to protect the Reactor Coolant System by controlling system conditions within specified operating limits. Certain systems are required to operate under emergency conditions.

This section considers systems in which component malfunctions, inadvertent interruptions of system operation, or a partial system failure may lead to a hazardous or unsafe condition. The extent of information provided for each system is proportional to the relative contribution of, or reliance placed upon, each system with respect to the overall plant operational safety.

The systems considered under this category are:

Chemical and Volume Control System This system provides for nuclear poison fluid injection, chemical additions for corrosion control, reactor coolant clean-up and degasification, reactor coolant make-up, reprocessing of water letdown from the Reactor Coolant System, and reactor coolant pump seal water injection.

Auxiliary Coolant System This system provides for transferring heat from reactor plant waters to the service water system and consists of the following three loops:

The residual heat removal loop removes residual and sensible heat from the core and reduces the temperature of the Reactor Coolant System during the second phase of plant cooldown.

The spent fuel pit loop removes from the spent fuel pit the heat generated by stored spent fuel elements.

The component cooling loop removes residual and sensible heat from the Reactor Coolant System, via the residual heat removal loop, during plant shutdown, cools the spent fuel pit water and the letdown flow to the Chemical and Volume Control System during power operation and provides cooling to dissipate waste heat from various primary plant components.

Sampling System This system provides the equipment necessary to obtain liquid and gaseous samples from the reactor plant systems.

Facility Service Systems These systems include fire protection, service water systems and auxiliary building ventilation systems.

Reactor Components Handling System This system provides for handling fuel assemblies, control rod assemblies, core structural components and material irradiation specimens.

Equipment and Decontamination Processes These procedures provide for the removal of radioactive deposits from primary system surfaces.

Primary Auxiliary Building Ventilation System This system maintains safe ambient operation temperatures and provides purging of the auxiliary building to the plant vent.

Control Room Ventilation System This system maintains a comfortable environment in the control room.

9.1 GENERAL DESIGN CRITERIA

The criteria which apply primarily to other systems discussed in other Sections are listed and cross-referenced because details of directly related systems and equipment are given in this Section. Those criteria which are specific to one of the Auxiliary and Emergency Systems are listed and discussed in the appropriate system design basis section.

9.1.1 RELATED CRITERIA

Reactivity Control Systems Malfunction

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

As described in Section 7, and justified in Section 14, the Reactor Protection Systems are designed to limit reactivity transients to DNBR $\geq$ 1.3 due to any single malfunction in the deboration controls.

Engineered Safety Features Performance Capability

Criterion: Engineered Safety Features such as the emergency core cooling system and the containment heat removal system shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public. (GDC 41)

Each of the auxiliary cooling systems which serves an emergency function provides sufficient capability in the emergency operational mode to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public.

Containment Heat Removal Systems

Criterion: Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure this system shall perform its required function, assuming failure of any single active component. (GDC 52)

Each of the auxiliary cooling systems which serves an emergency function to prevent exceeding containment design pressure, provides sufficient capability in the emergency operational mode to accomodate any single failure of an active component and still pe... its required function.

x

9.2 CHEMICAL AND VOLUME CONTROL SYSTEM

The Chemical and Volume Control System a) adjusts the concentration of chemical neutron absorber for chemical reactivity control, b) maintains the proper water inventory in the Reactor Coolant System, c) provides the required seal water flow for the reactor coolant pump shaft seals, d) processes reactor coolant effluent for reuse of boric acid and reactor makeup water, e) maintains the proper concentration of corrosion inhibiting chemicals in the reactor coolant and f) maintains the reactor coolant and corrosion activities to within design levels. The system is also used to fill and hydrostatically test the Reactor Coolant System.

During normal operation, this system has also provisions for supplying the following chemicals:

i) Regenerant chemicals to the deborating and evaporator condensate demineralizers

ii) Hydrogen to the volume control tank

iii) Nitrogen as required for purging the volume control tank

iv) Hydrazine and lithium hydroxide, as required, via the chemical mixing tank to the charging pumps suction.

9.2.1 DESIGN BASES

Redundancy of Reactivity Control

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

In addition to the reactivity control achieved by the rod cluster control (RCC) as detailed in Section 7, reactivity control is provided by the Chemical and Volume Control System which regulates the concentration of boric acid solution neutron absorber in the Reactor Coolant System. The system is designed to prevent, under anticipated system malfunction, uncontrolled or inadvertent reactivity changes which might cause system parameters to exceed design limits.

Reactivity Hold-Down Capability

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

Normal reactivity shutdown capability is provided by control rods, with boric acid injection used to compensate for the long term xenon decay transient and for plant cooldown. Any time that the plant is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection will always exceed that quantity required for the normal cold shutdown. This quantity will always exceed the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

The boric acid solution is transferred from the boric acid tanks by boric acid pumps to the suction of the charging pumps which inject boric acid into the reactor coolant. Any charging pump and boric acid transfer pump can be operated from diesel generator power on loss of off-site AC power. Boric acid can be injected by one charging pump and one boric acid transfer pump at a rate which shuts the reactor down with no rods inserted in less than sixteen minutes. In sixteen additional minutes, enough boric acid can be injected to compensate for xenon decay although xenon decay below the equilibrium operating level will not begin until approximately 12-15 hours after shutdown. If two boric acid pumps and two charging pumps are available, these time periods are halved. Additional boric acid is employed if it is desired to bring the reactor to cold shutdown conditions.

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

Reactivity Hot Shutdown Capability

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

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes. The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning of life of the initial core. The full length Rod Cluster Control (RCC) assemblies are divided into two categories comprising a control group and shutdown groups.

The control group, used in combination with chemical shim provides control of the reactivity changes of the core throughout the life of the core at power conditions. This group of RCC assemblies is used to compensate for short term reactivity changes at power that might be produced due to variations in reactor power requirements or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion and fission product buildup and decay.

Reactivity Shutdown Capability

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

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

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

Sufficient shutdown capability is also provided to maintain the core subcritical for the most severe anticipated cooldown transient associated with a single active failure, e.g., accidental opening of a steam bypass or relief valve. This is achieved with a combination of control rods and automatic boron addition via the Safety Injection System with the most reactive rod assumed to be fully withdrawn. Manually controlled boric acid addition is used to maintain the shutdown margin for the long term conditions of xenon decay and plant cooldown

Codes and Classifications

All pressure retaining components (or compartments of components) which are exposed to reactor coolant, comply with the following codes:

The tube side on both the regenerative and excess letdown heat exchangers are designed as ASME III, Class C. This designation is based on the following considerations: (a) each exchanger is connected to the primary coolant system by a 2" line, and (b) each is located inside the reactor containment. Analyses show that the accident associated with a 2" line break does not result in clad damage or failure. Additionally, a previously contaminated primary coolant, escaping from the primary coolant system during such accident is confined to the reactor containment building and no public hazard results.

9.2.2 SYSTEM DESIGN AND OPERATION

The Chemical and Volume Control System, shown in Figure 9.2-1 through 9.2-3, provides a means for injection of control poison in the form of boric acid solution, chemical additions for corrosion control, and reactor coolant cleanup and degasification. This system also adds makeup water to the Reactor Coolant System, reprocesses water letdown from the Reactor Coolant System, and provides seal water injection to the reactor coolant pump seals.

Overpressure protective devices are provided for system components whose design pressure and temperature are less than the Reactor Coolant System design limits.

System discharges from overpressure protective devices (safety valves) and system leakages are directed to closed systems. Effluents removed from such closed systems are monitored and discharged under controlled conditions.

System design enables post-operational hydrostatic testing to applicable code test pressures. The relief valves will be gaged during hydrostatic testing. The relief valves in systems that are hydrostatically tested after refueling operations will be set at the system design pressure.

During plant operation, reactor coolant flow through the letdown line from the reactor coolant loop cold leg on the suction side of the pump and is returned to the same cold leg on the discharge side of the pump via a charging line. An alternate charging connection is provided to the hot leg of another loop. An excess letdown line is also provided.

Each of the connections to the Reactor Coolant System has an isolation valve located close to the loop piping. In addition, a check valve is located downstream of each charging line isolation valve. Reactor coolant entering the Chemical and Volume Control System flows through the shell side of the regenerative heat exchanger where its temperature is reduced. The coolant then flows through a letdown orifice which reduces the coolant pressure. The cooled, low pressure water leaves the reactor containment and enters the auxiliary building where it undergoes a second temperature reduction in the tube side of the non-regenerative heat exchanger followed by a second pressure reduction by the low pressure letdown valve. After passing through one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filters and enters the volume control tank through a spray nozzle.

Hydrogen is automatically supplied, as determined by pressure control, to the vapor space in the volume control tank, which is predominantly hydrogen and water vapor. The hydrogen within this tank is, in turn, the supply source to the reactor coolant. Fission gases are periodically removed from the system by venting the volume control tank to the Waste Disposal System prior to a cold or refueling shutdown.

Next, the coolant flows to the charging pumps which raise the pressure above that in the Reactor Coolant System. The coolant then enters the containment, passes through the tube side of the regenerative heat exchanger, and is returned to the Reactor Coolant System.

The cation bed demineralizer, located downstream of the mixed bed demineralizers, is used intermittently to control cesium activity in the coolant and also to remove excess lithium which is formed from B^{10} (n, α) Li^7 reaction.

Boric acid is dissolved in hot water in the batching tank to a concentration of approximately 12 per cent by weight. The lower portion of the batching tank is jacketed to permit heating of the batching tank solution with low pressure steam. A transfer pump is used to transfer the batch to the boric acid tanks. Small quantities of boric acid solution are metered from the discharge of an operating transfer pump for blending with makeup water as makeup for normal leakage or for increasing the reactor coolant boron concentration during normal operation. Electric immersion heaters maintain the temperature of the boric acid tank's solution high enough to prevent precipitation.

During plant startup, normal operation, load reductions and shutdowns liquid effluents containing boric acid flow from the Reactor Coolant System through the letdown line and are collected in the holdup tanks. As liquid enters the holdup tanks, the nitrogen cover gas is displaced to the gas decay tanks in the Waste Disposal System through the waste vent header. The concentration of boric acid in the holdup tanks varies throughout core life from the refueling concentration to essentially zero at the end of the core cycle. A recirculation pump is provided to transfer liquid from one holdup tank to another.

Liquid effluent in the holdup tanks is processed as a batch operation. This liquid is pumped thr[illegible] the evaporator feed ion exchangers which primarily remove lithium hydroxide and fission-products such as cesium. It then flows through the ion exchanger filter and into the gas stripper where dissolved gases are removed from the liquid. The gases are vented to the Waste Disposal System. The liquid effluent from the gas stripper enters the boric acid evaporator.

The vapor produced in the boric acid evaporator leaves the evaporator condenser and is pumped through a condensate cooler where the distillate is cooled to the operating temperature of the evaporator condensate demineralizers. After non-volatile evaporator carry over is removed by one of the two evaporator condensate demineralizers it then flows through the condensate filter and accumulates in one of two monitor t[illegible] The dilute boric acid solution originally in the boric acid evaporator remains as the bottoms of the distillation process and is concentrated to approximately twelve per cent boric acid.

Subsequent handling of the condensate is dependent on the results of sample analysis. Discharge from the monitor tanks may be pumped to the primary water storage tank, recycled through the evaporator condensate demineralizers, returned to the holdup tanks for reprocessing in the evaporator train or discharged to the environment with the condenser circulating water within the allowable activity concentration as discussed in Section 11. If the sample analysis of the monitor tank contents indicates that it may be discharged safely to the environment, two valves must be opened to provide a discharge path. As the effluent leaves, it is continuously monitored by the waste disposal system liquid effluent monitor. If an unexpected increase in radioactivity is sensed, one of the valves in the discharge line to the condenser circulating water closes automatically and an alarm sounds in the control room.

Boric acid evaporator bottoms are discharged through a concentrates filter to the concentrates holding tank. Solution collected in the concentrates holding tank is sampled and then transferred to the boric acid tanks if analysis indicates that it meets specifications for use as boric acid. Otherwise the solution is pumped to the holdup tanks for reprocessing by the evaporator train.

The concentrated solution can also be pumped from the evaporator to the Waste Disposal System and finally placed in containers and mixed with cement. These containers can then be stored at the plant site for ultimate shipment off-site for disposal.

The deborating demineralizers can be used intermittently to remove boron from the reactor coolant near the end of the core life. When the deborating demineralizers are in operation, the letdown stream passes from the mixed bed demineralizers and then through the deborating demineralizers and into the volume control tank after passing through the reactor coolant filter.

During plant cooldown when the residual heat removal loop is operating and the letdown orifices are not in service, a flow path is provided to remove corrosion impurities and fission products. A portion of the flow leaving the residual heat exchangers passes through the non-regenerative heat exchanger, mixed bed demineralizers, reactor coolant filter and volume control tank. The fluid is then pumped, via the charging pump, through the tube side of the regenerative heat exchanger into the Reactor Coolant System.

Expected Operating Conditions

Tables 9.2-2, 9.2-3, and 9.2-5 list the system performance requirements, data for individual system components and reactor coolant equilibrium activity concentration. Table 9.2-4 supplements Table 9.2-5.

Reactor Coolant Activity Concentration

The parameters used in the calculation of the reactor coolant fission product inventory, including pertinent information concerning the expected coolant cleanup flow rate and demineralizer effectiveness, are presented in Table 9.2-4. The results of the calculations are presented in Table 9.2-5. In these calculations the defective fuel rods are assumed to be present at initial core loading and are uniformly distributed throughout the core and the fission product escape rate coefficients are therefore based upon an average fuel temperature.

The fission product activity in the reactor coolant during operation with small cladding defects* in 1% of the fuel rods is computed using the following differential equations:

For parent nuclides in the coolant,

$$\frac{dN_{wi}}{dt} = D\nu_i N_{C_i} - \left(\lambda_i + R\eta_i + \frac{B'}{B_o - tB'}\right) N_{wi}$$

* fuel rods containing pinholes or fine cracks

for daughter nuclides in the coolant,

$$\frac{dN_{wj}}{dt} = D\nu_j \; N_{C_j} - \left(\lambda_j + R\eta_j + \frac{B'}{B_o - tR'}\right) N_{wj} + \lambda_i N_{wi}$$

where:

N = population of nuclide

D = fraction of fuel rods having defective cladding

R = purification flow, coolant system volumes per sec.

B_o = initial boron concentration, ppm

B' = boron concentration reduction rate by feed and bleed, ppm per sec

η = removal efficiency of purification cycle for nuclide

λ = radioactive decay constant

ν = escape rate coefficient for diffusion into coolant

Subscript C refers to core

Subscript w refers to coolant

Subscript i refers to parent nuclide

Subscript j refers to daughter nuclide

Tritium is produced in the reactor from ternary fission in the fuel, irradiation of boron in the burnable poison rods (during initial fuel cycle only) and irradiation of boron, lithium and deuterium in the coolant. The deuterium contribution is less than 0.1 Ci per year and may be neglected. The parameters used in the calculation of tritium production rate are presented in Table 9.2-6.

Reactor Makeup Control

The reactor makeup control consists of a group of instruments arranged to provide a manually pre-selected makeup composition to the charging pump suction header or the volume control tank. The makeup control functions are to maintain desired operating fluid inventory in the volume control tank and to adjust reactor coolant boron concentration for reactivity and shim control.

Makeup for normal plant leakage is regulated by the reactor makeup control which is set by the operator to blend water from the primary water storage tank with concentrated boric acid to match the reactor coolant boron concentration.

The makeup system also provides concentrated boric acid or primary water or decrease the boric acid concentration in the Reactor Coolant System. To maintain the reactor coolant volume constant, an equal amount of reactor coolant at existing reactor coolant boric acid concentration is letdown to the holdup tanks. Should the letdown line be out of service during operation, sufficient volume exists in the pressurizer to accept the amount of boric acid necessary for cold shutdown.

Makeup water to the Reactor Coolant System is provided by the Chemical and Volume Control System from the following sources:

a) The primary water storage tank, which provides water for dilution when the reactor coolant boron concentration is to be reduced.

b) The boric acid tanks, which supply concentrated boric acid solution when reactor coolant boron concentration is to be increased.

c) The refueling water storage tank, which supplies borated water for emergency makeup.

d) The chemical mixing tank, which is used to inject small quantities of solution when additions of hydrazine or pH control chemical are necessary.

The reactor makeup control is operated from the control room by manually preselecting makeup compositon to the charging pump suction header or the volume control tank in order to adjust the reactor coolant boron concentration for reactivity control. Makeup is provided to maintain the desired operating fluid inventory in the Reactor Coolant System. The operator can stop the makeup operation at any time in any operating mode by remotely closing the makeup stop valves.

One primary water makeup pump and one boric acid transfer pump are normally
13 aligned for operation on demand from the reactor makeup control system.

A portion of the high pressure charging flow is injected into the reactor coolant pumps between the pump impeller and the shaft seal so that the seals are not exposed to high temperature reactor coolant. Part of the flow is the shaft seal leakage flow and the remainder enters the Reactor Coolant System through a labyrinth seal on the pump shaft. The shaft seal leakage flow cools the lower radial bearing, passes through the seals, is cooled in the seal water heat exchanger, filtered, and returned to the volume control tank.

Seal water inleakage to the Reactor Coolant System requires a continuous letdown of reactor coolant to maintain the desired inventory. In addition, bleed and feed of reactor coolant are required for removal of impurities and adjustment of boric acid in the reactor coolant.

Automatic Makeup

The "automatic makeup" mode of operation of the reactor makeup control provides boric acid solution preset to match the boron concentration in the Reactor Coolant System. The automatic makeup compensates for minor leakage of reactor coolant without causing significant changes in the coolant boron concentration.

Under normal plant operating conditions, the mode selector switch and makeup stop valves are set in the "Automatic Makeup" position. A preset low level signal from the volume control tank level controller causes the automatic makeup control action to open the makeup stop valve to the charging pump suction, open the concentrated boric acid control valve and the primary water makeup control valve. The flow controllers then blend the makeup stream according to the preset concentration. Makeup addition to the charging pump suction header causes the water level in the volume control tank to rise. At a preset high level point, the makeup is stopped; the primary water makeup control valve closes, the concentrated boric acid control valve closes and the makeup stop valve to charging pump suction closes.

Dilution

The "dilute" mode of operation permits the addition of a pre-selected quantity of primary water makeup at a pre-selected flow rate to the Reactor Coolant System. The operator sets the makeup stop valves to the volume control tank and to the charging pump suction in the closed position, the mode selector switch to "dilute", the primary water makeup flow controller set point to the desired flow rate, and the primary water makeup batch integrater to the desire quantity. If the dilution flow deviates ± 5 gpm from the preset flow rate, an alarm indicates the deviation. Opening the makeup stop valve to the volume control tank starts a primary water makeup pump. Makeup water is added to the volume control tank and then goes to the charging pump suction header. Excessive rise of the volume control tank water level is prevented by automatic actuation (by the tank level controller) of a three-way diversion valve, which routes the reactor coolant letdown flow to the holdup tanks. When the preset quantity of primary water makeup has been added, the batch integrater causes the reactor makeup water pump to stop and the primary water makeup control valve to close.

Boration

The "borate" mode of operation permits the addition of a pre-selected quantity of concentrated boric acid solution at a pre-selected flow rate to the Reactor Coolant System. The operator sets the makeup stop valves to the volume control tank and to the charging pump suction in the closed position, the mode selector switch to "borate", the concentrated boric acid flow controller set point to the desired flow rate, and the concentrated boric acid batch integrater to the desired quantity. Opening the makeup stop valve to the charging pumps suction starts the selected boric acid transfer pump, and the concentrated boric aci 's added to the charging pump suction he der. The total quantity added in most cases is so small that it has only a minor effect on the volume control tank level. When the preset quantity of concentrated boric acid solution has been added, the batch integrater causes the boric acid transfer pump to stop and the concentrated boric acid control valve to close.

The capability to add bor n to the reactor coolant is sufficient so that no limitation is imposed on the rate of cooldown of the reactor upon shutdown. The maximum rates of boration and the equivalent coolant cooldown rates are given in Table 9.2-2. One set of values is given for the addition of boric acid from a boric acid tank with one transfer and one charging pump operating. The other set assumes the use of refueling water but with two of the three charging pumps operating. The rates are based on full operating temperature and on the end of the core life when the moderator temperature coefficient is most negative.

Alarm Functions

The reactor makeup control is provided with alarm functions to call the operator's attention to the following conditions:

a) Deviation of primary water makeup flow rate from the control set point

b) Deviation of concentrated boric acid flow rate from the control set point

c) Low level (makeup initiation point) in the volume control tank when the reactor makeup control selector is not set for the automatic makeup control mode.

Charging Pump Control

Three positive displacement variable speed drive charging pumps are used to supply charging flow to the Reactor Coolant System.

The speed of each pump can be controlled manually or automatically. During normal operation, only one of the three pumps is automatically controlled. During normal operation, only one charging pump is operating and the speed is modulated in accordance with pressurizer level. During load changes the pressurizer level set point is varied automatically to compensate partially for the expansion or contraction of the reactor coolant associated with the $\bar{\bar{T}}_{avg}$ changes. $\bar{\bar{T}}_{avg}$ compensates for power changes by varying the pressurizer level set points in conjunction with pressurizer level for charging pump control. The level set points are varied between 20 and 60 percent of the adjustable range depending on the power level. Charging pump speed does not change rapidly with pressurizer level variations due to the reset action of the pressurizer level controller.

If the pressurizer level increases, the speed of the pump decreases, likewise if the level decreases, the speed increases. If the charging pump on automatic control reaches the high speed limit, an alarm is actuated and a second charging pump is manually started. The speed of the second pump is manually regulated. If the speed of the charging pump on automatic control does not decrease and the second charging pump is operating at maximum speed, the third charging pump can be started and its speed manually regulated. If the speed of the charging pump on automatic control decreases to its minimum value, an alarm is actuated and the speed of the pumps on manual control is reduced.

Components

A summary of principal component data is given in Table 9.2-3.

Regenerative Heat Exchanger

The regenerative heat exchanger is designed to recover the heat from the letdown stream by reheating the charging stream during normal operation. This exchanger also limits the temperature at the letdown orifices during periods when letdown flow exceeds charging flow by a greater margin than at normal letdown conditions.

The letdown stream flows through the shell of the regenerative heat exchanger and the charging stream flows through the tubes. The unit is made of austenitic stainless steel, and is of all-welded construction. The exchanger is designed to withstand 2000 step changes in shell side fluid temperature from 130°F to 552.2°F during the design life of the unit.

Letdown Orifices

One of the three letdown orifices controls flow of the letdown stream during normal operation and reduces the pressure to a value compatible with the non-regenerative heat exchanger design. Two of the letdown orifices are designed to pass normal letdown flow. The other orifice is designed to be used in conjunction with one normal letdown flow orifice for maximum purification flow at normal Reactor Coolant System operating pressure. The orifices are placed in and taken out of service remote manual operation of their respective isolation valves. One or both of the standby orifices may be used in parallel with the normally operating orifice in order to increase letdown flow when the Reactor Coolant System pressure is below normal. This arrangement provides a full standby capacity for control of letdown flow. Each orifice is an austentitic pipe containing a bored corrosion and erosion resistant insert.

Non-Regenerative (letdown) Heat Exchanger

The non-regenerative heat exchanger cools the letdown stream to the operating temperature of the mixed bed demineralizers. Reactor coolant flows through the tube side of the exchanger while component cooling water flows through the shell. The letdown stream outlet temperature is automatically controlled by a temperature control valve in the component cooling water outlet stream. The unit is a multiple-tube-pass heat exchanger. All surfaces in contact with the reactor coolant are austenitic stainless steel, and the shell is carbon steel.

Mixed Bed Demineralizers

Two flushable mixed bed demineralizers maintain reactor coolant purity. A lithium-7 cation resin and a hydroxyl form anion resin are initially charged into the demineralizers. Both forms of resin remove fission and corrosion products, and in addition, the reactor coolant causes the anion resin to be converted to the borate form. The resin bed is designed to reduce the concentration of ionic isotopes in the purification stream, except for cesium, yttrium, and molybdenum, by a minimum factor of 10.

Each demineralizer is sized to accommodate the maximum letdown flow. One demineralizer serves as a standby unit for use if the operating demineralizer becomes exhausted during operation.

The demineralizer vessels are made of austenitic stainless steel, and are provided with suitable connections to facilitate resin replacement when required. The vessels are equipped with a resin retention screen. Each demineralizer has sufficient capacity to enable refueling after operation for one core cycle with one per cent defective fuel rods.

Cation Bed Demineralizer

A flushable cation resin bed in the hydrogen form is located downstream of the mixed bed demineralizers and is used intermittently to control the concentration of lithium-7 which builds up in the coolant from the B^{10} (n, α) Li^7 reaction. The demineralizer also has sufficient capacity to maintain the cesium-137 concentration in the coolant below 1.0 μc/cc with one percent defective fuel. The demineralizer would be used intermittently to control cesium.

The demineralizer is made of austenitic stainless steel and is provided with suitable connections to facilitate resin replacement when required. The vessel is equipped with a resin retention screen.

Resin Fill Tank

The resin fill tank is used to charge fresh resin to the demineralizers. The line from the conical bottom of the tank is fitted with a dump valve and may be connected to any one of the demineralizer fill lines. The demineralizer water and resin slurry can be sluiced into the demineralizer by opening the dump valve. The tank, designed to hold approximately one-third the resin volume of one mixed bed demineralizer, is made of austenitic stainless steel.

Reactor Coolant Filter

The filter collects resin fines and particulates larger than 25 microns from the letdown stream. The vessel is made of austenitic stainless steel, and is provided with connections for draining and venting. Design flow capacity of the filter is equal to the maximum purification flow rate. Disposable synthetic filter elements are used. Bases being considered to determine when the reactor coolant filter will be replaced are: (1) a high pressure differential across the filter, (2) a set time limit after which the filter will be replaced, and (3) when a portable radiation monitor shows radiation in excess of established limits.

Volume Control Tank

The volume control tank collects the excess water released from zero power to full power, that is not accommodated by the pressurizer. It also receives the excess coolant release caused by the deadband in the reactor control temperature instrumentation. Overpressure of hydrogen gas is maintained in the volume control tank to control the hydrogen concentration in the reactor coolant at 25 to 35 cc per kg of water (standard conditions).

A spray nozzle is located inside the tank on the inlet line from the reactor coolant filter. This spray nozzle provides intimate contact to equilibriate the gas and liquid phases. A remotely operated vent valve discharging to the waste disposal system permits removal of gaseous fission products which are stripped from the reactor coolant and collected in this tank. The volume control tank also acts as a head tank for the charging pumps and a reservoir for the leakage from the reactor coolant pump controlled leakage seal. The tank is constructed of austenitic stainless steel.

Charging Pumps

Three charging pumps inject coolant into the Reactor Coolant System. The pumps are the variable speed positive displacement type, and all parts in contact with the reactor coolant are fabricated of austenitic stainless steel or other material of adequate corrosion resistance. These pumps have mechanical packing followed by a leakoff to collect reactor coolant before it can leak to the outside atmosphere. Pump leakage is piped to the drain header for disposal. The pump design prevents lubricating oil from contaminating the charging flow, and the integral discharge valves act as check valves.

Each pump is designed to provide the normal charging flow and the reactor coolant pump seal water supply during normal seal leakage. Each pump is designed to provide rated flow against a pressure equal to the sum of the Reactor Coolant System normal maximum pressure (existing when the pressurizer power operated relief valve is operating) and the piping, valve and equipment pressure losses at the design charging flows. The capacity of the three charging pumps permits operation at normal charging line flow with one reactor coolant pump shaft seal operating normally while other reactor coolant pumps are operating with floating ring seal flow.

Any one of the three charging pumps can be used to hydrotest the Reactor Coolant System.

Charging Pump Accumulators

A charging pump accumulator is attached to each charging pump outlet line to substantially reduce the outlet pressure pulses and reduce piping vibration.

Chemical Mixing Tank

The primary use of the chemical mixing tank is in the preparation of caustic solutions for pH control and hydrazine for oxygen scavenging.

The capacity of the chemical mixing tank is determined by the quantity of 35 per cent hydrazine solution necessary to increase the concentration in the reactor coolant by 10 ppm. This capacity is more than sufficient to prepare solution of pH control chemical for the Reactor Coolant System.

The chemical mixing tank is made of austenitic stainless steel.

Excess Letdown Heat Exchanger

The excess letdown heat exchanger cools reactor coolant letdown flow until the flow rate is equal to the nominal injection rate through the reactor coolant pump labyrinth seal, if letdown through the normal letdown path is blocked. The unit is designed to reduce the letdown stream temperature from the cold leg temperature to 195°F. The letdown stream flows through the tube side and component cooling water is circulated through the shell side. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel. All tube joints are welded. The unit is designed to withstand 2,000 step changes in the tube fluid temperature from 80°F to the cold leg temperature.

Seal Water Heat Exchanger

The seal water heat exchanger removes heat from two sources; reactor coolant pump seal water returning to the volume control tank and reactor coolant discharge from the excess letdown heat exchanger. Reactor coolant flows through the tubes and component cooling water is circulated through the shell side. The tubes are welded to the tube sheet because leakage could occur in either direction, resulting in undesirable contamination of the reactor coolant or component cooling water. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

The unit is designed to cool the excess letdown flow and the seal water flow to the temperature normally maintained in the volume control tank if all the reactor coolant pump seals are leaking at the maximum design leakage rate. 13

Seal Water Filter

The filter collects particulates larger than 25 microns from the reactor coolant pump seal water return and from the excess letdown heat exchanger flow. The filter is designed to pass the sum of the excess letdown flow and the maximum design leakage from the reactor coolant pump floating ring seals. The vessel is constructed of austenitic stainless steel and is provided with connections for draining and venting. Disposable synthetic filter elements are used.

Seal Water Injection Filters

Two filters are provided in parallel, each sized for the injection flow. They collect particulates larger than 5 microns from the water supplied to the reactor coolant pump seal.

Boric Acid Filter

The boric acid filter collects particulates larger than 25 microns from the boric acid solution being pumped to the charging pump suction line. The filter is designed to pass the design flow of two boric acid pumps operating simultaneously. The vessel is constructed of austenitic stainless steel and the filter elements are disposable synthetic cartridges. Provisions are available for venting and draining the filter.

Boric Acid Tanks

The boric acid tank capacities are sized to store sufficient boric acid solution for refueling enough boric acid solution for a cold shutdown shortly after full power operation is achieved. In addition, sufficient boric acid solution is available for cold shutdown if the most reactive RCC is not inserted. One tank supplies boric acid for reactor coolant makeup while recycled solutions from the concentrates holding tank is accumulated in the other tank.

The concentration of boric acid solution in storage is maintained between 11.5 and 13% by weight. Periodic manual sampling and corrective action is provided, if necessary, to ensure that these limits are maintained. Therefore, measured quantities of boric acid solution can be delivered to the reactor coolant to control the chemical poison concentration. The combination overflow and breather vent connection has a water loop seal to minimize vapor discharge during storage of the solution. The tank is constructed of austenitic stainless steel.

Boric Acid Tank Heaters

Two 1[illegible] capacity electric immersion heaters located near the bottom of each boric acid tank are designed to maintain the temperature of the boric acid solution at 165°F with an ambient air temperature of 40°F thus ensuring a temperature in excess of the solubility limit (for 20,000 ppm boron this is 130°F). The heaters are sheathed in austenitic stainless steel.

Batching Tank

The batching tank is sized to hold one week's makeup supply of boric acid solution for the boric acid tank. The basis for makeup is reactor coolant leakage of 1/2 gpm at beginning of core life. The tank may also be used for solution storage. A local sampling point is provided for verifying the solution concentration prior to transferring it to the boric acid tank or for draining the tank. The tank manway is provided with a removable screen to prevent entry of foreign particles. In addition, the tank is provided with an agitator to improve mixing during batching operations. The tank is constructed of austenitic stainless steel, and is not used to handle radioactive substances. The tank is provided with a steam jacket for heating the boric acid solution to 165°F.

Boric Acid Transfer Pumps

Two 100% capacity canned centrifugal pumps are used to circulate or transfer chemical solutions. Redundancy is thus provided for the pumps to permit maintenance during operation of the plant. The pumps circulate boric acid solution through the boric acid tanks and inject boric acid into the charging pump suction header.

Although one pump is normally used for boric acid batching and transfer and the other for boric acid injection, either pump may function as standby for the other. The design capacity of each pump is equal to the normal letdown flow rate. The design head is sufficient, considering line and valve losses, to deliver rated flow to the charging pump suction header when volume control tank pressure is at the maximum operating value (relief valve setting). All parts in contact with the solutions are austenitic stainless steel or other adequately corrosion-resistant material.

The transfer pumps are operated either automatically or manually from the main control room or from a local control center. The reactor makeup control operates one of the pumps automatically when boric acid solution is required for makeup or boration.

Boric Acid Blender

The boric acid blender promotes thorough mixing of boric acid solution and reactor makeup water from the reactor coolant makeup circuit. The blender consists of a conventional pipe fitted with a perforated tube insert. All material is austenitic stainless steel. The blender decreases the pipe length required to homogenize the mixture for taking a representative local sample.

Recycle Process

Holdup Tanks

Three holdup tanks contain radioactive liquid which enters the tank from the letdown line. Th liquid is released from the Reactor Coolant System during startup, shutdowns, load changes and from boron dilution to compensate for burnup. The contents of one tank are normally being processed by the gas stripper and evaporator train while another tank is being filled. The third tank is normally kept empty to provide additional storage capacity when needed.

The total liquid storage sizing basis for the holdup tanks is given in Table 9.2-3. The tanks are constructed of austenitic stainless steel.

Holdup Tank Recirculation Pump

The recirculation pump is used to mix the contents of a holdup tank or transfer the contents of a holdup tank to another holdup tank. The wetted surface of this pump is constructed of austenitic stainless steel.

Gas Stripper Feed Pumps

The three gas stripper feed pumps supply feed to the gas stripper boric acid evaporator trains from a holdup tank. The capacity of each pump is equal to the gas stripper-evaporator capacity. These canned centrifugal pumps are constructed of austenitic stainless steel.

Cation Ion Exchangers

Four cation flushable demineralizers remove cations (primarily cesium and lithium) from the holdup tank effluent. The resin is initially in the hydrogen form. The design flow rate is equal to the gas stripper boric acid evaporator processing rate. Experiments performed by Westinghouse indicate that the decontamination factor for cesium (See Table 9.2-4) is conservative. The demineralizer vessels are constructed of austenitic stainless steel and contain a resin retention screen.

Ion Exchanger Filters

These filters collect resin fines and particulates larger than 25 microns from the cation ion exchanger. The vessel is made of austenitic stainless steel and is provided with connections for draining and venting. Disposable synthetic filter elements are used. The design flow capacity is equal to the boric acid evaporator flow rate.

Gas Stripper Equipment

Two gas strippers are provided. Each removes nitrogen, hydrogen, and fission gases from the evaporator feed. The gas stripper equipment consists of a preheater, stripping column and associated pumps, piping and instrumentation.

The gas stripper preheater located upstream of the gas stripper heats the liquid effluent from the holdup tanks from ambient temperature, at the evaporator train processing rate, using the gas stripper bottoms which are cooled in the preheater. The preheater is a regenerative type shell and tube unit constructed of austenitic stainless steel.

The gas stripper consists of a hot well with heating coil to store stripped water, a stripping section packed with pall rings, a spray type liquid inlet header and an overhead auxiliary spray. Liquid flowing to the gas strippers is controlled to constant rate by a flow controller. The gas strippers are designed for the same flow rate as the evaporator and are designed to reduce the influent gas concentration by a factor of 10^5.

Two gas stripper bottom pumps per gas stripper, operated from level control, transfer effluent from the gas stripper hot wells to the boric acid evaporator via the gas stripper preheaters. Each centrifugal pump is rated at the evaporator processing rate. The pumps are austenitic stainless steel and one is an installed standby for the operating pump.

Boric Acid Evaporator Equipment

The boric acid evaporators are provided. The boric acid evaporators concentrate boric acid for reuse in the Reactor Coolant System. Borated water enters the evaporator and the liquid is concentrated to approximately 12 weight per cent boric acid. Vapors leave the evaporator and are condensed. The solids decontamination factor between the condensate and the bottoms is approximately 10^6. All evaporator equipment is constructed of austenitic stainless steel and is supplied as a unit. The boric acid evaporator equipment consists of the boric acid evaporator feed tank, two boric acid evaporator concentrates pumps, boric acid evaporator, boric acid evaporator condenser, two boric acid evaporator condensate pumps, boric acid evaporator condensate cooler, two vacuum pumps and associated piping and instrumentation.

The evaporator and condenser heat transfer area is sufficient to maintain the required feed rate. The evaporator is steam heated. Component cooling water flows through the tubes of the condenser.

The boric acid distillate coolers reduce the temperature of the condensate to approximately 100°F. The condensate flows through the shell and component cooling water through the tubes.

Evaporator Condensate Demineralizers

Two anion demineralizers remove any boric acid contained in the evaporator condensate. Hydroxyl based ion-exchange resin is used to produce evaporator condensate of high purity. Facilities are provided for regeneration of the resin. When regeneration is no longer feasible, the resin is flushed to the spent resin storage tank. The resin volume in each demineralizer is selected to keep resin regenerations to an average of once per month during a core cycle.

Condensate Filters

The filters collect resin fines and particulates larger than 25 microns from the boric acid evaporator condensate streams. The vessel is made of austenitic stainless steel, and is provided with a connection for draining and venting. Disposable synthetic filter elements are used. The design flow capacity of the filter is equal to the boric acid evaporator flow rate.

Monitor Tanks

Three monitor tanks permit continuous operation of the evaporator train. When one tank is filled, the contents are analyzed and either, reprocessed, discharged to the Waste Disposal System, or pumped to the primary water storage tank. These tanks contain a diaphragm membrane and are stainless steel.

Monitor Tank Pumps

Three monitor tank pumps discharge water from the monitor tanks. The pumps are sized to empty a monitor tank in approximately 2.0 hours. The pumps are constructed of austenitic stainless steel.

Primary Water Storage Tank

The primary water storage tank is used to store makeup water which is supplied from the monitor tanks and the water treatment plant. Makeup water from the tank discharges to the suction of the primary water makeup pumps. The tank contains a diaphragm membrane and is stainless steel.

Primary Water Makeup Pumps

Two primary water makeup pumps discharge from either the monitor tanks or the primary water storage tank. These pumps are used to feed dilution water to the boric acid blender and are also used to supply makeup water for intermittent flushing of equipment and piping.

Each pump is sized to match the maximum letdown flow. One pump serves as a standby for the other. These centrifugal pumps are constructed of austenitic stainless steel.

Concentrates Filter

A disposable synthetic cartridge type filter removes particulates larger than 25 microns from the evaporator concentrates. Design flow capacity of the filter is equal to the boric acid evaporator concentrates transfer pump capacity. The vessel is made of austenitic stainless steel.

Concentrates Holding Tank

The concentrates holding tank is sized to hold approximately the production of concentrates from one batch of evaporator operation. The tank is supplied with an electrical heater which prevents boric acid precipitation and is constructed of austenitic stainless steel.

Concentrates Holding Tank Transfer Pumps

Two holding tank transfer pumps discharge boric acid solution from the concentrates holding tank to the boric acid tanks. The canned centrifugal pumps are sized to empty the concentrates holding tank in 20 minutes. The wetted surfaces are constructed of austenitic stainless steel.

Deborating Demineralizers

When required, two anion demineralizers remove boric acid from the Reactor Coolant System fluid. The demineralizers are provided for use near the end of a core cycle, but can be used at any time. Hydroxyl based ion-exchange resin is used to reduce Reactor Coolant System boron concentration by releasing a hydroxyl ion when a borate ion is absorbed. Facilities are provided for regeneration. When regeneration is no longer feasible, the resin is flushed to the spent resin storage tank.

Each demineralizer is sized to remove the quantity of boric acid that must be removed from the Reactor Coolant System to maintain full power operation near the end of core life should the holdup tanks be full.

Electrical Heat Tracing

Electrical heat tracing is installed under the insulation on all piping, valves, line-mounted instrumentation, and components normally containing concentrated boric acid solution. The heat tracing is designed to prevent boric acid precipitation due to cooling, by compensating for heat loss.

Exceptions are:

a) Lines which may transport concentrated boric acid but are subsequently flushed with reactor coolant or other liquid of low boric acid concentration during normal operation

b) The boric acid tanks, which are provided with immersion heaters

c) The batching tank, which is provided with a steam jacket

d) The concentrates holding tank, which is provided with an immersion heater.

Duplicate tracing on s tions of the Chemical and Volume Contr..l System normally containing boric acid solution provides backup capacity if the operating tracing malfunctions.

Temperature detectors, alarm and control functions and electrical power requirements for the heat tracing which are p..ovided are not shown on the process flow diagram.

<u>Valves</u>

Valves that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to the Wast. Disposal System. All other valves have stem leakage control. Globe valves are installed wi.h flow over the seats when such an arrangement reduces the possibility of leakage. Basic material of construction is stainless steel for all valves except the batching tank steam jacket valves which are carbon steel.

Isolation valves are provided at all connections to the Reactor Coolant System. Lines entering the reactor containment also have check valves inside the containment to prevent reverse flow from the containment.

Relief valves are provided for lines and components that might be pressurized above design pressure by improper operation or component malfunction. Pressure relief for the tube side of the regenerative heat exchanger is provided by the auxiliary spray line isolation valve which i. designed to open when pressure under the seat exceeds reactor coolant pressure by 250 psi.

<u>Piping</u>

All Chemical and Volume Control System piping handling radioactive liquid is austenitic stainless steel. All piping joints and connections are welded, except where flanged connections are required to facilitate equipment removal for maintenance and hydrostatic testing. Piping, valves, equipment and line-mounted instrumentation, which normally contain concentrated boric solution, are heated by electrical tracing to ensure solubility of the boric acid.

9.2.3 SYSTEM DESIGN EVALUATION

Availability and Reliability

A high degree of functional reliability is assured in this system by providing standby components where performance is vital to safety and by assuring fail-safe response to the most probable mode of failure. Special provisions include duplicate heat tracing with alarm protection of lines, valves, and components normally containing concentrated boric acid.

The system has three high pressure charging pumps, each capable of supplying the normal reactor coolant pump seal and makeup flow.

The electrical equipment of the Chemical and Volume Control System is arranged so that multiple items receive their power from various 480 volt buses (See Figure 8.2-4). Each of the three charging pumps are powered from separate 480 volt buses. The two boric acid transfer pumps are also powered from separate 480 volt buses. One charging pump and one boric acid transfer pump are capable of meeting cold shutdown requirements shortly after full-power operation. In cases of loss of a-c power, a charging pump and a boric acid transfer pump can be placed on the emergency diesels if necessary.

Control of Tritium

The Chemical and Volume Control System is used to control the concentration of tritium in the Reactor Coolant System. Essentially all of the tritium is in chemical combination with oxygen as a form of water. Therefore, any leakage of coolant to the containment atmosphere carries tritium in the same proportion, as it exists in the coolant. Thus, the level of tritium in the containment atmosphere, when it is sealed from outside air ventilation, is a function of tritium level in the reactor coolant, the cooling water temperature at the cooling coils, which determines the dew point temperature of the air, and the presence of leakage other than reactor coolant as a source of moisture in the containment air.

There are two major considerations with regard to the presence of tritium:

a) Possible plant personnel hazard during access to the containment. Leakage of reactor coolant during operation with a closed containment causes an accumulation of tritium in the containment atmosphere. It is desirable to limit the accumulation to allow containment access for two hours per week for in-core instrumentation maintenance.

b) Possible public hazard due to release of tritium to the plant environment.

Neither of these considerations is limiting in this plant.

The concentration of tritium in the reactor coolant is maintained at a level which preludes personnel hazard during access to the containment. This is achieved by discharging part of the condensate from the boric acid recovery process to the river via the plant circulating cooling water.

In Section 11 it is shown that the tritium released to the environment in this manner is between 10^{-2} and 10^{-4} of 10 CFR 20 limits, and thus no public hazard would result.

Leakage Prevention

Quality control of the material and the installation of the Chemical and Volume Control valves and pipings which are designated for radioactive service, is provided in order to essentially eliminate leakage to the atmosphere. The components designated for radioactive service are provided with welded connections to prevent leakage to the atmosphere. However, flanged connections are provided in each charging pump suction and discharge, on each boric acid pump suction and discharge, on the relief valves inlet and outlet, on three-way valves and on the flow meters to permit removal for maintenance.

The positive displacement charging pumps stuffing boxes are provided with leakoffs to collect reactor coolant before it can leak to the atmosphere. All valves which are larger than 2 inches and which are designated for radioactive service at an operating fluid temperature above 212°F are provided with a stuffing box and lantern leakoff connections. Leakage to the atmosphere is essentially zero for these valves. All control valves are either provided with stuffing box and leakoff connections or are totally enclosed. Leakage to the atmosphere is essentially zero for these valves.

Diaphragm valves are provided where the operating pressure and the operating temperature permit the use of these valves. Leakage to the atmosphere is essentially zero for these valves.

Incident Control

The letdown line and the reactor coolant pumps seal water return line penetrate the reactor containment. The letdown line contains air-operated valves inside the reactor containment and two air-operated valves outside the reactor containment which are automatically closed by the containment isolation signal.

The reactor coolant pumps seal water return line contains one motor-operated isolation valve outside the reactor containment which is automatically closed by the containment isolation signal.

The four seal water injection lines to the reactor coolant pumps and the normal charging line are inflow lines penetrating the reactor containment. Each line contains a check valve inside the reactor containment to provide isolation of the reactor containment when a break occurs in these lines outside the reactor containment.

Malfunction Analysis

To evaluate system safety, failures or malfunctions were assumed concurrent with a loss-of-coolant accident and the consequences analyzed and presented in Table 9.2-7. As a result of this evaluation, it is concluded that proper consideration has been given to station safety in the design of the system.

If a rupture were to take place between the reactor coolant loop and the first isolation valve or check valve, this incident would lead to an uncontrolled loss of reactor coolant. The analysis of loss of coolant accidents is discussed in Section 14.

Should a rupture occur in the Chemical and Volume Control System outside the containment, or at any point beyond the first check valve or remotely operated isolation valve, actuation of the valve would limit the release of coolant and assure continued functioning of the normal means of heat dissipation from the core. For the general case of rupture outside the containment, the largest source of radioactive fluid subject to release is the contents of the volume control tank. The consequences of such a release are considered in Section 14

When the reactor is subcritical; i.e., during cold or hot shutdown, refueling and approach to criticality, the relative reactivity status (neutron source multiplication) is continuously monitored and indicated by BF_3 counters and count rate indicators. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate (approximately 480 ppm/hr), is slow enough to give ample time to start a corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical. The maximum dilution rate is based on the abnormal condition of two charging pumps operating at full speed delivering unborated makeup water to the Reactor Coolant System at a particular time during refueling when the boron concentration is at the maximum value and the water volume in the system is at a minimum.

At least two separate and independent flow paths are available for reactor coolant boration; i.e., either the charging line, or the reactor coolant pumps labyrinths. The malfunction or failure of one component will not result in the inability to borate the Reactor Coolant System. An alternate suply path is always available for emergency boration of the reactor coolant. As backup to the boration system, the operator can align the refueling water storage tank outlet to the suction of the charging pumps.

On loss of seal injection water to the reactor coolant pump seals, seal water flow may be reestablished by manually starting a standby charging pump. Even if the seal water injection flow is not reestablished, the plant can be operated indefinitely since the thermal barrier cooler has sufficient capacity to cool the reactor coolant flow which would pass through the thermal barrier cooler and seal leakoff from the pump volute.

Galvanic Corrosion

The only types of materials which are in contact with each other in borated water are stainless steels, Inconel, Stellite valve materials and Zircaloy fuel element cladding. These materials have been shown[1] to exhibit only an insignificant degree of galvanic corrosion when coupled to each other.

For example, the galvanic corrosion of Inconel versus 304 stainless steel resulting from high temperature tests (575°F) in lithiated, boric acid solution was found to be less than -20.9 mg/dm^2 for the test period of 9 days. Further galvanic corrosion would be trivial since the cell currents at the conclusion of the tests were approaching polarization. Zircaloy versus 304 stainless steel was shown to polarize at 180°F lithiated, boric acid solution in less than 8 days with a total galvanic attack of -3.0 mg/dm^2. Stellite versus 304 stainless steel was polarized in 7 days at 575°F in lithiated boric acid solution. The total galvanic corrosion for this couple was -0.97 mg/dm^2.

[1] WCAP 1844 "The Galvanic Behavior of Materials in Reactor Coolants" D. G. Sammarone, August, 1961.

As can be seen from the tests, the effects of galvanic corrosion are insignificant to systems containing borated water.

Boration during normal operation to compensate for power changes will be indicated to the operator from two sources; (a) the control rod movement and (b) the flow indicators in the boric acid transfer pump discharge line. When the emergency boration path is used, two indications to the operator are available. The charging line flow indicator will indicate boric acid flow since the charging pump suction is aligned to the boric acid transfer pump suction for this mode of operation. The change in boric acid tank level is another indication of boric acid injection.

9.2.4 MINIMUM OPERATING CONDITIONS

Minimum operating conditions will be specified in the Technical Specifications.

9.2.5 TESTS AND INSPECTIONS

The minimum frequencies for testing, calibrating and/or checking instrument channels for the Chemical and Volume Control System are presented in Table 9.2-8.

TABLE 9.2-1

CHEMICAL AND VOLUME CONTROL SYSTEM CODE REQUIREMENTS

Regenerative heat exchanger	ASME III*, Class C
Non-Regenerative heat exchanger	ASME III, Class C, tube side, ASME VIII, shell side
Mixed bed demineralizers	ASME III, Class C
Reactor coolant filter	ASME III, Class C
Volume control tank	ASME III, Class C
Seal water heat exchanger	ASME III, Class C, tube side, ASME VIII, shell side
Excess letdown heat exchanger	ASME III, Class C, tube side, ASME VIII, shell side
Chemical mixing tank	ASME VIII
Deborating demineralizers	ASME III, Class C
Cation bed demineralizer	ASME III, Class C
Seal water injection filters	ASME III, Class C
Holdup tanks	ASME III, Class C
Boric acid filter	ASME III, Class C
Gas stripper package	ASME III, Class C
Boric acid evaporator package	ASME III, Class C
Evaporator condensate demineralizers	ASME III, Class C
Concentrates filter	ASME III, Class C
Cation ion exchanger	ASME III, Class C
Ion exchanger filter	ASME III, Class C
Condensate filter	ASME III, Class C
Piping and valves	USAS B31.1**,

* ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

**USAS B31.1 - Code for Pressure Piping, and special nuclear cases where applicable.

TABLE 9.2-2*

CHEMICAL AND VOLUME CONTROL SYSTEM PERFORMANCE REQUIREMENTS

Plant design life, years	40
Seal water supply flow rate, gpm	32
Seal water return flow rate, gpm	12
Normal letdown flow rate, gpm	75
Maximum letdown flow rate, gpm	120
Normal charging pump flow (one pump), gpm	87
Normal flow to reactor coolant pumps, gpm	32
Normal charging line flow, gpm	55
Maximum rate of boration with one transfer and one charging pump, ppm/min	24
Equivalent cooldown rate to above rate of boration, °F/min	7.0
Maximum rate of boron dilution (maximum design letdown rate), ppm/hour	300
Two-pump rate of boration, using refueling water, ppm/min	7.4
Equivalent cooldown rate to above rate of boration, °F/min	2.1
Temperature of reactor coolant entering system at full power, °F	555.0
Temperature of coolant return to Reactor Coolant System at full power, °F	505.0
Normal coolant discharge temperature to holdup tanks, °F	127.0
Amount of 12% boric acid solution required to meet cold shutdown requirements shortly after full power operation, gallons	4000

* Volumetric flow rates in gpm are based on 127°F and 15 psig.

* Reactor coolant water quality is given in Table 4.2-4.

TABLE 9.2-3
CHEMICAL AND VOLUME CONTROL SYSTEM
PRINCIPLE COMPONENT DATA SUMMARY

	Quantity	Heat Transfer Btu/hr	Design Letdown Flow lb/hr	Letdown ΔT F°	Design Pressure psig, shell/tube	Design Temperature °F, shell/tube
Heat Exchangers						
Regenerative	1	10.5×10^6	37050	249	2485/2735	650/650
Letdown	1	14.8×10^6	59300	253	150/500	200/400
Seal Water	1	2.4×10^6	158,000	14	150/150	200/250
Excess Letdown	1	4.7×10^6	12,400	355	150/2485	200/650

	Quantity	Type	Capacity gpm	Head ft or psi	Design Pressure psig	Design Temperature °F
Pumps						
Charging	3	Pos. Displ.	98	2500 psi	3000	250
Boric acid transfer	2	Canned	75	235 ft	150	250
Hold up tank recirculation	1	Centrifugal	500	100 ft	75	200
Primary water makeup	2	Centrifugal	150	235 ft	150	250
Monitor tank	3	Centrifugal	60	235 ft	50	250
Concentrates holding tank transfer	2	Canned	40	150 ft	150	250
Gas stripper feed	3	Canned	12.5	200 ft	150	200
Gas stripper bottom	4	Canned	12.5	93 ft	75	300

	Quantity	Type	Volume	Design Pressure psig	Design Temperature °F
Tanks					
Volume control	1	Vertical	400 ft^3	75/15	250
Charging pump accumulator	3	Vertical	100 in^3	3000	250
Boric acid	2	Vertical	8000 gal	atmos.	250
Chemical mixing	1	Vertical	5.0 gal	150	250
Batching	1	Jacket btm.	400 gal	atmos.	250
Holdup	3	Vertical	8,500 ft^3	15	200
Primary water storage	1	Diaphragm	165,000	atmos.	150
Concentrates holding	1	Vertical	800 gal	atmos.	250
Monitor	3	Diaphragm	7,500 gal.	atmos.	150
Resin Fill	1	Open	8 cu. ft.	-	-

TABLE 9.2-3 (Continued)

	Quantity	Type	Resin Volume ft^3	Flow gpm	Design Pressure psig	Design Temperature °F
Demineralizers						
Mixed Bed	2	Flushable	30	120	200	250
Cation Bed	1	Flushable	12.0	40	200	250
Evaporator Feed	4	Flushable	120	12.5	200	250
Evaporator Condensate	2	Fixed	12.0	12.5	200	250
Deborating	2	Fixed	30	120	200	250

TABLE 9.2-4

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT FISSION PRODUCT ACTIVITIES

1.	Core thermal power, MWt	3216
2.	Fraction of fuel containing clad defects	0.01
3.	Reactor coolant liquid volume, cu ft	12,600
4.	Reactor coolant average temperature, °F	573
5.	Purification flow rate (normal), gpm	75
6.	Effective cation demineralizer flow, gpm	7
7.	Volume control tank volumes	
	a. Vapor, cu ft	270
	b. Liquid, cu ft	130
8.	Fission product escape rate coefficients:	
	a. Noble gas isotopes, sec^{-1}	6.5×10^{-8}
	b. Br, I and Cs isotopes, sec^{-1}	1.3×10^{-8}
	c. Te isotopes, sec^{-1}	1.0×10^{-9}
	d. Mo isotopes, sec^{-1}	2.0×10^{-9}
	e. Sr and Ba isotopes, sec^{-1}	1.0×10^{-11}
	f. Y, La, Ce and Pr isotopes, sec^{-1}	1.6×10^{-12}
9.	Mixed bed demineralizer decontamination factors:	
	a. Noble gases and Cs-134, 136, 137, Y-90 and Mo-99	1.0
	b. All other isotopes	10.0
10.	Cation bed demineralizer decontamination factor for Cs-134, 136, 137, Y-90 and Mo-99	10.0
11.	Volume control tank noble gas stripping fraction (closed system):	

Isotope	Stripping Fraction
Kr-85	2.[illegible] $\times 10^{-5}$
Kr-85m	2.7×10^{-1}
Kr-87	6.0×10^{-1}
Kr-88	4.3×10^{-1}
Xe-133	1.6×10^{-2}
Xe-133m	3.7×10^{-2}
Xe-135	1.8×10^{-1}
Xe-135m	1.0

TABLE 9.2-5

REACTOR COOLANT SYSTEM EQUILIBRIUM ACTIVITIES

Activation Products	μc/cc (573°F)
Mn-54	2.60×10^{-4}
Mn-56	5.60×10^{-2}
Co-58	7.80×10^{-3}
Fe-59	1.80×10^{-4}
Co-60	9.20×10^{-4}

Non-Volatile Fission Products (Continuous Full Power Operation)

	μc/cc (573°F)		μc/cc (573°F)
Br-84	2.71×10^{-2}	I-133	1.36
Rb-86	2.59	T-134	1.95×10^{-2}
Rb-89	5.99×10^{-2}	I-134	3.56×10^{-1}
Sr-89	2.80×10^{-3}	Cs-134	2.29×10^{-1}
Sr-90	1.37×10^{-4}	I-135	1.36
Y-90	2.79×10^{-4}	Cs-136	2.22×10^{-2}
Sr-91	1.32×10^{-3}	Cs-137	9.76×10^{-1}
Y-91	5.60×10^{-2}	Cs-138	4.41×10^{-2}
Mo-99	3.3	Ba-140	5.98×10^{-4}
I-131	1.75	La-140	6.22×10^{-4}
Te-132	1.85×10^{-1}	Ce-144	2.10×10^{-4}
I-132	5.99×10^{-1}	Pr-144	2.31×10^{-4}

Gaseous Fission Products	μc/cc (573°F)
Kr-85	4.42 ([illegible])
Kr-85m	1.66
Kr-87	0.83
Kr-88	2.59
Xe-133	2.09×10^{2}
Xe-135	8.84
Xe-138	1.14×10^{-1}

TABLE 9.2-6

TRITIUM PRODUCTION IN THE REACTOR COOLANT

Basic Assumptions:

Plant Parameters:

1.	Core thermal power	3216
2.	Coolant water volume	12,600 ft^3
3.	Core volume	1152.5 ft^3
4.	Core volume fraction	
	a. UO_2	.3023
	b. Zr + SS	.1035
	c. H_2O	.5942
5.	Plant full power operating times	
	a. Initial cycle	78 weeks (18 months)
	b. Equilibrium	49 weeks (11.3 months)
6.	Boron Concentrations (Peak hot full power equilibrium Xe)	
	a. Initial cycle	890 ppm
	b. Equilibrium cycle	825 ppm
7.	Burnable poison boron content (total-all rods)	18.1 kg
8.	Fraction of tritium in core (ternary fission + burnable boron) diffusing thru clad	0.30*
9.	Ternary fission yield	8×10^{-5} atoms/ fission

*The assumption that 30% of the ternary produced tritium diffuses into the coolant is based on the analysis made of fuel retention in the Saxton and the Yankee stainless clad fuel. This analysis indicated that the fuel retained 68% of the tritium produced in the fuel. Although data is not currently available on zircaloy clad fuel operating at the specific power anticipated for these reactors, it is reasonably certain that a significant portion of the tritium released by the fuel will not diffuse through the zircaloy possibly

TABLE 9.2-6 (Cont'd)

because of the formation of zirconium tritide. Shippingport data indicates that less than 1% of ternary tirtium produced is released to the coolant. Although this data cannot be used directly, it does indicate that zircaloy will reduce tritium diffusion.

10. Nuclear cross-sections and neutron fluxes

B^{10} (n, 2α) T	σ ; mb	(nv; n/cm^2-sec)
1 Mev ≤ E ≤ 5 Mev = 31.59 (Spectrum weighted)		5.04×10^{13}
E > 5 Mev = 75		7.4×10^{12}
Li^7 (n, nα) T (99.9% purity)		
3 Mev ≤ E ≤ 6 Mev = 39.1 (Spectrum weighted)		2.14×10^{13}
E > 6 Mev = 0.4		2.76×10^{12}

Li^6 (n, α) T (99.9% purity Li^7)

σ = 675 barns; nv = 2.14×10^{13} n/cm^2-sec

11. Cooling water flow: 7.5×10^5 gpm = 15×10^{14} cc/yr

II CALCULATIONS (per year)

A. Tritium from Core (Curies)	Initial Cycle	Equilibrium Cycle
1. Ternary Fission	11,450	11,450
2. B^{10} (n, 2α) T (in poison rods)	800	N.A.
3. B^{10} (n, α) Li^7 (n, nα) T (in poison rods)	1500	N.A.
4. Release fraction (0.30)		
5. Total release to Coolant	4125	3440
B. Tritium from Coolant (Curies)		
1. B^{10} (n, 2α) T	1130	780
2. Li^7 (n, nα) T (limit 2.2 ppm Li)	8.8	8.8

TABLE 9.2-6 (Cont'd)

3. Li^6 (n, α) T (purity of Li^7 = 99.9%)	8.8	8.8
4. Release Fraction (1.0)		
5. Total Release to Coolant	1147.6	797.6
C. Total Tritium in Coolant (Curies)	5273	4238

TABLE 9.2-7

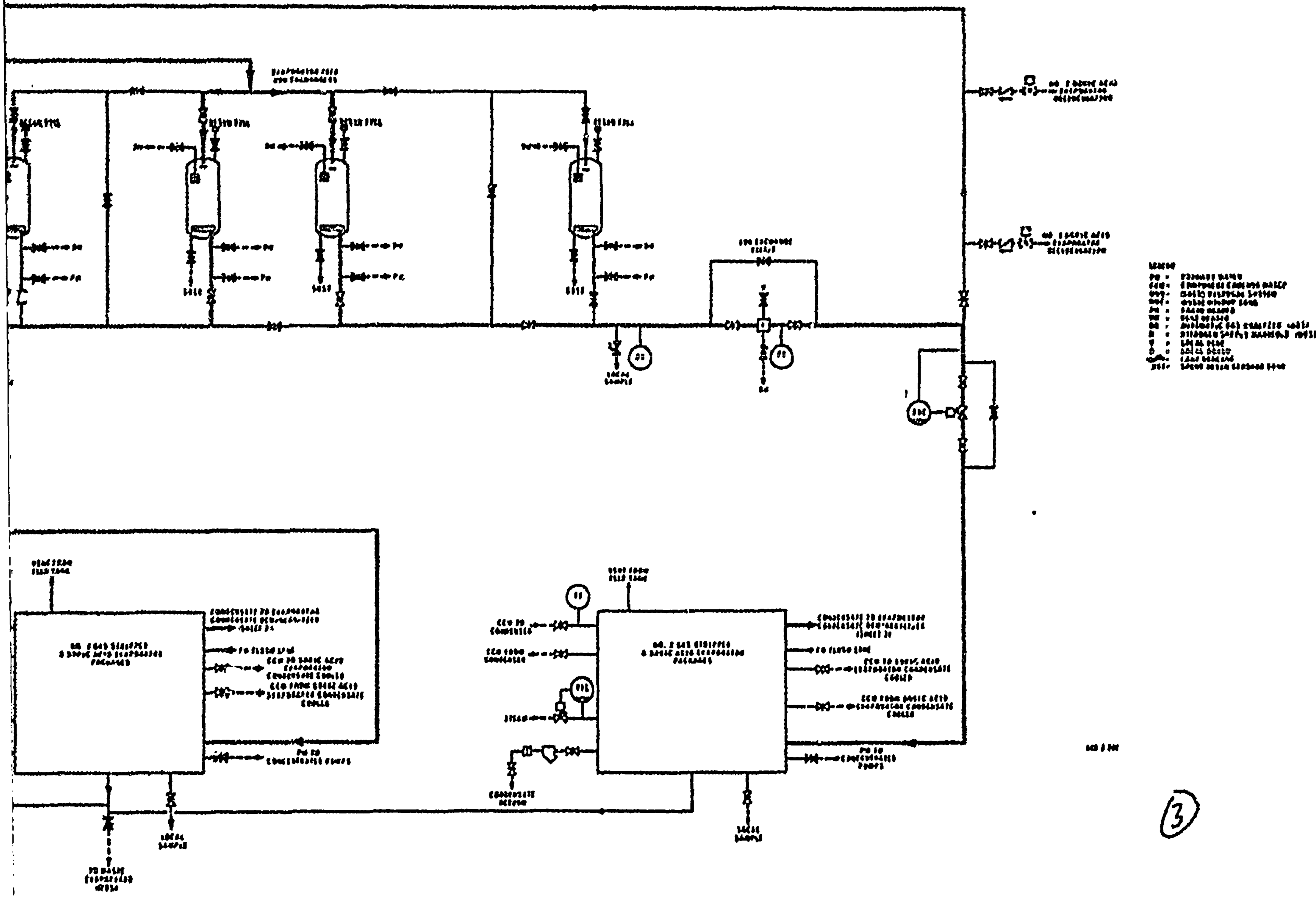
MALFUNCTION ANALYSIS OF CHEMICAL AND VOLUME CONTROL SYSTEM

	Component	Failure	Comments and Consequences
1)	Letdown line	Rupture in the line inside the reactor containment	The remote air-operated valve located near the main coolant loop is closed on low pressurizer level to prevent supplementary loss of coolant through the letdown line rupture. The containment isolation valves in the letdown line outside the reactor containment and also the orifice block valves are automatically closed by the containment isolation signal initiated by the concurrent loss-of-coolant accident. The closure of that valve prevents any leakage of the reactor containment atmosphere outside the reactor containment.
2)	Normal and alternate charging lines	See Above.	The check valves located near the main coolant loops prevent supplementary loss of coolant through the line rupture. The check valves located at the boundary of the reactor containment prevent any leakage of the reactor containment atmosphere outside the reactor containment.
3)	Seal water return line	See Above.	The motor-operated isolation valve located outside the containment is manually closed or is automatically closed by the containment isolation signal initiated by the concurrent loss-of-coolant accident. The closure of that valve prevents any leakage of the reactor containment atmosphere outside the reactor containment.

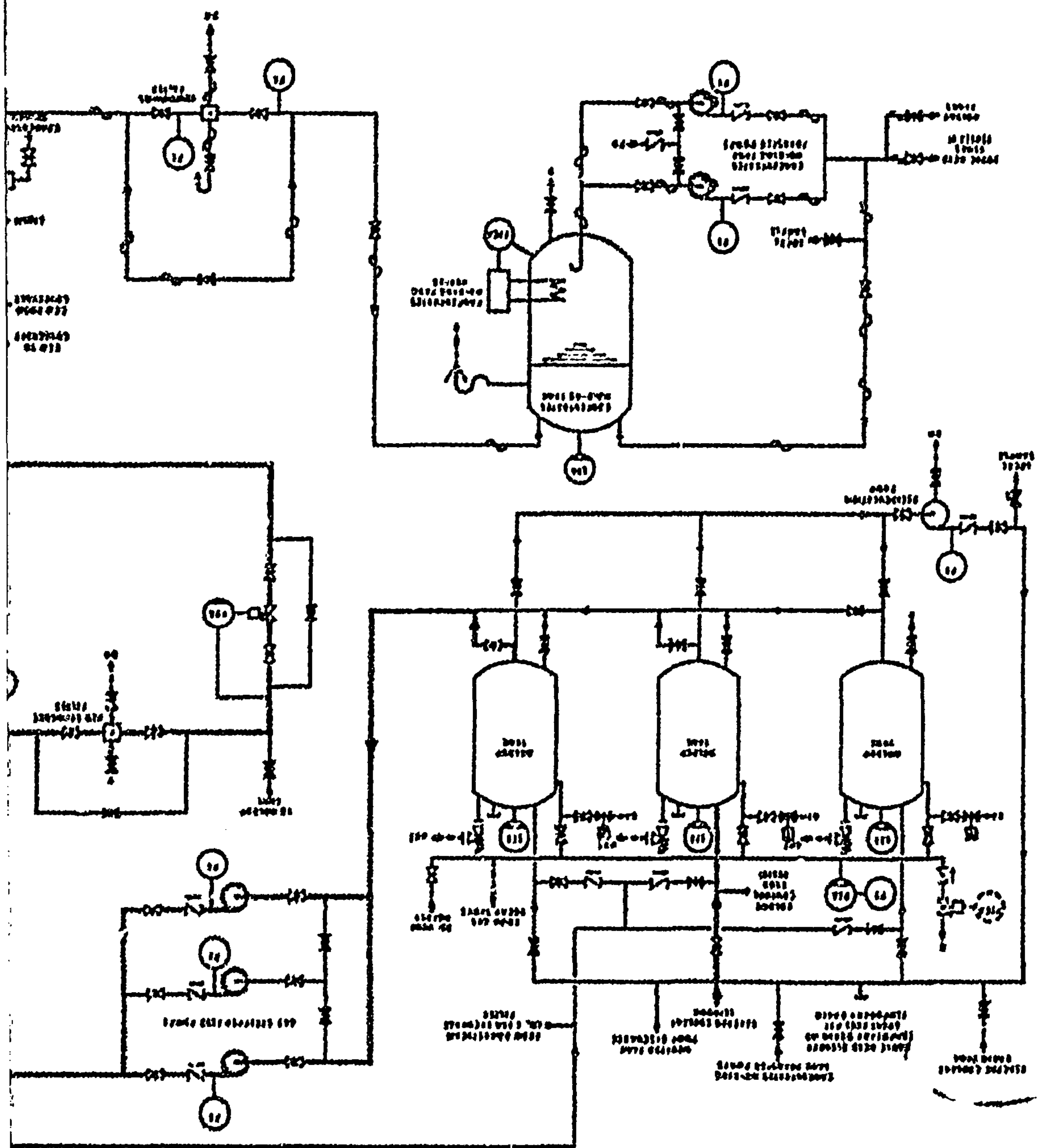
TABLE 9.2-8

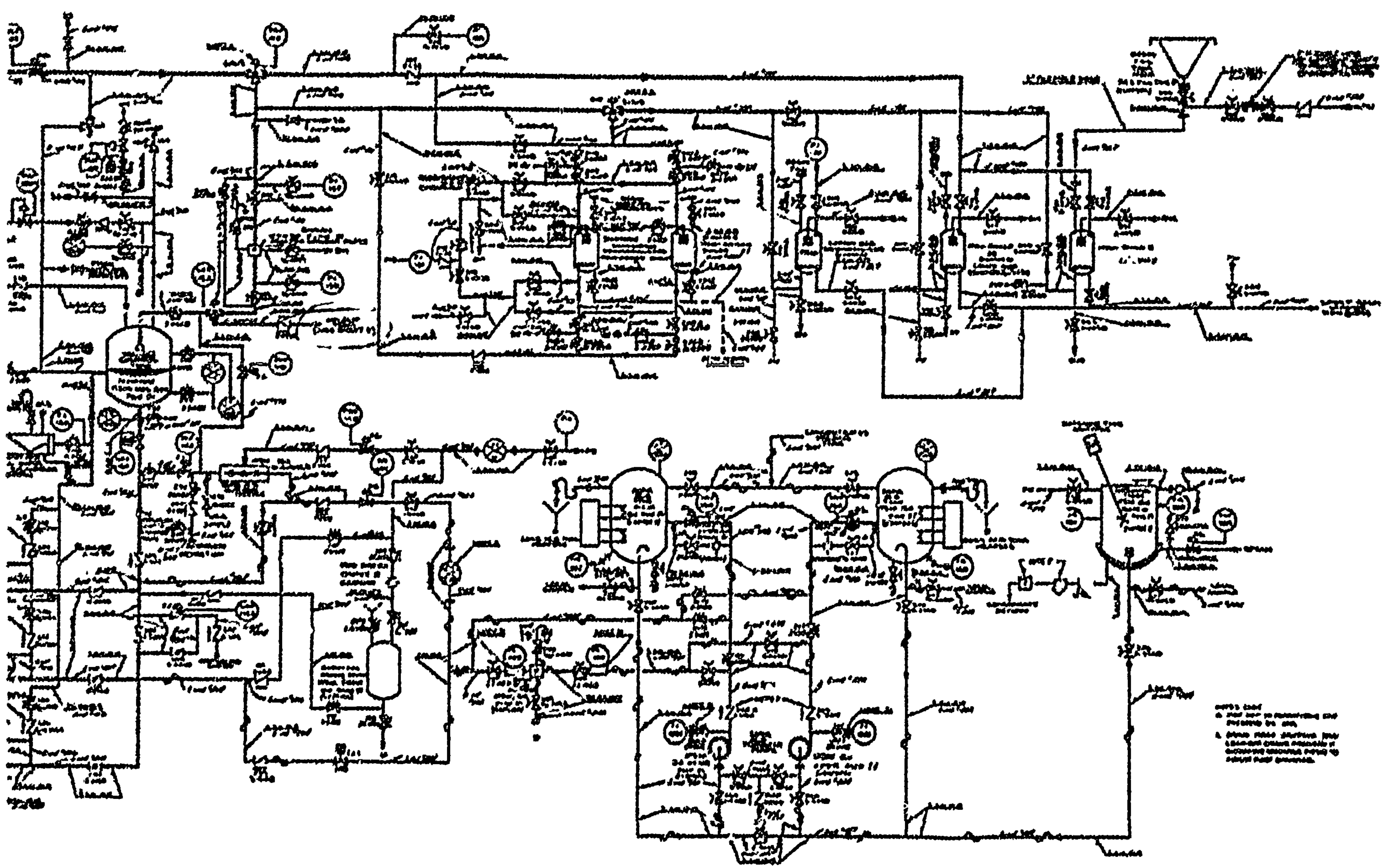
MINIMUM FREQUENCIES FOR TESTING, CALIBRATING AND/OR CHECKING INSTRUMENT CHANNELS

Channels	Action	Minimum Frequency
Charging flow	Calibrate	Each refueling
Boric acid tank level	Calibrate Check	Each refueling Each week
Volume control tank level	Calibrate Test	Each refueling 90 days
Boric acid control	Check	Each week
Valve temperature interlocks	Test	Each refueling
Pump-valve interlock	Check	Each refueling

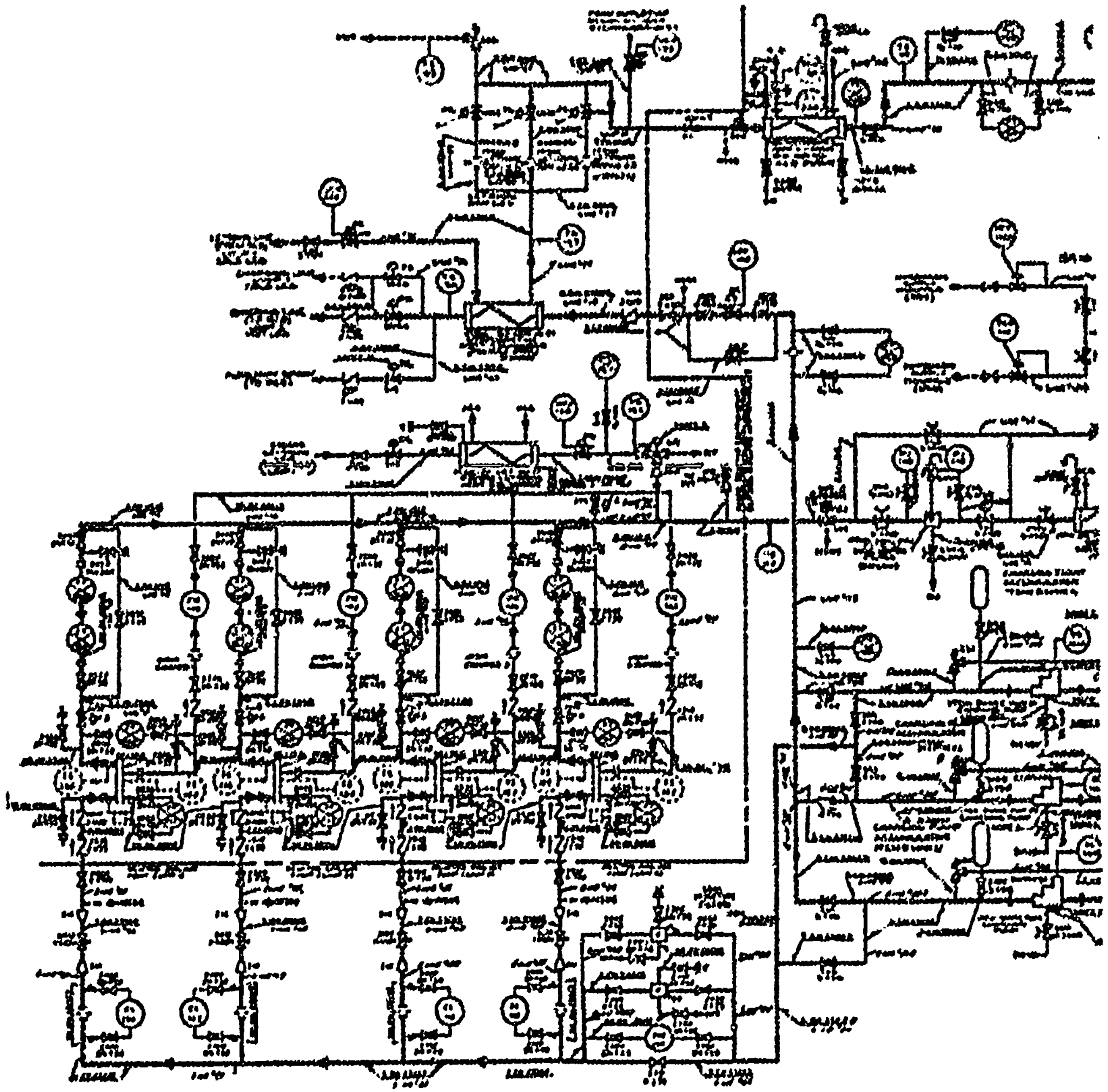


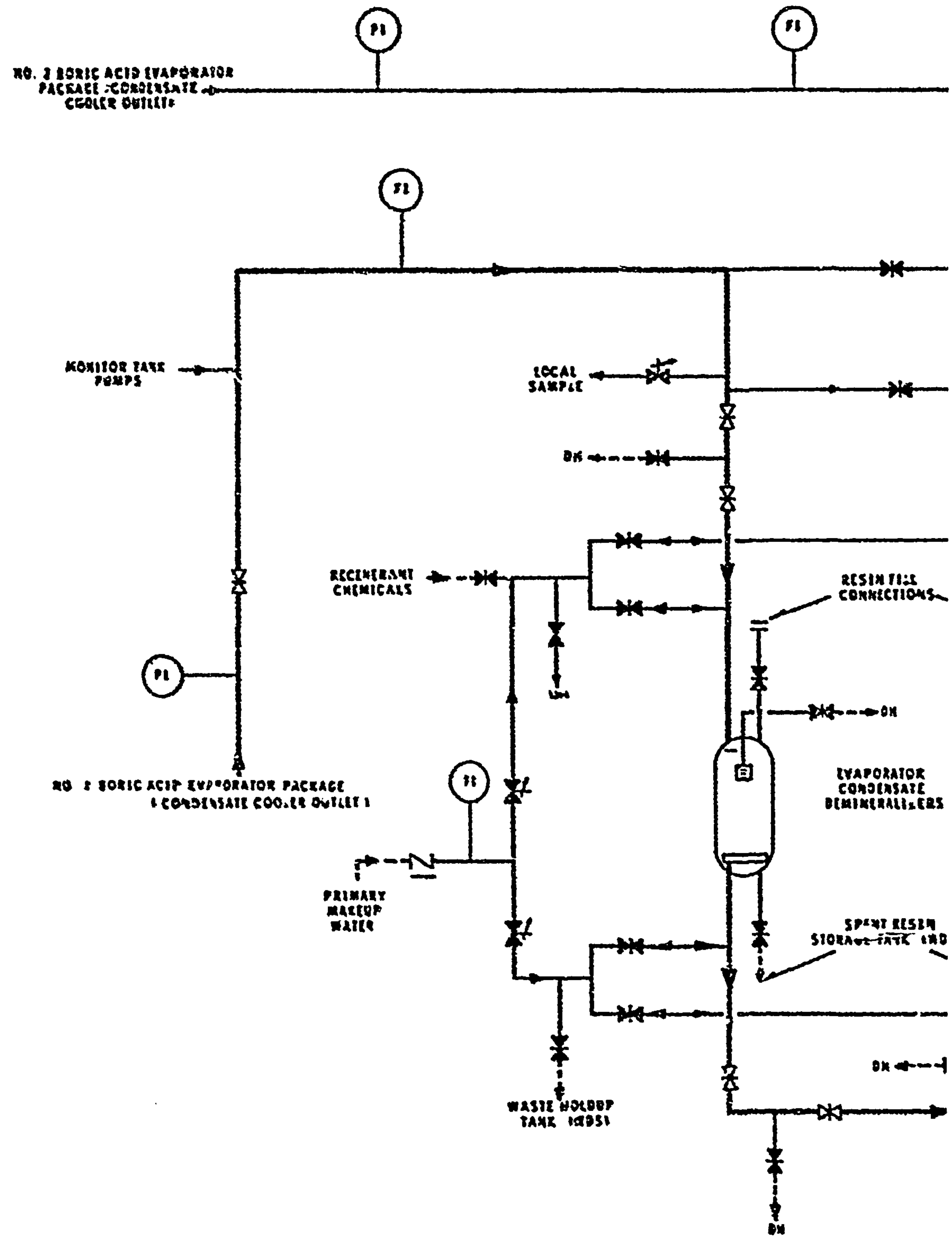
Chemical and Volume Control System – Sheet 2



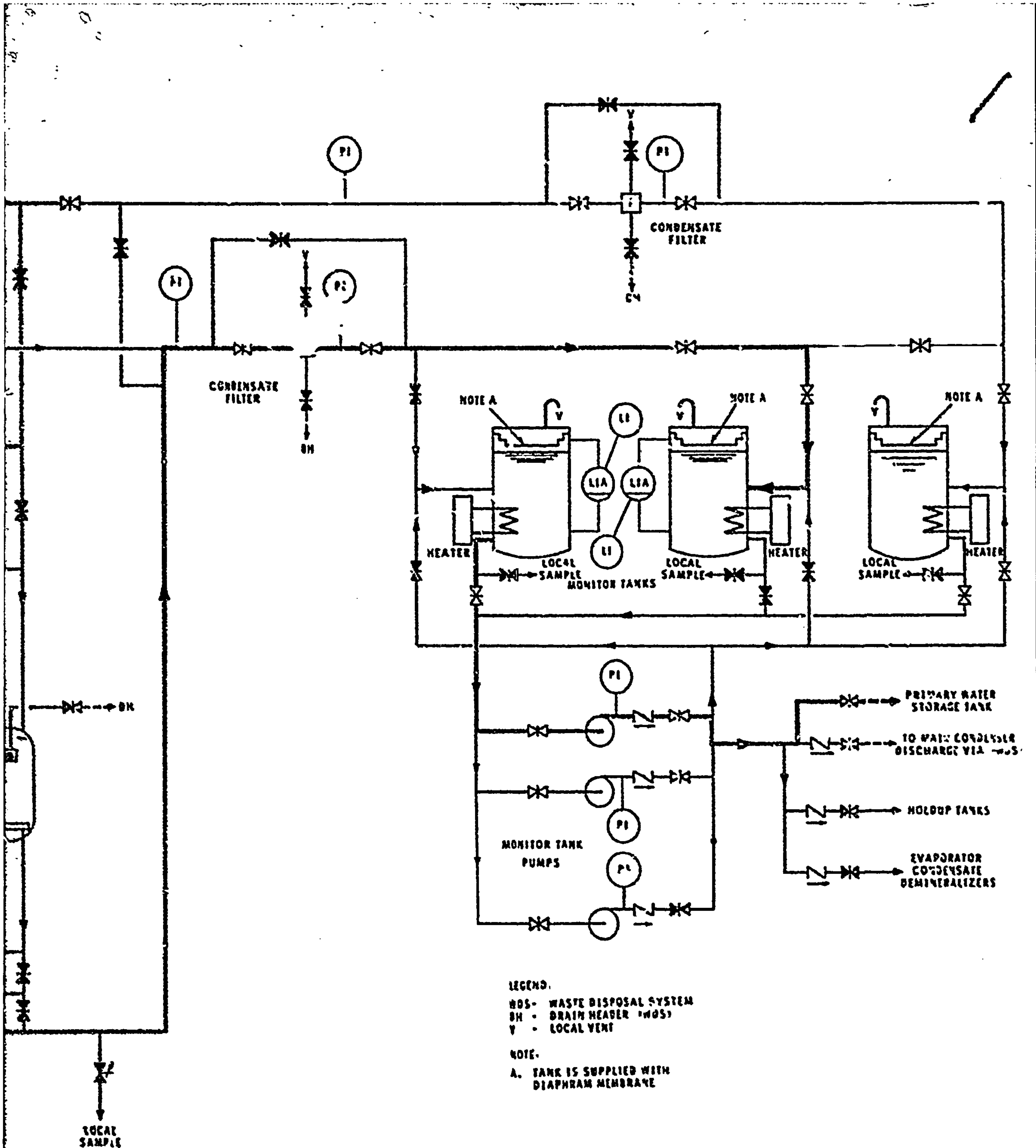


Chemical and Volume Control System - Sheet 1
Figure 9.2-1





FI
FI
NO. 2 BORIC ACID EVAPORATOR
PACKAGE (CONDENSATE
COOLER OUTLET)
FI
MONITOR TANK
PUMPS
LOCAL
SAMPLE
DR
REGENERANT
CHEMICALS
RESIN FILL
CONNECTIONS
DR
FI
NO. 2 BORIC ACID EVAPORATOR PACKAGE
(CONDENSATE COOLER OUTLET)
FI
EVAPORATOR
CONDENSATE
DEMINERALIZERS
PRIMARY
MAKEUP
WATER
SPENT RESIN
STORAGE TANK
DR
WASTE HOLDUP
TANK
DR



Chemical and Volume Control System - Sheet 3
Figure 9.2-3

9.3 AUXILIARY COOLANT SYSTEM

9.3.1 DESIGN BASIS

The Auxiliary Coolant System consists of three loops as shown in Figure 9.3-1; the component cooling loop, the residual heat removal loop, and the spent fuel pit cooling loop.

Performance Objectives

Component Cooling Loop

The component cooling loop is designed to remove residual and sensible heat from the Reactor Coolant System, via the residual heat removal loop, during plant shutdown; cool the letdown flow to the Chemical Volume and Control System during power operation; and to provide cooling to dissipate waste heat from various primary plant components.

Active loop components which are relied upon to perform the cooling function are redundant. Redundancy of components in the process cooling loop does not degrade the reliability of any system which the process loop serves.

The loop design provides for detection of radioactivity entering the loop from reactor coolant source and also provides for isolation means.

Residual Heat Removal Loop

The residual heat removal loop is designed to remove residual and sensible heat from the core and reduce the temperature of the Reactor Coolant System during the second phase of plant cooldown. During the first phase of cooldown, the temperature of the Reactor Coolant System is reduced by transferring heat from the Reactor Coolant System to the Steam and Power Conversion System.

All active loop components which are relied upon to perform their function are redundant.

The loop design provides means to detect radioactivity migration to the ultimate
13 heat sink environment and includes provisions which permit adequate action for continued core cooling when required, in the event radio-activity limits are exceeded.

The loop design precludes any significant reduction in the overall design reactor shutdown margin when the loop is brought into operation for decay heat removal or for emergency core cooling by recirculation.

The loop design includes provisions to enable periodic hydrostatic testing to applicable code test pressures.

Loop components, whose design pressure and temperature are less than the Reactor Coolant System design limits, are provided with overpressure protective devices and redundant isolation means.

Spent Fuel Pit Cooling Loop

The spent fuel pit cooling loop is designed to remove from the spent fuel pit the heat generated by stored spent fuel elements.

Loop design does not incorporate redundant active components. Alternate cooling capability can be made available under anticipated malfunctions or failures (expected fault conditions).

Loop piping is so arranged that failure of any pipeline does not drain the spent fuel pit below the top of the stored fuel elements.

The design basis for the loop provides the capability to totally unload the reactor vessel for maintenance or inspection at the time that 1/3 core already occupies the spent fuel storage pool.

Design Characteristics

Component Cooling Loop

Two pumps and one component heat exchanger are normally operated to provide cooling water for the components located in the auxiliary building and the reactor coolant pumps. The water is normally supplied to all components being cooled even though one of the components may be out of service.

Makeup water is taken from the primary water treatment plant, as required and delivered to the surge tank. A backup source of water is provided from the primary water make-up transfer pumps.

The operation of the loop is monitored with the following instrumentation:

a) A pressure detector on the line between the component cooling pumps and the component cooling heat exchangers

b) A temperature and flow indicator in the outlet line from the heat exchangers

c) A radiation monitor and temperature indicator on the main inlet line to the component cooling pumps.

Residual Heat Removal Loop

Two pumps and two residual heat exchangers perform the decay heat cooling functions for the reactor unit. After the Reactor Coolant System temperature and pressure have been reduced to 350°F and 350 psig, respectively, decay heat cooling is initiated by aligning pumps to take suction from the reactor outlet line and discharge through the heat exchangers into the reactor inlet line. If only one pump and one heat exchanger are available, reduction of reactor coolant temperature is accomplished at a lower rate.

The equipment utilized for decay heat cooling is also used for emergency core cooling during loss-of-coolant accident conditions. This is described in Section 6.

Spent Fuel Pit Cooling Loop

During normal conditions 1/3 of a core is stored in the pool. When 1/3 of a core is present, the pump and spent fuel heat exchanger will handle the load and maintain a pit water temperature less than 120°F. When 1-1/3 cores are stored, the pit is maintained below 150°F. The pool is initially filled with water from the refueling water storage tank.

Codes and Classifications

All piping and components of the Auxiliary Coolant System are designed to the applicable codes and standards listed in Table 9.3-4. The component cooling loop water contains a corrosion inhibitor to protect the carbon steel piping. Austenitic stainless steel piping is used in the spent fuel pit cooling loop, which contains borated water without corrosion inhibitor.

9.3.2 SYSTEM DESIGN AND OPERATION

Component Cooling Loop

Component cooling is provided for the following heat sources:

a) Residual heat exchangers (Auxiliary Coolant System, ACS)
b) Reactor coolant pumps (Reactor Coolant System)
c) Non regenerative heat exchanger (Chemical and Volume Control System, CVCS)
d) Excess letdown heat exchanger (CVCS)
e) Seal water heat exchanger (CVCS)
f) Boric acid recycle evaporators and condensate coolers (CVCS)
g) Sample heat exchangers (Sampling System)
h) Waste evaporator condenser (Waste Disposal System)
i) Waste gas compressors (Waste Disposal System

j) Reactor vessel support pads

k) Residual heat removal pumps (ACS)

l) Safety injection pumps (Safety Injection System, SIS)

m) Recirculation pumps

n) Spent fuel pit heat exchanger (ACS)

o) Charging pumps (CVCS)

At the reactor coolant pump, component cooling water removes heat from the bearing oil and the thermal barrier. Since the heat is transferred from the component cooling water to the service water, the component cooling loop serves as an intermediate system betwee the reactor coolant and service water cooling system. This double barrier arrangement reduces the probability of leakage of high pressure, potentially radioactive coolant to the service water system.

During normal full power operation, two component cooling pumps and one component cooling heat exchanger accomodate the heat removal loads. One standby pump provides 50% backup and a heat exchanger provides 100 per cent backup during normal operation. Three pumps and two heat exchangers are utilized to remove the residual and sensible heat during plant shutdown. If one of the pumps or one of the heat exchangers is not operative, safe shutdown of the plant is not affected, however, the time for cooldown is extended.

The surge tank accomodates expansion, contraction and in-leakage of water, and ensures a continuous component cooling water supply until a leaking cooling line can be isolated. Because the tank is normally vented to the atmosphere, a radiation monitor in the component cooling pump inlet header annunciates in the control room and closes a valve in the surge tank vent line in the unlikely event that the radiation level reaches a preset level above the normal background.

Residual Heat Removal Loop

The residual heat removal loop consists of heat exchangers, pumps, piping and the necessary valves and instrumentation. During plant shutdown, coolant flows from the Reactor Coolant System to the residual heat removal pumps, through the tube side of the residual heat exchangers and back to the Reactor Coolant System. The inlet line to the residual heat removal loop starts at the hot leg of one reactor coolant loop and the return line connects to the Safety Injection System Piping. The residual heat exchangers are used to cool the water during the latter phase of Safety Injection System operation. These duties are defined in Section 6 . The heat loads are transferred by the residual heat exchangers to the component cooling water.

During plant shutdown, the cooldown rate of the reactor coolant is controlled by regulating the flow through the tube side of the residual heat exchangers. Two remotely operated control valves downstream of the residual heat exchangers are used to control flow.

Double, remotely operated valving is provided to isolate the residual heat removal loop from the Reactor Coolant System. When Reactor Coolant System pressure exceeds the design pressure of the residual heat removal loop, an interlock between the Reactor Coolant System wide range pressure channel and the first inlet valve prevents the valve from opening. A remotely operated valve and two check valves isolate each line to the Reactor Coolant System cold legs from the residual heat removal loop.

Spent Fuel Pit Cooling Loop

The spent fuel pit cooling loop removes residual heat from fuel stored in the spent fuel pit. The loop is normally required to handle the heat load from 1/3 of the core freshly discharged from the reactor but it can safely accomodate the heat load from 1-1/3 cores, for which there is storage space available. The spent fuel is placed in the pit during refueling and is stored until it is shipped to a reprocessing facility.

The spent fuel pit is located outside the reactor containment and is not affected by any loss-of-coolant accident in the containment. The water in the pit is connected during refueling to that in the refueling canal by a valve. Only a very small amount of interchange of water occurs as fuel assemblies are transferred.

The spent fuel pit cooling loop consists of a pump, heat exchanger, filter, demineralizer, piping and associated valves and instrumentation. The pump draws water from the pit, circulates it through the heat exchanger and returns it to the pit. A second pump is used to circulate refueling water through the demineralizer and filter for purification. Component cooling water cools the heat exchanger. Redundancy of this equipment is not required because of the large heat capacity of the pit and the slow heat up rate as shown in Table 9.3-3. However, in the event of failure of the spent fuel pump, alternate connections are provided for connecting a temporary pump to the spent fuel pit loop. This consists of blind flange connections in the suction and discharge piping.

The clarity and purity of the spent fuel pit water is maintained by passing approximately 5 per cent of the loop flow through a filter and demineralizer. The spent fuel pit pump suction line, which is used to drain the pit, penetrates the spent fuel pit wall above the fuel assemblies. The penetration location prevents loss of water as a result of a possible suction line rupture.

Component Cooling Loop Components

Component Cooling Heat Exchangers

The two component cooling heat exchangers are of the shell and straight tube type. Service water circulates through the tubes while component cooling water circulates through the shell side. Parameters are presented in Table 9.3-1.

Component Cooling Pumps

The three component cooling pumps which circulate component cooling water through the component cooling loop are horizontal, centrifugal units. The pump casings are made from cast iron (ASTM 48) based on the corrosion-erosion resistance and the ability to obtain sound castings. The material thickness is dictated by high quality casting practice and ability to withstand mechanical damage and as such are substantially overdesigned from a stress level standpoint. Parameters are presented in Table 9.3-1.

Auxiliary Component Cooling Pumps

Two auxiliary component cooling pumps are started during the injection phase to protect the recirculation pump motors from the containment atmosphere. An auxiliary component cooling pump is connected to the motor shaft of each safety injection pump to cool the safety injection pump bearings. The auxiliary component cooling pumps are discussed in further detail in Section 6.2.

Component Cooling Surge Tank

The component cooling surge tank which accommodates changes in component cooling water volume is constructed of carbon steel. Parameters are presented in Table 9.3-1. In addition to piping connections, the tank has a flanged opening at the top for the addition of the chemical corrosion inhibitor to the component cooling loop.

Component Cooling Valves

The valves used in the component cooling loop are standard commercial valves constructed of carbon steel with bronze or stainless steel trim. Since the component cooling water is not normally radioactive, special features to prevent leakage to the atmosphere are not provided.

Self-actuated spring loaded relief valves are provided for lines and components that could be pressurized to their design pressure by improper operation or malfunction.

Component Cooling Piping

All component cooling loop piping is carbon steel with welded joints and connections except at components which might need to be removed for maintenance.

Residual Heat Removal Loop Components

Residual Heat Exchangers

The two residual heat exchangers located within the containment are of the shell and U-tube type with the tubes welded to the tube sheet. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell side. The tubes and other surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

Residual Heat Removal Pumps

The two residual heat removal pumps are vertical, centrifugal units with special seals to prevent reactor coolant leakage to the atmosphere. All pump parts in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

Residual Heat Removal Valves

The valves used in the residual heat removal loop are constructed of austenitic stainless steel or equivalent corrosion resistant material.

Stop valves are provided to isolate equipment for maintenance. Throttle valves are provided for remote and manual control of the residual heat exchanger tube side flow. Check valves prevent reverse flow through the residual heat removal pumps.

Two remotely operated series stop valves at the inlet with a pressure interlock isolate the residual heat removal loop from the Reactor Coolant System. The residual heat removal loop is isolated from the Reactor Coolant System by two series check valves and a remotely operated stop valve on the outlet line. Overpressure in the residual heat removal loop is relieved through a check valve to the low pressure letdown stream in the Chemical and Volume Control System.

Valves that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to the Waste Disposal System.

Manually operated valves have backseats to facilitate repacking and to limit the stem leakage when the valves are open.

Residual Heat Removal Piping

All residual heat removal loop piping is austenitic stainless steel. The piping is welded with flanged connections at the pumps.

Spent Fuel Pit Loop Components

Spent Fuel Pit Heat Exchanger

The spent fuel pit heat exchanger is of the shell and U-tube type with the tubes welded to the tube sheet. Component cooling water circulates through the shell, and spent fuel pit water circulates through the tubes. The tubes are austenitic stainless steel and the shell is carbon steel.

Spent Fuel Pit Pump

The spent fuel pit pump circulates water in the spent fuel pit cooling loop. All wetted surfaces of the pump are austenitic stainless steel, or equivalent corrosion resistant material. The pump is operated manually from a local station.

Refueling Water Purification Pump

The refueling water purification pump circulates water in a loop between the refueling water storage tank and the spent fuel pit demineralizer and filter. All wetted surfaces of the pump are austenitic stainless steel. The pump is operated manually from a local station.

Spent Fuel Pit Filter

The spent fuel pit filter removes particulate matter larger than 5 microns from the spent fuel pit water. The filter cartridge is synthetic fiber and the vessel shell is austenitic stainless steel.

Spent Fuel Pit Strainer

A stainless steel strainer is located at the inlet of the spent fuel pit loop suction line for removal of relatively large particles which might otherwise clog the spent fuel pit demineralizer.

Spent Fuel Pit Demineralizer

The demineralizer is sized to pass 5% of the loop circulation flow, to provide adequate purification of the fuel pit water for unrestricted access to the working area, and to maintain optical clarity.

Spent Fuel Pit Skimmer

A skimmer pump, strainer and filter are provided for surface skimming of the spent fuel pit water.

Spent Fuel Pit Valves

Manual stop valves are used to isolate equipment and lines, and manual throttle valves provide flow control. Valves in contact with spent fuel pit water are austenitic stainless steel or equivalent corrosion resistant material.

Spent Fuel Pit Piping

All piping in contact with spent fuel pit water is austenitic stainless steel. The piping is welded except where flanged connections are used at the pump, heat exchanger and filter to facilitate maintenance.

9.3.3 SYSTEM EVALUATION

Availability And Reliability

Component Cooling Loop

For component cooling of the reactor coolant pumps, the excess letdown heat exchanger and the residual heat exchangers inside the containment, most of the piping, valves, and instrumentation are located outside the primary system concrete shield at an elevation above the water level in the bottom of the containment at post-accident conditions. (The exceptions are the cooling lines for the reactor coolant pumps and reactor supports, which can be secured following the accident.) In this location the systems in the containment are protected against credible missiles and from being flooded during post-accident operations. Also, this location provides shielding which allows for maintenance and inspections to be performed during power operation.

Outside the containment, the residual heat removal pumps, the spent fuel heat exchanger, the component cooling pumps and heat exchangers and associated valves, piping and instrumentation are maintainable and inspectable during power operation. Replacement of one pump or one heat exchanger is practicable while the other units are in service.

Several of the components in the component cooling loop are fabricated from carbon steel. The component cooling water contains a corrosion inhibitor to protect the carbon steel. Welded joints and connections are used except where flanged closures are employed to facilitate maintenance. At least 10% of the com-
14 ponent cooling line welds inside the containment are 100% radiographed. The entire

system is seismic Class I and is housed in structures of the same classification. The components are designed to the codes given in Table 9.3-4. In addition the components are not subjected to any high pressures (See Table 9.3-1) or stresses. Hence a rupture or failure of the system is very unlikely.

During the recirculation phase following a loss-of-coolant accident, one of the three component cooling water pumps is required to deliver flow to the shell side of one of the residual heat exchangers.

Residual Heat Removal Loop

Two pumps and two heat exchangers are utilized to remove residual and sensible heat during plant cooldown. If one of the pumps and/or one of the heat exchangers is not operative, safe operation or safe cooldown of the plant is not affected, however, the time for cooldown is extended. The function of this equipment following a loss-of-coolant accident is discussed in Section 6.

Spent Fuel Pit Cooling Loop

This manually controlled loop may be shutdown safely for reasonable time periods, as shown in Table 9.3-3, for maintenance or replacement of malfunctioning components.

Leakage Provisions

Component Cooling Loop

Water leakage from piping, valves, and equipment in the system inside the containment is not considered to be generally detrimental unless the leakage exceeds the makeup capability. With respect to water leakage from piping, valves, and equipment outside the containment, welded construction is used where possible to minimize the possibility of leakage. The component cooling water could become contaminated with radioactive water due to a leak in any heat exchanger tube in the Chemical and Volume Control, the Sampling, or the Auxiliary Coolant Systems, or a leak in the cooling coil for the mechanical seal on a reactor coolant pump.

Tube or coil leaks in components being cooled would be detected during normal plant operation by the leak detection system described in Sections 4.2.7 and 6.7. Such leaks are also detected anytime by a radiation monitor located on the main return cooling line.

Leakage from the component cooling loop can be detected by a falling level in the component cooling surge tank. The rate of water level fall and the area of the water surface in the tank permit determination of the leakage rate. To assure accurate determinations, the operator would check that temperatures are stable.

The component which is leaking can be located by sequential isolation or inspection of equipment in the loop. If the leak is in the on-line component cooling water heat exchanger, the standby exchanger would be put in line and the leaking exchanger isolated and repaired. Overall leakage within the containment is limited to the value given in the Technical Specifications.

Should a large tube side to shell side leak develop in a residual heat exchanger, the water level in the component cooling surge tank would rise, and the operator would be alerted by a high water alarm. The atmospheric vent on the tank is automatically closed in the event of high radiation level at the component cooling water pump suction header. If the leaking residual heat exchanger is not isolated from the component cooling loop before the inflow completely fills the surge tank, the relief valve on the surge tank lifts. The discharge of this relief valve is routed to the auxiliary building waste holdup tank.

The severence of a cooling line serving an individual reactor coolant pump cooler would result in substantial leakage of component cooling water. However, the piping is small as compared to piping located in the missile protected area of the containment. Therefore, the water stored in the surge tank after a low level alarm together with makeup flow provides ample time for the closure of the valves external to the containment to isolate the leak before cooling is lost to the essential components in the component cooling loop.

The relief valves on the component cooling water lines downstream from each reactor coolant pump are designed with a capacity equal to the maximum rate at which reactor coolant can enter the component cooling loop from a severence type break of the reactor coolant pump thermal barrier cooling coil. The valve set pressure equals the design pressure of the component cooling piping.

The relief valves on the cooling water lines downstream from the sample, excess letdown, seal water, non-regenerative, spent fuel pit and residual heat exchangers are sized to relieve the volumetric expansion occurring if the exchanger shell side is isolated when cool, and high temperature coolant flows through the tube side. The set pressure equals the design pressure of the shell side of the heat exchangers.

The relief valve on the component cooling surge tank is sized to relieve the maximum flow rate of water which enters the surge tank following a rupture of a reactor coolant pump thermal barrier cooling coil. The set pressure equals the design pressure of the component cooling surge tank.

Residual Heat Removal Loop

During reactor operation all equipment of the residual heat removal loop is idle, and the associated isolation valves are closed. During the loss-of-coolant accident condition, water from the recirculation sump is recirculated through a loop inside the containment using the recirculation sumps and the residual heat exchangers. The residual heat removal pumps (which are located outside of the containment) serve as backup to the internal recirculation pumps.

Each of the two residual heat removal pumps is located in a shielded compartment with a floor drain. Piping conveys the drain water to an external [illegible] which is capable of handling the flow which would result from the [illegible]e of a residual heat removal pump seal. A 50 gpm sump pump discharges [illegible] the waste hold-up tank.

Spent Fuel Pit Cooling Loop

Whenever a leaking fuel assembly is transferred from the fuel transfer canal to the spent fuel storage pool, a small quantity of fission products may enter the spent fuel cooling water. A small purification loop is provided for removing these fission products and other contaminants from the water.

The probability of inadvertently draining the water from the cooling loop of the spent fuel pit is exceedingly low. The only means through such actions as opening a valve on the cooling line and leaving it open when the pump is operating. In the unlikely event of the cooling loop of the spent fuel pit being drained, the spent fuel storage pit itself cannot be drained and no spent fuel is uncovered since the spent fuel pit cooling connections enter near the top of the pit. With no heat removal the time for the spent fuel pit water to rise from 120°F to 180°F with 1/3 core stored in the pit is approximately 13.8 hours. The temperature and level indicators in the spent fuel pit would warn the operator of the loss of cooling. This slow heatup rate of the spent fuel pit would allow sufficient time to take any necessary action to provide adequate cooling while the cooling capability of the spent fuel pit cooling loop is being restored.

Assuming that the reactor had recently been refueled and one core region was placed in [illegible] uel pit, a fisson product decay period of approximately 47.5 days woul[illegible] ed after the spent fuel was placed in the pit before the natural hea[illegible] om the pit would be equivalent to the decay heat.

Incident Control

Component Cooling Loop

In the unlikely event of a pipe severance in the component cooling loop, backup is provided for post accident heat removal by the fan coolers.

Should the break occur outside the containment the leak could either be isolated by valving or the broken line could be repaired, depending on the location in the loop at which the break occured.

Once the leak is isolated or the break has been repaired, makeup water is supplied from the reactor makeup water tank by one of the primary makeup water pumps. If the loop drains completely before the leakage is stopped, it can be refilled by a primary makeup water pump in less than two hours.

If the break occurs inside the containment on a cooling water line to a reactor coolant pump, the leak can be isolated. Each of the cooling water supply lines to the reactor coolant pumps contains a check valve inside and remotely operated valves outside the containment wall. Each return line has remote operated valves outside the containment wall. The cooling water supply line to the excess letdown heat exchanger contains a check valve inside the containment wall and both supply and return lines have automatically isolated valves outside the containment wall which are closed during normal operation.

Should the break occur inside the containment and the leak can not be isolated the residual heat removal pumps and safety injection pumps, if required, are employed to recirculate uncooled spilled water to the core. Heat is removed from the core by boil off of the water to the containment with the fan coolers being used to condense the resulting steam. Connections are provided on the residual heat removal and safety injection pump cooling water lines so that an alternate source of cooling water can be provided.

Flow indication is provided on the component cooling return lines from the safety injection and residual heat removal pumps. Each of the component cooling supply lines to the residual heat exchangers has a noramlly closed remotely operated valve. If one of the valves fails to open at initiation of long-term recirculation, the valve which does open supplies a heat exchanger with sufficient cooling to remove the heat load.

That portion of the loop located outside the containment is considered to be a part of the reactor building isolation barrier.

Except for the normally closed makeup line and equipment vent and drain lines, there are no direct connections between the cooling water and other systems. The equipment vent and drain lines outside the containment have manual valves which are normally closed unless the equipment is being vented or drained for maintenance or repair operations.

Residual Heat Removal Loop

The residual heat removal loop is connected to the reactor outlet line on the suction side and to the reactor inlet line on the discharge side. On the suction side the connection is through two electric motor-operated gate valves in series with the first valve interlocked with reactor coolant system pressure. On the discharge side the connection is through two check valves in series with an electric motor operated gate valve. All of these are closed whenever the reactor is in the operating condition.

Spent Fuel Pit Cooling Loop

The most serious failure of this loop is complete loss-of-water in the storage pool. To protect against this possibility, the spent fuel storage pool cooling connections enter near the water level so that the pool cannot be either gravity-drained or inadvertently drained. For this same reason care is also exercised in the design and installation of the fuel transfer tube. The water in the spnet fuel pit below the cooling loop connections could be removed by using a portable pump.

Malfunction Analysis

A failure analysis of pumps, heat exchangers and valves is presented in Table 9.3-5.

9.3.4 MINIMUM OPERATING CONDITIONS

Minimum operating conditions for the Auxiliary Coolant System will be specified in the Technical Specifications.

9.3.5 TESTS AND INSPECTIONS

The residual heat removal pumps flow instrument channels are calibrated during each refueling operation.

The active components of the Auxiliary Coolant System are in either continuous or intermittent use during normal plant operation and no additional periodic tests are required. Periodic visual inspections and preventative maintenance are conducted following normal industrial practice.

Samples are analyzed to determine the amount of radioactivity in the reactor coolant. If the radioactivity level is high a reactor coolant sample is analyzed and an iodine extraction made and counted as an indication of defects in fuel cladding. The frequency of sampling for gross activity and for radiochemical analysis of the reactor coolant, as specified in Section 15, Technical Specifications, will be adequate to detect fuel clad defects to support plant operation as based on past experience.

TABLE 9.3-1

COMPONENT COOLING

LOOP COMPONENT DATA

Component Cooling Pumps

Quantity	3
Type	Horizontal Centrifugal
Rated capacity (each), gpm	3600
Rated head, ft H_2O	220
Motor horsepower, hp	250
Material	Cast iron
Design pressure, psig	150
Design temperature, °F	200

Component Cooling Heat Exchangers

Quantity	2
Type	Shell and straight tube
Design heat transfer, Btu/hr	31.4×10^6
Shell Side (component cooling water) –	
Operating inlet temp., °F	100.1
Operating outlet temp., °F	88.2
Design flow rate, lb/hr	2.66×10^6
Design temperature, °F	200
Design pressure, psig	150
Material	Carbon steel
Tube side (service water) –	
Operating outlet temperature, °F	75
Operating outlet temperature, °F	81.9
Design flow rate, lb/hr	4.55×10^6
Design pressure, psig	150
Design temperature, °F	200
Material	Admiralty

TABLE 9.3-1 (Continued)

Component Cooling Surge Tank

Quantity	1
Volume, gal.	2000
Normal water volume, gal.	1000
Design pressure, psig	100
Design temperature, °F	200
Construction material	Carbon steel

Auxiliary Component Cooling Pumps

Quantity	2
Type	Vertical centrifugal
Rated capacity, gpm	80
Rated head, ft H_2O	100
Motor horsepower, hp	
Casing material	Cast iron
Design pressure, psig	150
Design temperature, °F	200

Component Cooling Loop Piping and Valves

Design pressure, psig	150
Design temperature, °F	200

TABLE 9.3-2

RESIDUAL HEAT REMOVAL LOOP COMPONENT DATA

Reactor coolant temperature at startup of residual heat removal, °F	350
Time to cool reactor coolant system from 350°F to 140°F, hr (all equipment operational)	16
Refueling water storage temperature, °F	Ambient
Decay heat generation at 20 hrs after shutdown condition Btu/hr	61.6×10^6
Reactor cavity fill time, hr	1
Reactor cavity drain time, hr	4
H_3BO_3 concentration in refueling water storage tanks, ppm boron	2000-5000
Residual Heat Removal Pumps	
Quantity	2
Type	Vertical Centrifugal
Rated capacity (each), gpm	3000
Rated head, ft H_2O	350
Motor horsepower, hp	400
Material	SS
Design pressure, psig	600
Design temperature, °F	400
Residual Heat Exchangers	
Quantity	2
Type	Shell and U-tube
Design Heat Transfer (each), Btu/hr	30.8×10^6
Shell side (component cooling water)	
Operating inlet temp., °F	83.12
Operating outlet temp., °F	100.8
Design flow rate, lb/hr	2.46×10^6
Design temperature, °F	200
Design pressure, psig	150
Material	Carbon steel

TABLE 9.3-2 (Continued)

Tube side (Reactor coolant)	
Operating inlet temp., °F	135
Operating outlet temp.	113.5
Design flow rate, lb/hr	1.44×10^6
Design temperature, °F	400
Design pressure, psig	600
Material	Stainless steel
Residual Heat Removal Loop Piping and Valves	
A. Isolated loop	
Design pressure, psig	600
Design temperature, °F	400
B. Loop isolation	
Design pressure, psig	2485
Design temperature, °F	650

TABLE 9.3-3

SPENT FUEL COOLING LOOP COMPONENT DATA

System Cooling Capacity, Btu/hr	
Normal (1/3 core)	9.99×10^6
Maximum (1-1/3 cores)	22.94×10^6
Pit Water Heat Inertia, No Heat Removal	
Time to heat from 120 to 180°F, 1/3 core, hr	13.8
Time to heat from 142[1] to 180°F, 1-1/3 core, hr	2.9
Spent Fuel Pit Heat Exchanger	
Quantity	1
Type	Shell and U-tube
Design heat transfer, Btu/hr[(2)]	11.15×10^6
Shell side (Component cooling water)	
Operating inlet temp., °F	91.3
Operating outlet, temp., °F	101.5
Design flow rate, lb/hr	1.1×10^6
Design temperature, °F	200
Design pressure, psig	150
Material	Carbon steel
Tube side (spent fuel pit water)	
Operating inlet temp., °F	120
Operating outlet temp., °F	109.9
Design flow rate, lb/hr	1.1×10^6
Design temperature, °F	200
Design pressure, psig	150
Material	Stainless steel

Note 1: The temperature of equilibrium with cooling system operating

Note 2: Assumed pool water to heat exchanger at 120°F and cooling water to heat exchanger at 91.3°F.

TABLE 9.3-3 (Continued)

Maximum differential pressure across filter at rated flow (clean), psi	5
Maximum differential pressure across filter prior to replacement, psi	20
Filtration requirement	98% retention of particles above 5 microns
Spent Fuel Pit Skimmer Pump	
Quantity	1
Type	Horizontal Centrifugal
Rated capacity, gpm	100
Rated head, ft H_2O	50
Design pressure, psig	50
Design temperature, °F	200
Material	Stainless steel
Refueling Water Purification Pump	
Quantity	1
Type	Horizontal Centrifugal
Rated capacity, gpm	100
Rated head, ft H_2O	150
Design pressure, psig	150
Design temperature, °F	200
Material	Stainless steel
Spent Fuel Pit Cooling Loop Piping and Valves	
Design pressure, psig	150
Design temperature, °F	200
Spent Fuel Pit Skimmer Loop Piping and Valves	
Design pressure, psig	100
Design temperature, °F	200
Refueling Water Purification Loop Piping and Valves	
Design pressure, psig	150
Design temperature, °F	200

TABLE 9.3-3 (Continued)

Spent Fuel Pit Pump	
Quantity	1
Type	Horizontal Centrifugal
Material	Stainless steel
Rated capacity, gpm	2300
Rated head, ft H_2O	125
Motor horsepower	100
Design pressure, psig	150
Design temperature, °F	200
Spent Fuel Storage Pool	
Volume, ft^3	37,300
Boron concentration, ppm boron	2000 to 2500
Spent Fuel Pit Filter	
Quantity	1
Internal design pressure of housing, psig	200
Design temperature, °F	250
Rated flow, gpm	100
Maximum differential pressure across filter element at rated flow (clean cartridge), psi	5
Maximum differential pressure across the filter element prior to removing, psi	20
Filtration requirement	98% retention of particles down to 5 micron
Spent Fuel Pit Strainer	
Quantity	1
Rated flow, gpm	2300
Maximum differential pressure across the strainer element at rated flow (clean), psi	1
Perforation	Approx. 0.2"

TABLE 9.3-3 (Continued)

Spent Fuel Pit Demineralizer	
Quantity	2
Type	Flushable
Design pressure, psig	200
Design temperature, °F	250
Flow rate, gpm	100
Resin volume, cu. ft.	30
Spent Fuel Pit Skimmers	
Quantity	2
Flow per unit, gpm	50
Vertical fluctuation range:	
Floating, inch	4
Manual adjustment, feet	2
Spent Fuel Pit Skimmer Strainer	
Quantity	1
Type	Basket
Rated flow, gpm	100
Design pressure, psig	50
Design temperature, °F	200
Maximum differential pressure across strainer at rated flow, psi	1
Perforation, inch	1/8
Spent Fuel Pit Skimmer Filter	
Quantity	1
Type	Replaceable
Internal design pressure, psig	200
Design temperature, °F	250
Rated flow, gpm	100

TABLE 9.3-4

AUXILIARY COOLANT SYSTEM

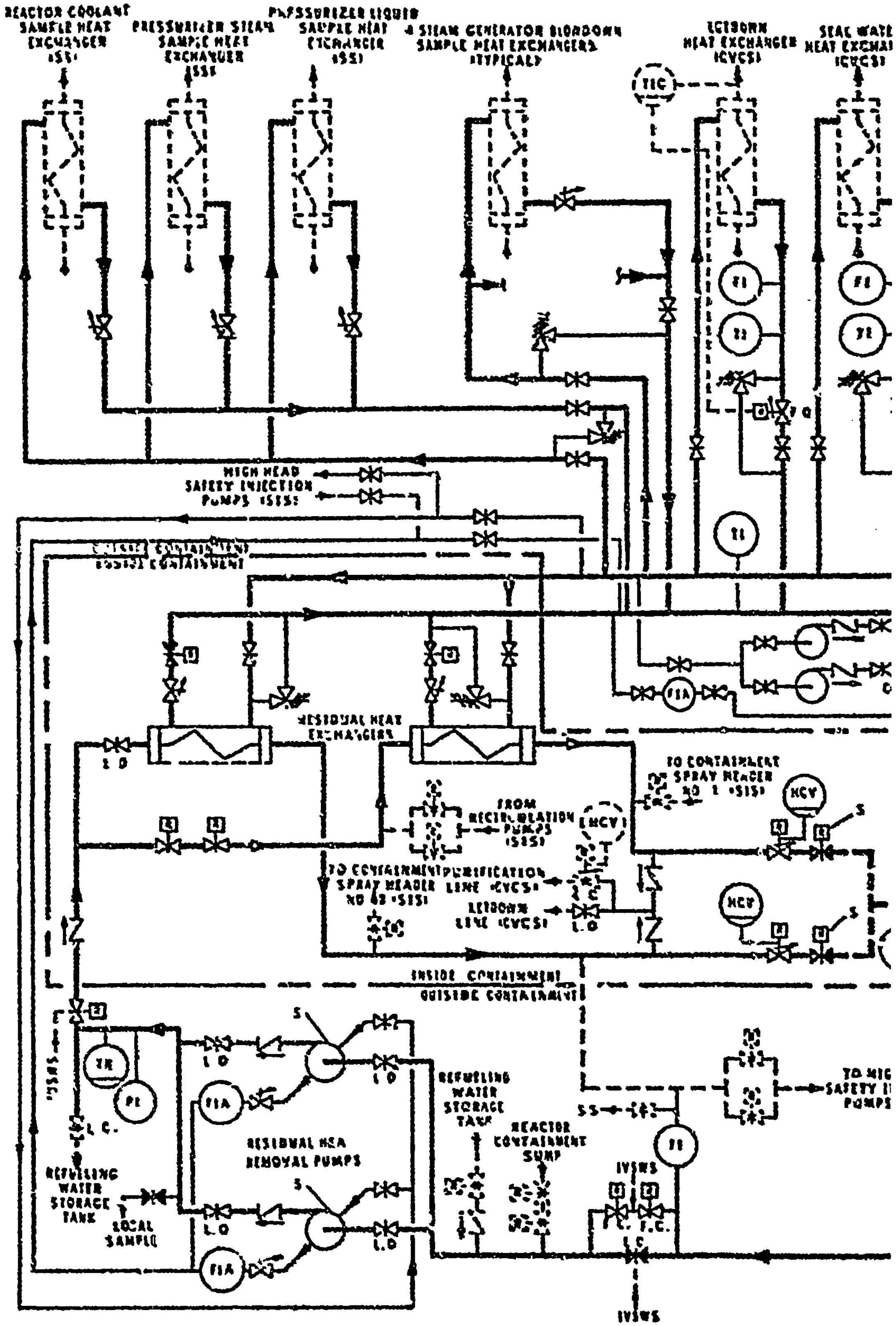
CODE REQUIREMENTS

Component cooling heat exchangers	ASME VIII
Component cooling surge tank	ASME VIII
Component cooling loop piping and valves	USAS B31.1
Residual heat exchangers	ASME III, Class C, tube side
	ASME VIII, shell side
Residual heat removal piping and valves	USAS B13.1
Spent fuel pit filter	ASME III, Class C
Spent fuel heat exchanger	ASME III, Class C, tube side
	ASME VIII, shell side
Spent fuel pit loop piping and valves	USAS B31.1

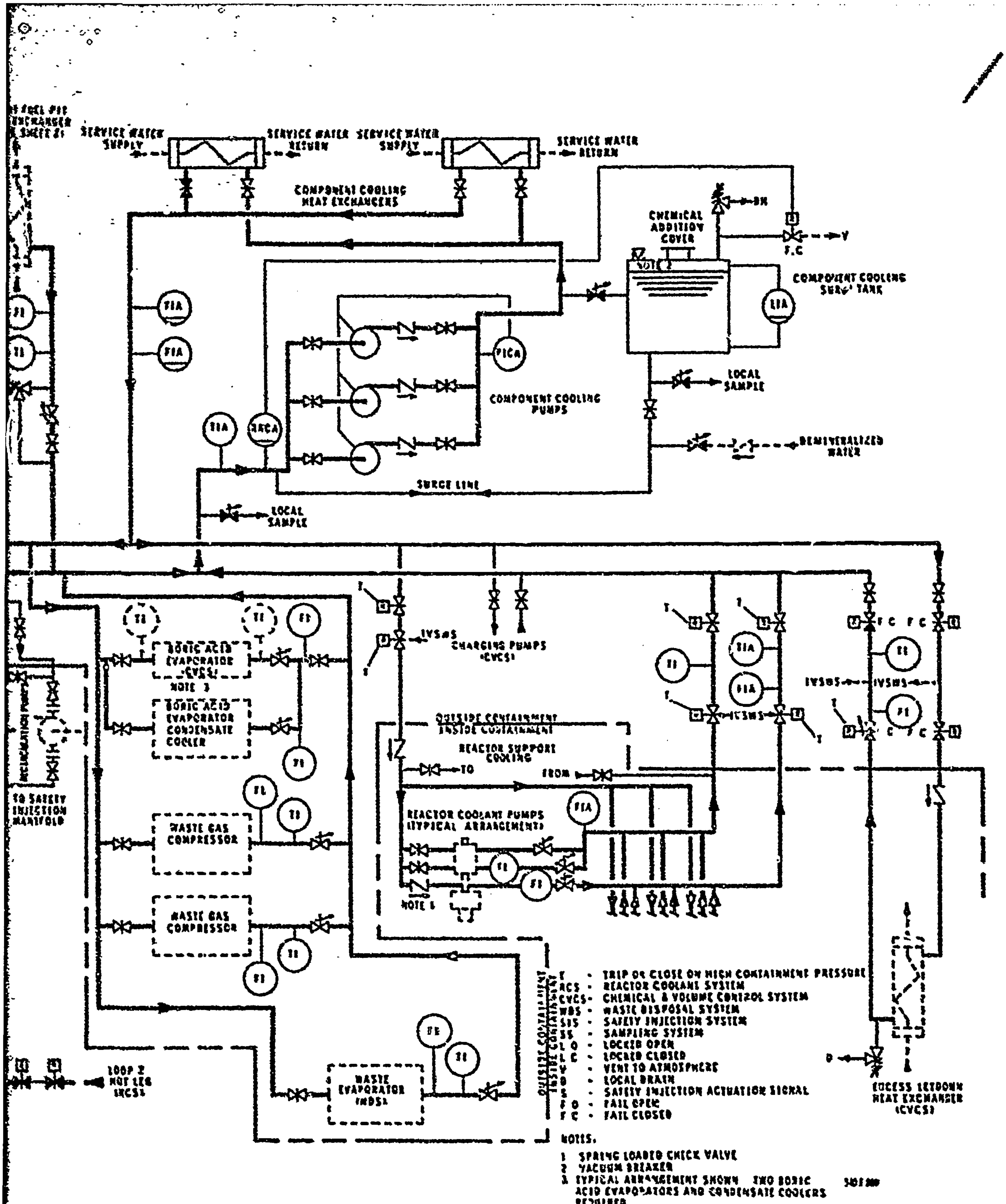
TABLE 9.3-5

FAILURE ANALYSIS OF PUMPS, HEAT EXCHANGERS, AND VALVES

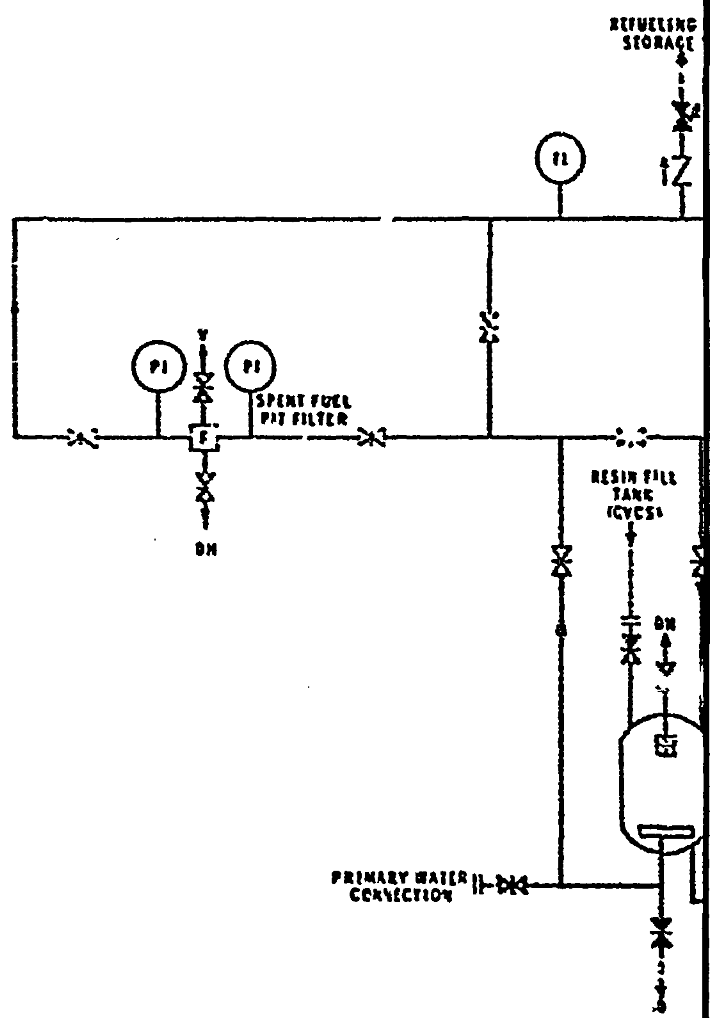
	Components	Malfunction	Comments and Consequences
1.	Component cooling water pumps	Rupture of a pump casing	The casting and shell are designed for 150 psi and 200°F which exceeds maximum operating conditions. Pump is inspectable and protected against credible missiles. Rupture is not considered credible. However, each unit is isolable. Two of the three pumps are needed to carry total pumping load.
2.	Component cooling water pumps	Pump fails to start	One operating pump supplies sufficient water for emergency cooling.
3.	Component cooling water pumps	Manual valve on a pump suction line closed	This is prevented by prestartup and operational checks. Further, during normal operation, each pump is checked on a periodic basis which would show if a valve is closed.
4.	Component cooling water	Normally open valve	The valve is checked open during periodic operation of the pumps during normal operation.
5.	Component cooling heat exchanger	Tube or shell rupture	Rupture is considered improbable because of the low operating pressures. Each unit is isolable. Second unit can carry total heat load for normal operation.
6.	Demineralized water makeup line check valve	Sticks open	The check valve is backed up by the manually operated valve. Manual valve is normally closed
7.	Component cooling heat exchanger vent or drain valve	Left open	This is prevented by prestartup and operational checks. On the operating unit such a situation is readily assessed by makeup requirements to system. On the second unit such a situation is ascertained during periodic testing.
8.	Component cooling water inlet valve to residual heat exchanger	Fails to open	There is one valve on each outlet line from each heat exchanger. One heat exchanger remains in service and provides adequate heat removal during long term recirculation. During normal operation the cooldown time is extended.



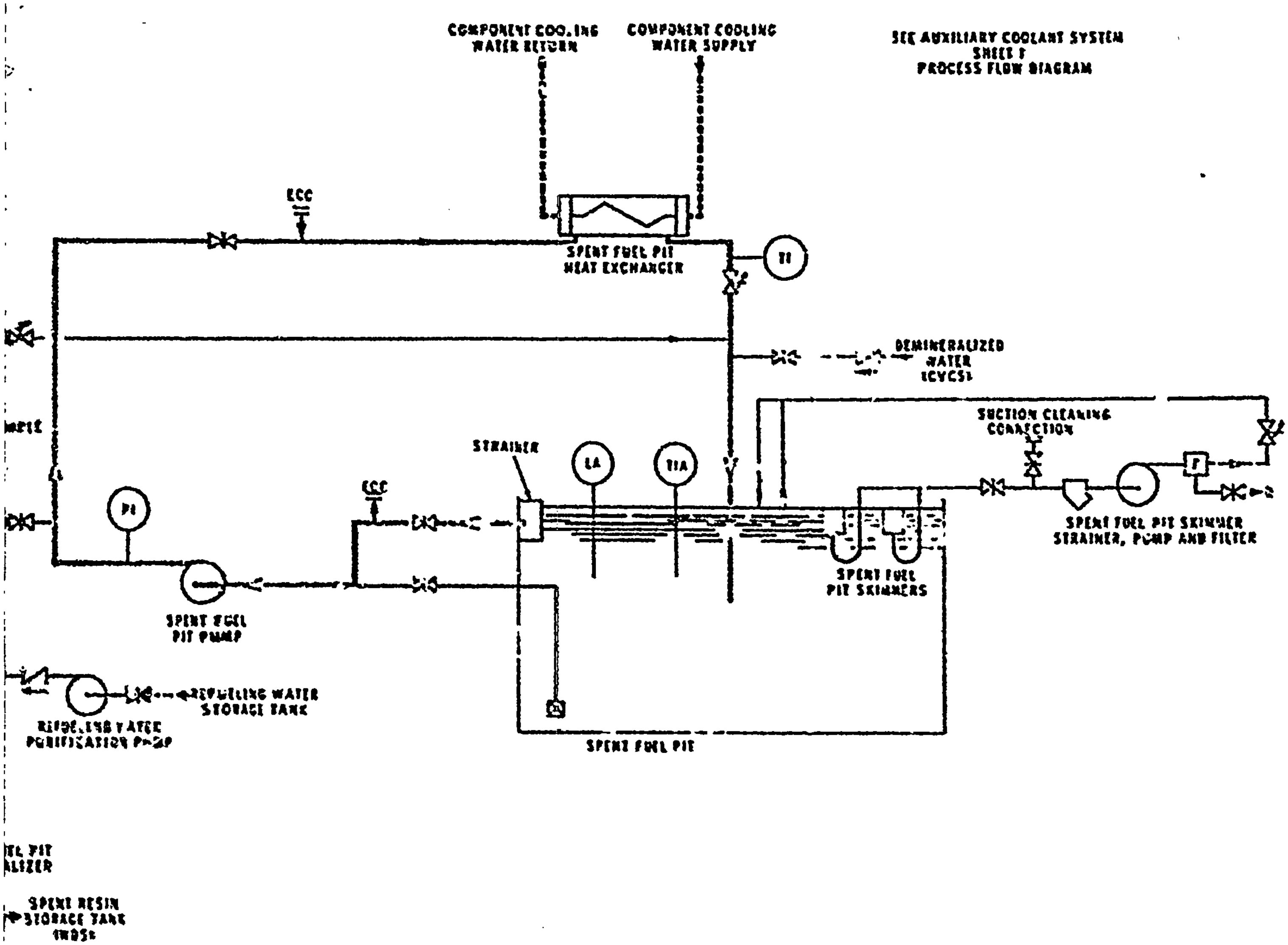
REACTOR COOLANT SAMPLE HEAT EXCHANGER (SS)
PRESSURIZER STEAM SAMPLE HEAT EXCHANGER (SS)
PRESSURIZER LIQUID SAMPLE HEAT EXCHANGER (SS)
4 STEAM GENERATOR BLOWDOWN SAMPLE HEAT EXCHANGERS (TYPICAL)
LETDOWN HEAT EXCHANGER (CVCS)
TIC
FI
TI
HIGH HEAD SAFETY INJECTION PUMPS (SIS)
OUTSIDE CONTAINMENT
INSIDE CONTAINMENT
RESIDUAL HEAT EXCHANGERS
FIA
L.O
FROM RECIRCULATION PUMPS (SIS)
HCV
TO CONTAINMENT SPRAY HEADER NO 1 (SIS)
TO CONTAINMENT SPRAY HEADER NO 2 (SIS)
PURIFICATION LINE (CVCS)
LETDOWN LINE (CVCS)
S
TR
PI
RESIDUAL HEAT REMOVAL PUMPS
L.C.
REFUELING WATER STORAGE TANK
LOCAL SAMPLE
REACTOR CONTAINMENT SUMP
IVSWS
F.C.
SS



Auxiliary Coolant System
Figure 9.3-1


REFUELING
STORAGE
FI
PI
PI
SPENT FUEL
PIT FILTER
DN
RESIN FILL
TANK
(CVCS)
DN
PRIMARY WATER
CONNECTION

CVCS - CHEMICAL & VOLUME CONTROL SYSTEM
ECC - EMERGENCY COOLING CONNECTION
DH - DRAIN HEADER (WASTE DISPOSAL SYSTEM)



Auxiliary Coolant System
Figure 9.3-2

9.4 SAMPLING SYSTEM

9.4.1 DESIGN BASIS

Performance Requirements

This system provides samples for laboratory analysis to evaluate reactor coolant, steam systems, and other reactor auxiliary systems chemistry during normal operation. It has no active emergency function. This system is normally isolated at the containment boundary.

Sampling system discharge flows are limited under normal and anticipated fault conditions (malfunctions or failure) to preclude any fission product releases beyond the limits of 10 CFR 20.

Design Characteristics

The system is capable of obtaining reactor coolant samples during reactor operation and during cooldown when the system pressure is low and the residual heat removal loop is in operation. Access is not required to the containment.

Sampling of other process coolants, such as tanks in the Waste Disposal System, is accomplished locally. Equipment for sampling secondary and non-radioactive fluids is separated from the equipment provided for reactor coolant samples. Leakage and drainage resulting from the sampling operations are collected and drained to tanks located in the Waste Disposal System.

Two types of samples are obtained by the system: high temperature -high pressure Reactor Coolant System and steam generators blowdown samples which originate inside the reactor containment, and low temperature -low pressure samples from the Chemical and Volume Control and Auxiliary Coolant Systems.

High Pressure - High Temperature Samples

A sample connection is provided from each of the following:

a) The pressurizer steam space
b) The pressurizer liquid space
c) Hot legs of loops 1 and 3
d) Blowdown from each steam generator

Low Pressure - Low Temperature Samples

A sample connection is provided from each of the following:

a) The mixed bed demineralizer inlet header
b) The mixed bed demineralizer outlet header
c) The residual heat removal loop, just downstream of the heat exchangers
d) The volume control tank gas space
e) The accumulator.

Expected Operating Temperatures

The high pressure, high temperature samples and the residual heat removal loop samples leaving the sample heat exchangers are held to a temperature of 130°F to minimize the generation of radioactive aerosols.

Codes and Standards

System component code requirements are given in Table 9.4-1.

9.4.2 SYSTEM DESIGN AND OPERATION

The Sampling System, shown in Figure 9.4-1, provides the representative samples for laboratory analysis. Analysis results provide guidance in the operation of the Reactor Coolant, Auxiliary Coolant, Steam and Chemical and Volume Control Systems. Analyses show both chemical and radiochemical conditions. Typical information obtained includes reactor coolant boron and chloride concentrations, fission product radioactivity level, hydrogen, oxygen, and fission gas content, corrosion product concentration, and chemical additive concentration.

The information is used in regulating boron concentrations adjustments, evaluating fuel element integrity and mixed bed demineralizer performance, and regulating additions of corrosion controlling chemicals to the systems. The Sampling System is designed to be operated manually, on an intermittent basis. Samples can be withdrawn under conditions ranging from full power to cold shutdown.

Reactor coolant liquid and steam sample lines, which are normally inaccessible or which require frequent sampling, are sampled by means of permanently installed tubing leading to the sampling room.

Sampling System equipment is located inside the auxiliary building with most of it in the sampling room. The delay coil and sample lines with remotely operated valves are located inside the reactor containment.

Reactor coolant hot leg liquid, pressurizer liquid and pressurizer steam samples originating inside the reactor containment flow through separate sample lines to the sampling room. Each of these connections to the Reactor Coolant System has a remote operated isolation valve located close to the sample source. The samples pass through the reactor containment to the auxiliary building, and into the sampling room, where they are cooled (pressurizer steam samples are condensed and cooled)) in the sample heat exchangers. The sample stream pressure is reduced by a manual throttling valve located downstream of each sample pressure vessel. The sample stream is purged to the volume control tank in the Chemical and Volume Control System until sufficient purge volume has passed to permit collection of a representative sample. After sufficient purging, the sample pressure vessel is isolated and then disconnected for laboratory analysis of the contents.

Alternately, liquid samples may be collected by bypassing the sample pressure vessels. After sufficient purge volume has passed to permit collection of a representative sample, a portion of the sample flow is diverted to the sample sink where the sample is collected.

The reactor coolant sample originating from the residual heat removal loop of the Auxiliary Coolant System has a remote operated, normally closed isolation valve located close to the sample source outside the containment. The sample line from this source is connected into the sample line coming from the hot leg at a point ahead of the sample heat exchanger. Samples from this source can be collected either in the sample vessels or at the sample sink as with hot leg samples.

Liquid samples originating at the Chemical and Volume Control System letdown line at demineralizer inlet and outlet pass directly through the purge line to the volume control tank. Samples are obtained by diverting a portion of the flow to the sample sink. If the pressure is low in the letdown line, the purge flow is directed to the chemical drain tank. The sample line from the gas space of the volume control tank delivers gas samples to the volume control tank sample pressure vessel in the sampling room. Purge flow for these samples is discharged to the vent header in the Waste Disposal System.

Samples of the steam generator liquid are obtained from the blowdown lines. These sample lines are routed by separate lines from each steam generator into the sample room. These lines are missile protected within the containment and are equipped with a remote operated isolation valve and manual isolation valve in each line immediately outside the containment. The remote operated valve is automatically closed upon receipt of a signal from the blowdown sample radiation monitor or the containment isolation system.

These blowdown lines are then routed on to the plant blowdown flash tank. The sample lines are taken off at an intermediate point inside the containment and routed to the sample room where the liquid is cooled and the pressure reduced. The sample lines are equipped with remote operated isolation valves. Each individual sample is then split into two routes: one goes to the sample sink to provide periodic samples for chemical analysis, the second goes to a conductivity cell, a radiation monitor and then to blowdown flash tank. This second line handles a continuous flow for a constant reading of conductivity and a constant monitoring for radiation.

The sample sink, which is contained in the laboratory bench as a part of the sampling hood, contains a drain line to the Waste Disposal System.

Local instrumentation is provided to permit manual control of sampling operations and to ensure that the samples are at suitable temperatures and pressures before diverting flow to the sample sink.

Components

A summary of principal component data is given in Table 9.4-2.

Sample Heat Exchangers

Seven sample heat exchangers reduce the temperature of samples from the pressurizer steam space, the pressurizer liquid space, each steam generator and the reactor coolant system liquid to 130°F before samples reach the sample vessels and sample sink. The tube side of the heat exchangers is austenitic stainless steel, the shell side is carbon steel.

The inlet and outlet tube sides have socket-weld joints for connections to the high pressure sample lines. Connections to the component cooling water lines are socket-weld joints. The samples flow at 0.42 gpm through the tube side and component cooling water from the Auxiliary Coolant System circulates through the shell side.

Delay Coil

The high pressure reactor coolant sample line contains a delay coil, consisting of coiled tubing, which has sufficient length to provide at least a 40 seconds sample transient time within the containment and an additional 20 seconds transit time from the reactor containment to the sampling hood. This allows for decay of short lived isotopes to a level that permits normal access to the sampling room.

Sample Pressure Vessels

The high pressure sample trains, the residual heat removal loop sample train and the volume control tank gas space sample train each contain sample pressure vessels which are used to obtain liquid or gas samples. The hot leg and the residual heat removal loop sample lines have a single sample pressure vessel in common. Integral isolation valves are furnished with the vessel and quick-disconnect coupling valves containing poppet-type check valves, are connected to nipples extending from the valves on each end. The vessels valves and couplings are austenitic stainless steel.

Sample Sink

The sample sink is located in a hooded enclosure which is equipped with an exhaust ventilator. The work area around the sink and the enclosure is large enough for sample collection and storage for radiation monitoring equipment. The sink perimeter has a raised edge to contain any spilled liquid.

The enclosure is penetrated by sample lines from the reactor plant, a demineralized water line, and steam system lines, all of which discharge into the sink. The sink and work area are stainless steel.

Piping and Fittings

All liquid and gas sample lines are austenitic stainless steel tubing and are designed for high pressure service. With the exception of the sample vessel quick-disconnect couplings and compression fittings at the sample sink, socket welded joints are used throughout the Sampling System. Lines are so located as to protect them from accidental damage during routine operation and maintenance.

Valves

Remotely operated stop valves are used to isolate all sample points and to route sample fluid flow inside the reactor containment. Manual stop valves are provided for component isolation and flow path control at all normally accessible Sampling System locations. Manual throttle valves are provided to adjust the sample flow rate as indicated on Figure 9.4-1.

Check valves prevent gross reverse flow of gas from the volume control tank i... the sample sink.

All valves in the system are constructed of austenitic stainless steel or equivalent corrosion resistant material.

Isolation valves provided outside the reactor containment which trip closed upon saturation of the containment isolation signal.

9.4.3 SYSTEM EVALUATION

Availability and Reliability

Neither automatic nor operator action is required of the Sampling System during an emergency or to prevent an emergency condition. The system is therefore designed in accordance with standard practices of the chemical processing industry.

Leakage Provisions

Leakage of radioactive reactor coolant from this system within the containment is evaporated to the containment atmosphere and removed by the cooling coils of the Recirculation Air Heating and Cooling System. Leakage of radioactive material from the most likely places outside the containment is collected by placing the entire sampling station under a hood provided with an offgas vent to the building exhaust. Liquid leakage from the valves in the hood is drained to the chemical drain tank.

Incident Control

The system operates on an intermittent basis, and under administrative manual control.

Malfunction Analysis

To evaluate system safety, the failures or malfunctions are assumed concurrent with a loss-of-coolant accident, and the consequences analyzed. The results are presented in Table 9.4-3. From this evaluation it is concluded that proper consideration has been given to station safety in the design of the system.

9.4.4 MINIMUM OPERATING CONDITIONS

Minimum operating conditions will be specified in the Technical Specifications.

9.4.5 TESTS AND INSPECTIONS

Typical examples for frequency of sample analyses is as follows:

a) Reactor coolant - radiochemical analysis - 5 days per week
b) Reactor coolant - boron concentration - 5 days per week
c) Refueling water - storage tank water - boron concentration - once a week.

TABLE 9.4-1

SAMPLING SYSTEM CODE REQUIREMENTS

Sample heat exchanger	ASME III*, Class C, tube side ASME VIII, shell side
Sample pressure vessels	ASME III, Class C
Piping and valves	USAS B31.1**

* ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

** USAS B31.1 - Code for Pressure Piping and special nuclear cases where applicable.

TABLE 9.4-2

SAMPLING SYSTEM COMPONENTS

Sample Heat Exchanger

General

Number	7
Type	Counter-flow
Design heat transfer rate (duty for 652.7°F sat. steam to 127°F liquid), Btu/hr	2.12×10^5

Shell

Design pressure, psig	150
Design temperature, °F	350
Component cooling water flow, gpm	14
Pressure loss at 14 gpm, psi	25
Operating cooling water	
temperature, in, °F	95
temperature, out (maximum) °F	125
Material	Carbon steel

Tubes

Tube diameter in., O.D.	3/8
Design pressure, psig	2485
Design temperature, °F	680
Sample flow, normal each lb/hr.	209
Maximum allowable pressure loss, each 209 lb/hr. psi	10
Operating sample temperature, in (maximum) °F	652.7
Operating sample temperature, out (maximum) °F	127
Material	Austenitic stainless steel

TABLE 9.4-2 (Continued)

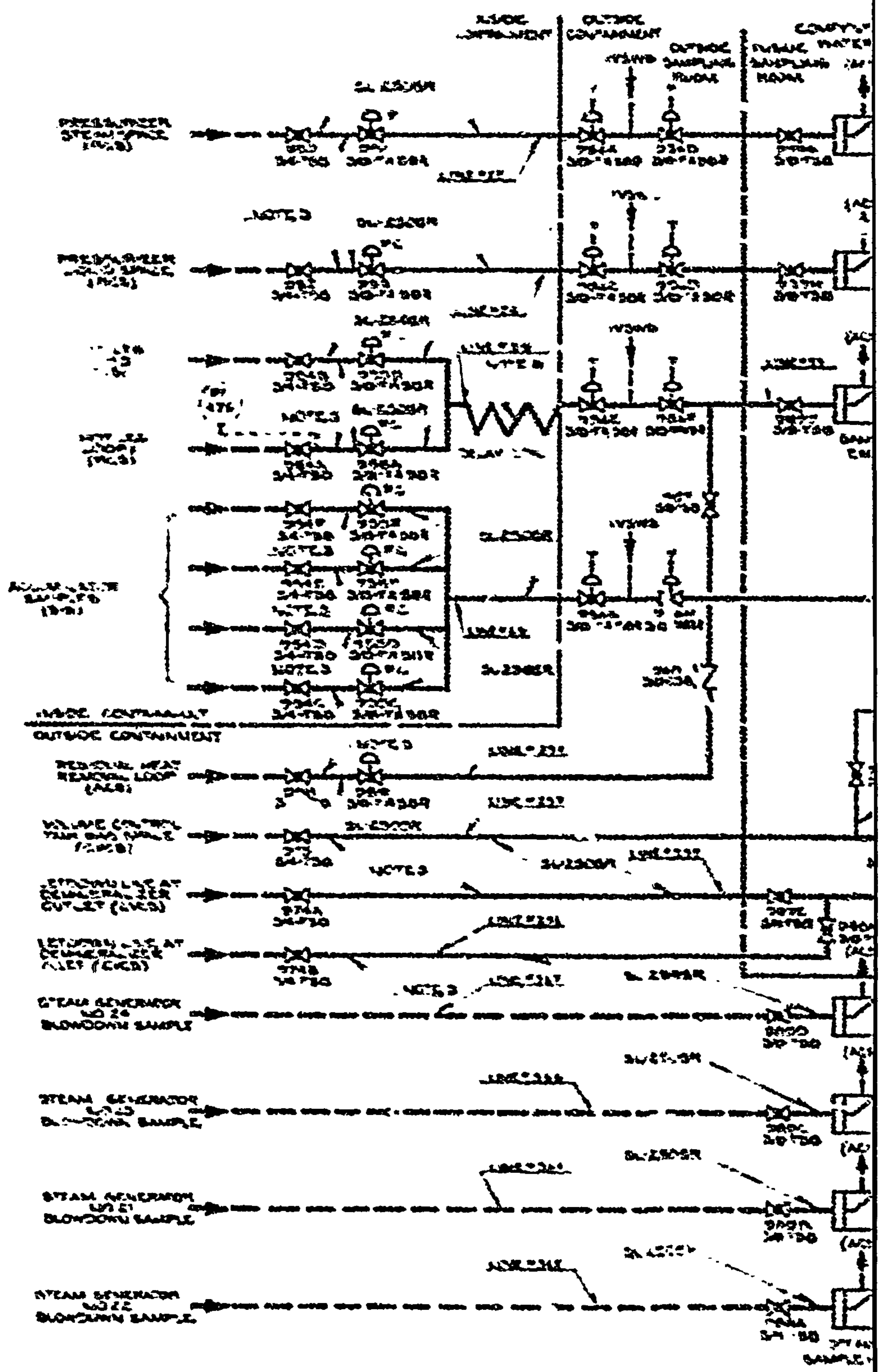
Sample Pressure Vessels

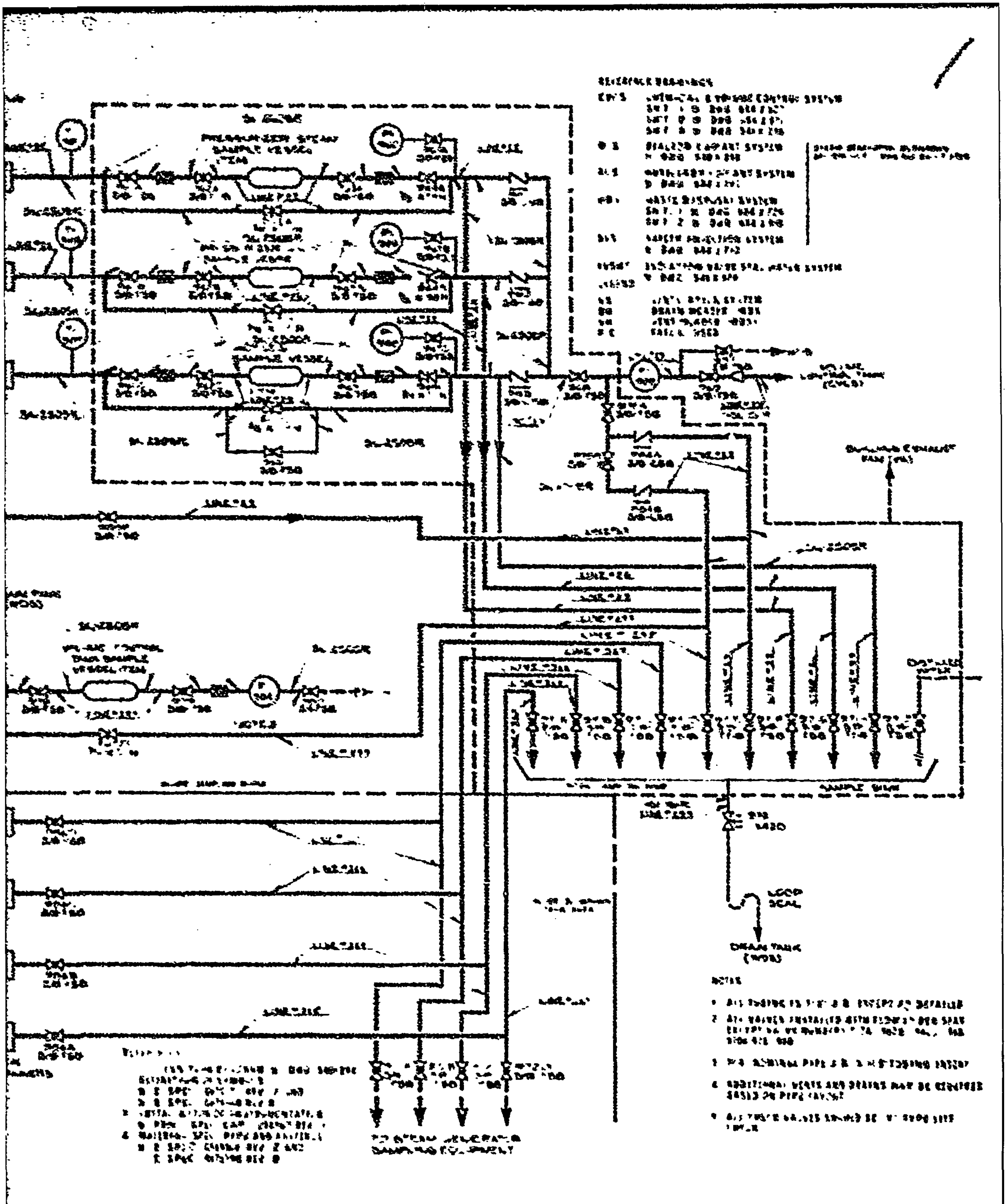
Number, total	8
Volume, pressurizer steam sample, 2 supplied ml	75
Volume, pressurizer liquid sample, 2 supplied ml	75
Volume, reactor coolant hot leg sample, 2 supplied ml	75
Volume, control tank sample, 2 supplied ml	75
Design pressure. psig	2485
Design temperature, °F	680

TABLE 9.4-3

MALFUNCTION ANALYSIS OF SAMPLING SYSTEM

Sample Chains	Malfunction	Comments and Consequences
Pressurizer steam space sample, pressurizer liquid space sample, or hot leg sample	Remote operated sampling valve inside reactor containment fails to close	Diaphragm - operated valve outside the reactor containment closes on containment isolation signal
Any sample chain	Sample line break inside containment	Same as above





Sampling System
Figure 9.4-1

9.5 FUEL HANDLING SYSTEM

The Fuel Handling System provides a safe effective means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves the plant after post-irradiation cooling.

The system is designed to minimize the possibility of mishandling or mal-operations that cause fuel damage and potential fission product release.

The Fuel Handling System consists basically of:

a) The reactor cavity, which is flooded only during plant shutdown for refueling

b) The spent fuel pit, which is kept full of water and is always accessible to operating personnel

c) The Fuel Transfer System, consisting of an underwater conveyor that carries the fuel through an opening between the areas listed in the discussion of plant containment.

9.5.1 DESIGN BASIS

Prevention of Fuel Storage Criticality

Criterion: Criticality in the new and spent fuel storage pits shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls. (GDC 66)

During reactor vessel head removal and while loading and unloading fuel from the reactor, boron concentration is maintained at not less than that required to shutdown the core to a $k_{eff} = 0.90$. This shutdown margin maintains the core at $k_{eff} <0.99$, even if all control rods are withdrawn from the core. Weekly checks of refueling water boron concentration ensure the proper shutdown margin.

The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The new and spent fuel storage pits have accommodations as defined in Table 9.5-1. In addition, the spent fuel pit has the required spent fuel shipping area. Borated water is used to fill the spent fuel storage pit at a concentration to match that used in the reactor cavity and refueling canal during refueling operations. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to assure $K_{eff} \leq 0.90$ even if unborated water was used to fill the pit.

Detailed instructions are available for use by refueling personnel. These instructions, the minimum operating conditions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.

Fuel and Waste Storage Decay Heat

Criterion: Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities and to waste storage tanks that could result in radioactivity release which would result in undue risk to the health and safety of the public. (GDC 67)

The refueling water provides a reliable and adequate cooling medium for spent fuel transfer and heat removal from the spent fuel pit is provided by an auxiliary cooling system. Natural radiation and convection is adequate for cooling the holdup tanks.

Fuel and Waste Storage Radiation Shielding

Criterion: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities. (GDC 68)

Adequate shielding for radiation protection is provided during reactor refueling by conducting all spent fuel transfer and storage operations under water. This permits visual control of the operation at all times while maintaining low radiation levels, less than 2.0 mr/hr, for periodic occupancy of the area by operating personnel. Pit water level is indicated, and water removed from the pit must be pumped out since there are no gravity drains. Shielding is provided for waste handling and storage facilities to permit operation within requirements of 10 CFR 20.

Gamma radiation is continuously monitored in the auxiliary building. A high level signal is alarmed locally and is annunciated in the control room.

Protection Against Radioactivity Release From Spent Fuel and Waste Storage

Criterion: Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity. (GDC 69)

All fuel and waste storage facilities are contained and equipment designed so that accidental releases of radioactivity directly to the atmosphere are monitored and do not exceed the guidelines of 10 CFR 100.

The reactor cavity, refueling canal and spent fuel storage pit are reinforced concrete structures with a seam-welded stainless steel plate liner. These structures are designed to withstand the anticipated earthquake loadings as Class I structures so that the liner prevents leakage even in the event the reinforced concrete develops cracks.

All vessels in the Waste Disposal System which are used for waste storage are designed as Class I equipment.

9.5.2 SYSTEM DESIGN AND OPERATION

The reactor is refueled with equipment designed to handle the spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Boric acid is added to the water to ensure subcritical conditions during refueling.

The Fuel Handling System may be generally divided into two areas: the reactor cavity which is flooded only during plant shutdown for refueling and the spent fuel pit which is kept full of water and is always accessible to operating personnel. These two areas are connected by the Fuel Transfer System consisting of an underwater conveyor that carries the fuel through an opening in the plant containment.

The reactor cavity is flooded with borated water from the refueling water storage tank. In the reactor cavity, fuel is removed from the reactor vessel, transferred through the water and placed in the fuel transfer system by a manipulator crane. In the spent fuel pit the fuel is removed from the transfer system and placed in storage racks with a long manual tool suspended from an overhead hoist. After a sufficient decay period, the fuel is removed from storage and loaded into a shipping cask for removal from the site.

New fuel assemblies are received and stored in racks in the new fuel storage area. New fuel is delivered to the reactor by lowering it into the spent fuel pit and taking it through the transfer system. The new fuel storage area is sized for storage of the fuel assemblies and control rods normally associated with the replacement of one-third of a core. The fuel for the initial core loading may be stored temporarily in the spent fuel storage pit or in both the spent fuel pit and the new fuel vault. The pit is kept dry during this period.

Major Structures Required for Fuel Handling

Reactor Cavity

The reactor cavity is a reinforced concrete structure that forms a pool above the reactor when it is filled with borated water for refueling. The cavity is filled to a depth that limits the radiation at the surface of the water to 2.0 milliroentgens per hour during fuel assembly transfer.

The reactor vessel flange is sealed to the bottom of the reactor cavity by a clamped, gasketed seal ring which prevents leakage of refueling water from the cavity. This seal is fastened and closed after reactor cooldown but prior to flooding the cavity for refueling operations.

The cavity is large enough to provide storage space for the reactor upper and lower internals, the control cluster drive shafts, and miscellaneous refueling tools.

The floor and sides of the reactor cavity are lined with stainless steel.

Refueling Canal

The refueling canal is a passageway extending from the reactor cavity to the inside surface of the reactor containment. The canal is formed by two concrete shielding walls, which extend upward to the same elevation as the reactor cavity. The floor of the canal is at a lower elevation than the reactor cavity to provide the greater depth required for the fuel transfer system tipping device and the control cluster changing fixture located in the canal. The transfer tube enters the reactor containment and protrudes through the end of the canal. Canal wall and floor linings are similar to those for the reactor cavity.

Refueling Water Storage Tank

The normal duty of the refueling water storage tank is to supply borated water to the refueling canal for refueling operations. In addition, the tank provides borated water for delivery to the core following either a loss-of-coolant or a steam line rupture accident. This is described in Chapter 6.

The capacity of the tank is based upon the requirement for filling the reactor cavity and refueling canal.

The water in the tank is borated to a concentration which assures reactor shutdown by at least 10% δk/k when all RCC assemblies are inserted and when the reactor is cooled down for refueling. Heating is provided to maintain the temperature above freezing.

The tank design parameters are given in Chapter 6.

Spent Fuel Storage Pit

The spent fuel storage pit is designed for the underwater storage of spent fuel assemblies, failed fuel cans if required, and control rods after their removal from the reactor.

The pit accomodations are listed in Table 9.5-1.

Spent fuel assemblies are handled by a long-handled tool suspended from an overhead hoist and mainipulated by an operator standing on the movable bridge over the pit.

The spent fuel storage pit is constructed of reinforced concrete having thick walls and is Class I seismic design. The entire interior basin face and transfer canal is lined with stainless steel plate. Hence, the probability of rupture of the pit is exceedingly low.

A storage rack is provided to hold spent fuel assemblies and is erected on the pit floor. Fuel assemblies are held in a square array, and placed in vertical cells. The racks are designed so that it is impossible to insert fuel assemblies in other than the prescribed locations, thereby ensuring the necessary spacing between assemblies. Control rod clusters are stored in place inside the spent fuel assemblies.

New Fuel Storage

New fuel assemblies and control rods are stored in a separate area whose location facilitates the unloading of new fuel assemblies or control rods from trucks. This storage vault is designed to hold new fuel assemblies in specially constructed racks and is utilized primarrily for the storage of the replacement fuel assemblies required for cycled loading. The assemblies which make up the remaining part of the first core are stored in the spent fuel pit which otherwise remains unused until the time of first refueling. The new fuel assemblies are stored in racks arranged to space the fuel assemblies the same amount as for spent fuel. For the initial fuel core, all of the new fuel assemblies may be stored in the spent fuel pit or may be, or in both storage areas as described above.

Decontamination facilities, consisting of an equipment and cas pit, are located adjacent to the spent fuel storage pit; cask handling and other tools can be cleaned and decontaminated in the cask decontamination pit. The outside surfaces of the casks are decontaminated, if required, by using steam, water, detergent solutions, and manual scrubbing to the extent required.

Major Equipment Required for Fuel Handling

Reactor Vessel Stud Tensioner

Stud tensioners are used to make up the head closure joint and during this process all studs are stretch tested to more than nominal working loads at every refueling.

The stud tensioner is a hydraulically operated (oil as the working fluid) device provided to permit preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. Stud tensioners were chosen in order to minimize the time required for the tensioning or unloading operations. Three tensioners are provided and they are applied simultaneously to three studs 120° apart. One hydraulic pumping unit operates the tensioners which are hydraulically connected in parallel. The studs are tensioned to their operational load in two steps to prevent high stresses in the flange region and unequal loadings in the studs. Relief valves are provided on each tensioner to prevent overtensioning of the studs due to excessive pressure. Charts indicating the stud elongation and load for a given oil pressure are included in the tensioner operating instructions. In addition, micrometers are provided to measure the elongation of the studs after tensioning.

Reactor Vessel Head Lifting Device

The reactor vessel head lifting device consists of a welded and bolted structural steel frame with suitable rigging to enable the crane operator to lift the head and store it during refueling operations. The lifting device is permanently attached to the reactor vessel head.

Reactor Internals Lifting Device

The reactor internals lifting device is a fixture providing the means to grip the top of the reactor internals package and to transfer the lifting load to the crane. Utilizing a long sling, the device is lowered onto the guide tube support plate of the internals and is manually bolted to the support plate by three bolts. The bolts are controlled by long torque tubes extending up to an operating platform on the lifting device. Bushings on the fixture engage guide studs mounted on the vessel flange to provide close guidance during removal and replacement of the internals package.

Manipulator Crane

The manipulator crane is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water. The bridge spans the reactor cavity and runs on rails set into the floor along the edge of the reactor cavity. The bridge and trolley motions are used to position the vertical mast over a fuel assembly in the core. A long tube with a pneumatic gripper on the end is lowered down out of the mast to grip the fuel assembly. The gripper tube is long enough so the upper end is still contained in the mast when the gripper end contacts the fuel. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. The fuel is transported while inside the mast tube to its new position.

All controls for the manipulator crane are mounted on a console on the trolley. The bridge is positioned on a coordinate system laid out on one rail. The electrical readout system on the console indicates the position of the bridge. The trolley is positioned with the aid of a scale on the bridge structure. The scale is read directly by the operator at the console. The drives for the bridge, trolley, and winch are variable speed and include a separate inching control on the winch. Electrical interlocks and limit switches on the bridge and trolley drives protect the equipment. In an emergency, the bridge, trolley, and winch can be operated manually using a handwheel on the motor shaft.

The suspended weight on the gripper tool is monitored by an electric load cell indicator mounted on the control console. A load in excess of 110 per cent of a fuel assembly weight stops the winch drive from moving in the up direction. The gripper is interlocked through a weight sensing device and also a mechanical spring lock so that it cannot be opened when supporting a fuel assembly.

Safety features are incorporated in the system as follows:

a) Travel limit switches on the bridge and trolley drives

b) Bridge, trolley, and winch drives which are mutually interlocked to prevent simultaneous operation of any two drives

c) A position safety switch, the GRIPPER TUBE UP position switch, which prevents bridge and trolley main motor drive operation except when it is actuated

d) An interlock which prevents the opening of a solenoid valve in the air line to the gripper except when zero suspended weight is indicated by a force gage. As back-up protection for this interlock, the mechanical weight actuated lock in the gripper prevents operation of the gripper under load even if air pressure is applied to the operating cylinder.

e) The EXCESSIVE SUSPENDED WEIGHT switch, which opens the hoist drive circuit in the up direction when the loading is excessive

f) An interlock on the hoist drive circuit in the up direction, which permits the hoist to be operated only when either the OPEN or CLOSED indicating switch on the gripper is actuated

g) An interlock of the bridge and trolley drives, which prevents the bridge drive from traveling beyond the east edge of the core unless the trolley is aligned with the refueling canal centerline. The trolley drive is locked out when the bridge is beyond the east edge of the core.

Suitable restraints are provided between the bridge and trolley structures and their respective rails to prevent derailing and the manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper in the event of a maximum potential earthquake.

Spent Fuel Pit Bridge

The spent fuel pit bridge is a wheel-mounted walkway, spanning the spent fuel pit which carries an electric monorail hoist on an overhead structure. The fuel assemblies are moved within the spent fuel pit by means of a long-handled tool suspended from the hoist. The hoist travel and tool, length are designed to limit the maximum lift of a fuel assembly to a safe shielding depth.

Fuel Transfer System

The fuel transfer system, shown in Figure 9.5-1 is an underwater air-motor driven conveyor car that runs on tracks extending from the refueling canal through the transfer tube and into the spent fuel pit. The conveyor car receives a fuel assembly in the vertical position from the manipulator crane. The fuel assembly is lowered to a horizontal position for passage through the tube, and then is raised to a vertical position in the spent fuel pit.

During plant operation, the conveyor car is stored in the refueling canal. A blind flange is bolted on the transfer tube to seal the reactor containment.

Rod Cluster Control Changing Fixture

A fixture is mounted on the reactor cavity wall for removing rod cluster control (RCC) elements from spent fuel assemblies and inserting them into new fuel assemblies. The fixture consists of two main components; a guide tube mounted to the wall for containing and guiding the RCC element, and a wheel-mounted carriage for holding the fuel assemblies and positioning fuel assemblies under the guide tube. The guide tube contains a pneumatic gripper on a winch that grips the RCC element and lifts it out of the fuel assembly. By repositioning the carriage, a new fuel assembly is brought under the guide tube and the gripper lowers the RCC element and releases it. The manipulator crane loads and removes the fuel assemblies into and out of the carriage.

Refueling Procedure

The following is general outline of the refueling procedure:

Preparation

a) The reactor is shut down and cooled to ambient conditions.
b) A radiation survey is made and the containment vessel is entered.
c) The control rod drive mechanism (CRDM) missile shield is removed to storage.
d) CRDM cables and cooling air ducts are disconnected from CRDM and removed to storage.
e) Reactor vessel head insulation and instrument leads are removed.
f) The reactor vessel head nuts are loosened with the hydraulic tensioners.
g) The reactor vessel head studs are removed to storage.
h) The canal drain holes are plugges and the fuel transfer tube flange is removed.
i) Checkout of the fuel transfer device and manipulator crane is started.
j) Guide studs are installed in three holes and the remainder of the stud holes are plugged.
k) The reactor vessel to cavity seal ring is clamped in place.
l) Final preparation of underwater lights and tools is made. Checkout of manipulator crane and fuel transfer system is completed.
m) The reactor vessel head is unseated and raised one foot with the plant crane.

x

n) The reactor cavity is filled with water to the vessel flange. The water is pumped into the reactor cavity by the residual heat removal pumps from the refueling water storage tank through the reactor vessel. The normal Residual Heat Removal System inlet valves from the Reactor Coolant System are closed.

o) The reactor vessel head is slowly lifted while water is pumped into the reactor cavity. The water level and vessel head are raised simultaneously keeping the water level just below the head.

p) When the reactor is filled, restore Residual Heat Removal System to normal operation.

q) The reactor vessel head is taken to the storage pedestal.

r) The control rod drive shafts are unlatched.

s) The reactor vessel internals lifting rig is lowered into position by the plant crane and latched to the support plate.

t) The reactor vessel internals are lifted out of the vessel and placed in the underwater storage rack.

u) The core is now ready for refueling.

Refueling

The refueling sequence is now started with the manipulator crane. The sequence for fuel assembies in non-control positions is as follows:

a) Spent fuel is removed from the center region of the core and placed into the fuel transfer system for removal to the spent fuel pit.

b) Partially spent fuel is transferred from the intermediate region of the core to the vacated positions in the center region.

x

c) Partially spent fuel is transferred from the outer region of the core to vacated positions in the intermediate region.

d) New fuel assemblies are brought in from the spent fuel pit through the transfer system and loaded into the outer region.

e) Whenever new fuel is added to the reactor core, a reciprocal curve of source neutron multiplication is recorded to verify the subcriticality of the core.

The refueling sequence is modified for fuel assemblies containing rod cluster control (RCC) elements, as required. If a transfer of the RCC elements between fuel assemblies is required, the assemblies are taken to the RCC change fixture to exchange the RCC elements from one assembly to another. Such an exchange is required whenever a spent fuel assembly containing RCC elements is removed from the core and whenever a fuel assembly is placed in or taken out of a control position during refueling rearrangement.

Reactor Reassembly

a) The fuel transfer car is parked and the fuel transfer tube isolation valve closed.

b) The reactor vessel internals package is picked up by the plant crane and replaced in the vessel. The reactor vessel internals' lifting rig is removed to storage.

c) The control rod drive shafts are relatched to the RCC elements.

d) The manipulator crane is parked.

x

e) The old seal rings are removed from the reactor vessel head, the grooves cleaned and new rings installed.

f) The reactor vessel head is picked up by the plant crane and positioned over the reactor vessel.

g) The reactor vessel head is slowly lowered as the water level is lowered. The water level is lowered by opening a valve at the residual heat removal pump discharge and water is pumped from the reactor cavity into the refueling water storage tank until it reaches the vessel flange level. The normal residual heat removal line is closed.

h) When the reactor vessel head is about one foot above the flange, the reactor cavity is completely drained. When the water in the reactor cavity reaches the vessel flange level, the valve at the residual heat removal pump is closed. The normal residual heat removal operation is restored and the remaining water in the reactor cavity is drained into the reactor coolant drain tank via the low point in the canal drain. The water is then pumped back into the refueling water storage tank by the reactor coolant drain tank pumps. The flange surface is manually cleaned.

i) The reactor vessel head is seated.

j) The guide studs are removed to their storage rack. The stud hold plugs are removed.

k) The head studs are replaced and retorqued.

l) The canal drain holes are unpluggged and the fuel transfer tube flange is replaced.

m) Electrical leads and cooling air ducts are reconnected to the CRDM's.

n) Vessel head insulation and instrumentation leads are replaced.

o) The reactor vessel to cavity seal ring is unclamped.

p) A hydrostatic test is performed on the reactor vessel.

q) Control rod drives are checked.

r) The CRDM missile shield is picked up with the plant crane and replaced.

s) Equipment access door is closed and sealed.

t) Pre-operational tests are performed.

9.5.3 SYSTEM EVALUATION

Underwater transfer of spent fuel provides essential ease and corresponding safety in handling operations. Water is an effective, economic and transparent radiation shield and a reliable cooling medium for removal of decay heat.

Basic provisions to ensure the safety of refueling operations are:

a) Gamma radiation levels in the containment and fuel storage areas are continuously monitored. These monitors provide an audible alarm at the initiating detector indicating an unsafe condition. Continuous monitoring of reactor neutron flux provides immediate indication and alarm of an abnormal core flux level in the control room.

X

b) Violation of containment integrity is not permitted when the reactor vessel head is removed unless the shutdown margin is maintained greater than 10% $\Delta k/k$.

c) Whenever new fuel is added to the reactor core, a reciprocal curve of source neutron multiplication is recorded to verify the subcriticality of the core.

Incident Protection

Direct communication between the control room and the refueling cavity manipulator crane is available whenever changes in core geometry are taking place.

This provision allows the control room operator to inform the manipulator crane operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

Malfunction Analysis

An analysis is presented in Section 14 concerning damage to one complete outer row of fuel elements in an assembly, assumed as a conservative limit for evaluating environmental consequences of a fuel handling incident.

Any suspected defective fuel assembly is placed in a failed fuel can and sealed to provide an isolated chamber for testing for the presence of fission products.

The failed fuel cans are stainless steel cylinders with lids that can be bolted in place remotely. An internal gas space in the lid provides for water expansion and for collection and sampling of fission product gases. Various remotely operable quick-disconnect fittings permit connection of the can to sampling loops for continuous circulation through the can.

If sampling shows the presence of fission products indicative of a cladding failure, the sampling lines are closed off by valves on the can and the encapsulated fuel assembly is removed to the spent fuel storage racks to await shipment. Design of the cans complies with federal regulation 10 CFR 72 so that the defective fuel can be stored and shipped while sealed in the failed fuel can.

9.5.4 MINIMUM OPERATING CONDITIONS

Minimum operating conditions will be specified in the Technical Specifications.

9.5.5 TESTS AND INSPECTIONS

Upon completion of core loading and installation of the reactor vessel head, certain mechanical and electrical tests are performed prior to initial criticality. The electrical wiring for the rod drive circuits, the rod position indicators, the reactor trip circuits, the in-core thermocouples, and the reactor vessel head water temperature thermocouple are tested at the time of installation. The tests are repeated on these electrical items before initial plant operation.

b) Violation of containment integrity is not permitted when the reactor vessel head is removed unless the shutdown margin is maintained greater than 10% $\Delta k/k$.

c) Whenever new fuel is added to the reactor core, a reciprocal curve of source neutron multiplication is recorded to verify the subcriticality of the core.

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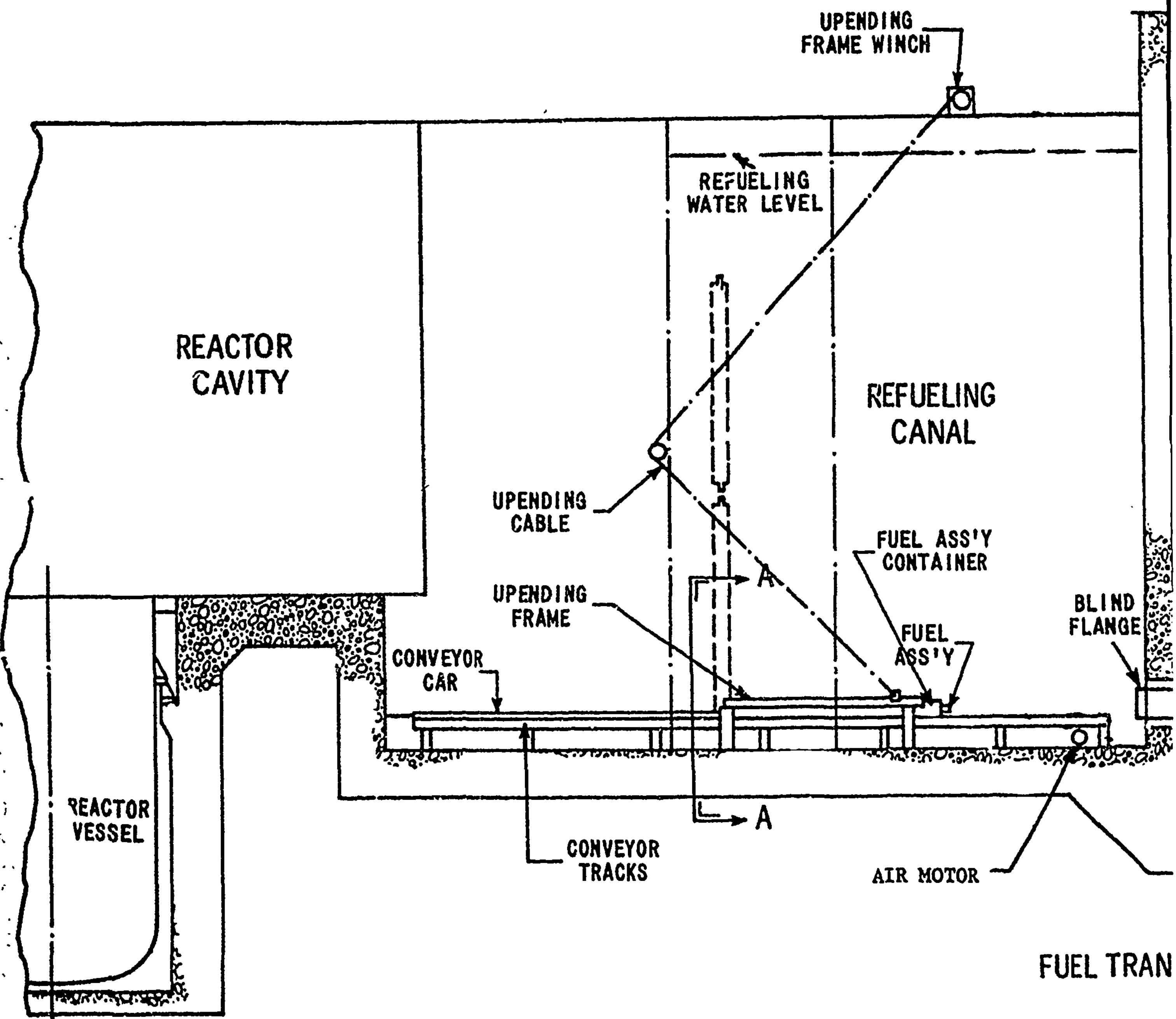
TABLE 9.5-1

FUEL HANDLING DATA

New Fuel Storage Pit	
Core storage capacity	1/3
Equivalent fuel assemblies	72
Center-to-center spacing of assemblies, in.	20.5
Maximum k_{eff} with unborated water	0.90
Maximum k_{eff} with unborated water for any degree of interspersed moderation	0.93
Spent Fuel Storage Pit	
Core storage capacity	1 - 1/3
Equivalent fuel assemblies	264
Number of space accommodations for failed fuel cans	2
Number of space accommodations for spent fuel shipping casks	1
Center-to-center spacing of assemblies, in.	20.5
Maximum k_{eff} with unborated water	0.90
Maximum k_{eff} with unborated water for any degree of interspersed moderation	0.93
Miscellaneous Details	
Width of refueling canal, ft.	3
Wall thickness for spent fuel storage pit, ft.	3 to 6
Weight of fuel assembly with RCC (dry), lb.	~1580
Capacity of refueling water storage tank, gal.	355,000
Minimum contents of refueling water storage tank for Safety Injection or Spray System Operability, gal.	350,000
Quantity of water required for refueling, gal.	350,000


UPENDING
FRAME WINCH
REFUELING
WATER LEVEL
REACTOR
CAVITY
REFUELING
CANAL
UPENDING
CABLE
FUEL ASS'Y
CONTAINER
UPENDING
FRAME
A
BLIND
FLANGE
FUEL
ASS'Y
CONVEYOR
CAR
REACTOR
VESSEL
A
CONVEYOR
TRACKS
AIR MOTOR

FUEL TRAN

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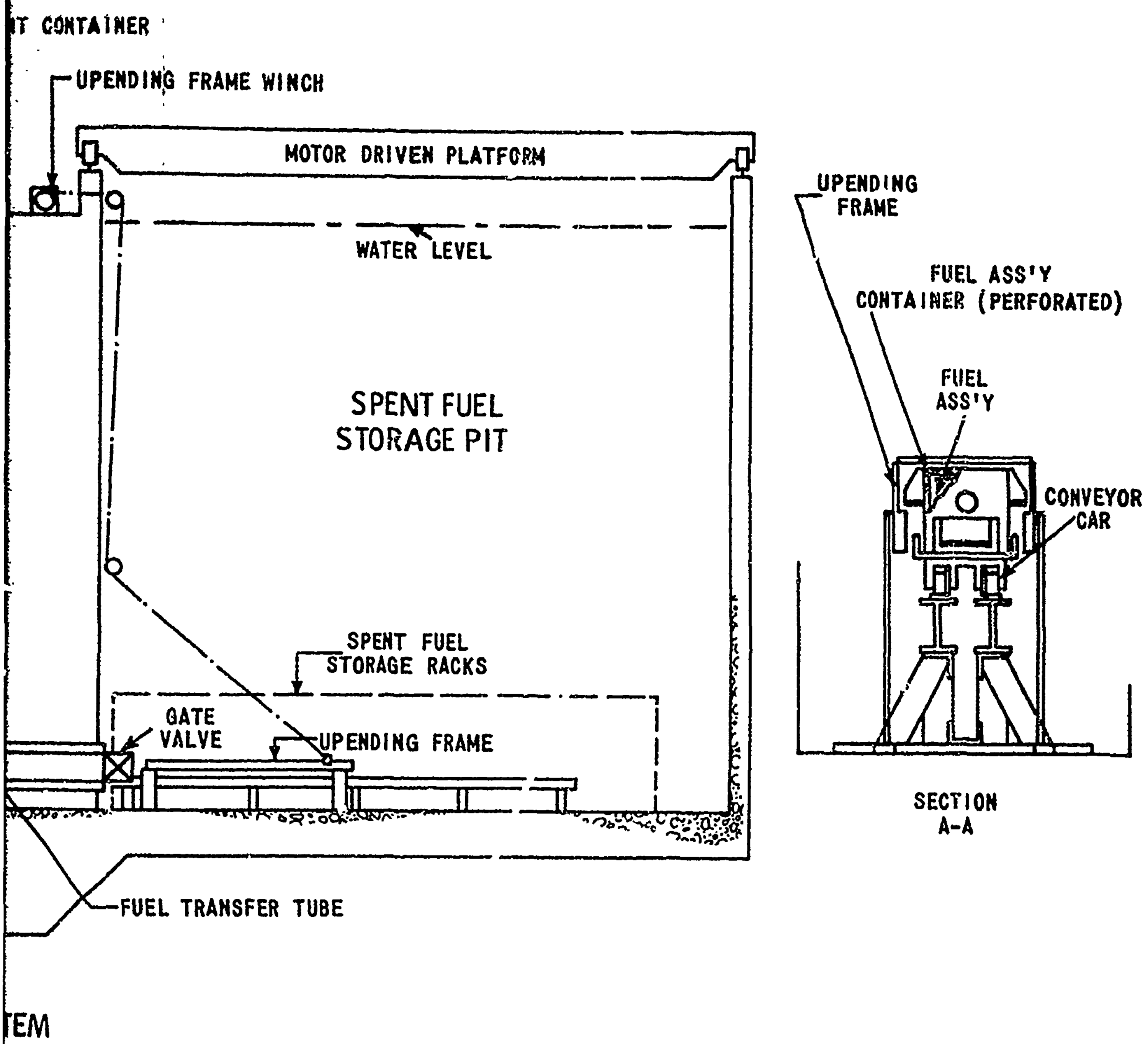


FUEL TRANSFER SYSTEM
FIG. 9.5-1

9.6 FACILITY SERVICE SYSTEM

9.6.1 SERVICE WATER SYSTEM

Design Basis

The service water system is designed to supply cooling water from the Hudson River to various heat loads in both the primary and secondary portions of the plant. Provision is made to ensure a continuous flow of cooling water to those systems and components necessary for plant safety either during normal operation or under abnormal and accident conditions. Sufficient redundancy of active and passive components is provided to insure that cooling is maintained to vital loads for short and long periods. The system also provides water required for cleaning the traveling screens and raw water to the condensate make up flash evaporator.

System Design and Operation

The service water system flow diagram is shown in Figure 9.6.1-1. Six identical vertical, centrifugal sump-type pumps, each having a capacity of 5000 gpm at 220 ft TDH, supply service water to two independent discharge headers, each header being supplied by three of the pumps. An automatic, continuous, rotary-type strainer is in the discharge of each pump, and is capable of removing solids down to 1/8 inch diameter. Each header is connected to an independent supply line. Either of the two supply lines can be used to supply the essential loads, with the other line feeding the non-essential loads. The essential loads are the containment ventilation cooling coils; the containment ventilation fan motor coolers; instrument air compressors; turbine oil, boiler feed pump turbine oil and the seal oil coolers; and the diesel generator coolers.

Water is drawn from the river and passes under a debris wall, through a coarse screen and finally a fine traveling band screen. To keep the intake free of ice, warm water is circulated from the condenser discharge canal to a point ahead of the coarse screens. Electric heaters are provided in the driving head of the traveling screens to prevent icing of the screen panels. Each main circulating water - p is installed in an individual chamber while the service water pumps are in a common chamber with one intake normally operating. A second full flow service water intake is constructed and provided with a temperary screen. This intake is brought into operation when required. Openings are also provided between the main circulating water pump chambers either side and the service water pump chamber. These two openings can be closed by gates, but are normally open.

The service water pump can therefore obtain water through four separate intakes each equipped with means to prevent debris from entering the pumps, and each capable of supplying all the water required for the service water pumps. Even if the main circulating pump intake were 90% blocked, that intake alone would be capable of supplying all water required for the service water pumps at design conditions.

The intake structure is designed as Seismic Class I, and is therefore not subject to collapse under earthquake loading.

The non-essential loads are the comp· ent cooling heat exchangers and the remaining steam-generator plant services. By manual valve operation, the essential loads can be transferred to the supply line carrying the non-essential loads and vice versa.

The essential loads are those which must be supplied with cooling water immediately in the event of a blackout and/or loss-of-coolant accident. The cooling water for these loads is supplied by the nuclear service water header. The non-essential loads are those which are supplied with cooling water from the conventional service water header by manually starting a service water pump when required following a loss-of-coolant accident.

During normal operation, the essential loads are supplied by one of the three pumps available. The non-essential loads are supplied by two of the three pumps provided.

Following a simultaneous incident and blackout, the cooling water requirement for all five fan cooling units and the other essential loads can be supplied by any two of the three service water pumps on the header designated to supply the nuclear and essential secondary load supply lines. Any two of these three pumps can be powered by the emergency diesels as described in Section 8. These emergency powered pumps are those necessary and sufficient to meet blackout and emergency conditions. Either one of the two sets of three pumps can be placed on the diesel starting logic.

The containment ventilation cooling units are supplied by individual lines from the containment service water header. Each inlet line is provided with a manual shutoff valve and drain valve. Similarly, each discharge line from the cooler is provided with a manual shutoff valve. This allows each cooler to be isolated individually for leak testing of the system or to be drained and maintained open to the atmosphere during the integrated leak tests of the containment. The ventilation cooler discharge lines will be monitored for radioactivity by routing a small bypass flow from each through redundant radiation monitors. Upon indication of radioactivity in the effluent, each cooler discharge line would be monitored individually to locate the defective cooling coil. However, since the cooling coils and service water lines are completely closed inside the containment, no contaminated leakage is expected into these units. The service water system pressure at locations inside the containment is below the containment design pressure of 47 psig.

During no mal plant operation, flow through the cooling units will be throttled for containment temperature contro.. purposes by a valve on the common discharge header from the cooling units. Two independe..., full flow isolation valve open automatically in the event of a high conta'nment pressure signal to by-pass the control valve. Both valves fail in the open position upon loss of air pressure and either valve is capable of passing the full flow required for all five fan cooling units.

Should there be a failure in the piping or valves at the header supplying water to the containment cooling coils, one of the two series isolation valves in the center of the header can be manually closed and service will continue on the side of the header opposite the failure. The supply line attached to this side of the header now supplies the essential loads, whether or not it did so before the failure.

Likewise, operation of at least one component cooling heat exchanger is assured despite the failure of any single active or passive component in the system from the service water pumps to the heat exchangers themselves.

Following a simultaneous incident and blackout, the component cooling heat exchangers are not needed during the injection phase. thus they are normally fed from th -essential supply line. At the beginning of the recirculation phase at least one .. ponent cooling heat exchanger is placed in service by the start-up of one of the service water pumps on the non-essential header.

The emergency diesel-driven generator units are supplied with cooling water from the essential supply line on a continuous basis. One of the two parallel modulating control valves in the common discharge line from the diesel coolers is flow controlled during normal operation, and on a high conta..nment building pressure signal, both valves open fully to insure a sufficient supply of cooling water to the diesels. The inlet valving is arranged so that each of the three diesels can be served by either of the supply lines and, furthermore, the failure of single passive or active component will not result in the loss of all diesel power.

Design Evaluation

The non-essential service water system is not required for the maintenance of plant safety immediately following an accident. The essential portion of the service water system is designed to prevent the single failure of any active component used during this injection p[illegible] the Safety Injection System (Section 6.2) or the single failure of an [illegible]e or inactive component of the remaining engineered safety features from [illegible]ing the ability of the engineered safety features to perform their function in the event of an accident.

Sufficient pump capacity is included to provide design service water flow under all conditions and the headers are arranged in such a way that even loss of a complete header does not jeopardize plant safety.

Tests and Inspections

Each service water pump will undergo a hydrostatic test in the shop in which all wetted parts will be subjected to a hydrostatic pressure of one and one-half times the shut-off head of the pump. In addition, the normal capacity vs. head tests will be made on each pump.

All valves in the service water system will undergo a shop hydrostatic test of 250 psi on the body and 175 psi on the seat. Service water system design pressure is 150 psig. 14

All service water piping will be hydrostatically tested in the field at 225 psig or one and one-half times design. The welds in shop fabricated service water piping will be liquid penetrant or magnetic particle inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII.

Electrical components of the service water system are tested periodically.

Supplement 14
8/70

TABLE 9.6-1

ESSENTIAL SERVICE WATER REQUIREMENTS @ 75°F

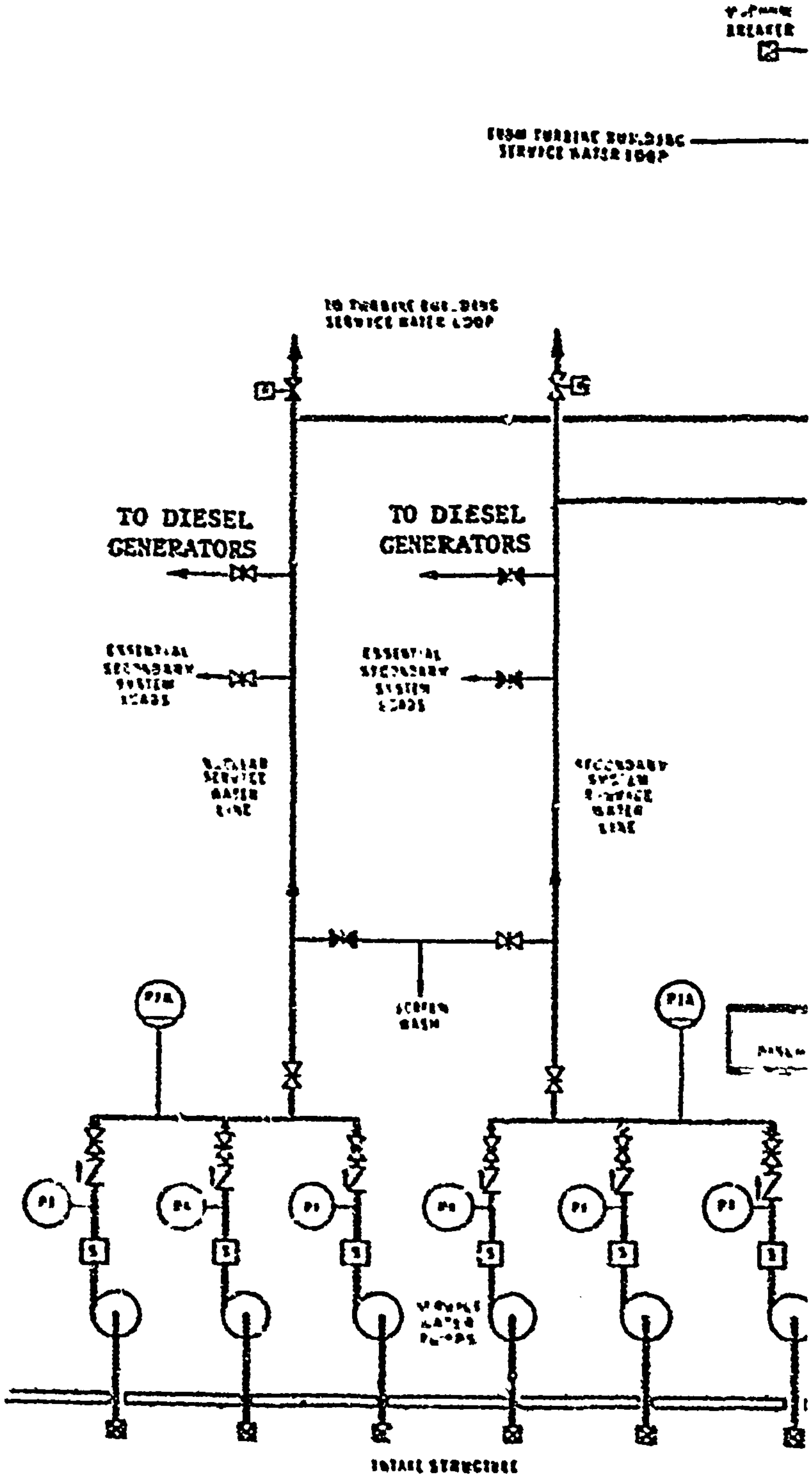
	Service (Number)	Flow Each (GPM)	Normal Flow Total	MCA Flow 1st Hr.	MCA Flow After 1st. Hr.
	Containment cooling coils (5)	2,000 MCA	1,160	10,000 (1)	6,000 (1)
13	Component coolers (2) *	7,000*	7,000*	-	5,000*
	Diesel generators (3)	400	1,200	1,200	200
	Turbine oil coolers (2)	1,100	1,100	1,100	-
	Seal oil coolers and Steam generator feed pump oil coolers	220	220	200	-
	Radiation sample cooler	80	80	80	80
	Air compressor heat exchanger	65	65	65	65
	Service water pump strainer blowdown	100	100	100	100
13	Total Flows	10,965	10,925	12,745	1[illegible],445

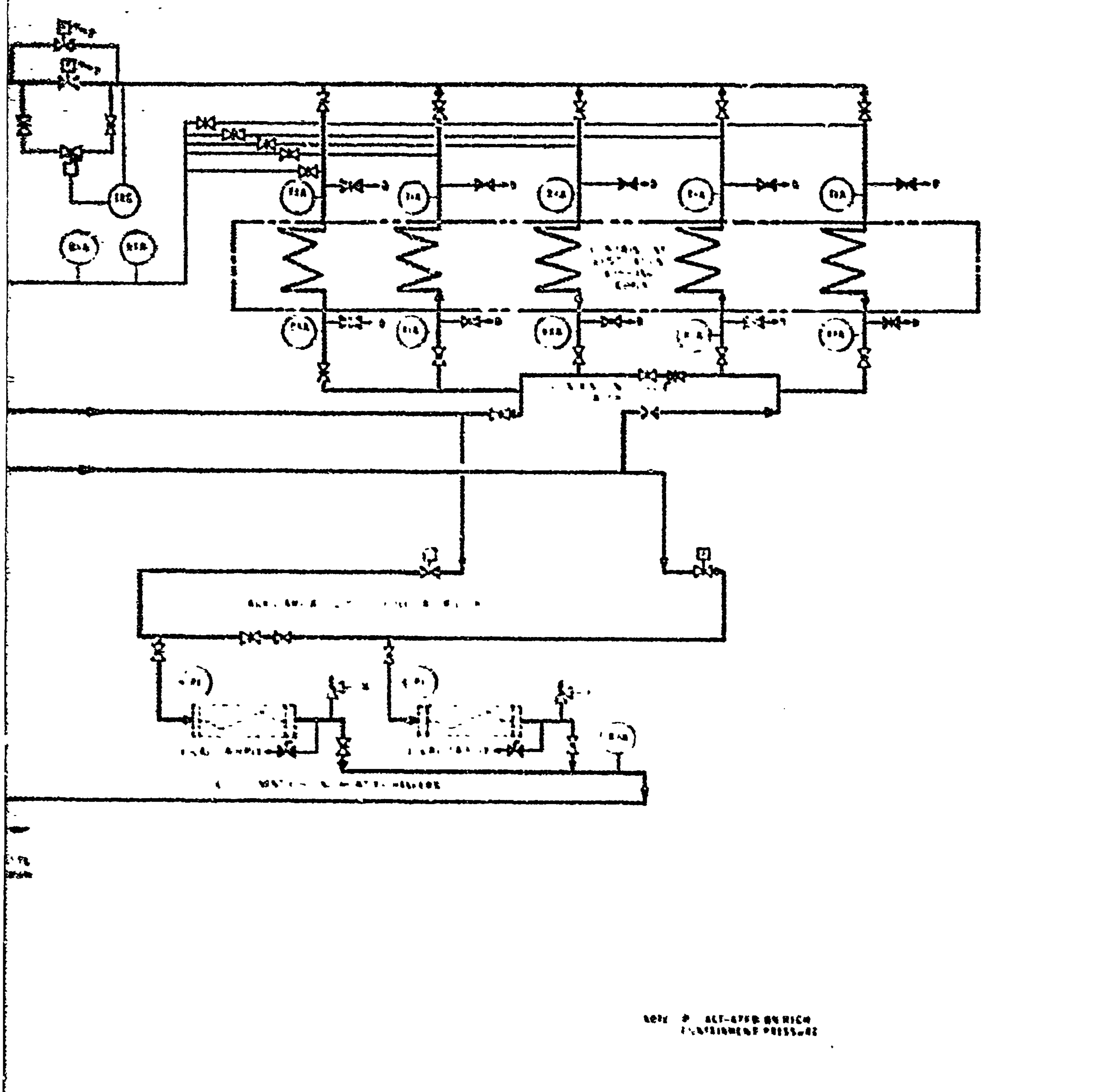
Item marked * is supplied from other header of service water system.

	Normal Flow	MCA Flow 1st Hr.	MCA Flow After 1st Hr.
Total flows for essential service header	3,925	12,745	7,445
No. of pumps required	1	2	2
Required pump capacity	3,925	6,373	3,723

The pump will run out on their curve to meet the flows.

(1) Only 6,000 gpm required after isolation of 2 coils and fans: only 3 fans are required.


TO DIESEL GENERATORS
TO DIESEL GENERATORS



Service Water System
Figure 9.6-1

Name and Function	Professional Education and Special Training	Avail. Class.
Albert C. Hamilton, Asst. Medical Director, Medical Dept., Con Edison	Doctor of Medicine Sup-rvisor of Medical Services for Indian Point since 1962. Fellow, Industrial Medical Association	Readi able persc medic
Roy E. Albert, Medical Consultant	Doctor of Medicine Institute of Environmental Medicine, Sterling Forest, N. Y. Professor of Environmental Medicine, full time. N. Y. University Medical Center	R adi able perso medic
Manfred Blum Physician	Doctor of Medicine Insturctor of Medicine New York University School of Medicine. Supervisor, Environmental Radiation Laboratory New York University Medical Center Physician Incharge Diagnostic Isotope Laboratory, University Hospital, New York	Readi avail site medic

.E 11.2-13 (Continued)

y tion	Provisions for Meeting Cost	Hours of On-Site Duty	Off-Site Availability Location and Distance and Distance from Site	Hospital Staff Appointment
ail- ite	Employee of Con Edison		Around-the-clock Con Edison 4 Irving Place N.Y. C., N. Y. (40 miles)	
ail- ite	Fee for Services		Around-the-clock, Institute of Environmental Medicine, Sterling Forest, N. Y. (15 miles)	Attending Physician, University Hospital
off- nnel	Fee for Services		Around-the-clock New York University Medical Center New York City, N.Y. (40 miles)	Assistant visiting Physician attending Physician Endocurine Clinic Bellevue Hospital, New York. Clinical Assistant Professor University Hospital. Member, Radiation Casualty Team, New York University Medical Center, University Hospital.

T

Name and Function	Professional Education and Special Training	Avail Class
Marcus A. Rothschild Physician	Doctor of Medicine Instructor of Medicine New York University School of Medicine Assistant Professor Medicine, New York University School of Medicine, New York, N.Y.	Readi able off-s perso medic
Phillip Braunstein Physician	Doctc of Medicine Insturctor in Radiology New York Medical College Assistant Professor Radiology New York University Medical Center	Readi able perso medic
Joseph G. Hernberg	Doctor of Medicine Assistant Professor Radiology New York University Medical Center	Readi able perso

l.2-13 (Continued)

y ion	Provisions for Meeting Cost	Hours of On-Site Duty	Off-Site Availability Location and Distance and Distance from Site	Hospital Staff Appointment
il-	Fee for Services		Around-the-clock New York University Medical Center University Hospital New York City, N.Y. (40 miles)	Chief, Radioisotope Service, V.A. Hospital New York, N.Y., Member, Radiation Casualty Team, New York University Medical Center, University Hospital.
il- ite	Fee for Services		Around-the-clock New York University Medical Center University Hospital New York City, N.Y., (40 miles)	Assistnat Attending in Radiology Flower and 5th Avenue, and Metropoltian Hospitals, Member, Radiation Casualty Team, New York University Medical Center, University Hospital.
il- ite edical	Fee for		Around-the-clock New York University Medical Center University Hospital, New York City N.Y. (40 miles)	Member, Radiation Casualty Team, New York University, University Hospital.

TABLE

Name and Function	Professional Education and Special Training	Avail Class
Ramesn Chandra Physicist	Ph.D. Nuclear Physics Assistant Professor, Radiology, Responsible for Physics in Diagnostic Isotope Department in New York University and Bellevue Hospital	Readi able profe
Colin G. Orton Physicist	Ph.D. Medical Physics Senior Physicist, Radiology Department, New York University Medical Center Assistant Professor, Radiology, New York University Member, Health Physics Society	Readi able profe
Christopher H. Marshall Physicist	Ph.D. Radiation Physics and Biology Assistant Professor, Radiology, Radiation Safety Officer, New York University Medical Center	Readi able profe

-13 (Continued)

ity ation	Provisions for Meeting Cost	Hours of On-Site Duty	Off-Site Availability Location and Distance and Distance from Site	Hospital Staff Appointment
vail- site tal			Round-the-clock New York University Medical Center, New York City, N.Y. (40 miles)	Member, Radiation Casualty Team New York University Medical Center, University Hospital.
vail- site tal			Round-the-clock New York University Medical Center University Hospital New York City, N.Y. (40 miles)	Member, Radiation Casualty Team New York University Medical Center, University Hospital
vail- site tal			Round-the-clock New York University Medical Center University Hospital New York City, N.Y. (40 miles)	Member, Radiation Casualty Team, New York University Medical Center, University Hospital

9.6.2 FIRE PROTECTION SYSTEM

Design Basis

Fire Protection

Criterion: The facility is designed so that the probability of fires and explosions and the potential consequences of such events does not rusult in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features. (GDC 3)

Fire prevention in all areas of the nuclear-electric plant is provided by structure and component design which optimizes the containment of combustible materials and maintains exposed combustible materials below their ignition temperature in the design atmosphere. Fire control requires the capability to isolate or remove fuel from an igniting source, or to reduce the combustibles temperature below the ignition point, or to exclude the oxidant, and preferably, to provide a combination of the three basic control means. The latter two means are fulfilled by providing fixed or portable fire fighting equipment of capacities proportional to the energy that might credibly be released by fire.

This station is designed on the basis of limiting the use of combustible materials in construction and of using fire-resistant materials to the greatest extent possible.

The fire protection system is designed to achieve the following objectives:

a. Provide automatic fire detection in those areas where the fire danger is greatest.

b. Provide fire extinguishment by fixed systems of water and foam, actuated automatically or manually in those areas where the fire danger is greatest.

c. Provide manually operated fire extinguishing equipment, including fire hose reels and portable equipment, including CO_2, dry chemical and pressurized water of the wheeled and hand carried type for use by personnel.

d. The fire protection system is designed to equal or exceed the standards of the National Fire Protection Association.

Description of Fire Protection System

The fire protection system for Indian Point Generating Station, Unit No. 2, represents an extension of the fire protection system for Unit No. 1 of the same station, as shown in Figure 9.6.2-1, Unit No. 1 Piping Diagram Fire Protection System, Figure 9.6.2-2, Yard Fire Protection Piping.

The source for all fire protection is city water, which is taken either directly from the 1,500,000 gallon water storage tank or from the 16 in. mains connected to the Catskill Water Supply. The minimum volume in the city water storage tank is maintained at 1,435,000 gallons by an automatic level control.

There is a high-pressure hydrant system which is sustained by two 1500 gpm fire main booster pumps and is maintained by two 100 gpm fire main pressure maintenance pumps. One 1500 gpm pump starts automatically when the pressure drops to 125 psi and the other starts automatically when the pressure drops to 120 psi. Both are activated by pressure switches and continue to run until shut off manually. Starting of each pump will be annuciated by an alarm on the supervisory panel board alarm section. If the pressure drops to 106 psi, an emergency alarm sounds.

One 100 gpm pump starts automatically when the pressure drops to 135 psi and should the pressure continue to drop the other 100 gpm pump starts automatically at 130 psi. When the pressure returns to 146 psi the standby pump will stop automatically. When the pressure goes up to 151 psi the normal running pump shuts off automatically.

Both the two-125HP motors for the 1500 gpm pumps and the two-20HP motors for the 100 gpm pumps are fed from separate electrical light and power bus sections of Unit No. 1; bus sections No. 3 and No. 4.

The standpipe system is supplied from the high pressure main and is backed by gravity feed from the 20,000 gallon roof tank. 15,000 gallons of the roof tank capacity is reserved for fire system use. The tank is supplied with city water by two fill pumps.

As shown in Figure 9.6.2-2 Yard Fire Protection Piping, the existing fire protection system for Unit No. 1 is extended to protect Unit No. 2 in the following ways:

1. Addition of 15 hose reels in strategic spots in the turbine building, using 75 ft lengths of 1-1/2 in. hose.

2. Addition of deluge valves for following equipment:

   a) Hydrogen seal oil unit
   b) Boiler feed pump oil console
   c) Lube oil storage tank
   d) Lube oil reservoir
   e) Unit auxiliary transformer
   f) Main transformers, No. 21 and No. 22
   g) Station auxiliary transformer

3. Addition of eight yard hydrants.

Fire protection is provided to the exterior plant areas by yard fire hydrants as previously enumerated. The yard piping, as shown in Figure 9.6.2-2, Yard Fire Protection Piping, consists of an underground extension to the existing fire protection system. Valved branches from this underground system supply interior fire protection systems in the enclosed sections of the plant, including the turbine building. Sectionalizing valves in the yard piping system are provided to permit partial pipe line isolation without interruption of service to the entire system during maintenance or future extension of facilities.

The station service and main transformers are protected by separate water spray systems, automatically operated by heat actuated devices. The water supply for these systems is provided from the main fire protection header in the turbine building. Each system includes a deluge valve, heat actuated fire protection devices, supervisory air system, spray nozzles and a fixed piping system.

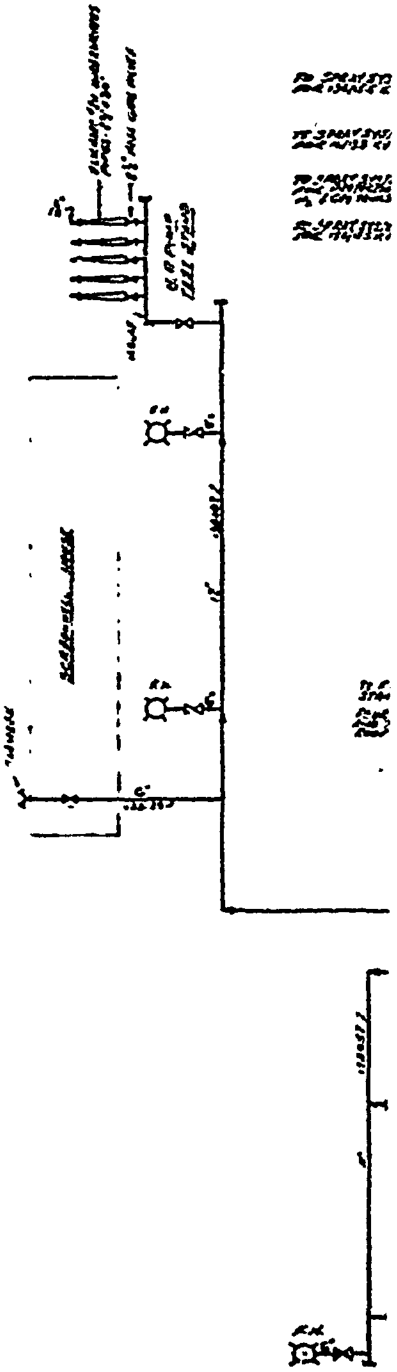
Since the reactor containment has little combustible equipment, no special fire protection systems are required within the structure. The bearing oil systems for the reactor coolant pumps are self-contained. In view of the small amount of combustible materials in the containment, the maximum surface temperature of equipment, and appropriate techniques for controlling fires, no remotely operated or special systems are deemed necessary. Portable dry chemical extinguishers are provided for use during maintenance periods.

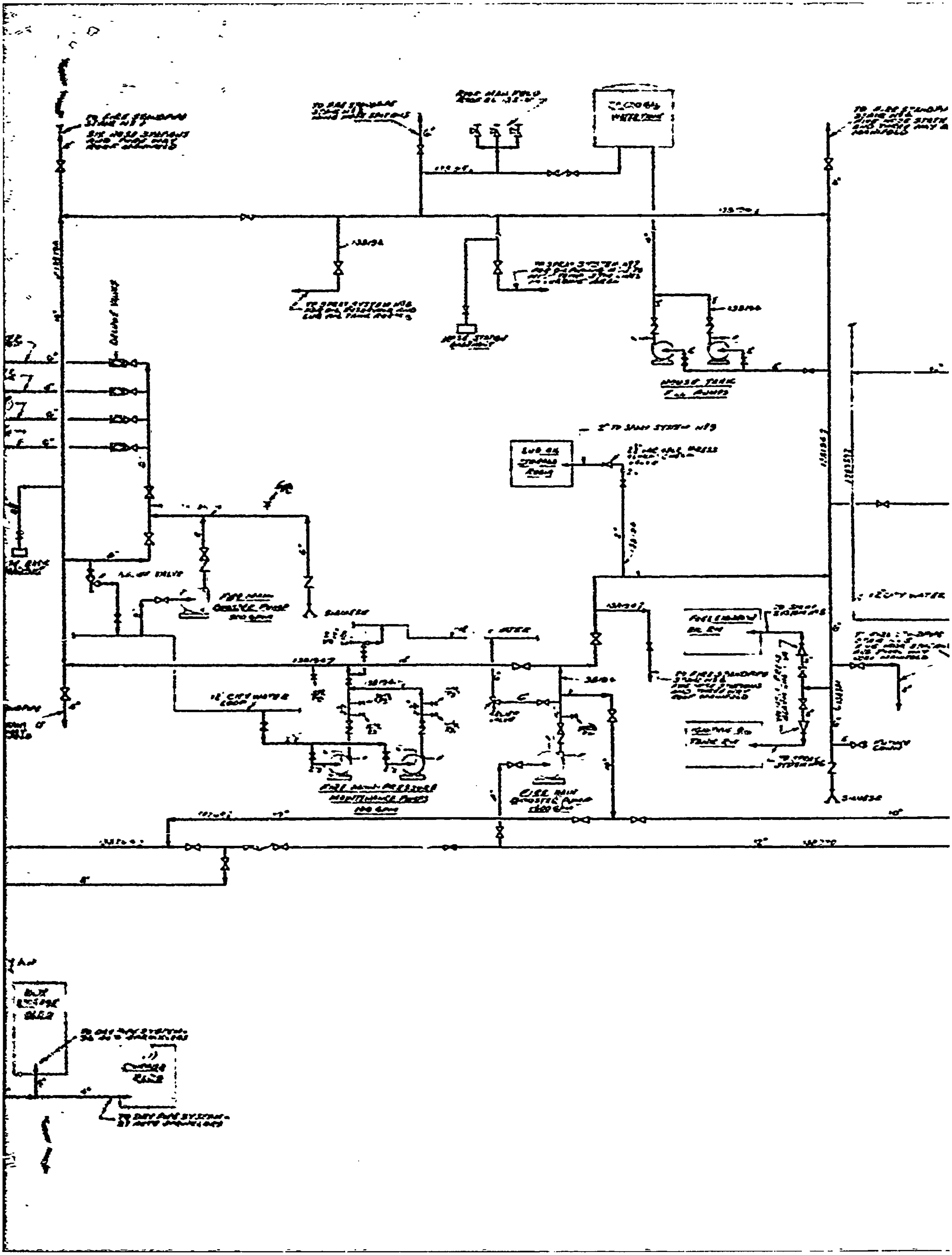
The primary auxiliary building is also provided with dry chemical, portable fire extinguishers in order to minimize the spread of radioactive contamination in the event of fire.

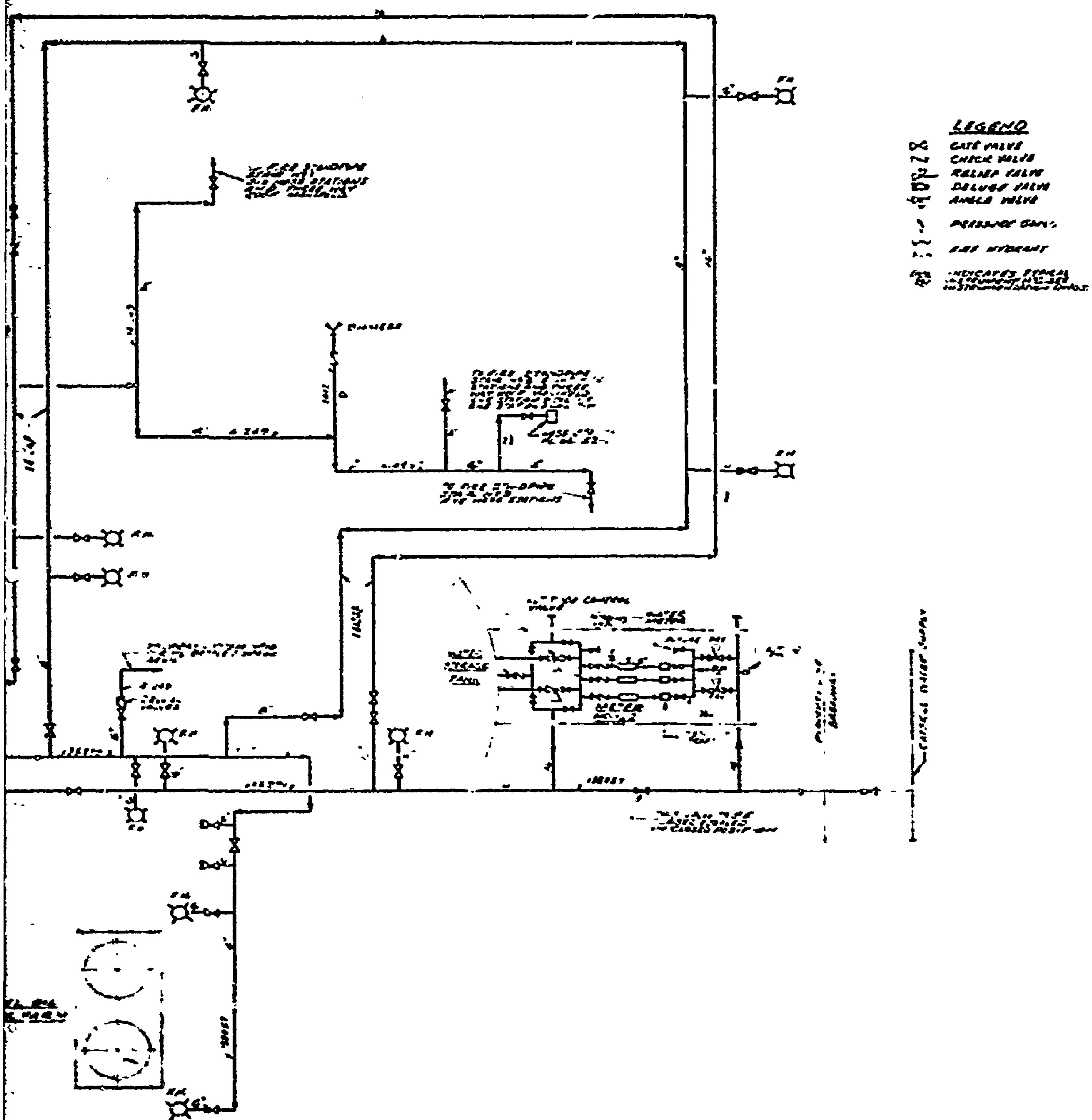
Fire protection for the fuel storage building consists of dry chemical extinguishers.

A main fire protection header in the turbine building supplies the turbine building piping loop and the deluge systems previously enumerated. This header is supplied with water through an extension of the fire protection system of Unit No. 1.

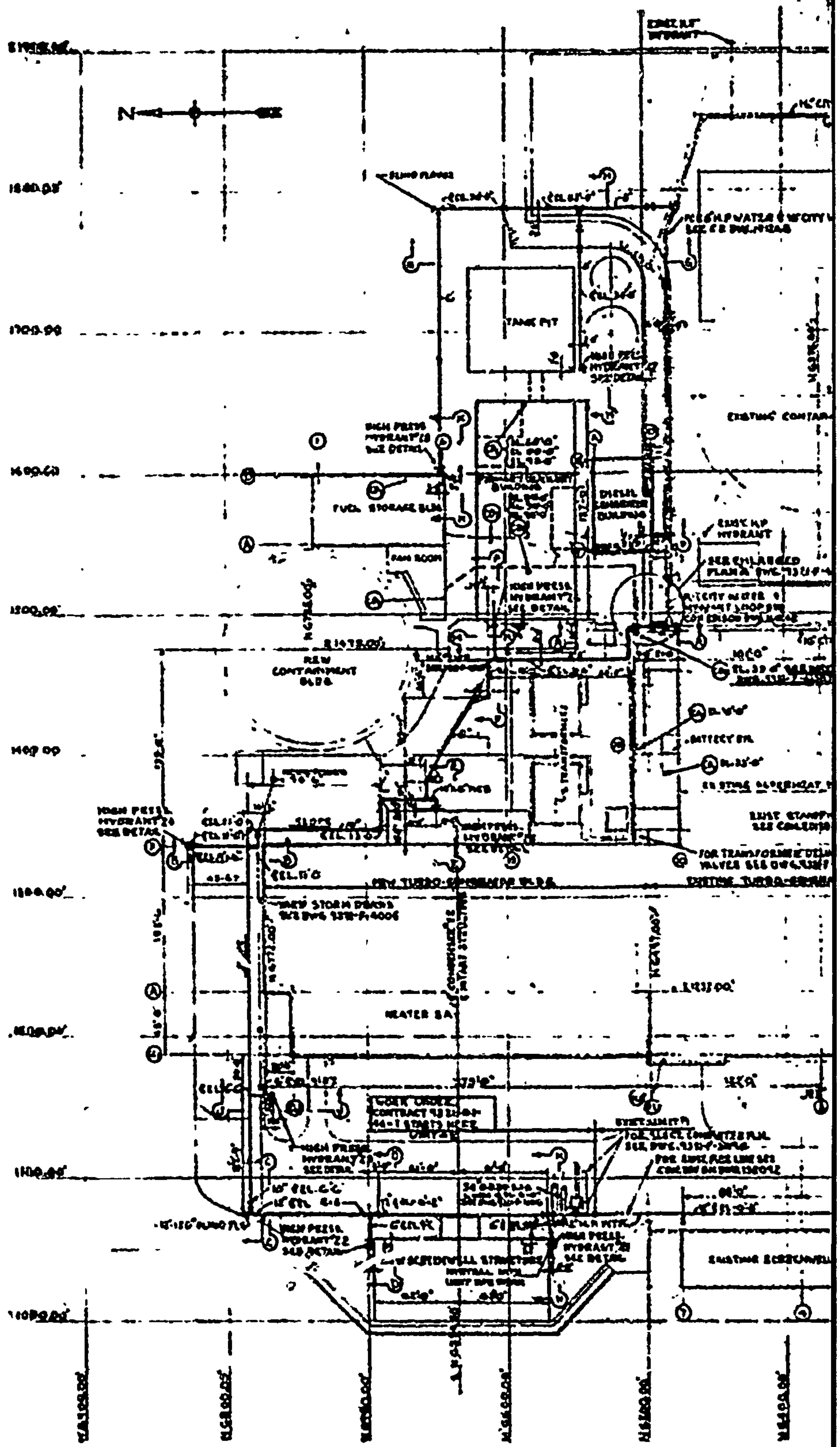
Portable fire extinguishers of various types are distributed within the plant, including turbine hall.




SIAMESE

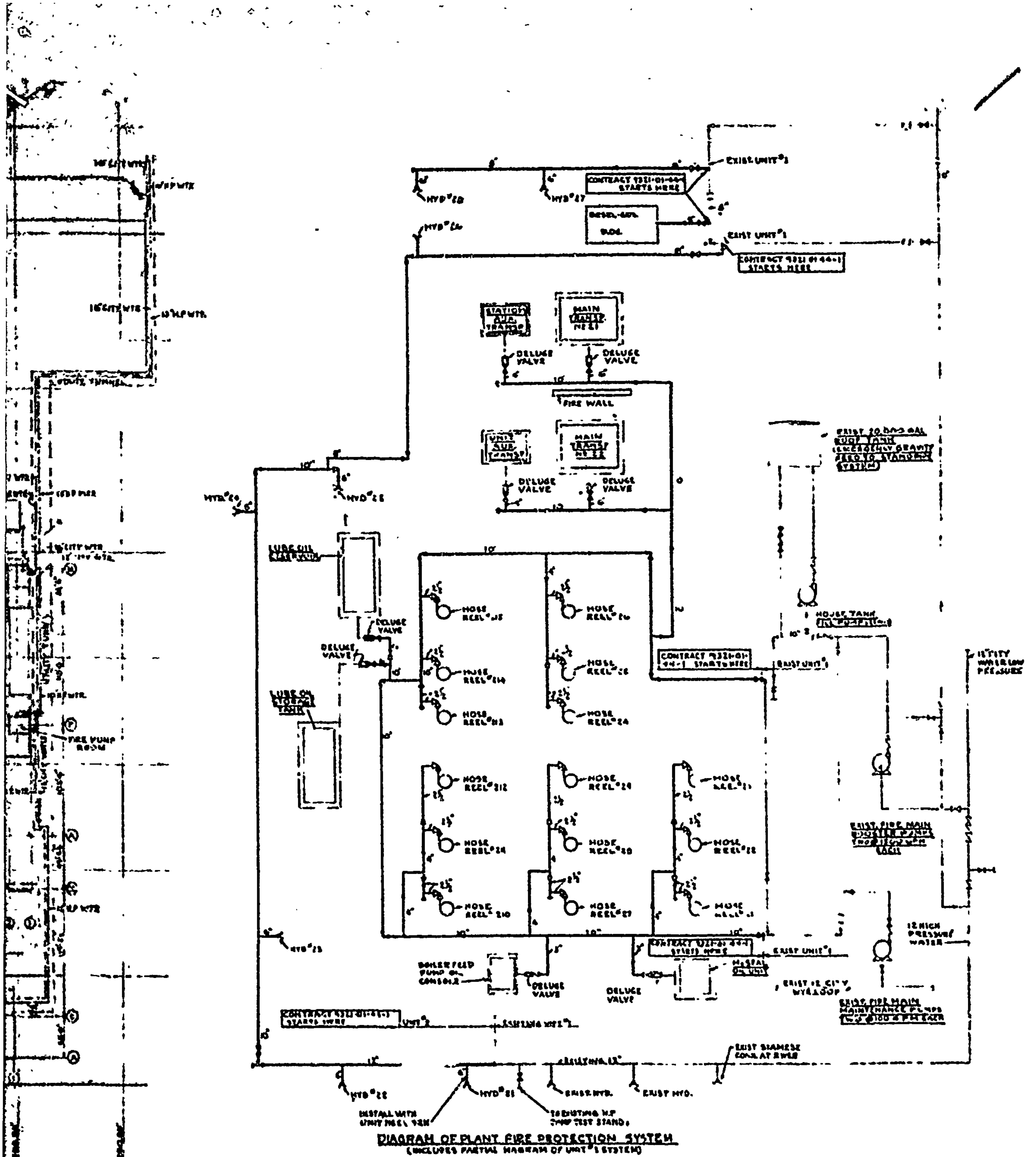


Indian Point Unit No. 1 Piping Diagram Fire Protection System
Figure 9.6.2-1



TANK PIT
NEW CONTAINMENT BLDG
FUEL STORAGE BLDG
NEW TURBO-GENERATOR BLDG

PLOT PLAN



Yard Fire Protection System
Figure 9.6.2-2

9.6.3 COMPRESSED AIR SYSTEMS

Instrument Air System

The instrument air system is designed such that the instrument air shall be available under all operating conditions, all essential systems requiring air during or after an accident shall be self supporting, all controls shall fail to a safe position on loss of power, and after an accident, the air system shall be re-established.

To meet the design criteria the following design features have been incorporated. Duplicate compressors are installed with duplicate dryers and filters throughout. In addition, a backup supply is taken from the station air system. Those items essential for safe operation and safe cooldown core provided with air reserves or gas bottles. These supplies will enable the equipment to fuction in a safe manner until the air supply is re-established. The controls are specified to fail to a safe position on loss of air or electrical power. The compressors and essential sections of the air supply system have been designed to operate after seismic shock. The compressors are located in a seismic Class I building. The non-essential header has a flow restrictor in it to limit flow in a break to the capacity of one compressor.

The system is served by two 225 scfm Worthington teflon-ring compressors. The compressors, filters and air dryers are located on the ground floor of the control building, a Class I, seismic structure.

Each compressor discharges into a common air receiver. The instrument air system is backed-up from the Unit No. 2 service air system.

The service air backup is filtered through an oil vapor and droplet type of equipment before entering the instrument air system. The instrument air receiver outlet enters one of two full capacity 225 scfm refrigeration type dryers which reduces the dewpoint from saturation to plus 35F. This system is used for all indoor services where it is anticipated that the ambient temperature will not go below 50°F. Those services which are used for outdoor instrumentation and for lines which leave the control house and/or turbine room and enter the yard area to serve the primary auxiliary building and containment building are served through a dessicant type dryer which further reduces the dewpoint to minus 40F, in order to be compatible with the lowest expected outdoor temperature. This dessicant dryer is rated at 150 scfm and is a dual tower type dryer. However, in the event that the transfer mechanism would fail during cycling of the dryer, a non-regenerative dessicant dryer is provided to automatically bypass this dryer so as to provide a four-hour continued air supply to pertinent Class I services at minus 40F dewpoint. An air filter set is provided on the discharge of this dryer in order to filter out any dessicant which may be carried over by possible flotation of the bed. A moisture detector is provided on the common discharge of the refrigerant dryers to notify the control room operator in the event of malfunction of the refrigerant dryer.

The instrument air compressors may be operated in two modes. One mode provides for the standby compressor to come on automatically in the event of low pressure in the common air receiver. The other mode of operation provides for simultaneous running of both compressors in order to provide continuity of service to Class I areas in the event of outage of the conventional plant instrument air header. A restriction orifice is provide so as to limit the flow to the capacity of one instrument air compressor into a possible line break in the secondary plant air header. Upon notification of this break, a valve is provided to isolate the secondary plant and prevent pressure decay in the primary plant header.

Valve position lights in the control room advise the operator as to the status of all emergency bypass or make-up control valves. A manual local reset solenoid valve is provided at each emergency valve so as to require the attention of an operator at the equipment. All air and oil filters are dual type to provide maintenance during operation.

Service Air System

The service air system is supplied by a Worthington Corporation two-stage 625 SCFM compressor located in the turbine room. The air is discharged through an aftercooler and moisture separator at 100 psig and 110F. The maximum discharge pressure will be 125 psig. The cooling water for the aftercooler pressure will be 125 psig. The cooling water for the aftercooler and compressor jacket is supplied from a closed cooling water system which contains treated city water.

The compressor is controlled by the solenoid underloaded valves which are energized through a pressure switch arrangement in automatic or hand (manual) modes. In the automatic mode, the compressor will run in single or two-stage operation and unload at a predetermined pressure setting with motor and compressor stopped. In manual mode, the compressor will start and stop at predetermined pressure settings; but the motor continues to run. High water and high air temperature switches are connected to the control annunciator.

The service air system furnishes compressed air for pneumatic tools and miscellaneous cleaning and maintenance purposes throughout the secondary and primary plants Alcohol pots are provided in the air lines passing outside the turbine building.

This system is backed-up by the Unit No. 1 service air system through a manually operated valve interconnection to the Unit No. 2 air receiver. The size of the connection is equal to the Unit No. 2 supply pipe.

The system also provides for an automatic emergency supply to the Unit No. 2 instrument air system through an oil vapor filtering arrangement. In addition, an automatic emergency supply is supplied to the containment building weld channel and penetration pressurization system. The air is first filtered and then dried to -40F dew point.

9.6.4 HEATING SYSTEM

The heating system for Unit No. 2 represents an extention of the heating system for Unit No. 1 of the Indian Point Generating Station. Three shop assembled package boilers, each rated at 40,000 lb of steam per hour at 265 psig, are presently installed. Two of these boilers are capable of supplying steam for miscellaneous services when the reactor system of Unit No. 1 is not in operation.

Two additional package boilers, each rated at 50,000 lb of steam per hour, are being installed to supply steam for Unit No. 2 and to be interconnected with the distribution header of the three boilers for Unit No. 1. The 10-inch main steam header from these new boilers links the existing steam header to Unit No. 2 and also to Unit No. 3, so that output from any of the five package boilers may be made available for the heating requirements of Unit No. 1, Unit No. 2 or Unit No. 3.

With respect to Unit No. 2 there are separate piping circuits for the unit heater steam supply to the east side and the west side of the turbine hall, including the heater bay. An extension from the circuit to the east side of the turbine hall serves the turbine oil storage tanks for both clean and dirty oil storage. Other heating services extend to the fan room, the fuel storage building, the containment building, the primary auxiliary building, the tank pit, the primary water storage tank and the refueling water storage tank.

Provision is made for the following heating services:

1) Containment Building

   a) Four steam unit heaters
   b) Valves with hose bibs for maintenance purposes

2) Primary Auxiliary Building

   a) Electric strip heaters
   b) Steam unit heaters
   c) Air makeup steam tempering units

3) Purge System Containment Building

   a) Air makeup steam tempering units

4) Fuel Storage Building

   a) Steam unit heaters for standby heating
   b) Air makeup steam tempering units

5) Waste Tank Storage Pit

   a) Air makeup steam tempering units

6) Fan Room

   a) One steam unit heater

9.6.5 PLANT COMMUNICATIONS SYSTEMS

A Gai-tronic Communication System provides separate and independent page and party-line communications channels. The page channel is regulated from the control room and is used to call personnel as well as to issue plant-wide instructions over the plant area loudspeakers. The party-line system is common to all handset stations and consists of two channels. One channel is common to both the nuclear and conventional areas of the plant, while the remaining channel serves the nuclear area only. Up to five parties can communicate with each other at one time over a common party-line channel.

A call from one plant area to another is initiated by a page call using a plant area handset. The handset has a receiver and speaker and provides an open communications line to the control room. A page call using a handset is transmitted via one of the party-line channels to the control room and is heard over the control room loudspeaker. An operator in the control room uses the page channel to alert personnel in the desired plant area to contact the person who initiated the call. Handsets are used to communicate from one plant area to another over the party-line system. All communications must go through the control room and are monitored over a control room loudspeaker.

Both the page channel and the party-line channel common to both the nuclear and conventional areas of the plant can be merged with a page channel and a party-line channel of unit number 1, respectively, from the unit number 2 control desk for inter-plant communication.

A Bell dial phone is located near various handsets and is used to supplement the Gai-tronic Communication System.

For a period of inaccessibility to the control room during a hot shutdown, there are Gai-tronic handsets located at each of the vital areas, external to the control room, that are required to bring the plant to a hot shutdown. Both party-line channels are available for communications.

9.7 EQUIPMENT AND SYSTEM DECONTAMINATION

9.7.1 DESIGN BASIS

Activity outside the core could result from fission products from defective fuel elements, fission products from tramp uranium left on the cladding in small quantities during fabrication, products of n - γ or n - p reactions on the water or impurities in the water, and activated corrosion products. Fission products in the reactor coolant associated with normal plant operation and tramp uranium are generally removed with the coolant or in subsequent flushing of the system to be decontaminated. The products of water activiation are not long lived and may be removed by natural decay during reactor cooldown and subsequent flushing procedures. Activated corrosion products are the primary source of the remaining activity.

The corrosion products contain radioisotopes from the reactor coolant which have been absorbed on or have diffused into the oxide film. The oxide film, essentially magnetite (Fe_3O_4) with oxides of other metals including Cr and Ni, can be removed by chemical means presently used in industry.

Water from the primary coolant system and the spent fuel pit is the primary potential source of contamination outside of the corrosion film of the primary coolant system. The contamination could be spread by various means when access is required. Contact while working on primary system components could result in contamination of the equipment, tools and clothing of the personnel involved in the maintenance. Also, leakage from the system during operation or spillage during maintenance could contaminate the immediate areas and could contribute to the contamination of the equipment, tools, and clothing.

9.7.2 METHODS OF DECONTAMINATION

Surface contaminates which are found on equipment in the primary system and the spent fuel pit that are in contact with the water are removed by conventional techniques of flushing and scrubbing as required. Tools are decontaminated by flushing and scrubbing since the contaminates are generally on the surface only of non-porous materials. Personnel and their clothing are decontaminated according to the standard health physics requirements.

Those areas of the plant which are susceptable to spillage of radioactive fluids are painted with a sealant to facilitate decontamination that may be required. Generally washing and flushing of the surfaces are sufficient to remove any radioactivity present.

The corrosion films generally are tightly adhering surface contaminates, and must be removed by chemical processes. The removal of these films is generally done with the aid of commercial vendors who provide both services and formulations. Since decontamination experience with reactors is continually being gained, specific procedures may change for each decontamination case. For corrosion films, the APAC (alkaline permanganate-diammonium citrate) treatment, or an organic acid variation of the APAC treatment is considered to be the most effective for removal.

Portable components may be cleaned with a combina ion of chemical and ultrasonic methods if required.

9.7.3 DECONTAMINATION FACILITIES

Decontamination facilities on site consist of an equipment pit and a cask pit located adjacent to the spent fuel storage pit. In the stainless steel lined equipment pit, fuel handling tools and other tools can be cleaned and decontaminated.

In the cask decontamination pit, the outside surfaces of the shipping casks are decontaminated, if required, by using steam, water detergent solutions, and manual scrubbing to the extent required. When the outside of the casks are decontaminated, the casks are removed by the auxiliary building crane and hauled away.

For the personnel, a decontamination shower and washroom is located adjacent to the Radiation Control Area (RCA) locker room. Personnel decontamination kits with instructions for their use are in the RCA locker room.

9.8 PRIMARY AUXILIARY BUILDING VENTILATION SYSTEM

9.8.1 DESIGN BASIS

The Primary Auxiliary Building Ventilation System is designed to accomplish the following:

a) Provide sufficient circulation of filtered air through the various rooms and compartments of the building to remove equipment heat and maintain safe ambient operating temperatures.

b) Control flow direction of airborne radioactivity from low activity areas toward higher activity areas.

c) Provide purging of the building to the plant vent for dispersion to the environment.

The air exhausted by the system is filtered, monitored and diluted so that off-site dose during normal operation will not exceed one-tenth of that permitted by 10 CFR 20.

9.8.2 SYSTEM DESIGN AND OPERATION

The Primary Auxiliary Building Ventilation System (See Figure 5.3-1) is composed of the following systems:

a) Make-up air handling system complete with fan, bypass dampers, filters, heating coils and supply ductwork.

b) Exhaust system complete with fans, ductwork, roughing filters and HEPA filters.

c) Make-up air tempering unit for the waste storage tank pit.

Design parameters for the system components are given in Table 9.8-1.

Branch supply ducts direct make-up air to the various floors at the east end of the building, from where it flows to the rooms and compartments. Air is exhausted from each of the building compartments through ductwork designed to make the supply air sweep across the room as it travels to the room exhaust register. The air then flows to the exhaust fan inlet plenum, and is drawn by the operating exhaust fan through rouging filters and HEPA filters before discharge to the plant vent. The exhaust system has been designed to insure that air flows from the "clean" end of the building through the "hot" areas.

Ventilating air exhausted from the waste storage tank pit is arranged to by-pass the primary auxiliary building system and flow directly into the exhaust fan inlet plenum. Makeup air to this area is tempered with a steam heating coil.

There are four fans in the Containment Building Purge System and Primary Auxiliary Building Ventilation System. The two exhaust fans (containment building purge and/or primary auxiliary building exhaust fans 21 & 22) are common to both the Containment Building Purge System and Primary Auxiliary Building Ventilation System. The supply fan in each of the ventilation systems operates only in its individual ventilation system.

The four fans operate in pairs, with the primary auxiliary building supply fan running with either of the exhaust fans, with the other exhaust fan serving as back-up, or the containment building purge supply fan running with either of the exhaust fans with the other as back-up, or with all four fans running simultaneously. The interlocking for the fans is such that in the event of a trip of any of the fans, the corresponding fan in that pair will stop.

The selection of the desired pair or pairs of fans is manual, using a selector switch located on the fan room control panel. All four fans can be started and stopped by the single control switch located on the fan room control panel. Each fan has indicating lights on the fan room control panel and in the main control room. An auto-trip alarm is also provided. In addition, each of the fans have a "jog" pushbutton located on the fan room control panel for testing.

A 50,000 cfm dilution fan is provided for reducing the concentration of radioactivity in the exhaust air discharge through the plant vent. This fan will start automatically upon a high activity signal from the stack gas monitor, and continue to run until manually stopped from a control switch on the fan room control panel. Indicating lights are provided on both the fan room control panel and in the main control room. An auto trip alarm is also provided to indicate that the fan is operating.

TABLE 9.8-1

PRIMARY AUXILIARY BUILDING VENTILATION SYSTEM COMPONENT DATA

System	Units Installed	Units Capacity	Units Required for Normal Operation
***EXHAUST**			
Fans, Standard Conditions	2	53,000 cfm	1
Fan Pressure	-	8.5 in. H_2O	-
Fan Motors	2	100 HP	1
Plenums	2	53,000 cfm	1
Roughing Filters	2	53,000 cfm	1
HEPA Filters	2	53,000 cfm	1
SUPPLY TEMPERING UNIT (PAB)			
Fans, Standard Conditions	1	48,100 cfm	1
Fan Pressure	1	[illegible].5 in. H_2O	1
Fan Motor	1	25 HP	1
Filters	1	48,100 cfm	1
Coils	1	48,100 cfm	1
TEMPERING UNIT (WASTE STORAGE TANK PIT)			
Coil complete with motor-operated dampers	1	5,100 cfm	1

*These two exhaust fans are used interchangeably and/or as backup for:

(1) Ventilation of Primary Auxiliary Building.

(2) Containment Building Purge System.

9.9 CONTROL ROOM VENTILATION SYSTEM

9.9.1 DESIGN BASIS

The Control Room Ventilation System is designed to accomplish the following:

a) Maintain 75 F D.B. and approximately 50% R.H. in the control room.

b) Permit cleanup of airborne particulate radioactivity entering the control room with normal makeup air flow and by infiltration.

9.9.2 SYSTEM DESIGN AND OPERATION

The Control Room Ventilation System is composed of the following equipment:

a) A direct expansion, water-cooled air conditioning unit complete with fan, steam heating coil and roughing filter.

b) A filter unit consisting of case, roughing filter, HEPA filters, charcoal filters and a booster fan with a capacity of approximately 1840 cfm.

c) Duct system complete with dampers and controls to give four (4) system booster fan operating conditions.

The Control Room Ventilation System will operate as follows:

a) Normal Conditions

1) With outside air makeup

Will supply cooling or heating for the control room atmosphere as required, using fresh outside air makeup and with the charcoal filter unit bypassed.

2) With building air makeup

Same as 1), except that pneumatic dampers will be positioned to provide makeup air from the control building interior instead of from outside.

b) Fallout or Accident Conditions

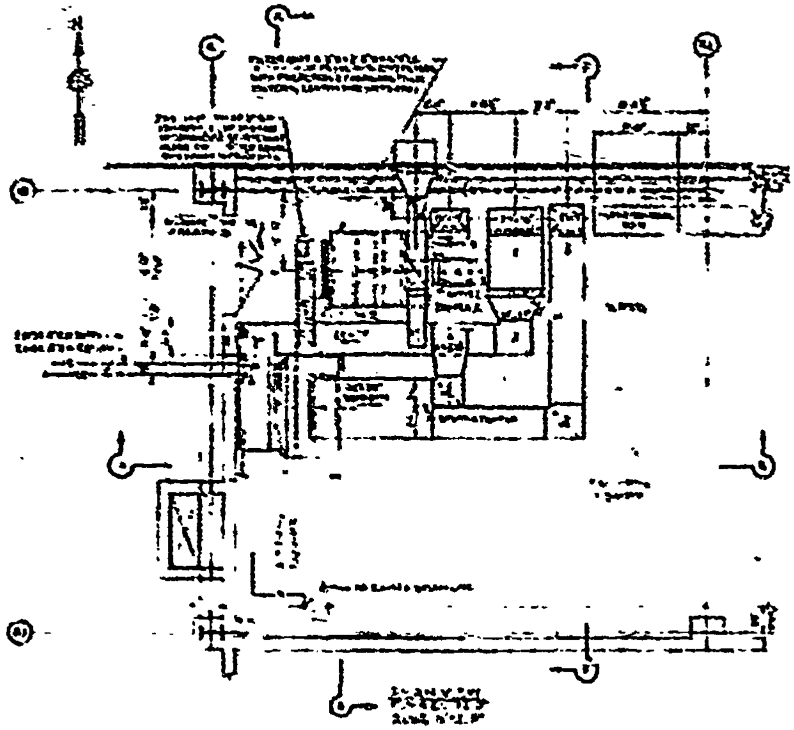
1) With outside air makeup filtered

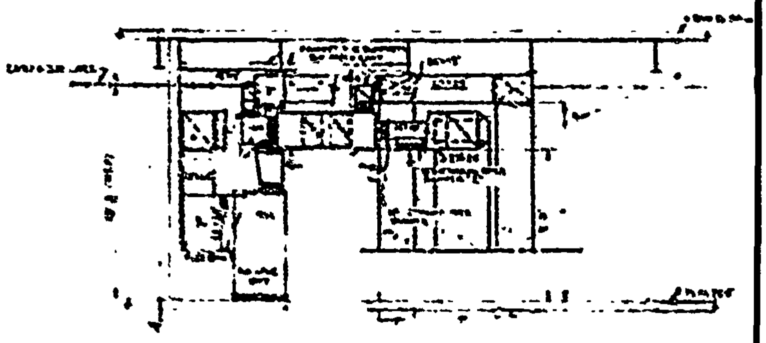
The booster fan will start and pneumatic dampers will be positioned to permit outside air and a portion of recirculated room air to flow through the charcoal filter unit.

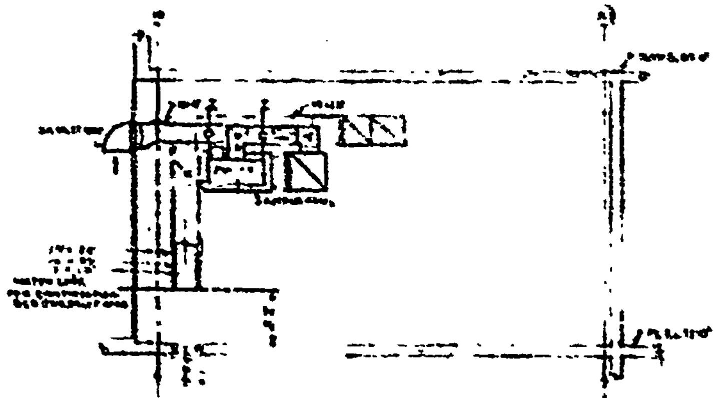
2) With no makeup air

The pneumatic dampers will be positioned to close both the outside air makeup and building air makeup ducts. Approximately 1840 cfm will be directed through the charcoal filter unit to remove airborne particulate radioactivity that may have entered the control room through infiltration.

All these operations can be performed manually from the control room. In addition, in the event of safety injection signal the control room dampers will automatically reposition and start the booster fan to place the charcoal filter unit in service, for system operating mode b)2).







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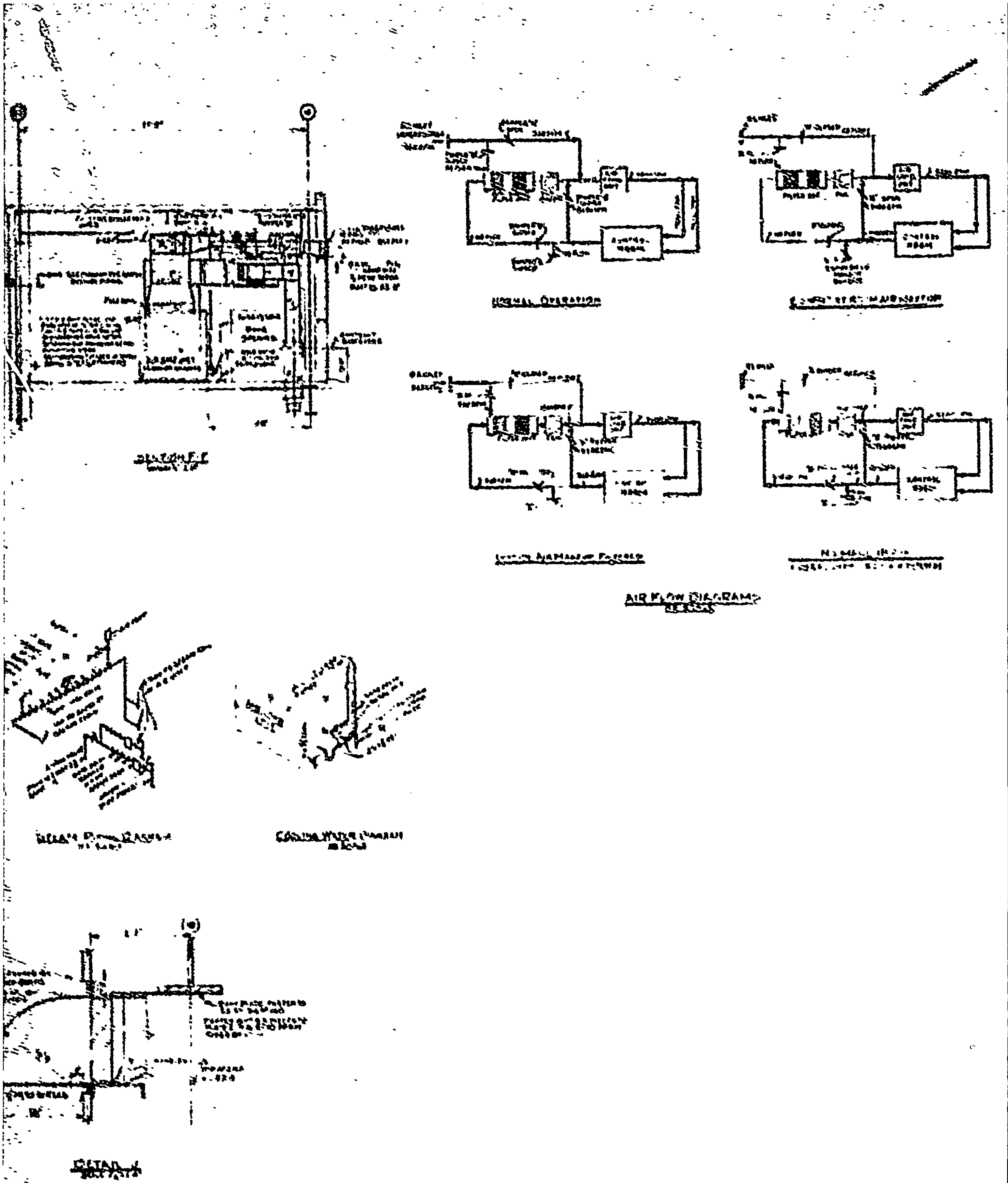

Control Building Heating, Ventilation and Air Conditioning System
Figure 9.4-1

Chap 10

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STEAM AND POWER CONVERSION SYSTEM

10. STEAM AND POWER CONVERSION SYSTEM

10.1 DESIGN BASIS

10.1.1 PERFORMANCE OBJECTIVES

The turbine-generator systems consist of components of conventional design, acceptable for use in large power stations. The equipment is arranged to provide high thermal efficiency with no sacrifice to safety. The component design parameters are given in Table 10.1-1.

The Steam and Feedwater System is designed to remove heat from the reactor coolant in the four steam generators, producing steam for use in the 1,021,793 KW turbine generator. The Steam and Feedwater System can receive and dispose of, in its cooling systems and through atmospheric relief valves, the total heat existent or produced in the Reactor Coolant System following an emergency shutdown of the turbine generator from a full load condition.

All of the equipment in the turbine-generator systems is designed to produce a maximum calculated gross output of 1,068,701 KW, should the Nuclear Steam Supply System prove to have capacity in the future for such operation.

The system design provides means to monitor and restrict radioactivity discharge to normal heat sinks or the environment such that the limits of 10 CFR 20 are not exceeded under normal operating conditions nor in the event of anticipated system malfunctions.

One turbine and two electric driven auxiliary feed pumps are provided to ensure that adequate feedwater is supplied to the steam generators for reactor decay heat removal under all circumstances, including loss of power and normal heat sink. Feedwater flow can be maintained until either, power is restored, or reactor decay heat removal can be accomplished by other systems. Auxiliary feedwater pumps and piping are designed as Class I seismic components.

10.1.2 LOAD CHANGE CAPABILITY

The transmission system directs the power conversion system to provide load changes up to generation step load increases of 10% and ramp increases of 5% per minute within the load range of 15-100% without reactor trip subject to possible xenon limitations late in core life. Similar step and ramp load reductions are possible within the range of 100-15% of full load. The Reactor Coolant System will accept a complete loss of load from full power with reactor trip. In addition, the turbine bypass and steam dump system makes it possible to accept a step load decrease of 50% of full power without reactor trip.

10.1.3 FUNCTIONAL LIMITS

The system design incorporates backup means (power relief and code safety valves) of heat removal under any loss of normal heat sink (e.g., Condenser isolation, circulating water loss of flow) to accommodate reactor shutdown heat rejection requirements. System atmospheric discharges under normal operations are made only if the releases are acceptable within the limits of 10 CFR 20. All such discharges to the atmosphere will be monitored for acceptable radiation levels.

10.1.4 SECONDARY FUNCTIONS

The Steam and Power Conversion System provides steam for the auxiliary steam driven feedwater pump and for the operation of the air ejector. The turbine bypass system is designed to dissipate the heat in the reactor coolant following a full load trip. This heat is removed by means of the steam bypass through the turbine generator to the condenser circulating water and by steam dump through the atmospheric power relief and safety valves in the event of loss of vacuum in the condenser.

TABLE 10.1-1

STEAM AND POWER CONVERSION SYSTEM COMPONENT DESIGN PARAMETERS

Turbine-Generator	
Turbine Type	Four element, tandem-compound six - flow exhaust
Turbine Capacity (KW)	
Maximum guaranteed	1,021,793
Maximum calculated	1,068,701
Generator Rating (KVA)	1,125,600
Turbine Speed (rpm)	1800
Condensers	
Type	Radial flow, single pass, divided water box, deaerating
Number	3
Condensing Capacity (lbs of steam/hr)	7,230,000
Condensate Pumps	
Type	8 stage, vertical pit type, centrifugal
Number	3
Design Capacity (each-gpm)	7860
Motor Type	Vertical Induction
Motor Rating (hp)	3000
Feedwater Pumps	
Type	High Speed, barrel coring, single stage, centrifugal
Number	2
Design Capacity (each-gpm)	15,600
Pump Drive	Horizontal steam turbine
Drive Rating (hp)	8300

TABLE 10.1-1 (Cont'd)

Emergency Feedwater Source	360,000 gallons assured reserve in 600,000 gallon condensate tank. Alternate supply from 1,500,000 gallon city water tank.
Emergency Feedwater Pumps	3 (one steam turbine driven, two electric motor driven)
Design Capacity (gpm)	800 (turbine driven) 400 (motor driven)

10.2 SYSTEM DESIGN AND OPERATION

The Steam and Power Conversion System Process Flow Diagram is shown on Figure 10-1.

Heat Balance diagrams at loads of 1,068,701 KW(e) 1,021,793 KW(e); 766,350 KW(e); 510,897 KW(e); and 255,448 KW(e) are shown on Figures 10-2 through 10-6.

10.2.1 MAIN STEAM SYSTEM

The main steam system conducts steam in a 28 in. pipe from each of the four steam generators within the reactor containment through a swing disc type isolation valve and a swing disc type non-return valve to the turbine stop and control valves. The isolation and non-return valves are located outside of the containment. The four lines are interconnected local to the turbine. The design pressure of this system is 1085 psig at 600°F. A steam flow-meter (flow venturi) is provided in the line from each steam generator, upstream of the isolation and non-return valves, to meter steam flow from each steam generator. Steam flow signals are used by the automatic feed-water flow control system (See Section 7). The flow venturi also serves to limit steam flow rate in the event of a steam line break downstream of the venturi. Steam pressure is measured upstream of the isolation and non-return valves.

The isolation valves contain free swinging discs which are normally held up out of the main steam flow path by an air piston. These valves are automatically closed upon receipt of a signal from the steam line break protection system as described in Section 7. The isolation valves are designed to close in less than five seconds.

The non-return valves are activated upon reverse flow of steam in case of accidental pressure reduction in any steam generator or its piping.

The system is classified as Class I for seismic design up to the isolation valves.

The steam break incident is analyzed in Section 14.

Turbine Steam Bypass

Excess steam generated by the reactor coolant system is bypassed, during conditions described below, from the four 28 in. main steam lines ahead of the turbine stop valves directly to the condensers by means of two 20 in. main steam bypass lines. One bypass line runs on either side of the turbine. From each 20 in. line six 8 in. lines are taken, each with an 8 in. bypass control valve installed. Each bypass valve discharges into a 10 in. pipe which is connected by a manifold with one other 8 in. bypass valve and discharges into a 12 in. manifold. Each 12 in. manifold is taken to a separate section of the condenser where it discharges into the condenser through a perforated diffuser. Each bypass valve has a capacity of 500,000 lb/hr when the steam generator is operating at full load (1022 MW(e) pressure 770 psia). The total capacity of all 12 bypass valves when operated with 770 psia in the steam generators is 6,000,000 lb/hr (40% of the steam generator steam flow at full load). The large number of small size valves is installed to limit the maximum steam flow should one valve stick open. A potential hazard in the form of an uncontrolled plant cooldown is thus eliminated. Local manually operated isolation valves are provided at each control valve.

On a turbine trip with reactor trip, the pressure in the steam generators rises. To prevent overpressure without main steam safety valve operation, the twelve turbine steam bypass valves open, discharging to the condenser for several minutes. The operation of the valves is initiated by the error signal from the reactor coolant average temperature. In the event of a turbine trip, all valves open fully in three seconds. After initial opening, the valves are modulated by the T_{avg} signal to reduce the average temperature and to maintain it at the no-load value. This is further described in Section 7.

After a normal orderly shutdown of the turbine generator leading to plant cooldown, the operator may select pressure control for more accurate maintenance of no-load conditions using the bypass valves to release steam generated by the residual heat. Plant cooldown, programmed to minimize thermal transients and based on residual heat release is effected by a gradual manual adjustment of this pressure setpoint until the cooldown process is transferred to the residual heat removal system.

During start-up, hot standby service or physics testing, the bypass valves may be controlled manually from the pressure controllers located on the main control board.

The twelve temperature controlled valves open on turbine trip or large load rejection. All twelve valves are prevented from opening on loss of condenser vacuum; they are also blocked on trip of the associated cooling water circulating pump.

Steam Dump to Atmosphere

If the condenser heat sink is not available during a turbine trip, excess steam, generated as a result of reactor coolant system sensible heat and core decay heat, is discharged to the atmosphere.

There are four 6 in. by 10 in. and one 6 in. by 8 in. code safety valves located on each of the four 28 in. main steam lines outside the reactor containment and upstream of the non-return valves. Discharge from each of the 20 safety valves is carried to atmosphere through individual vent stacks. The five safety valves in each main steam line are set to relieve at 1065, 1080, 1095, 1110, and 1120 psig, respectively. The total relieving capacity of all 20 valves is 14,619,000 lb. per hour.

In addition, four 6 in. power operated relief valves are provided which are capable of releasing the sensible and core decay heat to the atmosphere. These valves are automatically controlled by pressure or may be manually operated from the main control board and are capable of releasing ten per cent of the equivalent nominal rated steam flow (1,390,000 lb. per hour of steam at 1020 psi pressure). One power operated relief valve is located on each main steam line, upstream of the swing disc trip valve. Discharge from each of the four power relief valves is carried to atmosphere through individual vent stacks. In addition, the power operated relief valves may be used to release the steam generated during reactor physics testing, operator license training and plant hot standby operation, if the main condenser is not available.

Steam For Auxiliaries

The steam for the auxiliary turbine driven feedwater pump is obtained from two of the 28 in. steam generator outlet mains, upstream of the swing disc trip valves. The steam is reduced to 600 psig for the turbine by a pressure reducing control valve.

Auxiliary steam for the turbine gland steam supply control valve, the three steam-jet air ejectors, the reheater section of the six moisture separator-reheaters, the three priming ejectors and supplementary steam for the main feed pump turbines is obtained from branches on the steam lines ahead of the turbine stop valves. Pressure reducing stations are used for the priming and main air ejectors. The temperature control valves are located in the auxiliary steam line to the reheaters. The design pressure and temperature for this system are 1085 psig and 600°F, respectively.

Steam from six extraction openings in the turbine casings is piped to the shells of the three parallel strings of feed-water heaters. The first point extraction originates at the high pressure turbine casing and supplies steam to the shell of the No. 6 (high pressure) feed-water heater. The second point extraction originates in the high pressure turbine exhaust piping ahead of the moisture separators, and supplies steam to the No. 5 (low pressure) feed-water heater. The third, fourth, fifth and sixth point extractions all originate at the low pressure turbine casings and supply steam to the No. 4, No. 3, No. 2, and No. 1 (all low pressure) feed-water heaters, respectively.

To prevent turbine overspeed from backflow of flashed condensate from the heaters after a turbine trip, non-return valves are provided in all but the two lowest pressure extraction steam lines. The non-return valves in the two high pressure first and second point extraction lines are air cylinder operated valves which are closed automatically upon a signal from the turbine trip circuit. The non-return valves in the third and fourth point intermediate pressure extraction lines are swing check type valves with balancing counterweights and are also air cylinder operated. The low pressure fifth and sixth point extraction lines are located entirely in the condenser shells and do not contain non-return valves.

Steam Generator Blowdown

Each steam generator is provided with two 2 in. bottom blowdown connections for shell solids concentration control. The two connections are at the same level, but on opposite sides of the shell. Piping from the two connections join to form a 2 in. blowdown header for each steam generator. The bottom of each steam generator is also provided with a drain connection which discharges into the blowdown line.

Each blowdown line is provided with a hand shutoff valve and two diaphragm operated trip valves. Each blowdown line includes, in addition to these shutoff valves, a manually operated needle-type flow control valve for blowdown flow adjustment. A steam generator sample line is taken from the blowdown line inside containment. A small flow from each sample line is combined and is monitored for radiation. In the event of a high radiation signal, both diaphragm valves in the sample and blowdown lines will close automatically.

Blowdown from all four steam generators passes to the blowdown flash tank. The flashed vapor is discharged to atmosphere while the condensate drains by gravity through a service water discharge line into the circulating water discharge canal.

If drains from the blowdown tank become contaminated, the blowdown tank drains may be diverted to the waste disposal system for processing.

The normal full load blowdown rate from four steam generators is approximately 29,000 lb. per hour or 0.2 per cent of feedwater flow. The design basis is 50,000 lb. per hour for four steam generators to provide for occasionally higher blowdown rates should they be required to reduce solids concentration.

A flash steam evaporator provides distillate for makeup water to the reactor coolant system and the condensate system. With the main turbine operating at 1022 MW(e), this evaporator will produce 56,000 lb/hr or 87,000 lb/hr at extended rating.

10.2.2 TURBINE-GENERATOR

The turbine has a guaranteed capability of 1,021,793 kw at 1.5 in. Hg absolute exhaust pressure with zero per cent makeup and six stages of feedwater heating. The unit operates at 1800 rpm with steam supplied ahead of the main stop valves at 730 psia, 508°F and 0.25 per cent moisture. Steam is admitted to the turbine through four stop valves and four control valves. The expected throttle flow at 1,021,793 kw is 12,529,696 lb. of steam per hour.

The turbine (TC6F44) is a four casing, tandem compound, six flow exhaust unit with 44 inch last row blades as shown in Figure 10-7. The turbine consists of one double flow HP element in tandem with three double flow LP elements. Steam after passing through the stop and control valves, passes through the high pressure turbine, and then through the moisture separator reheaters.

There are six, horizontal-axis, cylindrical shell, combined moisture separator, steam reheater assemblies. Steam from the exhaust of the HP turbine element enters each assembly at one end. Internal manifolds in the lower section distribute the wet steam. The steam then rises through a wire mesh moisture separator where the moisture is removed and drained to a drain tank. The steam leaving the wire mesh separator flows over a tube bundle where it is reheated. This reheated steam leaves through nozzles in the top of the assemblies and flows to the LP turbines. The tube bundle is supplied with main steam from ahead of the turbine throttle valves which condenses in tubes and leaves as condensate. Condensate from the reheater assemblies flows to the high pressure heater.

The turbine oil system consists of a high pressure hydraulic control system and a low pressure lubrication system. Oil is also used to seal the generator shaft seals to prevent hydrogen leakage from the generator into the turbine room. The oil pump mounted on the main trubine shaft normally supplies all oil requirements. A motor driven auxiliary oil pump supplies the oil required during turbine startup and whenever there is low pressure in the bearing oil header. The auxiliary unit is a centrifugal pump driven by a 150 HP motor. Oil is supplied at 300 psig to the hydraulic control mechanisms. A motor driven bearing oil pump is also provided which supplies oil whenever there is a low pressure in the bearing oil header. This pump is a centrifugal type with a 75 HP motor. During startup, these auxiliary oil pumps supply all the oil while the main pump acts against a closed check valve. An AC motor driven oil pump is provided for turning gear and emergency operation. A DC motor driven oil pump operated from the station battery is included for additional backup to assure lubricating oil to the machine. An AC motor driven generator seal oil pump is furnished for normal operation with a DC motor driven backup pump to assure confinement of the hydrogen within the generator.

A continuous bypass turbine oil purification system is provided to remove contaminents from the oil.

To maintain shaft alignment while the unit is down, a motor driven turning gear is provided.

The turbine is coupled to a single, hydrogen inner cooled generator and rotating rectifier exciter. The generator is rated at 1,125,600 kva, 3 phase, 60 cycles, 22 kv and 90 per cent power factor, and 75 psig hydrogen pressure. It has sufficient capability to accept gross kilowatt output of the steam turbine with its control valves wide open at 730 psia saturated 1 % moisture rated steam conditions.

10.2.3 TURBINE CONTROLS

High pressure steam enters the turbine through four stop valves and four governing control valves. The four main stop valves are designed for the specific operating conditions. Each stop valve is a single seated, oil operated, spring closing valve controlled primarily by the turbine over speed trip device. The turbine overspeed trip pilot is actuated by one of the following to close the stop valves:

Turbine thrust bearing trip
Low bearing oil pressure trip
Low condenser vacuum
Solenoid trip
Overspeed trip
Hand trip

Each stop valve has limit switches which operate position lights on the main control board. There are also similar limit switches in the electrical interlock system to operate the turbine trip auxiliary relay and the reactor trip breakers.

Test switches on the main control board permit test closure of each valve. The valve operation can be observed from within the turbine front end enclosure. Periodic tests exercise the stop valves and ensure their ability to close during an emergency.

Before a stop valve can be opened, the pressure across the valve must be equalized. This is done by opening a small bypass valve around each of the stop valves.

Four hydraulically operated valves of the single seated plug type open and close in sequence to control steam admission to the turbine. They are actuated by the turbine speed governor which is responsive to turbine speed, and which includes:

A speed changer or synchronizing device

A load limit device which must be reset after operation of the overspeed trip before the control valves can be opened

A second load limit device without reset is furnished to give redundancy of load cutback following a rod drop

The governing emergency trip valve, actuated when the stop valves are tripped, to close the control valves

An auxiliary governor, responsive to the rate of turbine speed increase, to close the control valves

A motor controlled hydraulic pilot valve is provided for each control valve to test the operation of the valve. Test switches with indicating lights are provided on the main control board turbine section. Removable strainers are located in each control valve body to protect the valves and turbine from foreign material in the steam. Temporary fine mesh strainers are installed during initial operation.

The normal governing devices, which operate through hydraulic relays to operate the control valves are as follows:

The governor handwheel at the unit

The governor synchronizing motor, which is controlled by a switch on the electrical section of the main control board and is used for raising or lowering turbine speed or load

The load limit handwheel at the un[illegible]

The load limit motor, which is controlled by a switch on the turbine section of the main control board and by a reactor control rod drop run back signal (This is further described in Section 7)

The pre-emergency device functions similar to the normal governing devices by operating the control valves in case of abnormal operating conditions in the auxiliary governor. This pre-emergency device closes the control valves on rapid increase in turbine speed. The control valves will be actuated by either the speed governor or load limit, and the device delivering the lowest oil pressure will be in control. Pressure gages on the main control board indicate the oil pressure from these devices.

The emergency devices which will trip the stop valves, the control valves, and the air relay dump valve are as follows:

- Overspeed emergency governor
- Solenoid trip (also actuated from reactor trip, electrical faults and a manual push button)
- Low condenser vacuum trip
- Low bearing oil trip
- Thrust bearing trip
- Hand trip at unit

The mechanical overspeed trip mechanisms consists of an eccentric weight mounted in the end of the turbine shaft, which is balanced in position by a spring until the speed reaches approximately 108% of rated speed (the tripping speed). Its centrifugal force then overcomes the restraining spring and the eccentric weight flies out striking a trigger which trips the overspeed trip valve and releases the autostop fluid to drain. The resulting decrease in autostop pressure causes the governing emergency trip valve to release the control oil pressure, and this closes the main stop and governing valves. An air pilot valve used to control the extraction non-return valves is also actuated from the autostop pressure.

The autostop valve is also tripped when any one of the protective devices is actuated. The protective devices include a low bearing oil pressure trip, a solenoid trip, a thrust bearing trip and a low vacuum trip. These devices are all included in a separate assembly, but connected hydraulically to the overspeed trip valve. An additional protective feature includes a turbine trip following a reactor trip.

Trip of the turbine-generator, when unit load is greater than 50% of full load, initiates a reactor trip to prevent excessive reactor coolant temperature and/or pressure.

A dropped rod control cluster assembly signal as indicated by either a rapid decrease in nuclear flux or by the rod bottom on-off controllers, initiates automatic turbine load cutback. This is further described in Section 7.

10.2.4 CIRCULATING WATER SYSTEM

Hudson River water is used for the condenser circulating water. River water flows under the floating debris skimmer wall into six separate screen wells through heavy steel trash bar racks. The trash bar racks protect the travelling screens against damage from heavier debris. The water flows through travelling screens where the smaller debris is removed. When the screens become sufficiently dirty to require cleaning, a pressure differential signal is dispatched to the control room and cleaning is automatically inaugurated. Each screen well is provided with stop logs to allow dewatering of any individual screen well for maintenance purposes.

The water from each individual screen well flows to a motor driven, vertical, mixed flow condenser circulating pump. Each of the six condenser circulating pumps provides 140,000 gpm and 21 ft. TDH when operating at 292 rpm and is located in an individual pump well, thus, tying a section of the condenser to an individual pump. The circulating water is piped to the condensers and is discharged back into the river at a distance far enough away from the intake to minimize recirculation. To protect the travelling screens against ice during freezing water conditions, warm discharge water from the condenser may be recirculated ahead of the screens to raise the inlet water temperature. Radient heaters are also supplied for the screens.

Sodium hypochlorite is intermittently injected into the circulating water to prevent the buildup of bacterial slime on the travelling water screens, condenser tubes, and piping.

Sodium hypochlorite containing approximately 13-1/2 per cent by weight of available chlorine is stored in two 4000 gallon tanks in the hypochlorite room of Unit No. 1 screen well house.

10.2.5 CONDENSER AND AUXILIARIES

Three surface type, single pass, radial flow condensers are provided with bolted divided water boxes at both ends. Fabricated steel water boxes and shell construction is used. Hotwell design is for four minute storage while operating at maximum turbine throttle flow with free volume for condensate surge protection. The hotwells are longitudinally divided to facilitate condenser tube leakage detection. Each half is provided with separate conductivity measurement devices. In the event of high conductivity (high salinity) in a hotwell it will be manually isolated and the condensate dumped overboard instead of normally providing suction for the condensate pumps described below. The deaerating hotwells are provided to reduce the residual oxygen in the condensate to less than 0.01 cc per liter. The Admiralty metal tubes are welded into silicon-bronze tube sheets Water box manholes are provided for access. Provision is made for condensing the main feedpump drive turbine exhaust. The condensers have steam turbine bypass condensing arrangements to condense turbine bypass steam for controlled startups and to condense residual and decay heat steam following a shutdown.

Three motor driven, eight-stage, one-third capacity, vertical, pit type, centrifugal condensate pumps are provided, each taking suction from the condenser hotwells. The condensate pumps discharge into three separate parallel streams of feedwater heaters and provide the suction supply to the feedwater pumps.

One four element, two-stage air ejector with separate inter- and common after-condensers is provided for each condenser. For normal air removal, one air ejector unit is required per condenser. The ejectors function by using main steam. The air ejector exhaust is monitored for radiation. In the event of a steam generator leak and subsequent presence of radioactive contaminated steam in the secondary system, the radioactive non-condensable gases which concentrate in the air ejector effluent will be detected by this radiation monitor. A high activity level signal automatically diverts the exhaust gases from the vent stack to the containment.

For initial condenser shell side air removal, three non-condensing priming ejectors are provided, each having a capacity of 900 cfm. During periods of plant shutdown where decay heat is involved, this apparatus may be used. The main ejectors will also be operated at the same time to ensure that the effluent is monitored for radiation.

10.2.6 CONDENSATE AND FEEDWATER SYSTEM

The condensate and feedwater system is designed to supply a total of approximately 13,283,282 lb. of feedwater per hour to the four steam generators at a turbine load of 1022 MW(e). This system, as shown on Figure 10-1, is comprised of:

A condensate system which collects and transfers condensed steam and the drains from five feedwater heaters through five stages of feedwater heating to the suction of the steam generator feed pumps.

A condensate makeup and surge system which maintains a normal water level in the condenser hot wells.

A heater drain system which collects and transfers to the suction of the steam generator feed pumps the drains from No. five and six feedwater heaters and six moisture separator-reheaters

A feedwater system which delivers the condensate and heater drains through one stage of feedwater heating to the steam generators

An auxiliary feedwater system which provides a flow of water from the condensate tank to the steam generators when the main steam generator feed pumps are unavailable. The flow is equivalent to that required for makeup due to reactor core decay heat removal requirements.

Condensate System

The condensate system transfers condensate and low pressure heater drains from the condenser hot well through five stages of feedwater heating to the suctions of the steam generator feed pumps.

Three 1/3 size condensate pumps take suction through individual 24 in. lines from the bottom of the condenser hot well. The pumps discharge into a common header which carries the condensate through three steam jet air ejector condensers, arranged in parallel, and through one gland steam condenser.

The condensate pumps are eight stage, vertical, pit-type pumps. Each pump is rated at 7860 gpm and 1150 ft TDH when operating at 1170 rpm. Shaft sealing is accomplished with a standard packed stuffing box. The pump bearings are lubricated by the pumped liquid. Each pump is driven, through a solid coupling, by a 3000 HP, vertical, solid shaft, induction motor having an open dripproof enclosure.

The condensate pumps are operated by manual controls on the main control board.

To maintain condenser vacuum and turbine steam seals during start-up, shutdown, and at very low loads, an 8 in. condensate recirculation line, containing a diaphragm operated valve, is provided to maintain minimum flow through the air ejector condensers and gland steam condenser. The recirculation line originates at the condensate header downstream of the gland steam condenser and terminates at the condenser hot well. The diaphragm operated recirculation valve is automatically controlled by the minimum flow required by the air ejector condensers.

The 24 in. condenser header divides into three 14 in. lines downstream of the gland steam condensers. These lines carry the condensate through the tube sides of three parallel strings of two LP feedwater heaters. The flow to the remaining three strings of three LP heaters is through a common 24 in. pipe. An 18 in. bypass is taken to the flash evaporator for cooling the product. The condensate is returned to the common stream before the remaining LP heaters. A manually controlled valve in the 24 in. main regulates flow to the flash evaporator condenser. After the No. 5 feedwater heater, the three condensate lines join into a common header. The heater drain pump discharge enters this header and then continues on to the suction of the steam generator feed pumps.

Each parallel string of feedwater heaters may be taken out of service by closing a manual gate valve at the inlet to the string of heaters, and at the outlet to the string of heaters.

The condensate makeup and surge systems operate to maintain normal water level in the condenser hot well.

The makeup system connects the 600,000 gal. capacity water storage tank to a diffusing pipe in the condenser shell. This line contains a diaphragm operated valve which automatically opens on low level in the condenser hot well to pass makeup water from the tank to the condenser. An isolating valve will close the condenser makeup when the storage tank level reaches 360,000 gal. This will ensure a reserve of condensate for the auxiliary feedwater pumps to hold the plant at hot shutdown for 24 hours following a trip at full power.

The surge or condensate draw-off system connects the condensate pump discharge header to the condensate storage tank. This line contains a diaphragm operated valve which automatically opens on high level in the condenser hot well to pass excess condensate from the condensate pump discharge header to the condensate storage tank.

Hot well levels are indicated on the main control board. Should the automatic make-up valve or the surge valve become inoperative, it may be isolated from its respective system and the hot well level controlled from the control room by remote manual positioning. The condenser hotwells contain 114,000 gallons which is equal to approximately 5.5 min. condensate flow at 1022 MWe load.

The drains from the No. 6 feedwater heater flow to the heater drain tank. Normal condensate level is maintained in the No. 6 heater by diaphragm operated level control valves.

The drains from the No. 5 feed-water heaters flow by gravity directly to the heater drain tank. There are no level control valves in the drains from these heaters.

The heater drain tank also receives drains from the shells of moisture separators through separate gravity flow drain lines. Air cylinder operated swing check type non-return valves in these drain lines close on turbine trip.

Two half size heater drain pumps pump the drains from the drain tank into the condensate header ahead of the steam generator feed pumps. Both pumps discharge through diaphragm operated level control valves.

The heater drain pumps are fourteen-stage, vertical, enclosed suction-type pumps. Each pump is rated at 4150 gpm and 720 ft TDH when operating at 1170 rpm. Each pump is driven through a solid coupling by a 1000 hp, vertical, solid shaft, induction motor, having an open drip proof enclosure.

The heater drain pumps are operated by manual controls on the main control board. A heater drain pump is automatically stopped if the water level drops to extreme low in the drain tank, or if the flow falls below a set minimum. After the pump has stopped, the water level in the heater drain tank will increase. An alarm sounds in the control room on both tank low level and pump low flow.

When an extremely high level occurs in the heater drain tank, a diaphragm operated valve opens to discharge the excess condensate from the heater drain tank directly to the shell of a condenser. An alarm sounds in the control room. The heater drain tank has a 5660 gal. storage capacity at normal water level or approximately 3/4 minute storage of drains at the normal full load of 1022 MW(e).

Drains from the Nos. 4, 3 and 2 feed-water heaters normally flow through diaphragm operated level control valves to the shells of the next lowest pressure feed-water heater. On high level in any heater, a separate high level drain from the heater discharges directly to the condenser.

Drains from the No. 1 feed-water heaters normally flow through diaphragm operated level control valves to the condenser. When a high level occurs in the heaters, a separate high level drain for each heater discharges to the condenser.

Feedwater System

Main Feedwater System

Two half size steam driven main feed water pumps increase the pressure of the condensate for delivery through one stage of feed-water heating and the feed-water regulating valves to the steam generators.

The main feed-water pumps are single-stage, horizontal, centrifugal pumps with barrel casings. Each pump is rated at 15,600 gpm and 1700 ft. TDH when operating at 4950 rpm. Shaft sealing is accomplished by seal water injection. Bearing lubrication for both the pump and its turbine drive is accomplished by an integral lubricating oil system mounted on the pump base. Normal circulation of the lubricating oil is by a motor driven pump. The lubricating oil system includes a reservoir, a cooler, and two motor driven oil pumps. Each main feedwater pump is driven through a flexible gear type coupling by an 8350 HP horizontal steam turbine using steam from the discharge of the three rehater moisture separators on one side of the turbine hall. The main feedwater pumps are operated automatically by the feed control system. Manual controls are also provided on the main control board for remote operation and testing during normal operation. During normal startup of the plant, the steam generator feed pumps are started locally. A minimum flow control system is provided to ensure that at least a 3000 gpm flow is being handled by each pump at all times.

Low suction pressure reduces the turbine speed to maintain suction pressure. Normal speed is regained when the suction pressure and flow is re-established. High discharge pressure reduces turbine speed to prevent excessive pressure in the feed piping.

An automatic bypass is provided around the low pressure heaters to ensure sufficient suction pressure at the feed pumps during a transient when flashing may occur in the heater drain tanks and effect the drain pumps performance.

High feed water pump bearing temperatures are alarmed in the main control room but do not automatically stop the pump.

The two main feed-water pumps operate in series with the condensate pumps, discharging through check valves and motor operated gate valves into a common header. The feed water then flows through the three parallel, high pressure feed-water heaters and flow into a common header. Four parallel 18 in. lines containing the feed-water metering and regulating stations, feed the four steam generators.

Shutoff valves at the inlets and outlets of the feed-water heaters are provided to permit a heater to be taken out of service. Bypass lines are provided around the heaters to allow operation when a heater is out of service for maintence.

The steam generator feed-water metering and regulating stations measure, indicate, record and control the water level in each of the four steam generators. A conventional three element system receives flow and load signals from the reactor protection system through isolation amplifiers and compares the difference between steam and feed-water flows to adjust the level set point. The deviation of level measurement from this set point positions the feed-water control valve accordingly. Totalized steam flow controls the speed of the feed pump turbines.

Reactor trip is actuated on a coincidence of steam flow - feedwater flow mismatch, coupled with low level in the corresponding steam generator. A reactor trip is also initiated on a coincidence of two out of three very low water level signals from any one steam generator.

Whenever this reactor trip occurs, the feed-water valves move to the fully opened position to provide an additional heat sink for the reduction of reactor coolant temperature to the no-load average temperature value. The valves remain fully open until either one of the following conditions occurs, at which time the respective valve, or valves, fully close:

Abnormally high steam generator level

Average temperature error signal (between measured T_{avg} and the no load T_{ref}) reduces to a preset level which is equal to the value at which the turbine steam bypass valves begin to close.

The feed-water control system is an electronic analog instrumentation system.

Readout and control equipment is as follows:

A wide range level recorder (one two-pen for each of two pairs of steam generators), calibrated for cold conditions in the steam generator, permits observation of level over essentially the full height of each steam generator shell.

A direct reading, two-pen recorder in the control room records steam and feedwater flows in pounds per hour for each steam generator.

Each flow channel and each narrow range level channel is indicated on the main control board.

One manual control station is provided for each feedwater controller. This unit consists of an auto/manual transfer switch and an analog output control which serves at the valve position signal when in "Manual." The "Automatic" set point is pre-set but adjustable in the instrument rack.

Other manual control stations are used to position auxiliary feed valves.

Auxiliary Feedwater System

The Auxiliary Feedwater System supplies high pressure feedwater to the steam generators in order to maintain a water inventory for removal of heat energy from the reactor coolant system by secondary side steam release in the event of inoperability of the main feedwater system. The head generated by the pumps is sufficient to deliver feedwater into the steam generators at safety valve pressure. Redundant supplies are provided by using two pumping systems, using different sources of power for the pumps.

The capacity of each system is set so that the four steam generators will not boil dry nor will the primary side relieve fluid through the pressure relief valves, following a loss of main feedwater flow.

One system utilizes a steam turbine driven pump, with the steam capable of being supplied from two of the steam generators. This system supplies 800 gpm of feedwater (200 gpm to each steam generator). The estimated performance characteristic of the pump is given in Figure 10-8.

Steam to drive the turbine is supplied from two of the main steam lines upstream of the isolation valves at steam generator outlet pressure and is reduced to the 600 psi turbine design pressure by a pressure reducing control valve.

The pump is started and feedwater is supplied to all four steam generators through individual feed regulating valves, controlled from the main control board or local to the valve. The drive is a single stage turbine, capable of quick starts from cold standby and is directly connected to the pump. The turbine is started by opening the pressure reducing valve between the turbine supply steam header and the main steam lines. The turbine sleeve journal bearings are ring oil lubricated water cooled. The pump uses oil flinger lubricated ball bearings.

The other system utilizes two motor-driven pumps with ring lubricated ball bearings. Each pump has a capacity of 400 gpm and the discharge piping is arranged so that each pump supplies two steam generators. The estimated performance characteristic for the pumps is given in Figure 10-9.

The motors are of open drip-proof design with ball bearings. Electrical power is automatically obtained from the diesel generators in the event of complete loss of power.

The auxiliary feed pumps are located in the area of the main steam and feedwater penetration immediately outside the reactor containment.

Flow measurement devices are installed in the discharge lines to each steam generator with indicators on the control board. These provide the operator with the information necessary to properly route the discharge flow through the two remote manual discharge valves.

The distribution piping is seismic Class I throughout and is designed to ensure that a single fault will not restrict the system function.

The water supply source for this system is redundant. The main source is by gravity feed from the condensate storage tank. This tank is sized to meet the normal operating and maintenance needs of the turbine cycle systems: however, a minimum water level will be maintained, equivalent to the steam generation due to 24 hours of residual heat generation at hot shutdown conditions.

A point for connection to the pumps from an alternate supply of water is provided for long-term cooling. This supply is from the 1.5 million gallon tank for plant storage of city water.

The auxiliary feedwater pumps are automatically started on receipt of any of the following signals:

Steam driven feedwater pump.

1) Low-low water level in any two of the four steam generators.

2) Loss of outside power concurrent with a unit trip

Motor driven feedwater pumps.

1) Low-low water level in any steam generator

2) Automatic trip of either main feed pump as indicated by loss of auto stop oil pressure in the turbine control system.

3) Safety Injection Signal

4) Loss of outside power concurrent with a unit trip.

In the event of a complete loss of offsite power, the power is supplied by the diesel generator as described in Section 8.

In addition, both the steam driven pump and the motor driven pumps can be manually started from the control room and locally at the pump.

System Chemistry

Steam generator water chemistry is maintained within the limits given in Table 4.2-3. Hydrazine is added to the condensate for oxygen control and morphaline to maintain the pH at the optimum value for the materials of construction for the system. Trisodium phosphate is injected into the steam generator through the main feedline in the event of residual hardness build-up in the steam generator. No other additives are contemplated.

No radiation shielding is required for the components of the steam and power conversion system. Continuous access to the components of this system is possible during normal operation.

Under normal operating conditions, there is no radioactive contaminants present in the steam and power conversion system. It is possible for this system to become contaminated through steam generator tube leaks. In this event any contaminant is detected by monitoring of the steam generator shell side sample points and the air ejector off-gas. Operation with a steam generator tube leak is discussed in Section 14.

10.2.7 CODES AND CLASSIFICATIONS

The pressure retaining components or compartments of components comply, at least, with the codes detailed in Table 10.2-1.

TABLE 10.2-1

CODES AND CLASSIFICATIONS

System Pressure Vessels and Pump Casing	ASME Boiler and Pressure Vessel Code, Section VIII
Steam Generator Vessel	ASME Boiler and Pressure Vessel Code, Section III, Class C*
System Valves, Fittings, and Piping	USAS Section B31.1 Pressure Piping Code

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

10.3 SYSTEM EVALUATION

10.3.1 SAFETY FEATURES

Trips, automatic control actions and alarms will be initiated by deviations of system variables within the Steam and Power Conversion System. Appropriate corrective action is taken as required to protect the Reactor Coolant System. The more significant malfunctions or faults which cause trips, automatic actions or alarms in the steam and power conversion system are:

a) Turbine Trip

1. Generator/electrical faults.
2. Low condenser vacuum.
3. Thrust bearing failure.
4. Low lubricating oil pressure.
5. Turbine overspeed.
6. Reactor trip.
7. Manual trip.

b) Automatic Control Actions

1. High level in steam generator stops feedwater flow.
2. Normal and low level in steam generator modifies feedwater flow by continuous proportional control.

c) Principal Alarms

1. Low pressure at feedwater pump suction.
2. Low vaccum in condenser.
3. Thrust bearing failure.
4. Low lubricating oil pressure.
5. Turbine overspeed.
6. Low level in steam generator.
7. High level in steam generator.

A reactor trip from power requires subsequent removal of core decay heat. Immediate decay heat removal requirements are satisfied by the steam bypass to the condensers. Thereafter, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption.

Normally, the capability to return feedwater flow to the steam generators is provided by operation of the turbine cycle feedwater system. In the unlikely event of a complete loss of offsite electrical power to the station, and concurrent reactor trips decay heat removal would continue to be assured by the steam-driven, and two motor-driven (via emergency generator) auxiliary steam generator feed-water pumps, and steam dumped to atmosphere via the main steam safety and power relief valves. In this case feedwater is available from the condensate storage tank by gravity feed to the auxiliary feedwater pumps. The minimum 360,000 gallons of water in the condensate storage tank is adequate for decay heat removal for a period of at least 24 hours. A back-up source of feedwater is available from the city water storage tank.

The analysis of the effects of loss of full load on the Reactor Coolant System is discussed in Section 14.

10.3.2 SECONDARY-PRIMARY INTERACTIONS

Following a turbine trip, the control system reduces reactor power output immediately by a reactor trip. Steam is bypassed to the condenser and there is no lifting of the main safety valves. In the event of failure of a main feedwater pump, the auxiliary feedwater pumps are automatically started and the second main feedwater pump remaining in service will carry approximately 65 per cent of full load feedwater flow. If both main feedwater pumps fail, the reactor will be tripped, as a result of steam generator low-low level or steam-feedwater flow mismatch and the auxiliary feedwater pumps started. If Reactor Coolant System conditions reach trip limits, the reactor will trip.

Pressure relief is required at the system design pressure of 1085 psig. The first safety valve is set to relieve at 1065 psig. Additional safety valves are set at pressures up to 1120 psig, as allowed by the ASME Code. The pressure relief capacity is equal to the steam generation rate at maximum calculated conditions.

The evaluation of the capability to isolate a steam generator to limit the release of radioactivity in the event of a steam generator tube leak is presented in Section 14. The steam break accident analysis is also presented in Section 14.

10.3.3 SINGLE FAILURE ANALYSIS

A single failure analysis has been made for all active components of the system which have an emergency function. The analysis, which is presented in Table 10.3-1 shows that the failure or malfunction of any single active component will not reduce the capability of the system to perform its emergency function.

TABLE 10.3-1

SINGLE FAILURE ANALYSIS

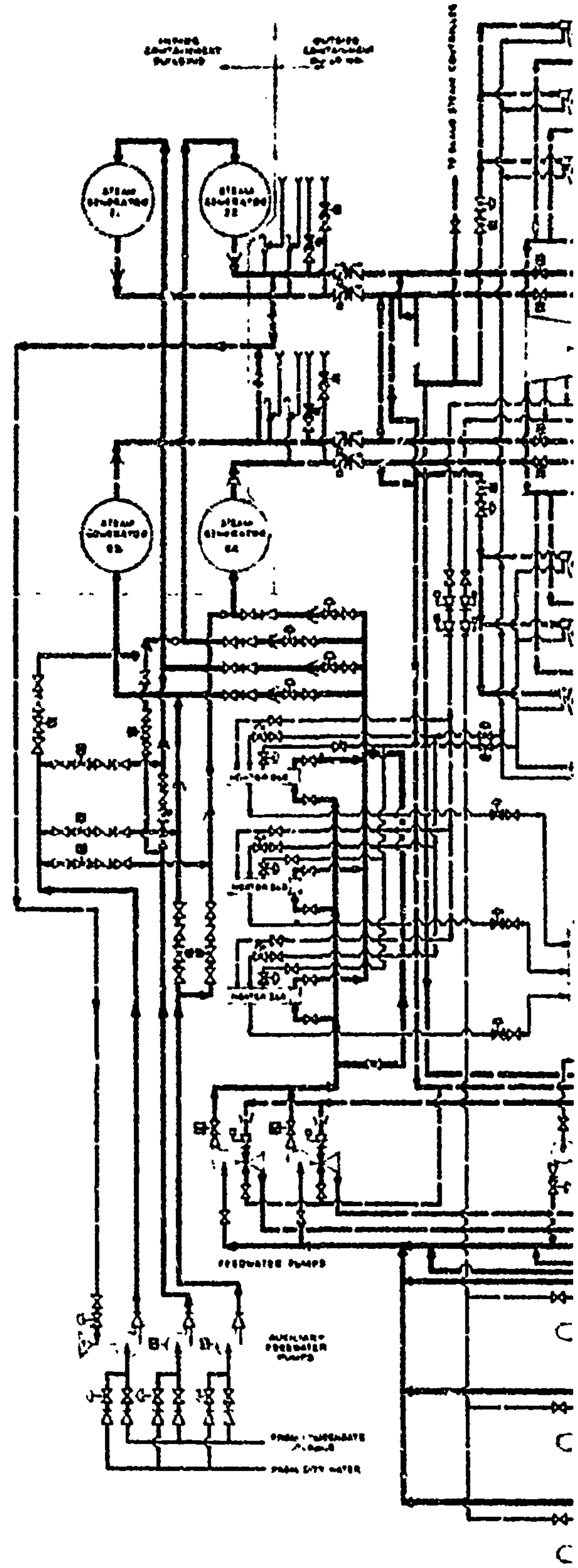
Component or System	Malfunction	Comments and Consequences
Auxiliary Feedwater System	Auxiliary Feedwater pump fails to start (following loss of main feedwater)	The Auxiliary Feedwater System comprises one turbine driven and two motor driven pumps. The turbine pump is twice the capacity of a motor driven pump and one motor driven pump has sufficient capacity to prevent relief of fluid through the primary side relief valves. Thus adequate redundancy of auxiliary feedwater pumps is provided.
Steam Line Isolation System	Failure of Steam line isolation valve to close (following a main steam line rupture)	Each steam line contains an isolation valve and a non-return valve in series. Hence a failure of an isolation (or non-return) valve will not permit the blowdown of more than one steam generator irrespective of the steamline rupture location.
Turbine Bypass System	Bypass valve sticks open (following operation of the bypass system resulting from a turbine trip)	The turbine bypass system comprises 12 bypass valves. Hence one valve can only pass < 4% of the steam generator steam flow and there is no hazard in the form of an uncontrolled plant cooldown if a bypass valve sticks open.

10.4 TESTS AND INSPECTIONS

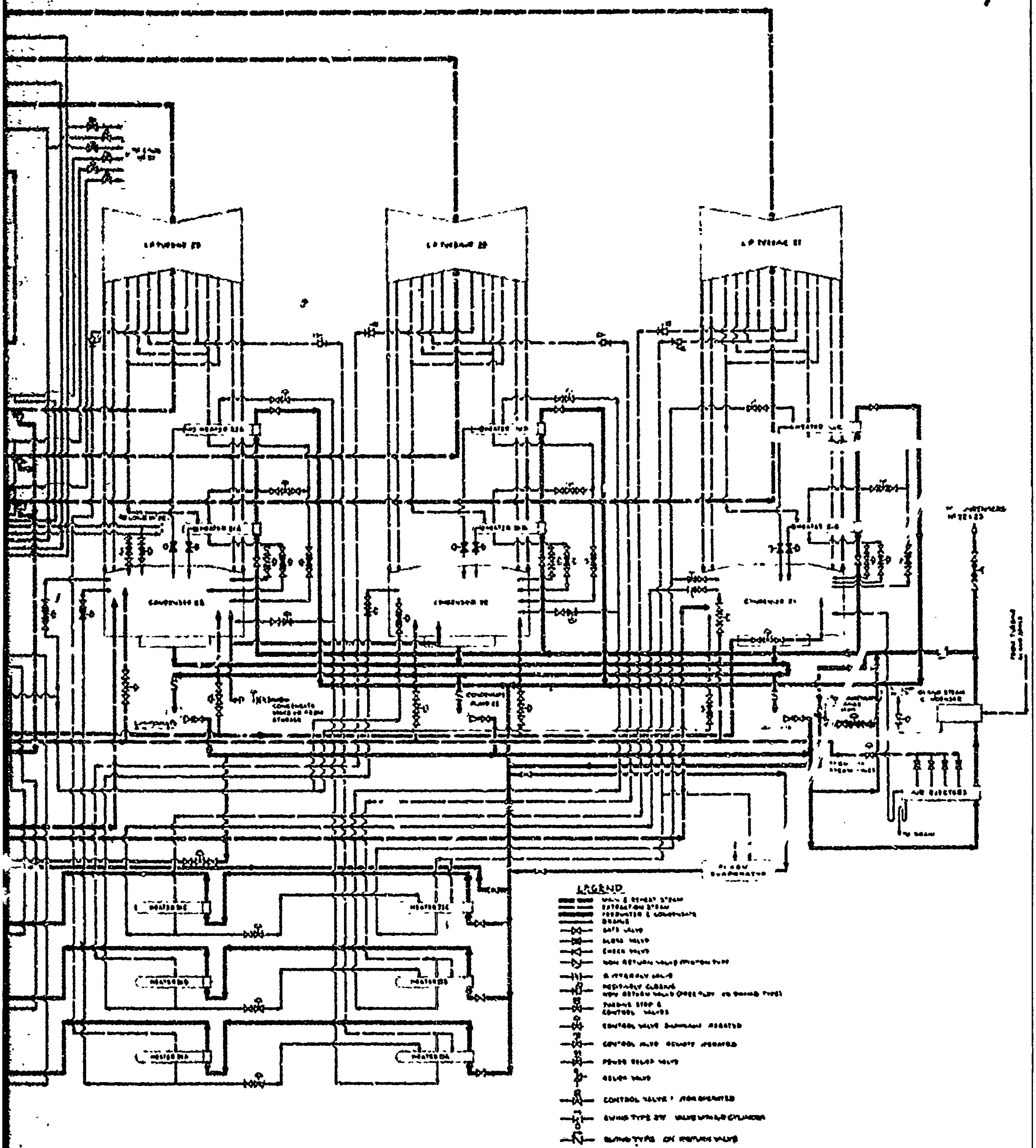
The main steam isolation valves will be tested at regular intervals as specified in the Technical Specifications. Closure time of 5 seconds will be verified.

The main steam isolation valves serve to limit an excessive reactor coolant system cooldown rate and resultant reactivity insertion following a main steam break incident. Their ability to close upon signal should be verified at periodic intervals. A closure time of 5 seconds from receipt of closing signal was selected as being consistent with expected response time for instrumentation as detailed in the steam line break incident analysis.

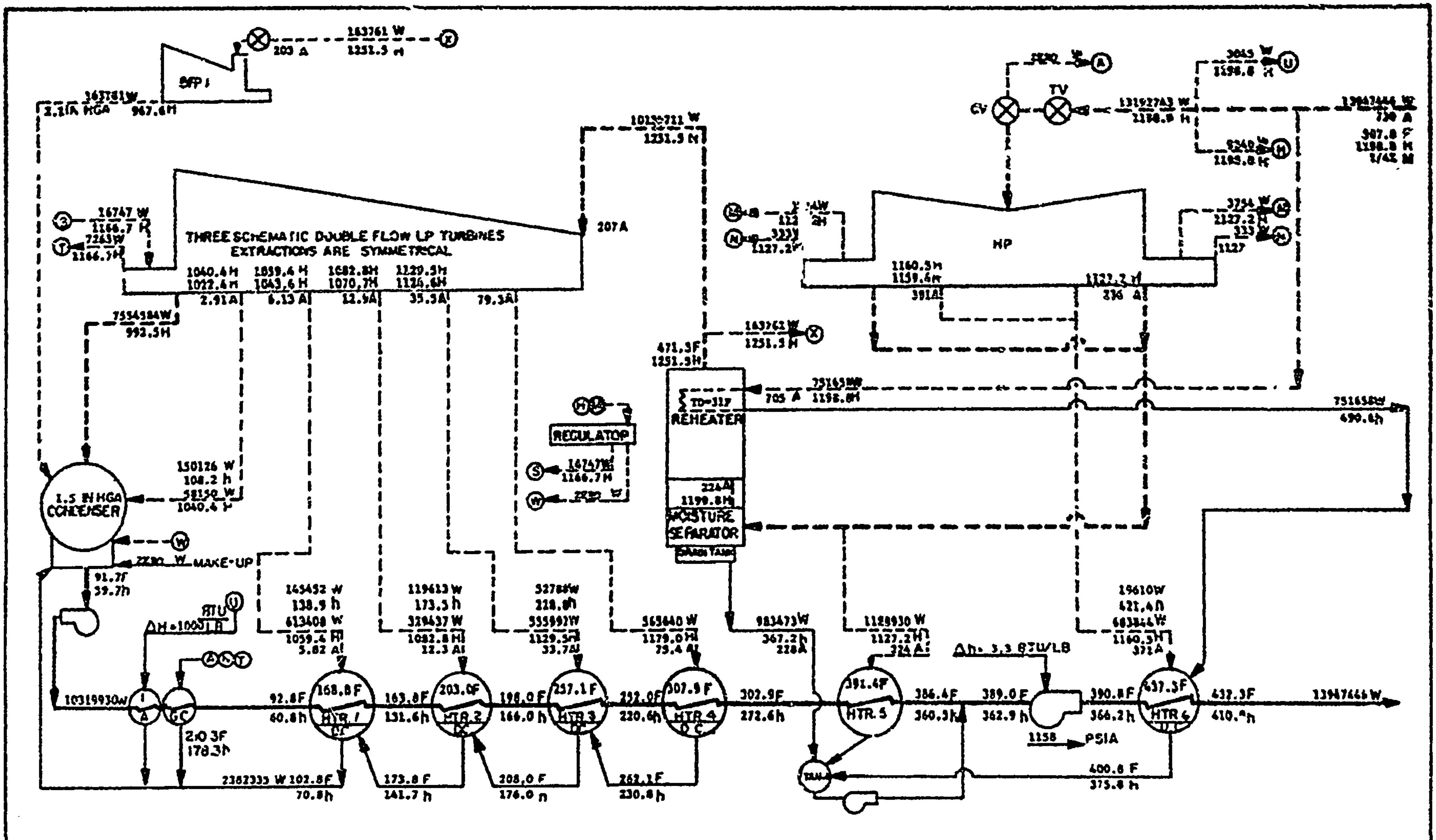
The auxiliary feedwater pumps can be tested at any time. Each pump will deliver water from the condensate storage tank through its feedwater control valves to the feedwater line to the steam generators. Verification of correct operation will be made both from instrumentation within the main control room and by direct visual observation of the pump. The frequency of testing is specified in the Technical Specifications.



FEEDWATER PUMPS
AUXILIARY FEEDWATER PUMPS
FROM CITY WATER



Steam Condensate - Drains Systems Flow Diagram
Figure 10-



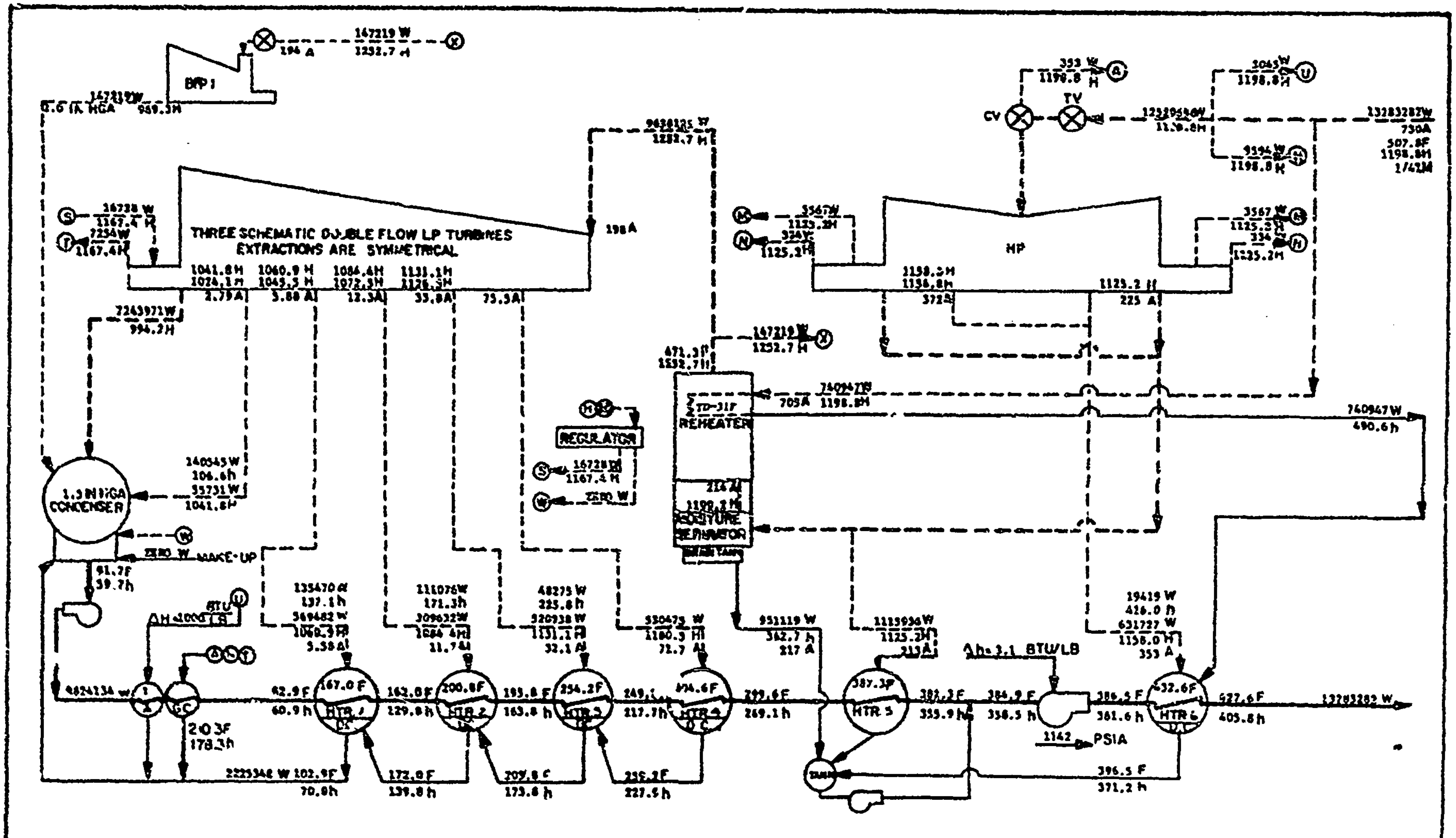
NET HEAT RATE $= \frac{13947446\ (1198.8-410.9)}{1068701} = 10283$ BTU/KW HR	TEP = 992.3 BTU/LB ELEP = 973.4 BTU/LB MECH LOSS = 3645 KW ELECT LOSS = 11940 KW 0.90PF 75 #H2 FWP POWER = 13424 KW FWP EFF = 83%
(1) CALCULATIONS ARE BASED ON NO RADIATION LOSSES TO HEATERS OR EXTRACTION PIPING LOCATED IN THE CONDENSER NECK. (2) PRIMARY VALVE AND ABOVE HEAT RATES ARE CALCULATED ON LOCUS OF VALVE POINTS STEAM GEN FLOW AT MAX CALC. IS NOT GUARANTEED. MAX GUAR. SG FLOW = 13203282 LB/HR MAX.CALC. SG FLOW = 13947446 LB/HR.	

1021793 KW TURB-GEN UNIT TC6F-44 IN.
730 PSIA - 507.8 F 1.5 IN. HGA
1125600 KVA 0.90 PF 22000 VOLTS 75 #H2

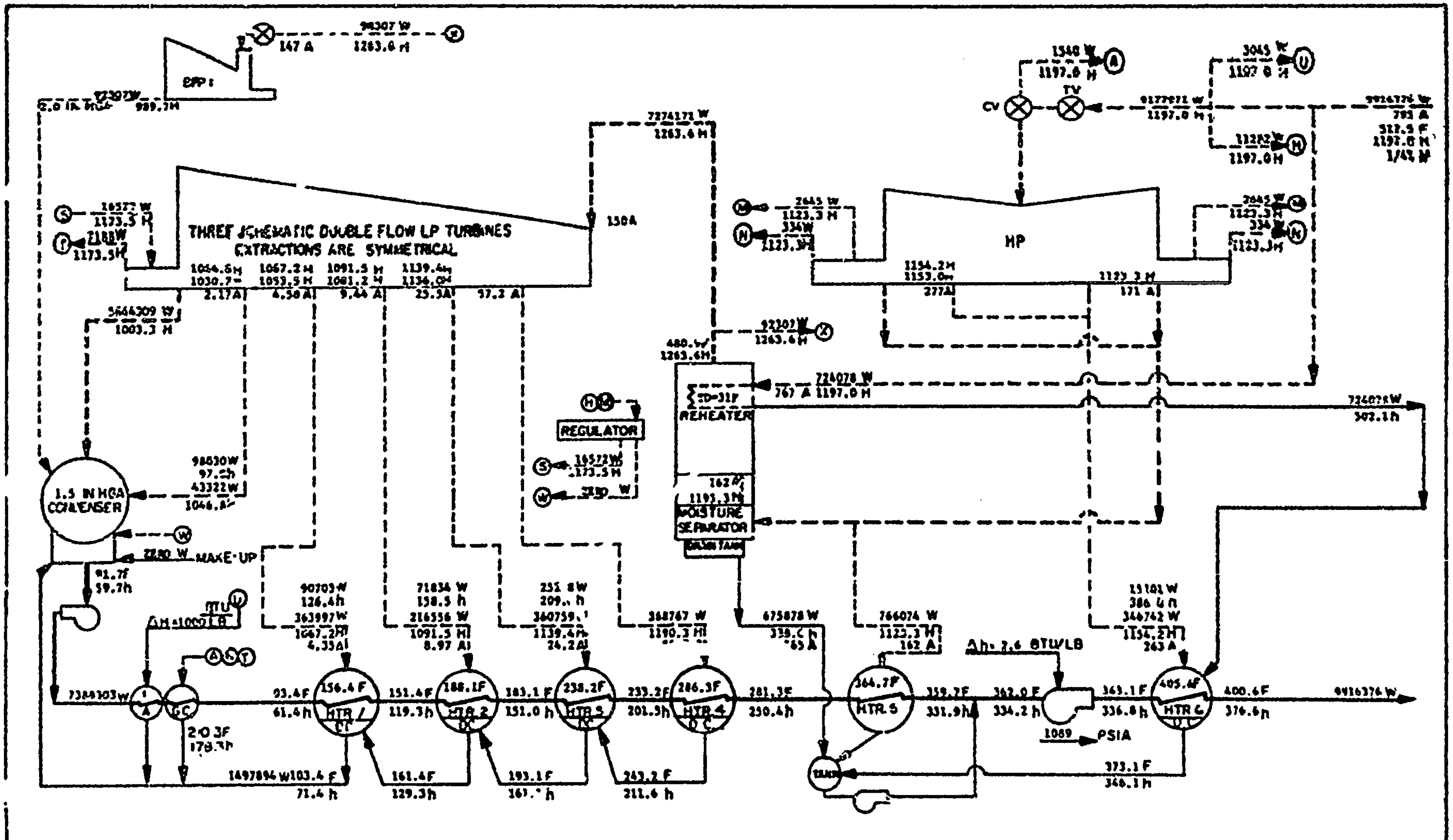
1068701 KW NET LOAD HEAT BALANCE

MAXIMUM CALCULATED - NOT GUARANTEED

FIGURE 10-2



$\frac{13283282\ (1198.8-405.8)}{1021793} = 10309$ BTU/KW HR …NS ARE BASED ON NO RADIATION LOSSES TO HEATERS OR EXTRACTION PIPING LOCATED IN CONDENSER NECK. … VALVE AND ABOVE HEAT RATES ARE CALCULATED ON LOCUS OF VALVE POINTS STEAM GEN FLOW AT MAX CALC. IS NOT GUARANTEED. VALS.GUAR. SG. FLOW = 13283282 LB/HR MAX CALC. SG. FLOW = 13947446 LB/HR.	TEP = 994.0 BTU/LB ELEP = 976.7 BTU/LB MECH LOSS = 3648 KW ELECT LOSS = 12661 KW 0.90 PF 75 #H2 FWP POWER = 12226 KW FWP EFF = 88%	1021793KW TURB-GEN UNIT TC6F-44IN 730PSIA - 507.8F 1.5IN HGA 1125600 KVA 0.90 PF 22000 VOLTS 75 #H2 1021793 KW NET LOAD HEAT BALANCE MAXIMUM GUARANTEED FIGURE 10-3



NET HEAT RATE $= \dfrac{9916376\ (1197.0-376.6)}{766350} = 10616$ BTU/KW HR

(1) CALCULATIONS ARE BASED ON NO RADIATION LOSSES TO HEATERS OR EXTRACTION PIPING LOCATED IN THE CONDENSER NECK
(2) PRIMARY VALVE AND ABOVE HEAT RATES ARE CALCULATED ON LOCUS OF VALVE POINTS

STEAM GEN FLOW AT MAX CALC. IS NOT GUARANTEED
MAX GUAR. S.G. FLOW = 13283282 LB/HR MAX.CALC. S.G. FLOW = 13947446 LB/HR.

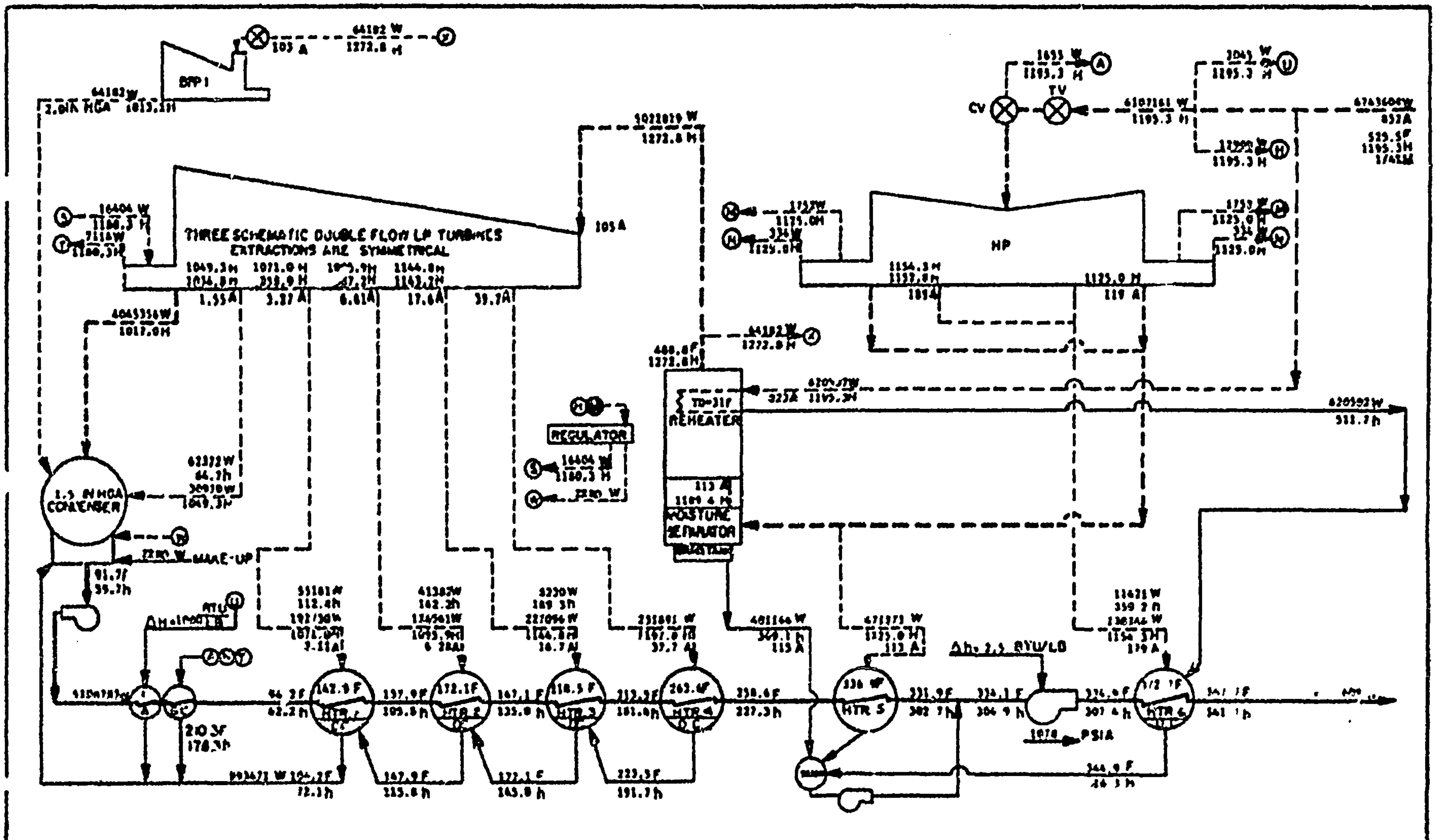
TEP = 1003.0 BTU/LB
ELEP = 995.0 BTU/LB
MECH LOSS = 3648 KW
ELECT LOSS = 9180 KW
0.90 PF 75 # H_2
FWP POWER = 7411 KW
FWP EFF = 85%

1021793 KW TURB-GEN UNIT TCGF-44 IN
730 PSIA - 507.8 F 1.5 IN HGA
1125600 KVA 0.90 PF 22000 VOLTS 75 # H_2

766350 KW NET LOAD HEAT BALANCE

75% LOAD

FIGURE 10-4



NET HEAT RATE = 6763604 (1195.3−341.7) / 510697 = 11267 BTU/KW HR

(1) CALCULATIONS ARE BASED ON NO RADIATION LOSSES TO HEATERS OR EXTRACTION PIPING LOCATED IN THE CONDENSER NECK
(2) PRIMARY VALVE AND ABOVE HEAT RATES ARE CALCULATED ON LOCUS OF VALVE POINTS

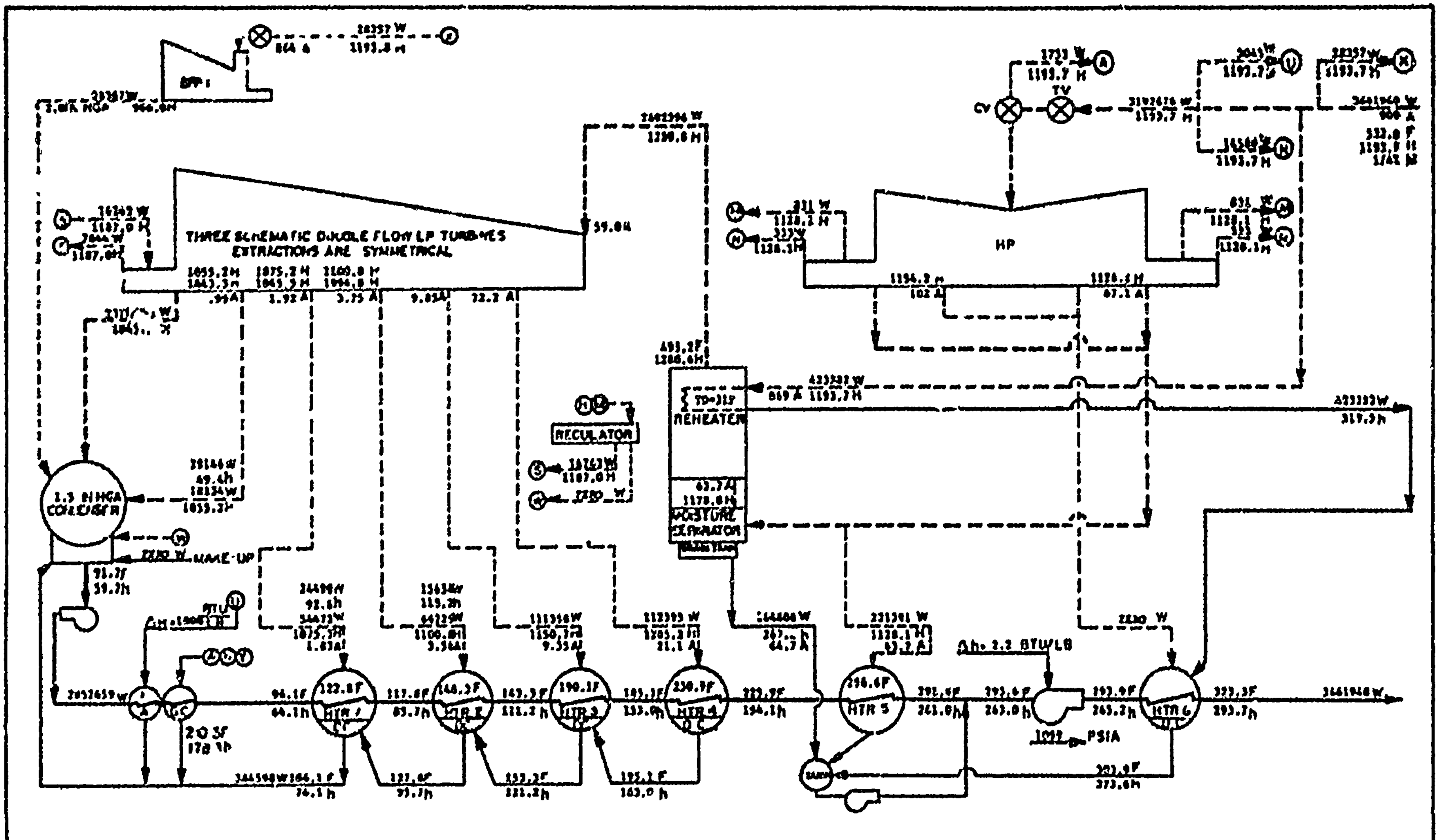
STEAM GEN FLOW AT MAX CALC. IS NOT GUARANTEED
VAV GUAR. SG. FLOW=13283282 LB/HR MAX CALC. SG. FLOW= 13967446 LB/HR.

TEP = 1016.6 BTU/LB
ELEP = 1010.9 BTU/LB
MECH LOSS = 3648 KW
ELECT LOSS = 6573 KW
0.90 PF 75 #H_2
POWER = 4683 KW
EFF = 77 %

1021793 KW TURB-GEN UNIT TC6F-44 IN
730 PSIA 507 °F 1.5 IN HGA
1135600 KVA 0.9 PF 22000 VOLTS 75 #H_2

510697 KW NET LOAD HEAT BALANCE

FIGURE 10-5



NET HEAT RATE = 3661940 (1193.7-293.7) / 255448 = 12907 BTU/KW HR

(1) CALCULATIONS ARE BASED ON NO RADIATION LOSSES TO HEATERS OR EXTRACTION PIPING LOCATED IN THE CONDENSER NECK
(2) PRIMARY VALVE AND ABOVE HEAT RATES ARE CALCULATED ON LOCUS OF VALVE POINTS

STEAM GEN FLOW AT MAX CALC IS NOT GUARANTEED
MAX GUAR. SG FLOW = 13203243 LB/HR MAX CALC. SG FLOW = 13947446 LB/HR.

TEP = 1044.7 BTU/LB
ELEP = 1034.2 BTU/LB
MECH LOSS = 3448 KW
ELECT LOSS = 5032 KW
0.90 PF 75 # H_2
FWP POWER = 2391 KW
FWP EFF = 80%

1031793 KW TURB-GEN UNIT TC6F-44 IN.
730 PSIA - 507.8 F 1.5 IN HGA
1125600 KVA 0.90 PF 22900 VOLTS 75 # H_2

255448 KW NET LOAD HEAT BALANCE

25% LOAD

FIGURE 10-6

Indian Point
Final Facility Description and Safety Analysis Report
Dated: October 15, 1968

Ex 4

Figure 10-7, Titled "Turbine Generator Building General Arrangement – Cross Section"

The above was **redacted** by NRC staff as **sensitive information** to allow release of this document to the public.

Indian Point
Final Facility Description and Safety Analysis Report
Dated: October 15, 1968

E-4

Figure 10-7a, Titled "Turbine Generator Building General Arrangement – Operating Floor"

The above was **redacted** by NRC staff as **sensitive information** to allow release of this document to the public.

STEAM-TURBINE DRIVEN AUXILIARY FEEDWATER PUMP ESTIMATED PERFORMANCE CHARACTERISTIC

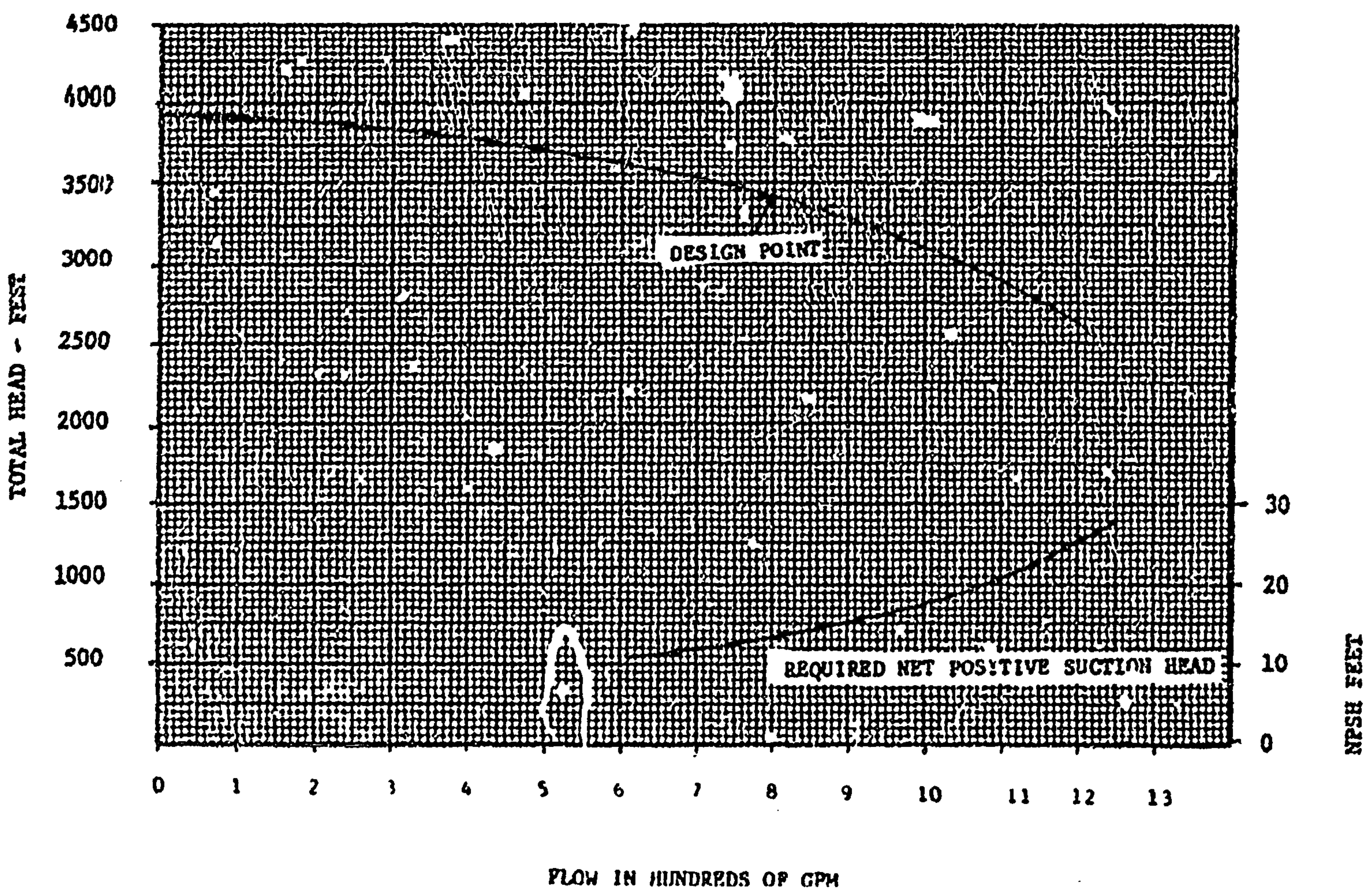


FIGURE 10-8

MOTOR - DRIVEN AUXILIARY FEEDWATER

PUMP ESTIMATED PERFORMANCE CHARACTERISTIC

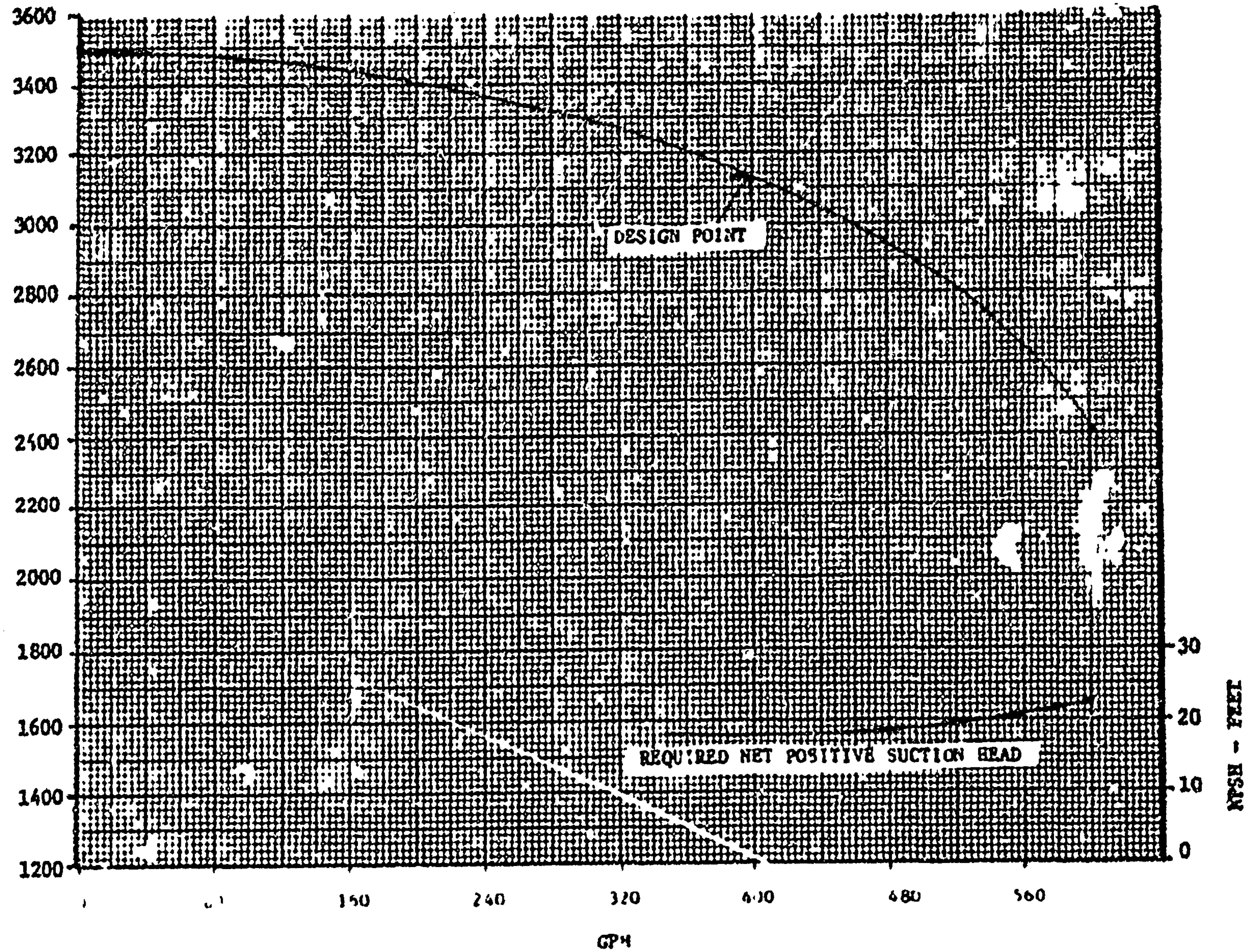


FIGURE 10-9

Chap 11

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BY QUESTION RESPONSES IN
VOLUMES 5 AND 6

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11. WASTE DISPOSAL AND RADIATION PROTECTION SYSTEM

11.1 WASTE DISPOSAL SYSTEM

11.1.1 DESIGN BASES

Control of Releases of Radioactivity to the Environment

Criterion: The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control must be justified (a) on the basis of 10 CFR 20 requirements, for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurence (GDC 70).

Liquid, gaseous, and solid waste disposal facilities are designed so that discharge of effluents and off-site shipments are in accordance with applicable governmental regulations.

Radioactive fluids entering the Waste Disposal System are collected in sumps and tanks until determination of subsequent treatment can be made. They are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown if necessary. Before any attempt is made to discharge radioactive waste, they are processed as required and then released under controlled conditions. The system design and operation are characteristically directed toward minimizing releases to unrestricted areas. Discharge streams are appropriately monitored and safety features are incorporated to preclude releases in excess of the limits of 10 CFR 20.

The bulk of the radioactive liquids discharged from the Reactor Coolant System are processed and retained inside the plant by the Chemical and Volume Control System recycle train. This minimizes liquid input to the Waste Disposal System which processes relatively small quantities of generally low-activity level wastes. The processed water from waste disposal, from which most of the radioactive material has been removed, is discharged through a monitored line into the circulating water discharge.

Radioactive gases are pumped by compressors through a manifold to one of the gas decay tanks where they are held a suitable period of time for decay. Cover gases in the nitrogen blanketing system are re-used to minimize gaseous wastes. During normal operation, gases are discharged intermittently at a controlled rate from these tanks through the monitored plant vent. The system is provided with discharge controls so that the release of radioactive effluents to the atmosphere is controlled within the limits set in the Technical Specifications.

The spent resins from the demineralizers, the filter cartridges and the concentrates from the evaporators are packaged and stored on-site until shipment off-site for disposal. Suitable containers are used to package these solids at the highest practical concentrations to minimize the number of containers shipped for burial.

All solid waste is placed in suitable containers and stored on-site until shipment off-site is made for disposal.

11.1.2 SYSTEM DESIGN AND OPERATION

The Waste Disposal System Process Flow Diagrams are shown in Figures 11.1-1 and 11.1-2, and Performance Data are given in Table 11.1-1.

The Waste Disposal System collects and processes all potentially radioactive primary plant wastes for removal from the plant site within limitations established by applicable governmental regulations. Fluid wastes are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown if necessary, before any attempt is made to discharge them and they are then released under controlled conditions. A radiation monitor is provided to maintain surveillance over the release operation, but the permanent record of activity releases is provided by radiochemical analysis of known quantities of waste. The system is capable of processing all wastes generated during continuous operation of the primary system assuming that fission products corresponding to defects in one per cent of the fuel cladding, escape into the reactor coolant.

As secondary functions, system components supply hydrogen and nitrogen to primary system components as required during normal operation, and provide facilities to transfer fluids from inside the containment to other systems outside the containment.

System Description

Liquid Processing

During normal plant operation the Waste Disposal System processes liquids from the following sources:

a) Equipment drains and leaks
b) Radioactive chemical laboratory drains
c) Decontamination drains
d) Demineralizer regeneration
e) Floor drains

The system also collects and transfers liquid drained from the following sources directly to the Chemical and Volume Control System for processing:

a) Reactor coolant loops
b) Pressurizer relief tank
c) Reactor coolant pump secondary seals
d) Excess letdown during startup
e) Accumulators
f) Valve and reactor vessel flange leakoffs

The valve and reactor flange leakoff liquids flow to the reactor coolant drain tank and are discharged directly to the CVCS holdup tanks by the reactor coolant drain pumps which are operated automatically by a level controller in the tank. These pumps also return water from the refueling canal and cavity to the refueling water storage tank.

Where plant layout permits, waste liquids drain to the waste holdup tank by gravity flow. Other waste liquids including floor drains drain to the sump tank and are discharged to the waste holdup tank by pumps operated automatically by a level controller in the tank.

If preliminary analysis by sampling indicates that the liquid is suitable for discharge, it is pumped from the waste holdup tank to the waste condensate tanks where its activity can be determined for record by isolating, sampling and analyzing before it is discharged through the radiation monitor to the condenser circulating water.

Liquids requiring cleanup before release are processed in batches by the waste evaporator. The concentrated bottoms are discharged to the drumming room where they are packaged and stored until removal to a burial facility. The condensate is routed to one of two waste condensate tanks. When one tank is filled, it is isolated and sampled for analysis while the second tank is in service. If analysis confirms the activity level is suitable for discharge, the condensate is pumped through a flow meter and a radiation monitor to the condenser circulating water discharge. Otherwise it is returned to the waste holdup tank for reprocessing. Although the radiochemical analysis forms the basis for recording activity releases, the radiation monitor provides surveillance over the operation by preventing the discharge valve from opening if the liquid activity level exceeds that which can be safely discharged.

Gas Processing

During plant operations, gaseous wastes will originate from:

a) Degassing reactor coolant and purging the volume control tank

b) Displacement of cover gases as liquid accumulates in various tanks

c) Equipment purging

d) Sampling operations and automatic gas analysis for hydrogen and oxygen in cover gases.

During normal operation the Waste Disposal System supplies nitrogen and hydrogen from standard cylinders to primary plant components. Two headers are provided, one for operation and one for backup. The pressure regulator in the operating header is set for 100 psig discharge and that in the backup header at 90 psig. When the operating header is exhausted, its discharge pressure will fall below 100 psig and an alarm will alert the operator. The second tank will come into service automatically at 90 psig to ensure a continuous supply of gas. After the exhausted header has been replaced, the operator manually sets the operating pressure back to 100 psig and the backup pressure at 90 psig. This operation is identical for both the nitrogen supply and the hydrogen supply.

Most of the gas received by the Waste Disposal System during normal operation is cover gas displaced from the Chemical and Volume Control System holdup tanks as they fill with liquid. Since this gas must be replaced when the tanks are emptied during processing, facilities are provided to return gas from the decay tanks to the holdup tanks. A backup supply from the nitrogen header is provided for makeup if return flow from the gas decay tanks is not available. Since the hydrogen concentration may exceed the combustible limit during this type of operation, components discharging to the vent header system are restricted to those containing no air or aerated liquids and the vent header itself is designed to operate at a slight positive pressure (0.5 psig minimum to 2.0 psig maximum) to prevent in-leakage. On the other hand, out-leakage from the system is minimized by using Saunders patent diaphragm valves, bellows seals, self contained pressure regulators and soft-seated packless valves throughout the radioactive portions of the system.

Gases vented to the vent header flow to the waste gas compressor suction header. One of the two compressors is in continuous operation with the second unit instrumented to act as backup for peak load conditions. From the compressors, gas flows to one of the four large gas decay tanks. The control arrangement on the gas decay tank inlet header allows the operator to place one large tank in service and to select a second large tank for backup. When the tank in service becomes pressurized to 110 psig, a pressure transmitter automatically opens the inlet valve to the backup tank, closes the inlet valve to the filled tank, and sounds an alarm to alert the operator of this event so that he may select a new backup tank. Pressure indicators are supplied to aid the operator in selecting the backup tank.

Gas held in the decay tanks can either be returned to the Chemical and Volume Control System holdup tanks, or discharged to the atmosphere if the activity concentration is suitable for release. Generally, the last tank to receive gas will be the first tank emptied back to the holdup tanks in order to permit the maximum decay time before releasing gas to the environment. However, the header arrangement at the tank inlet gives the operator freedom to fill, reuse or discharge gas to the environment simultaneously without restriction by operation of the other tanks.

Six additional small gas decay tanks are supplied for use during degassing of the reactor coolant prior to a cold shutdown. The reactor coolant fission gas activity inventory is distributed equally among the six tanks through a common inlet header. With this arrangement assuming 1% defective fuel rods the activity inventory in any one tank will be less than 2.0×10^4 curies of equivalent Xe-133 with this arrangement.

A radiation monitor in the sample line to the gas analyzer checks the gas decay tank activity inventory each time a sample is taken for hydrogen-oxygen analysis. An alarm warns the operator when the inventory limit is approached so that he may place another tank in service.

Before a tank can be emptied to the environment, its contents must be sampled and analyzed to verify sufficient decay and to provide a record of the activity to be released, and only then discharged to the plant vent at a controlled rate through a radiation monitor in the vent. Samples are taken manually by opening the isolation valve to the gas analyzer sample line and permitting gas to flow to the gas analyzer where it can be collected in one of the Sampling System gas sample vessels. After sampling, the isolation valve is closed. During release a trip valve in the discharge line is closed automatically by a high activity level indication in the plant vent.

During operation, gas samples are drawn periodically from tanks discharging to the waste gas vent header as well as from the particular large gas decay tank being filled at the time, and automatically analyzed to determine their hydrogen and oxygen content. The hydrogen analysis is for surveillance since the concentration range will vary considerably from tank to tank. There should be no significant oxygen content in any of the tanks, and an alarm will warn the operator if any sample shows 2% by volume of oxygen. This allows him time to isolate the tank before the combustible limit is reached. Another tank is placed in service while the operator locates and eliminates the source of oxygen. Discharged gases are released from the plant vent and diluted in the atmosphere due to the turbulence in the wake of the containment building in addition to the effects of normal dispersion. Using the model developed in Section 14 and evaluating the yearly meteorological data from Section 2, a sector average dispersion factor of 1.38 x 10^{-5} sec/m^3 has been calculated at the site boundary. Using the MPC values of 10CFR20, the corresponding allowable yearly release is:

Xe-133 and Kr-85	68.) curies
I-131	229 Ci

Based on 1% fuel defects, the maximum expected annual release is 5310 curies Kr-85, 4520 curies Xe-133. It is expected that the release of iodine in the waste gas will be negligible.

Solids Processing

The Waste Disposal System is designed to package all solid wastes in standard 55 gallon drums for removal to burial facilities. Concentrates from the waste evaporator will be pumped into a battery of 6 drums previously filled with a mixture of vermiculite and cement. This type of operation is similar to that successfully used over the past few years at Brookhaven National Laboratories. After filling, the drums are moved to a shielded storage area by a bridge and trolley crane until a sufficient number have accumulated for shipment. The same crane is then used to place the drums on the carrier for removal to a burial site. Maximum dose rate for unshielded drums is 1 R/hr at one meter. Lead shielding is provided for each drum to reduce the dose rate in work areas to 1mR/hr at a one meter range.

Spent resins will be packaged in a similar manner. Normally resin will be stored in the spent resin storage tank for a period of 1 to 6 months for decay. Resin is removed from in the storage tanks first by bubbling nitrogen through the tank to agitate the resin and then pumping water through the tank at a controlled rate to sluice the slurry to the drumming room. There it is received in a battery of 30 gallon containers enclosed in lead shielding approximately a 55 gallon drum in size. The slurry will enter the 30 gallon drum and be de-watered by an internal screen designed to retain the resin. Sluice water returns to the waste holdup tank via the sump tank. These drums are handled in a fashion identical to the concentrates drums. Maximum do.. rate from a 55 gallon size resin container is 1R/hr at one meter. The same lead shielding used around the concentrates drums also reduces dose rates to 1mR/hr at aone meter range for this container. The basis for all dose rate calculations is one cycle of operation with one percent defective fuel.

Miscellaneous solid wastes, such as paper, rags, and glassware, are compressed into 55 gallon drums by a hydraulically operated baler located in the drumming room. Filled drums are stored in a shielded area in the drumming room.

Components

Codes applying to components of the Waste Disposal System are shown in Table 11.1-2. Components summary data are shown in Table 11.1-3. Waste Disposal System components are located in the auxiliary building except for the reactor coolant drain tank which is in the containment and the waste holdup tank which is in the liquid holdup tank vault.

The seismic classification of the Waste Disposal System Components are included in Appendix A.

Regenerant Tank

The regenerant tank is austenitic stainless steel and provides facility to batch caustic used to regenerate anion exchange resins.

Chemical Drain Tank

The chemical drain tank is a vertical cylinder of austenitic stainless steel and collects drainage from the hot section of the chemistry laboratory. After analysis, the tank contents are pumped to the waste holdup tanks or to the waste condensate tanks.

Reactor Coolant Drain Tank

The reactor coolant drain tank is a horizontal cylinder with spherically dished heads. The tank is all welded austenitic stainless steel. This tank serves as a drain surge for the Reactor Coolant System and other equipment located inside the reactor containment.

Waste Holdup Tank

The waste holdup tank is the central collection point for radioactive liquid waste. The tank is stainless steel of welded construction.

Sump Tank and Pumps

The sump tank serves as a collecting point for waste discharged to the basement level drain header. It is located at the lowest point in the auxiliary building. All floor drains enter this tank through a loop seal to prevent back flow of gas from the tank. Two horizontal centrifugal pumps transfer liquid waste to the waste holdup tank. All wetted parts of the pumps are stainless steel. The tank is all welded austenitic stainless steel.

Spent Resin Storage Tank

The spent resin storage tank retains resin discharged from the primary plant demineralizers. Normally, resins are stored in the tank from 1 to 6 months for decay of short-lived isotopes and then the tank is emptied. However, the contents can be removed at any time, if sufficient shielding is provided for the spent resin shipping vessel. A layer of water is maintained over the resin surface as a precaution against resin degradation due to heat generation by decaying fission. Resin is removed from the tank by first backflushing with nitrogen to loosen the resin bed and then flushing the resin out with water entering the bottom of the tank. The tank is all welded austenitic stainless steel.

Gas Decay Tanks

Four large and six small welded carbon steel tanks are provided to contain compressed waste gases (hydrogen, nitrogen, and fission gases). After a period for radioactive decay, these gases may be released at a controlled rate to the atmosphere through the plant vent. All discharges to the atmosphere will be monitored.

Compressors

Two compressors are provided for continuous removal of gases from equipment discharging to the plant vent header. These compressors are of the water-sealed centrifugal displacement type. Operation of the compressors is automatically controlled by radioactive waste gas vent header pressure. Construction is carbon steel. A mechanical seal is provided to maintain outleakage of compressor seal water at a negligible level.

Waste Evaporator Package

The evaporator concentrates dissolved and suspended solids in the liquid wastes. It consists of a batch tank, concentrator, distillate tank, a reagent tank, circulating pumps and control panel.

Surfaces exposed to radioactive liquids are austenitic stainless steel except for the heat transfer surfaces which are admiralty metal.

Waste Condensate Tanks

Two tanks collect waste evaporator condensate. The contents are sampled and analyzed for radioactivity and for chemical waste before discharge. The condensate is transferred by one of two waste condensate pumps to the waste holdup tank if the activity is high or to the condenser circulating water if the activity is sufficiently low. These tanks are constructed of all-welded stainless steel.

Baler

A hydraulically operated baler is used to pack compressible solid wastes into 55 gallon drums. The baler is operated manually from a local station and is supplied with a dust shroud to prevent escape of radioactive particulate matter. The shroud vents to the building exhaust system.

Nitrogen Manifold

Nitrogen, used as cover gas in the vapor space of various components is supplied from a dual manifold. A pressure controller, which automatically switches from one manifold to the other, assures a continuous supply of gas.

Hydrogen Manifold

Hydrogen is supplied to the volume control tank to maintain the hydrogen concentration in the reactor coolant. The hydrogen is supplied from a dual manifold. A pressure controller, which automatically switches from one manifold to the other, assures a continuous supply of gas.

Gas Analyzer

An automatic gas analyzer with a nominal one hour recycle time is provided to monitor the concentrations of oxygen and hydrogen in the cover gas of tanks discharging to the radiogas vent header. Upon indication of a high oxygen level, an alarm sounds to alert the operator.

Pumps

Pumps used throughout the system for draining tanks and transferring liquids are shown in Figures 11.1-1 and 11.1-2.

The wetted surfaces of all pumps are stainless steel.

Piping

Piping carrying liquid wastes is stainless steel while all gas piping is carbon steel. Piping connections are welded except where flanged connections are necessary to facilitate equipment maintenance.

Valves

All valves exposed to gases are carbon steel. All other valves are stainless steel. All valves have stem leakage control. Globe valves are installed with flow over the seats when such an arrangement reduces the possibility of leakage.

Stop valves are provided to isolate each piece of equipment for maintenance, to direct the flow of waste through the system, and to isolate storage tanks for radioactive decay.

Relief valves are provided for tanks containing radioactive waste if the tanks might be overpressurized by improper operation or component malfunction. Tanks containing wastes which contain oxygen and are normally of low activity concentrations are vented into the auxiliary building exhaust system.

11.1.3 DESIGN EVALUATION

Liquid Wastes

Liquid Wastes are generated primarily by plant maintenance and service operations and consequently, the quantities and activity concentrations of influents to the system, Tables 11.1-4 and 11.1-5, are estimated values. Therefore, considerable operational margin has been assigned between the estimated system load and the design capability as indicated in Table 11.1-4, Estimated Discharges to Waste Disposal. A conservative estimate of system ability to limit dissolved and suspended activity released in the liquid phase is summarized in Table 11.1-5, Estimated Liquid Release by Isotope. This tabulation is based on the following assumptions.

1. All liquid waste is initially at peak reactor coolant activity concentrations based on continuous full power operation with 1% defective fuel clad.

2. Allow 500 minutes for decay, the time required to process a 1000 gallon batch at 2 gallons per minute.

3. Concentrate the waste to a bottoms activity concentration of 40 μc/cc, the drumming facility design limit.

4. Divide by the waste evaporator DF of 10^6 which yields 4×10^{-5} μc/cc in the waste condensate.

5. Multiply by the quantity released, 169,050 gal. per year (Table 11.1-4), to obtain the total estimated annual release of 25.2 mc/yr excluding tritium.

6. Add to this the activity released through waste disposal by the CVCS monitor tanks. This is estimated to be less than 1 mc/yr.

7. The tritium estimates in Table 11.1-5 assume that one third of all tritium formed in the fuel (the predominant source) diffuses through the zirc clad and eventually becomes available for dispersal to the environment. This is believed to be an overestimate for zirc clad fuel.

To maintain the concentration of tritium in the reactor coolant at a level which precludes hazard to personnel during access to the containment, part of the condensate from the boric acid recovery process is discharged to the river via the plant Circulating Water System.

If the tot... ...t of tritium released to the reactor coolant over a year is released ... the Circulating Water System, the annual average concentration of tritium released to the environment is between 10^{-2} and 10^{-4} of 10 CFR 20 limits (See Table 11.1-6).

Gaseous Wastes

Gaseous wastes consist primarily of hydrogen stripped from coolant discharged to the CVCS holdup tanks during boron dilution, nitrogen and hydrogen gases purged from the CVCS volume control when degassing the reactor coolant, and nitrogen from the closed gas blanketing system. The gas decay tank capacity will permit 45 days decay of waste gas before discharge. Table 11.1-7 contains an estimate of annual noble gas activity release based on the following assumptions:

For Xe-133:

1. The quantity of Xe-133 removed from the plant over a core cycle is determined assuming all gaseous waste is initially at peak reactor coolant activity concentration based on 1% defective fuel clad, and 3216 Mwt power with a 60 hour load reduction to 50% power each weekend.

2. Using the same reactor coolant activity concentrations as in (1), the total Xe-133 removed to the Waste Disposal System by degassing the Reactor Coolant System for the cold shutdowns are combined. The cold shutdowns occur at the following times: (a) during the second week of operation, (b) at the peak xenon level and (c) during refueling.

4. Sum items 1, 2 and 3 to obtain the total Xe-133 removed to the Waste Disposal System and allow for 45 days decay to obtain the total estimated annual release of Xe-133.

For Kr-85:

1. Since there is not significant decay of Kr-85 during the operating periods involved, the total Kr-85 that enters the reactor coolant during the core cycle three cold shutdowns and four hot shutdowns is determined and it is assumed that it eventually is released through the Waste Disposal System.

In comparison to Kr-85 and Xe-133, there will be no significant activity release after 45 days of decay from the remaining gaseous wastes since the isotopes half lives are short and/or the quantities present in the reactor coolant are small.

In the event of a pipe or tank rupture, the maximum anticipated quantity of waste gas, that could be released from any one tank in the system is less than 2.0×10^4 curies of equivalent Xe-133 which would result in a dose of less than 0.5 rem beyond the site exclusion boundary.

Solid Wastes

Solid wastes consist of waste liquid concentrates, spent resins and miscellaneous materials such as paper, and glass ware. All solid wastes are packaged in 55 gallon drums for removal to a burial facility. It is estimated that 30 to 50 drums of each type waste will be produced per year for a total of 90 to 150 drums.

11.1.4 MINIMUM OPERATING CONDITIONS

Minimum operating conditions for the Waste Disposal System are enumerated in the Technical Specifications (Section 15).

TABLE 11.1-1

WASTE DISPOSAL SYSTEM

PERFORMANCE DATA

Plant Design Life	40 years
Normal process capacity, liquids	2 gpm
Load factor[(1)]	17%
Annual liquid discharge[(2)]	
Volume	169,050 gal.
Activity	25.2 mc
Annual gaseous discharge	
Activity	9,850 c
Annual solids shipped for burial	
Volume (55 gallon units)	90–150

(1) Batch operation

(2) estimate based on Table 11.1-4

TABLE 11.1-2

WASTE DISPOSAL COMPONENTS CODE REQUIREMENTS

Component	Code
Chemical Drain Tank	No code
Reactor Coolant Drain Tank	ASME III,[1] Class C
Sump Tank	No code
Spent Resin Storage Tanks	ASME III,[1] Class C
Gas Decay Tanks	ASME III,[1] Class C
Waste Holdup Tank	No code
Water Condensate Tank	No code
Regenerant Tank	No code
Waste Evaporator	No code
Waste Filter	No code
Piping and Valves	USAS-B31.1[2] Section 1

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(1) ASME III American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section IV, Nuclear Vessels

(2) USAS-B31.1-Code for pressure piping US American Standards Association and special nuclear cases where applicable.

TABLE 11.1-3

COMPONENT SUMMARY DATA

Tanks	Quantity	Type	Volume	Design Pressure	Design Temperature	Material
Reactor Coolant Drain	1	Horiz	350 gal	25 psig	267°F	ss
Regenerant Tank	1	Vert	400 gal	Atm	180°F	ss
Chemical Drain	1	Vert	375 gal	Atm	180°F	ss
Sump Tank	1	Vert	375 gal	Atm	150°F	ss
Waste Holdup	1	Horiz	3300 ft^3	Atm	150°F	ss
Spent Resin Storage	1	Vert	300 ft^3	100 psig	150°F	ss
Waste Condensate	2	Vert	1000 gal	Atm	180°F	ss
Gas Decay (large)	6	Vert	525 ft^3	150 psig	150°F	cs
Gas Decay (small)	6	Vert	60 ft^3	150 psig	150°F	cs

Pumps	Quantity	Type	Flow gpm	Head ft	Design Pressure	Design Temp	Materi
Reactor Coolant Drain (A)	1	Horiz cent canned	150	175	100 psig	267°F	ss
Reactor Coolant Drain (B)	1	Horiz cent canned	50	175	100 psig	267°F	ss
Chemical Drain	1	Horiz cent$^{(2)}$	20	100	100 psig	180°F	ss
Regenerant	1	Horiz cent$^{(2)}$	20	100	100 psig	180°F	ss
Sump Tank	2	Horiz cent$^{(2)}$	20	100	100 psig	180°F	ss
Waste Evaporator Feed	1	Horiz cent$^{(2)}$	20	100	100 psig	180°F	ss

TABLE 11.1-3 (Continued)

Waste Condensate	2	Horiz cent(2)	20	100	100 psig	180°F	ss

Miscellaneous	Quantity	Capacity	Type
Waste Evaporator	1	2 gpm	-
Waste Gas Compressors	2	20 CFM	horiz(2) cent

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(1)Wetted surfaces only

(2)Mechanical seal provided

TABLE 11.1-4

ESTIMATED LIQUID DISCHARGE TO WASTE DISPOSAL

Source	Peak Weekly Discharge, gal. At Power	Refueling	Total Annual Discharge, gal.
Laboratories	300	300	15,600
Equipment drains, leaks	2100	2100	109,200
Decontamination	800*	700	16,100
Demineralizer regeneration	1380	--	28,750
Totals	5130	3100	169,050
Load factor(1)	26%	16%	17%

(1) Based on 2 gpm process capability.

* Includes decontamination of spent fuel shipping cask.

TABLE 11.1-5

ESTIMATED LIQUID RELEASE BY ISOTOPE

Isotope	Annual Release μc	Isotope	Annual Release μc
H 3	4.238×10^{9}	I-131	5.52×10^{3}
Mn 54	9.08×10^{-1}	Te 132	5.6×10^{2}
Mn 56	1.94×10^{1}	I 132	1.76×10^{2}
Co 58	2.34×10^{1}	I 133	3.35×10^{3}
Co 60	2.99	I 134	7.62×10^{0}
Sr 89	9.09	I 135	1.88×10^{3}
Sr 90	4.46×10^{-1}	Cs 134	7.44×10^{2}
Y 90	8.32×10^{-1}	Cs 136	7.10×10^{1}
Sr 91	2.37	Cs 137	3.19×10^{3}
Y 91	1.82×10^{1}	Ba 140	1.88
Y 92	3.52×10^{-1}	La 140	1.75
MO 99	9.92×10^{3}	Ce 144	6.85

Total (Tritium excluded) 25.2×10^{3} μc

TABLE 11.1-6
TIRTIUM RELEASE INTO CIRCULATING WATER SYSTEM

	Initial Cycle	Equilibirum Cycle
Total tritium in coolant	5.273×10^9 μc/yr	4.238×10^9 μc/yr
Annual average concentration in Circulating Water System (total tritium/15×10^{14} cc/yr flow)	3.52×10^{-6} μc/cc	2.82×10^{-6} μc/cc
Fraction of MPC (10 CFR 20 Table II) (μc/cc/3×10^{-3} μc/cc)	1.17×10^{-3}	9.04×10^{-4}

TABLE 11.1-7

ESTIMATED ANNUAL GASEOUS RELEASE BY ISOTOPE TO ATMOSPHERE

Isotope		Activity Release to Environment Curies/yr
H3		Neglibible
Kr 85		5310
Kr 85m, 87, 88		Negligible
Xe 133		4540
Xe 133m, 135, 135m, 138		Negligible
	Total	9850

11.2 RADIATION PROTECTION

11.2.1 DESIGN BASES

Monitoring Radioactivity Releases

Criterion: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive. (GDC 17)

The containment atmosphere, the plant vent, the containment fan-cooler's service water discharge, the Waste Disposal System liquid effluent, the condenser air ejectors, and steam generators blowdown are monitored for radioactivity during normal operations, from anticipated transients, and from accident conditions.

All gaseous effluent from possible sources of accidental releases of radioactivity external to the reactor containment (e.g., the spent fuel pit and waste handling equipment) will be exhausted from the plant vent which is monitored. All accidental spills of liquids are maintained within the auxiliary building and collected in a drain tank. Any contaminated liquid effluent discharged to the condenser circulating water canal is monitored. For the case of leakage from the reactor containment under accident conditions the plant area radiation monitoring system supplemented by portable survey equipment to be kept in the Health Physics office area should provide adequate monitoring of accident releases. The details of the procedures and equipment to be used in the event of an accident are specified in Sections 11.2.2 and 12.3. The formulation of these details will consider the requirements for notification of plant personnel, the utility load dispatcher and local authorities.

Monitoring Fuel and Waste Storage

Criterion: Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels (GDC 18).

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas to detect inadequate cooling and to detect excessive radiation levels. Radiation monitors are provided to maintain surveillance over the release operation, but the permanent record of activity releases is provided by radiochemical analysis of known quantities of waste.

The spent fuel pit cooling loop flow is monitored to assure proper operation, as shown in Section 9.

A controlled ventilation system removes gaseous radioactivity from the atmosphere and fuel storage and waste treating areas of the auxiliary building and discharges it to the atmosphere via the plant vent. Radiation monitors are in continuous service in these areas to actuate high-activity alarms on the control board annunciator, as described in Section 11.2.3.

Fuel and Waste Storage Radiation Shielding

Criterion: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities. (GDC 68)

Auxiliary shielding for the Waste Disposal System and its storage components is designed to limit the dose rate to levels not exceeding .75 mr/hr in normally occupied areas, to levels not exceeding 2.0 mr/hr in intermittently occupied areas and to levels not exceeding 15 mr/hr in limited occupancy areas.

Gamma radiation is continuously monitored in the auxiliary building. A high level signal is alarmed locally and annunciated in the control room.

Protection Against Radioactivity Release from Spent Fuel and Waste Storage

Criterion: Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity. (GDC 69)

All waste handling and storage facilities are contained and equipment designed so that accidental releases directly to the atmosphere are monitored and will not exceed the limits of 10 CFR 100; refer also to Sections 11.1.2, 14.2.2, and 14.2.3. The components of the Waste Disposal System are not subjected to any high pressures (see Table 11.1-3) or stresses and are Class I seismic design. In addition, the tanks which have a design pressure greater than atmospheric pressure, piping and valves of the system are designed to the codes given in Table 11.1-2. Hence the probability of a rupture or failure of the system is exceedingly low.

11.2.2 SHIELDING

Design Basis

Radiation shielding is designed for operation at maximum calculated thermal power and to limit the normal operation radiation levels at the site boundary below those levels allowed for continuous non-occupational exposure. The plant is capable of continued safe operation with 1% fuel element defects.

In addition, the shielding provided ensures that in the event of a hypothetical accident, the integrated off-site exposure due to the contained activity does not result in any harmful off-site radiation exposures.

Operating personnel at the plant are protected by adequate shielding, monitorin and operating procedures. Each area in the plant is classed according to the dose rate allowable in the area. The allowable dose rate is based on the expected frequency and duration of occupancy. All plant areas capable of personnel occupancy are classified as one of the six zones of radiation level listed in Table 11.2-1. The shielding arrangement is shown on Figures 11.2-1 through 11.2-3.

Typical Zone 0 areas are the turbine building and turbine plant service areas and the control room. Typical Zone I areas are the offices, auxiliary building work stations and corridors, and the outer surfaces of the containment and auxiliary building. Zone II areas would include the surface of the refueling water at refueling and the operating deck of the containment during reactor shutdown. Areas designated Zone III include the sampling room, reactor cavity area after shutdown, and reactor containment penetration areas, including ventilation, steam line and electrical penetrations. Typical Zone IV areas include areas within the auxiliary building such as charging pump areas, evaporation area, heat exchanger areas, and valve operator areas. Typical Zone V areas are within the regions adjacent to the Reactor Coolant System at power operation and the demineralizer and volume control tank spaces.

All high radiation areas are appropriately marked and isolated in accordance with 10 CFR 20 and other applicable regulations.

The shielding is divided into five categories according to function. These functions include the primary shielding, the secondary shielding, the accident shielding, the fuel transfer shielding, and the auxiliary shielding.

Primary Shield

The primary shield is designed to:

1. Reduce the neutron fluxes incident on the reactor vessel to limit the radiation induced increase in transition temperature.

2. Attentuate the neutron flux sufficiently to prevent excessive activation of plant components.

3. Limit the gamma fluxes in the reactor vessel and the primary concrete shield to avoid excessive temperature gradients or dehydration of the primary shield.

4. Reduce the residual radiation from the core, reactor internals and reactor vessel to levels which will permit access to the region between the primary and secondary shields after plant shutdown.

5. Reduce the contribution of radiation leaking to obtain optimium division of the shielding between the primary and secondary shields.

Secondary Shield

The main function of the secondary shielding is to attentuate the radiation originating in the reactor and the reactor coolant. The major source in the reactor coolant is the Nitrogen -16 activity (83 μc/cc maximum), which is produced by neutron activation of oxygen during passage of the coolant through the core. The secondary shield will limit the full power dose rate outside the containment building to less than 0.75 mr/hr.

Accident Shield

The main purpose of the accident shield is to ensure safe radiation levels outside the containment building following a maximum credible accident.

Fuel Handling Shield

The fuel handling shield is designed to facilitate the removal and transfer of spent fuel assemblies and control rod clusters from the reactor vessel to the spent fuel pit. It is designed to attentuate radiation from spent fuel, control clusters, and reactor vessel internals to less than 2.0 mr/hr at the refueling cavity water surface and less than 0.75 mr/hr in areas adjacent to the spent fuel pit.

Auxiliary Shielding

The function of the shielding is to protect personnel working near various system components in the Chemical and Volume Control System, the Residual Heat Removal System, the Waste Disposal System and the Sampling System. The shielding provided for the auxiliary building is designed to limit the dose rate to less than 0.75 mr/hr in normally occupied areas, and at or below 0.2 mr/hr in intermittently occupied areas.

Shielding Design

Primary Shield

The primary shield consists of the core baffle, water annuli, barrel-thermal shield (all of which are within the reactor vessel), the reactor vessel wall, and a concrete structure surrounding the reactor vessel.

The primary shield immediately surrounding the reactor vessel consists of an annular reinforced concrete structure extending from the base of the containment to an elevation of 69 feet. The lower portion of the shield is a minimum thickness of 6 feet of regular concrete ($\rho = 2.3\ g/cm^3$) and is an integral part of the main structural concrete support for the reactor vessel. It extends upward to join the concrete cavity over the reactor. The reactor cavity, which is approximately rectangular in shape, extends upward to the operating floor with vertical walls 4 ft. thick, except in the area adjacent to fuel handling, where the thickness is increased to 6 ft. A shielding collar is provided at each point where the eight reactor coolant pipes penetrate the primary shield.

The primary concrete shield is air cooled to prevent overheating and dehydration from the heat generated by radiation absorption in the concrete. Eight "windows" have been provided in the primary shield for insertion of the out-of-core nuclear instrumentation. Cooling for the primary shield concrete and the nuclear instrumentation is provided by 12,000 cfm cooling air.

The primary shield neutron fluxes and design parameters are listed in Table 11.2-2.

Secondary Shield

The secondary shield surrounds the reactor coolant loops and the primary shield. It consists of the annular crane support wall, the operating floor, and the reactor containment structure. The containment structure also serves as the accident shield.

The lower portion of the secondary shield above grade consists of the 4 ft -6 in cylindrical portion of the reactor containment and a 3 ft concrete annular crane support wall surronding the reactor coolant loops. The secondary shield will attenuate the radiation levels in the primary loop compartment from a value of 25 rem/hr to a level of less than 0.75 mr/hr outside the reactor containment building. Penetrations in the secondary shielding are protected by supplemental shields.

The secondary shield design parameters are listed in Table 11.2-3.

Accident Shield

The accident shield consists of the 4 ft. -6 in. reinforced concrete cylinder capped by a hemispherical reinforced concrete dome of a 3 ft. -6 in. thickness. This shielding includes supplemental shields in front of the containment penetrations.

The equipment access hatch is shielded by a 3 ft. -6 in. thick concrete shadow shield and a 1 ft. -6 in. concrete roof to reduce scattered dose levels in the event of loss of reactor coolant accident accompanied by a complete core meltdown.

The accident shield design parameters are listed in Table 11.2-4.

Fuel Handling Shield

The refueling cavity, flooded to elevation 93.7 feet during refueling operations, provides a temporary water shield above the components being withdrawn from the reactor vessel. The water height during refueling is approximately 24 1/2 ft. above the reactor vessel flange. This neight ensures that a minimum of 10 1/2 ft. of water will be above the active fuel of a withdrawn fuel assembly. Under these conditions, the dose rate is less than 2.0 mr/hr at the water surface.

The refueling canal is a passageway connected to the reactor cavity and extending to the inside surface of the reactor containment. The canal is formed by two concrete walls each 6 ft. thick, which extends upward to the same height as the reactor cavity. During refueling the canal is flooded with borated water to the same height as the reactor cavity.

The spent fuel assemblies and control rod clusters are remotely removed from the reactor containment through the horizontal spent fuel transfer tube and placed in the spent fuel pit. Concrete, 6 ft. thick, shields the spent fuel transfer tube. This shielding is designed to protect personnel from radiation during the time a spent fuel assembly is passing through the main concrete support of the reactor containment and the transfer tube.

Radial shielding during fuel transfer is provided by the water and concrete walls of the fuel transfer pit. An equivalent of 6 feet of regular concrete is provided to insure a maximum dose value of 0.75 mr/hr. in the areas adjacent to the spent fuel pit.

Spent fuel is stored in the spent fuel pit which is located adjacent to the containment building. Shielding for the spent fuel storage pit is provided by 6 feet thick concrete walls and is flooded to a level such that the water height is greater than 13 ft. above the spent fuel assemblies.

The refueling shield design parameters are listed in Table 11.2-5.

Auxiliary Shielding

The auxiliary shield consists of concrete walls around certain components and piping which process reactor coolant. In some cases, the concrete block walls are removable to allow personnel access to equipment during maintenance periods. Periodic access to the auxiliary building is allowed during reactor operation. Each equipment compartment is individually shielded so that compartments may be entered without having to shut down and, possibly, to decontaminate the adjacent system.

The shield material provided throughout the auxiliary building is regular concrete (ρ = 2.3 g/cm^3). The principal auxiliary shielding provided is tabulated in Table 11.2-6.

11.2.3 RADIATION MONITORING SYSTEM

The Radiation Monitoring System is designed to perform two basic functions:

a. Warn of any radiation health hazard which might develop.
b. Give early warning of a plant malfunction which might lead to a health hazard or plant damage.

Instruments are located at selected points in and around the plant to detect, compute and record the radiation levels. In the event the radiation level should rise above a desired setpoint, an alarm is initiated in the control room. The automatic Radiation Monitoring System operates in conjunction with regular and special radiation surveys and with chemical and radiochemical analyses performed by the plant staff. Adequate information and warning is thereby provided for the continued safe operation of the plant and assurance that personnel exposure does not exceed 10 CFR 20 limits.

The only components of thos system which are located in the containment are the detectors for certain area monitoring channels. They would not be expected to operate following a major loss-of-coolant accident and are not designed for this purpose. Components of all other area and process monitoring channels are designed for post-accident operation.

The components of the radiation Monitoring System are designed according to the following environmental conditions:

a. Temperature - an ambient temperature range of 40 to 120°F*
b. Humidity - 0 to 100% relative humidity*
c. Pressure - Components in the auxiliary building and control room are designed for normal atmospheric pressure. Area monitoring system components inside the containment are designed to withstand test pressure.
d. Radiation - Process and area radiation monitors are of a nonsaturating design so that they "peg" full scale if exposed to radiation levels up to 100 times full scale indicaton. Process monitors are located in areas where the normal and post-accident background radiation levels will not affect their usefulness.

The Radiation Monitoring System is divided into the following subsystems:

a. The Process Radiation Monitoring System monitors various fluid streams for indication of increasing radiation levels.

b. The Area Monitoring System monitors area radiation in various parts of the plant.

c. Environmental Radiation Monitoring System monitors radiation in the area surrounding the plant.

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*Equipment located in the control room area may be specified for smaller temperature and humidity ranges because of the controlled environment provided by the heating and ventilating system.

PROCESS RADIATION MONITORING SYSTEM

This system consists of ten channels which monitor radiation levels in various plant operating systems. The output from each channel detector is transmitted to the Radiation Monitoring System cabinets located in the control room area where the radiation level is indicated by a meter and recorded by a multipoint recorder. High radiation level alarms are annunciated on the main control room board and indicated on the Radiation Monitoring System cabinets.

Each channel contains a completely integrated modular assembly, which includes the following:

a) Level Amplifier

Amplifies the energy of the radiation pulse to provide a discriminated output to the log level amplifier.

b) Log Level Amplifier

Accepts the shaped pulse of the level amplifier output, peforms a log integration, (converts total pulse rate to a logarithmic analog signal) and amplifies the resulting output for suitable indication and recording.

c) Power Supplies

Power supplies are contained in each drawer for furnishing the positive and negative voltages for the transistor circuits, relays and alarm lights and for providing the high voltage for the detector.

d) Test-Calibration Circuitry

These circuits provide a precalibrated analog signal to perform channel test, and a solenoid operated radiation check source to verify the channel's operations. An annunciator light on the main control board indicates when the channel is in the test calibrate mode.

e) Radiation Level Meter

This meter, mounted on the drawer, has a scale calibrated logarithmically in counts per minute from 10^1 to 10^6, and 10^1 to 10^6. The level signal is also recorded by the recorder.

f) Indicating Lights

These lights indicate high-radiation alarm levels and circuit failure. An annunciator on the main control board is actuated on high radiation.

g) Bi-stable Circuits

Two bistable circuits are provided, one to alarm on high radiation (actuation point may be set at any level within the range of the instruments), and one to alarm on loss of signal (circuit failure).

h) A remotely operated long half-life radiation check source is furnished in each channel. The energy emission ranges are similar to the radiation energy spectra being monitored. The source strength is sufficient to cause approximately mid-range indication of the detector unit.

The system Process Radiation Monitoring system consists of the following radiation monitoring channels:

1) <u>Containment and Plant Vent - Air Particulate Monitor (R-11)</u>

The monitor is provided to measure air particulate gamma radioactivity in the containment and to ensure that the release rate through the containment vent during purging is maintained below specified limits. It also provides a backup to the vent gas monitor.

High radiation level for the channels initiates closure of the containment purge supply and exhaust duct valves and pressure relief line valves.

This monitor has a measuring range of 10^{-9} to 10^{-6} microcuries per cubic centimeter.

This channel takes a continuous air sample from either the containment atmosphere, or the plant vent. The sample is drawn outside the containment or the plant vent ductwork in a closed, sealed system monitored by a scintillation counter - filter paper detector assembly. The filter paper collects all particulate matter greater than 1 micron in size, on its constantly moving surface, and is viewed by a photomultiplier-scintillation crystal combination. The sample is returned to the containment or plant vent, depending on which sample is being monitored, after it passes through the series connected (Ch R-12) gas monitor.

The detector assembly is in a completely enclosed housing. The detector is a hermetically-sealed photomultiplier tube - scintillation crystal (NaI) combination. A preamplifier transmits the pulse signal to the Radiation Monitoring System cabinets in the control room. The filter paper has a 25-day minimum supply at normal speed. Lead shielding is provided to reduce the background level to where it does not interfere with the detector's sensitivity. The filter paper mechanism, an electromechanical assembly which controls the filter paper movement, is provided as an integral part of the detector unit.

2. Containment and Plant Vent Radioactive Gas Monitor (R-12)

One monitor is provided to measure gaseous gamma radioactivity in the containment, to ensure that the radioactivity release rate during purging is maintained below specified limits and to serve as a backup to the vent gas monitor. High gas radiation level initiates closure of the containment purge supply and exhaust duct valves and pressure line relief valves.

This monitor has a measuring range of 10^{-6} to 10^{-3} microcuries per cubic centimeter.

This channel takes the continuous air sample from the containment atmosphere, or the plant vent after it passes through the air particulate monitor, and draws the sample through a closed, sealed system to the gas monitor assembly. The sample is constantly mixed in the fixed, shielded volume, where it is viewed by the Geiger-Mueller tube. The sample is then returned to the containment or the plant vent depending on which sample is being monitored.

The detector assembly is in a completely enclosed housing containing a gamma sensitive Geiger-Mueller tube mounted in a constant gas volume container. Lead shielding is provided to reduce the background level to a point where it does not interfere with the detector's sensitivity. A preamplifier and impedance matching circuit is mounted at the detector. Its output is transmitted to the Radiation Monitoring System cabinets in the control room.

The Containment Air Particulate and Radioactive Gas Monitors (Ch-R-11 and R-12) have assemblies that are common to both channels. They are described as follows:

a) The flow control assembly includes a pump unit and selector valves that provides a representative sample (or a "clean" sample) to the detector.

b) The pump unit consists of:

1. A pump to obtain the air sample.

2. A flowmeter to indicate the flow rate.

3. A flow control valve to provide flow adjustment.

4. A flow alarm assembly to provide low and high flow alarm signals.

c) Selector valves are used to direct the desired sample to the detector for monitoring and to block flow when the channel is in maintenance or "purging" condition.

d) A pressure sensor is used to protect the system for high pressure. This unit automatically closes an inlet and outlet valve upon a high pressure condition.

e) Purging is accomplished with a valve control arrangement whereby the normal sample flow is blocked and the detector purged with a "clean" sample. This facilitates detector calibration by establishing the background level and aids in verifying sample activity level.

f) The flow control panel in the control room Radiation Monitoring racks permits remote operation of the flow control assembly. By operating a sample selector switch on the control panel, either the containment or plant stack sample may be monitored.

g) A sample flow rate indicator is calibrated linearly (from 0 to 14) cubic feet per minute.

Alarm lights are actuated by the following:

a. Flow alarm assembly (low or high flow)
b. The pressure sensor assembly (high pressure)
c. The filter paper sensor (paper drive malfunction)
d. The pump power control switch (pump motor on)

3) <u>Plant Vent Gas Monitor</u> (R-14)

The Plant Vent Gas Monitor detects radiation passing through the plant vent to the atmosphere. It consists of four, thin-walled, self-quenching type Geiger-Mueller tubes (high sensitivity beta-gamma detector) operated in parallel together with an impedance matching network. A preamplifier and impedance matching circuit is used to match the signal source to the cable and transmit a pulse signal to the control room.

Remote indication and annunciation are provided on the Waste Disposal System control board. On high radiation level alarm the gas release valve in the Waste Disposal System is automatically closed.

This monitor has a maximum sensitivity of 5×10^{-7} microcuries per cubic centimeter.

4) Condenser Air Ejector Gas Monitor (R-15)

This channel monitors the discharge from the air ejector exhaust header of the condensers for gaseous radiation which is indicative of a primary to secondary system leak. The gas discharge is routed to the vent stack.

On high radiation level alarm, the condenser exhaust gases are diverted from the vent stack to the containment through a blower.

A gamma sensitive Geiger-Mueller tube is used to monitor the gaseous radiation level. The detector is inserted into an in-line fixed volume container which includes adequate shielding to reduce the background radiation to where it does not interfere with the detectors maximum sensitivity. This monitor has a maximum sensitivity of 10^{-6} microcuries per cubic centimeter.

5) Containment Fan Cooling Water Monitor (R-16 and R-23)

During a loss-of-coolant accident, these channels monitor the containment fan cooling water for radiation indicative of a leak from the containment atmosphere into the cooling water. A small bypass flow from each of the heat exchangers is mixed in a common header and monitored by a single scintillation detector mounted in a holdup tank assembly. Upon indication of a high radiation level each heat exchanger is individually sampled to determine which unit is leaking. This sampling sequence is achieved by manually selecting the desired unit to be monitored and allotting sufficient time for sample equilibrium to be established. (Approximately 1 minute)

The measuring range of this monitor is 10^{-5} to 10^{-2} microcuries per cubic centimeter.

6) Component Cooling Liquid Monitor (R-17)

This channel continuously monitors the component cooling loop of the Auxiliary Coolant System for radiation indicative of a leak of reactor coolant from the Reactor Coolant System and/or the residual heat removal loop in the Auxiliary Coolant System. A scintillation counter is located in an in-line well. A high-radiation level alarm signal initiates closure of the valve located in the component cooling surge tank vent line to prevent gaseous radiation release.

The measuring range of this monitor is 10^{-5} to 10^{-2} microcuries per cubic centimeter.

7) Waste Disposal System Liquid Effluent Monitor (R-18)

This channel continuously monitors all Waste Disposal System liquid releases from the plant. Automatic valve closure action is initiated by this monitor to prevent further release after a high-radiation level is indicated and alarmed. A scintillation counter and holdup tank assembly monitors these effluent discharges. Remote indication and annunciation are provided on the Waste Disposal System control board.

The measuring range of this monitor is 10^{-5} to 10^{-2} microcuries per cubic centimeter.

8) <u>Waste Disposal System Gas Analyzer Monitor (R-20)</u>

This detector is located in the sample line from the waste gas decay tanks to the gas analyzer, and monitors the waste gas decay tank activity inventory each time a sample is taken for hydrogen-oxygen analysis. A high radiation level alarm warns the operator when the decay tank inventory limit is approached, so that another tank may be placed in service.

Because of the extremely high levels of radiation in the waste gas decay tanks, an ion chamber mounted in a specially constructed sampler, is used as the detector. The ion chamber shell is in direct contact with the gas sample being monitored. The measuring range for this monitor is 1.4×10^{1} to 1.4×10^{5} curies stored in the gas decay tank.

9) <u>Steam Generator Liquid Sample Monitor (R-19)</u>

This channel monitors the liquid phase of the secondary side of the steam generator for radiation, which would indicate a primary-to-secondary system leak, providing backup information to that of the condenser air removal gas monitor. Samples from the bottom of each of the four steam generators are mixed in a common header and the common sample is continuously monitored by a scintillation counter and holdup tank assembly. Upon indication of a high-radiation level, each steam generator is individually sampled in order to determine the source. This sampling sequence is achieved by manually selecting the desired unit to be monitored and allotting sufficient time for sample equilibrium to be established. (Approximately 1 minute)

A remote indicator panel, mounted at the detector location, indicates the radiation level and high-radiation alarm.

A high-radiation level signal will close the isolation valves in the blowdown line , and sample lines.

The measuring range of this monitor is 10^{-5} to 10^{-2} microcuries per cubic centimeter.

In channels R-16, R-18, R-19, and R-23, a photomultiplier tube - scintillation crystal (NaI) combination, mounted in a hermetically sealed unit, is used for liquid effluent radiation detection. Lead shielding is provided to reduce the background level so it does not interfere with detector's sensitivity. The in-line, fixed volume container is an integral part of the detector unit. In channel R-17, the detector is inserted into a well in the component cooling system piping.

In addition, grab samples from the monitor tanks are analyzed to determine the amount of tritium in the effluent discharged to the Waste Disposal System.

During normal plant operation the tritium level in the reactor coolant will be limited to 2.5 μc/cc, this will ensure a tritium activity in the refueling water of <1 μc/cc. With a containment purge rate of 40,000 cfm, the maximum concentration of tritium in the containment air will be less than 1/5 of MPC.

The basis for this concentration is determined from the assumption that the refueling water evaporation rate is 100 lb/hr., the containment is purged for 2 hours at the rate of 40,000 cfm prior to access, and that the purge continues during the refueling operation at 40,000 cfm.

During normal plant operation, grab samples from the containment and auxiliary building areas will be analyzed for tritium as required.

GROSS FAILED FUEL DETECTORS

The Gross Failed Fuel Detector (GFFD) is based on the principle of measuring the delayed neutron activity from fission products in the primary coolant after having allowed the decay of N-17 neutron activity. Piping to the detector is connected to the hot leg of the reactor coolant loop (Figure 11.2-6). The fluid passes a delay coil and a sample cooler befroe it reaches the coil with the neutron detector mounted inside. Finally the fluid drains into the volume control tank. The proper delay time (about 60 sec) to the detector can be adjusted by regulating the rate of water flow. The length of tubing between the coils will be designed to match the required time delay between the coils.

Figure 11.2-7 shows the block diagram of the GFFD. The delayed neutron signal is displayed on a recorder in the control room. An alarm signal is provided, which indicates an excessive signal level.

Some experience has been obtained on the operational reliability of GFFD presently in operation on various plants (RGE, Zorita and Saxton). Based on this limited amount of information, the system is capable of providing data useful to the designer, once the data has been analyzed and correlated to plant operating conditions. However, at the present stage of development, sufficient experience has not been accumulated to satisfactorily prove the performance and reliability of the detector. It is therefore premature to impose operating requirements for this detector which could unduely restrict plant operation. The GFFD to be installed at IPP-2 will be used during power operation to gain further experience with its operational reliability and potential in assisting the plant operator in assessing fuel behavior.

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AREA RADIATION MONITORING SYSTEM

This system consists of nine channels which monitor radiation levels in various areas of the plant. These areas are listed as follows:

Channel	Area Monitor
R-1	Control Room
R-2	Containment
R-3	Criticality Monitor
R-4	Charging Pump Room
R-5	Spent Fuel Building
R-6	Sampling Room
R-7	In-Core Instrument Area
R-8	Drumming Station
R-10	Containment (Special)

Each channel (R-1 through R-8) consists of fixed position gamma sensitive G-M tube detector. The detector output is amplified and the log count-rate is determined by the integral amplifier at the detector. The radiatio level is indicated locally at the detector, and at the Radiation Monitoring System cabinets where it is also recorded. High-radiation alarms are displayed on the main annunciator, the Radiation Monitoring cabinets, and at the detector location. The control room annunciator provides a single window which alarms for any channel detecting high-radiation. Verification of which channel has alarmed is done at the Radiation Monitori System cabinets. A remotely operated, long half-life radiation check source is provided in each channel. The source strength is sufficient to produce approximately mid-range indication.

The computer-indicator module amplifies the radiation level signal, computed by the log level preamplifier, for indication and recording. The module also provides controls for actuation of the channel check source.

A meter is mounted on the front of each computer indicator module and is calibrated logarithmically from 0.1 mr/hr to 10 R/hr.

A remote meter, calibrated logarithmically from 0.1 mr. to 10 R/hr is mounted at the detector assembly.

Radiation Monitoring System cabinet alarms consist of a red indicator light for high radiation and an amber light to annunciate detector or circuit failure. The remote meter and alarm assembly at the detector contains a red indicator light and a buzzer type alarm annunciator actuated on high radiation.

Channel R-10 area monitor is provided to monitor the containment atmosphere following a highly unlikely loss-of-coolant accident. This monitor will be mounted in the steamline penetration area and will assist the operator to quickly determine if the accident consequences are either more or less serious than predicted. The range of channel R-10 will be from 10 mr/hr, to 2.5×10^2 R/hr. The principal isotopes to be monitored
13 will be fission products. The drawer for this equipment will be mounted in the RMS area monitor rack.

Radiation Monitoring System Control Room Cabinet

All of the control room system equipment is centralized in three cabinets. High reliability and ease of maintenance are emphasized in the design of this system. Sliding channel drawers are used for rapid replacement of units, assemblies, and entire channels. It is possible to completely remove the various chassis from the cabinet, after disconnecting the cables from the rear of these units.

Recorder

A 24 point strip chart recorder is provided in the Radiation Monitoring System cabinets in the control room. Each monitoring channel is sequentially recorded. The print rate is 20 seconds per point. Chart speed is adjustable.

Monitor Sensitivity

The sensitivity of the radiation monitors is given in Table 11.2-7. The sensitivity range is based on the first isotope listed in the last column. Calibration curves showing count rate versus radiation activity (μc/cc) are provided for all isotopes to be measured.

Operating Conditions

Table 11.2-8 indicates the detector operating condition during normal operation. Where fluid temperature is too high for the monitor, a cooling device with temperature indication is included. The different operating temperature ranges are within the design limits of the sensors.

The relation of the radiation monitoring channels to the systems with which they are associated is given in the sections describing those systems. Routine test and recalibrations will ensure that the channels operate properly.

Health Physics Program

Personnel Monitoring

The official and permanent record of accumulated external radiation exposure received by individuals is obtained principally from the interpretation of the film badge. The direct reading dosimeter provides day-by-day indication of external radiation exposure.

All persons subject to occupational radiation exposure are issued beta-gamma film badges and are required to wear such film badges at all times while within the Controlled Area. A neutron sensitive film packet is issued to personnel whenever a significant neutron exposure is possible. Personnel who are issued film badges pick them up at the Security Room on entering the Controlled Area and deposit them at the Security Room on leaving the Controlled Area.

Special or additional film badges are issued as may be required under unusual conditions. These devices are issued at the discretion of health physics personnel.

The film badges are processed on a routine basis at monthly intervals. The film badge of any individual is processed when an integrated dose of 1 rem is indicated by other measurements and when it appears that an overexposure may have occurred.

A self-reading dosimeter is issued in addition to a film badge to certain individuals whose work conditions make a day-to-day indication of exposure desirable. Dosimeters are read, recorded and re-zeroed regularly. Dosimeter records furnish the exposure data for the administrative control of radiation exposure.

Each individual is required to examine his dosimeter at frequent intervals while in radiation areas. Any individual who receives a dosimeter reading greater than 150 mrem is required to report his exposure to health physics personnel who will record the reading and rezero the dosimeter.

Personnel Protective Equipment

All personnel entering the Radiation Control Area are required to wear protective clothing. The nature of the work to be done is the governing factor in the selection of protective clothing to be worn by individuals. The protective apparel available are shoe covers, head covers, gloves and coveralls or lab coats. Additional items of specialized apparel such as plastic suits, face shields, and respirators are available for operations involving contamination. In all cases, health physics personnel shall evaluate the radiological conditions and specify the required items of protective clothing to be worn.

Respiratory protective devices are required in any situation arising from plant operations in which an airborne radioactive area exists or is expected to exist in excess of applicable limits. In such cases, the airborne concentrations are monitored by health physics personnel and the necessary protective devices specified according to concentration and type of airborne contaminants present.

Respiratory devices available for use include:

- Half-face respirator (filter type, negative pressure)
- Full-face respirator (filter or gas canister, negative pressure)
- Atmosphere supplying respirators (pressure demand, or continuous flow)
- Self-contained breathing apparatus

Self-contained breathing apparatus will be used in any situation involving exposure to gaseous activity or oxygen deficient atmospheres.

The appropriate type of respiratory protection equipment required will be determined from 10CFR20, Section 20.103 and Appendix E.

Facilities and Access Provisions

The plant site is divided into two categories, the Conventional Area and the Controlled Area.

The Radiation Control Area encompasses all plant areas in which radioactive materials and radiation are present in excess of limits specified in 10CFR20, 20.202, b2. Access to the Controlled Area is limited to those persons authorized for entry by the General Superintendant. Entry to and exit from the Controlled Area is through the Security Room only.

Any area in which radioactive materials and radiation are present shall be surveyed, classified and conspicuously posted with the appropriate radiation caution sign.

The general arrangement of the service facilities is designed to provide adequate personnel decontamination and change areas.

The Controlled Area locker room is employed as a protective clothing change area. A supply of clean protective clothing is maintained in this area. Locker space in the Controlled Area locker room is assigned to plant personnel for storage of personal clothing.

Geiger-Mueller count rate meters, "Friskers," are located in the laundry lobby, the washroom are , the Security Room, the fuel handling building, and the reactor containment. All personnel will survey themselves before leaving the Controlled Area.

A shower, drying and washroom is located adjacent to the Controlled Area locker room. Personnel decontamination equipment is available at the laundry lobby, and in the Controlled Area Decontamination and First Aid Rooms.

The fuel storage building has facilities to handle the decontamination of large items of equipment. This decontamination area contains a decontamination tank and service facilities.

Administrative and physical security measures are employed to prevent unauthorized entry of personnel to any high radiation area. These measures include the following:

> Areas in which radiation levels are so high that individuals might receive doses in excess of 100 mrem in 1 hr. are barricaded and conspicuously posted as "high radiation areas." Administrative controls require the issuance of a Radiation Work Permit prior to entry to any high radiation area.
>
> Locations where the above value exceeds 1 rem in 1 hour are conspicuously posted, and in addition, locked doors are provided to prevent unauthorized entry. Keys to these doors are kept under special administrative control. The locks on these doors are arranged so that personnel can leave from the high radiation area side without a key or other special device.

The limits for removable surface contamination in the Controlled Area are as follows:

Removable radioactive surface contaminations:

Beta-Gamma	1000 dpm/100 cm^2
Alpha (Uranium and Thorium)	1000 dpm/100 cm^2
Alpha (Other)	100 dpm/100 cm^2

When the above levels are exceeded, the area is posted as a "Contaminated Area." Additionally, all personnel are required to wear appropriate protective clothing for entry. The areas involved will be decontaminated as soon as possible to prevent the spread of contamination. Decontamination will be carried out under the direction of health physics personnel.

Radiation Instrumentation

Laboratory facilities are provided for the health physics and chemistry groups. These facilities include both laboratory and calibration rooms. A Health Physics Control Station is equipped to analyze routine air samples and contamination wipe surveys. Two gas flow windowless proportional counters are used for this purpose. The control station also serves as a central location for portable radiation survey instruments, respiratory protection equipment and contamination control supplies.

The Health Physics Equipment is supplemented with a 400 channel gamma spectrometer and an external gas flow proportional counter.

Count rate meters, "Friskers," with Geiger-Mueller type probes are located in the following plant areas:

Controlled Area Washroom Area
Fuel Handling Building
Primary Auxiliary Building
Reactor Containment

An eight channel beta-gamma portal monitor is located in the Security Room as a final check on personnel leaving the Controlled Area.

The types of portable radiation survey instruments available for routine monitoring functions are listed in Table 11.2-9.

Survey instruments are calibrated quarterly. Calibration and maintenance records are provided for each instrument.

The portable radiation survey instruments listed in Table 11.2-9 are available for use off-site during and following any possible accidental release of radioactivity from the facility. In addition to those listed, the following equipment is also available:

1) Fixed particulate coolector 0.8 micron filter - activated charcoal or plain. Nominal air flow rate of 1 cfm.

2) Hurricane Air Sampler 4 inch diameter, 0.8 micron filter - plain. Nominal air flow rate of 10 cfm.

3) Proportional Counter - 2π geometry 50% yield for alpha and 68% yield for beta. Lead shielded to minimize the effect of background radiation.

4) Gamma Scintillation Spectrometer - can be used to serve one of three functions; a) G.M. detector counting rate meter b) Scintillation detector counter rate meter with a 2 inch by 1 inch thallium activated sodium iodine crystal sealed to a photomultiplier tube. c) Spectrometer when used in this mode the crystal and photomultiplier tube are utilized. In addition, the window and anti-coincidence circuit select only the pulses from certain energies from 0.03 to 3.0 Mev. The instrument has six ranges up to a limit of 10^{+5} cpm.

MEDICAL FACILITIES

Medical facilities, consisting of a decontamination room and an examination room (Figure 11.2-4), are located in the Nuclear Service Building for Unit No. 1 (Figure 11.2-5) adjacent to the containment sphere, but outside the external concrete biological shield. This centralized facility, described below, will serve Unit No. 2. Each room contains an area of about 330 square feet.

Decontamination Room

The Decontamination Room is specially designed for the care of personnel contaminated by radioactivity. It has been used on infrequent occasions when skin contamination was not cleared by measures at the work site. It is designed to allow the admission of several mixed ambulatory and non-ambulatory patients. Patients will be sent to the decontamination room under the following conditions:

(1) they are uninjured, but their skin contamination is greater than 200 counts per minute, measured at contact with a GEIGER-MUELLER (G.M.) Portable Survey Meter; or

(2) they are injured and contaminated; or

(3) they require nasal, ear, or oral decontamination; or

(4) they are injured and contaminated and require emergency treatment before being moved to the hospital.

The floor is of stainless steel and the walls and ceilings are of special vinyl tile material. In the center of the Decontamination Room there is a fixed pedestal-type stainless steel table which has been adapted for decontamination and treatment of non-ambulatory patients. This fixture offers the additional advantage of allowing copious irrigation in cleansing wounds without the spread of radioactive contaminants to other parts of the room. Stainless steel fixtures in this room for use by ambulatory patients include a double-bowl sink at waist level for treatment of hands and arms and another double-bowl sink at knee level for treatment of feet and legs.

A shower area is situated at one end of the room for use by patients before they leave the decontamination area to enter the dispensary area. At the shower exit, in the vestibule area, radiation monitoring is performed to determine possible residual activity. A clean clothing cabinet is located immediately at the entrance of the dispensary area from the monitoring vestibule.

All liquid waste from the fixtures in the Decontamination Room drain into a laundry waste collection tank located in the Nuclear Service Building adacent to the Valve Room. The liquids in the tank are sampled and monitored. If the activity levels are above the maximum permissable concentration (MPC) as defined in 10CFR20, the liquids are pumped to the main waste collection tank for holdup or treatment. If the liquids in the laundry waste collection tank are below MPC, they are drained or pumped to the discharge canal. Solid wastes from the Decontamination Room are placed in containers, shielded with lead if necessary and moved to the bailing station for drum disposal or burial according to "Procedures for Handling Solid Radioactive Wastes". These procedures conform to 10CFR20 and ICC regulations for handling radioactive material.

Examining Room

A conventional examining room is located in the Dispensary Area and can be used for first aid after decontamination of ambulatory and litter cases. It can also be used for general clinic services. Medical and first aid supplies are kept in the Dispensary and are passed through as they are needed into the Decontamination Room.

An alternate site for medical supplies and an alternate treatment facility for possible radiation casualties is located in the regular Medical Bureau in the Indian Point Service Building on the Indian Point property about 0.7 miles from Nuclear Plant Unit No. 2. Access to this facility is by either interior or exterior roads.

The medical unit consist of: a surgical pack; burn pack; wound pack; ten litters; 24 blankets; a portable radioisotope decontamination kit (See Appendix 11A) which holds protective clothing; decontamination equipment; containers and supplies; and a plastic inflatable tub which can be inflated by a hand pump similar to a inflatable life raft. The inflatable tub is provided for personnel decontamination. A drainage spout at one end of the plastic tub removes the decontaminated water to a storage container. A portable kit similar to the radioisotope decontamination kit is available for transfer of an injured radiation casualty to an outside medical facility.

All of the above treatment facilities, medical supplies, and equipment are available on an around-the-clock basis. The key to these facilities is under the juridication of the licensed Senior Reactor Operator on shift. The medical and surgical supplies available for immediate control and emergency treatment of injured personnel are listed in Table 11.2-11.

The equipment and supplies available for immediate gross decontamination of personnel, including injured, are shown in Table 11.2-12.

MEDICAL PROCEDURES AND TECHNIQUES

The duties of the industrial medical department in a nuclear power station are defined as follows:

1. Radiation health protection.

2. Decontamination Procedure.

3. Plan for medical aspects of potential radiation accidents.

4. Routine medical services. The latter have constituted the major medical activity and will not require further description in this presentation.

1. Radiation Health Protection

To prevent radiation injury, the Health Physics Section has the basic responsibility for monitoring of radiation and personnel protection. The Medical Department responsibility is confined to preplacement examination to eliminate certain abnormally radiosensitive individuals, follow-up examinations and special examinations in case of exposure.

a. Radiation Monitoring

Radiation monitoring systems include instruments to measure exposure levels and devices to warn when permissible levels are exceeded. Radiation monitoring instruments are of two types; personnel and area monitoring. Personnel monitoring instruments can be subdivided into those that can be read directly and those that require supplemental processing and are a measure of exposure levels over a long period of time. The pocket dosmieter and the hand and foot counter are examples of direct reading instruments. The film badge provides a longer record of individual exposure to radiation and is usually developed on a monthly basis. Area monitoring is the use of instruments to determine the levels of radiation in an area and to give alarm when permissible levels are exceeded.

b. Medical Examination

Preplacement and follow-up examinations are important in preventing radiation injury. The preplacement examination eliminates certain abnormally radiosensitive individuals such as those with abnormal blood counts, splenomegaly and lymphadonopathy. Also those workers with a systemic disease, organic or functional, that might prove a hazard in the event of a radiation incident should not be allowed to work in a nuclear power plant.

(1) Preplacement Examination

Preplacement examinations are required for employees when assigned to the Indian Point Nuclear Plant. This examination includes the following:

a) Complete occupational history.

b) Complete medical history.

c) A description of any unusual exposure to radiation resulting from previous occupation, accident, diagnostic study and x-ray or isotope treatment.

d) Complete physical examination.

e) Special examination on vision including color and visual acuity.

The physician performs careful opthalmoscopic examination of the lens in a darkened room, using a plus 8 or 12 diopter lens. Slit-lamp examination is performed when indicated.

f) Routine urinalysis, complete blood count, chest x-ray and electrocardiogram.

(2) Special Examinations

Special examinations when required are made when employees are potentially exposed to radiation, chemical or other particular hazards. Special re-examination of those employees who have exceeded 3 rems exposure "in any quarter" may be required before re-entry to radiation exposure.

(3) Re-examination

Routine re-examination is scheduled on an annual basis. Every effort is directed to early detection of medical conditions even though these bear no relation to radiation exposure levels, e.g., the discovery of early cataract, latent coronary heart disease or a neoplasm of the lung.

(4) Medical Records

Records of physical examinations and laboratory tests are retained for an indefinite period under radiation protection laws.

c. Procedures For Treatment And Decontamination Of Contaminated Personnel

Procedures for treating and decontaminating affected personnel depend on the types of contamination received. Among the types treated are: a) Skin decontamination of non-injured ambulatory personnel; b) Personnel with coexistent injury; c) Nasal, ear, and oral cases; and eye decontamination cases. Each of the procedures for treatment and decontamination are described below.

(1) Skin Decontamination Procedure in Non-Injured Ambulatory Personnel

In the Medical Decontamination Room the following instructions will be posted:

Scrub the hands for a period of eight to ten minutes using a hand brush with the special soap powder and tepid water. Light pressure should be exerted on the brush. One should not press so hard that the bristles are bent out of shape. There should be at least three complete changes of soap powder and water. All surfaces should be covered with a minimum of four brush strokes. A convenient routine for hand decontamination is to start by scrubbing one thumb, being certain to brush all surfaces and then proceed to the webbed spaces between the thumb and the index finger and similarly over the other fingers. Then scrub the palm and dorsal surface. The hands should be thoroughly rinsed and checked by a survey meter. The brush and towels should be discarded in a radioactive waste can suitably labeled.

Be sure to clean the fingernails, toenails, nostrils, scalp, ears, and body folds. After the entire body is washed, and the contamination removed to the maximum extent for the method employed, the body must be re-surveyed with a portable radiation monitoring instrument.

(2) Decontamination Procedures in Personnel with Coexistent Injury

Emergency facilities are available in the Decontamination Room in planning for the care of injured contaminated personnel. When contaminated individuals with injury are brought to the Medical Decontamination Room, special care should be taken not to spread further contamination in the medical area. All persons concerned shall work safely so as not to further contaminate themselves or spread contamination from one patient to another. Monitoring shall be made to determine the risk of exposure to external radiation. Precaution shall be taken by assigned personnel to avoid internal contamination of the mouth and nose.

In the event of an extremity wound, the absorption of radioactive contaminants can be diminished by the proper application of a tourniquet between the wound and the body. This can be accomplished by the trained First Aid Attendant. The usual precaution in regard to continuous application of a tourniquet will be observed.

Decontamination of the involved area surrounding a wound will be completed in the Medical Decontamination Room.

Definitive care of the wound is the responsibility of the physician. Monitoring of all materials will be accomplished. Blood samples, urine and stool specimens will be collected for bio-assay.

Seriously injured persons presenting fractures, burns, hemorrhage or shock shall not be moved without supervision of the First Aid Attendant. However, urgent requirements for radiation safety may necessitate the removal of the injured individual as soon as possible. Cautious attempts on the premises should be made to remove contaminated clothing and accomplish decontamination. There may be cases when the need for immediate medical care will take precedence over decontamination procedures. First Aid Attendants can apply tourniquets to control hemorrhage, use temporary splints to prevent moving of fractured limbs and treat shock to the best of their ability.

In serious situations, immediately place a call for an ambulance and contact the on-call physician. In general, hospitalization can be expedited by contacting the Medical Department.

If a patient is not readily movable to an outside hospital, a surgical team could be brought to the Decontamination Room of the nuclear medical facility to perform emergency treatment and removal of impregnated radioactive materials.

A litter would be set on the floor and portable lead bricks would be build up at the sides to a level of 2 to 3 feet to provide protective shielding for operating personnel. Portable glass shielding (lead) bricks would roof over the lead bricks and provide a visible work field as well as more adequate screening than from the ordinary lead glass. A lead container is available for storage of radioactive material which might be removed from the injured person.

(3) Nasal, Ear and Oral Decontamination Procedures

In the event of contamination without injury, flush thoroughly with water and refer to the Medical Decontamination Room. The First Aid Attendant will check the local situation, monitoring will be accomplished by a member of the Health Physics Group, the Medical Department will be notified and a decision will be made concerning referral to an outside specialist.

When contamination is present with concomitant injury to the nose, ear or mouth, the individual will be taken to the Medical Decontamination Room where the First Aid Attendant will check the local situation, monitoring will be accomplished by a member of the Health Physics Group, the Medical Department will be notified and a decision will be made concerning referral to an outside specialist.

(4) Eye Decontamination

Eye contamination without injury requires immediate simple flush with water and the individual is referred to the Medical Decontamination Room. The First Aid Attendant will check the local situation, monitoring will be accomplished by a member of the Health Physics Group, the Medical Department will be notified and decision will be made concerning referral to an outside specialist.

In the event of eye contamination with injury, the eye itself should not be disturbed, but decontamination of the surrounding area should be performed as much as is practical. Refer to the Medical Decontamination Room where the First Aid Attendant will check the local situation, monitoring will be accomplished by a member of the Health Physics Group, the Medical Department will be notified and a decision will be made concerning referral to an outside specialist.

For inhalation exposures the retention, clearance and translocation of inhaled particles are determined by physiological processes.[1] The ICRP standard lung model assumes that with "soluble" particles, 25 per cent are exhaled, 50 per cent are deposited in the upper respiratory tract and transported to the gastro-intestinal tract, and 25 per cent are taken up in the body via the lower respiratory tract. It is thought that "insoluble" compounds follow the same pattern, except that only 12.5 per cent is taken up in body fluids via the lower respiratory tract. An additional 12.5 per cent is assumed to be transported up the respiratory tract and swallowed within 24 hours. The case study following inhalation of radio-nuclides should include whole body counts, and if indicated, urine bioassay and fecal analysis. Follow-up will be contained for a sufficient period of time so that the biological half life can be determined accurately.

QUALIFICATIONS OF MEDICAL PERSONNEL

A description of qualifications of on-site and off-site medical personnel is given in Table 11.2-13.

The professional staff at University Hospital, New York University Medical Center, New York City would be under the supervision of Roy E. Albert, M.D. Dr. Albert is a recognized expert in nuclear medicine. He has formulated policy and training procedures for the attending and resident staff at University Hospital. The hospital staff has the capability to provide the number and types of physicians to handle radiation casualties.

Transport of Injured Personnel

Arrangements for transporting injured personnel provide for the immediate treatment of personnel on a round-the-clock basis. The equipment and supplies used for in-transit emergency treatment are the following:

(1) Verplanck Volunteer Ambulance is completely equipped in accordance with Red Cross recommendations.

(2) The Portable Radioisotope Decontamination Kit located in the Medical Bureau facility in the Indian Point Service Building and supplies from the nuclear medical facility at the Indian Point Station will also accompany a patient to the receiving hospital. The kit holds protective clothing, decontamination equipment, containers and supplies as listed in Appendix 11A, Table 11A-1.

(3) The feasibility of emergency evacuation by helicopter air ambulance service has been investigated. A heliport has been set up at Indian Point on the pier on the Hudson River, and the area will be cleared on notification of an emergency. Heliports are available at the hospital at Brookhaven Lab, and at 60th Street in Manhattan where an ambulance will meet the helicopter to transport the patient to University Hospital. It is expected that a private helicopter service will be utilized if air evacuation is to be used. Conversion of the helicopter to an ambulance configuration takes approximately 10 minutes, and a round trip from the 60th Street heliport to Indian Point takes approximately 50 minutes.

Standing orders for emergency procedures are a part of the Radiation Health Procedures for the Indian Point Station nuclear medical facility. Health Physics personnel and First Aid men of Con Edison will accompany the patient in the ambulance. Instructions for transporting injured persons who have become contaminated are the following:

(1) Transportation of highly contaminated persons is to be done in such a way that maximal distance will separate the patient and the attendants.

(2) A shielding of lead bricks, as required, is to be placed between the front seat of the ambulance and the patient to protect the driver and attendants from radiation.

(3) Special lead shielding of the affected part of the patient's body with lead sheeting is to be applied, if required, to further decrease radiation exposure from a contaminated arm or leg.

Verplank Volunteer Ambulance Corps, Verplanck, New York, is located approximately 3 miles from Indian Point Station. An ambulance is available round-the-clock. The average time to respond to calls is 15 minutes.

HOSPITAL FACILITIES

Non-Injured Contaminated Personnel

It is planned that further decontamination of non-injured personnel would be completed in the Decontamination Room of the Indian Point nuclear medical facility.

Contaminated Injured Personnel

Arrangments were completed with University Hospital, New York University Medical Center, New York City, in November 1963, for handling contaminated injured personnel. University Hospital is approximately 40 miles from Indian Point. The "Guide for Management of Radioactive Casualties at University Hospital" is included as Appendix 11B. This guide designates the radiation casualty team, the method of admission, evaluation of patient, operating room procedure, treatment, radiation hazard areas, storage of decontamination equipment and monitors, and levels of exposure to hospital personnel.

The Medical Department of the Consolidated Edison Company has set forth the following basic principles in handling radioactive contamination cases admitted to a hospital.

(1) The hospital should be notified before a contaminated patient is sent for admission. The location of the accident, number of people involved, and estimate of the extent of injury of patients should be furnished to the hospital. The person at the hospital should instruct the caller exactly where to bring the patients.

(2) The single error of commission to be avoided at all cost in handling contaminated patients in the hospital is that of bringing them into the hospital in such a manner as to spread high levels of radioactivity along the route.

(3) Persons exposed only to external radiation present no unusual admission problem since they are not radioactive.

(4) It is economically unsound to reserve one or more rooms solely for radiation accidents, since these accidents are extremely rare as compared to all other emergencies. A section of the hospital normally reserved for isotope treatment and study would provide an experienced facility to handle such contaminated emergencies.

(5) A hospital radiation team should be organized with a radiological safety officer who has had experience either in the department of radiology or the radioisotope laboratory. Technicians from the x-ray department and the radioisotope laboratory would make valuable auxiliary personnel. One or more physicians, nurses and a trained ambulance attendant should be part of such a team.

(6) A supply kit for radiological first aid should be provided. In the event of a contaminated case from the Indian Point Station, a portable radioisotope decontamination kit will accompany the patient. The equipment used to conduct survey and monitoring would be part of equipment normally found in the radioisotope laboratory of a hospital.

(7) Emergency decontamination apparatus is recommended for use with standard hospital stretchers. This equipment is described in Appendix 11C.

AVAILABILITY OF OFF-SITE MEDICAL FACILITIES

In the arrangements with University Hospital, New York University Medical Center, New York City, no limitations are visualized on admission or treatment of radiation accident casualties, length of stay for contaminated patients, extent of contamination and radiation levels associated with injuries, or the types of injuries or illness.

HOSPITAL PROCEDURES

The "Guide for Management of Radioactive Casualties at University Hospital" has been attached (see Appendix 11B) and would be followed by the staff at University Hospital.

a. This guide and orders are not posted but the telephone operator at the University Hospital has a standing list of persons to contact in the event that radiation casualties are admitted to the hospital.

b. The administrative and professional staff have been instructed about the plan for management of radioactive casualties.

PLAN FOR POTENTIAL RADIATION ACCIDENTS

(1) Radiation Accident Planning

Basically, the immediate accident program within the installation will include the following general instructions:

a. Evacuate exposed personnel from accident area and provid first aid.
b. Notify Production Supervision, Medical Department and Health Physics.
c. Close off radiation area.
d. Confine and survey all possibly contaminated persons.
e. Evaluate the situation.
   1) Contamination exposure
   2) Neutron and gamma exposure.
   3) Total level of radiation exposure.

Details of guidance for emergency exposure during recue and recovery activity are outlined in "Standards for Radiation Protection, AEC Appendix 0524."(2)

(2) Diagnosis and Treatment of Injury

For the details of this aspect of a medical program, reference is made to the handbook entitled "Medical Aspects of Radiation Accidents" published by the United States Atomic Energy Commission,[3] and dose classification into Groups I and V, of clinical radiation injury groups, by Thoma and Wald[4].

Biological monitoring may be used to determine the absorbed quantity of both radioactive and non-radioactive elements. Measurements are made of radioactivity in the urine, blood or the quantity of radioactive material excreted or inhaled. If there has been neutron exposure, heparinized blood shall be examined to determine the presence of radioactive sodium-24. Whole body counting is an important technique for the measurement of very low body burdens of gamma emiting nuclides.

(3) Hospital Planning

See "Contaminated Injured Personnel" on Page 11.2-39.

11.2.4 EVALUATION

The whole body gamma dose in the control room under accident conditions is calculated assuming that a complete core meltdown occurs resulting in the release of the following sources to the reactor containment: (Per TID-14844).

a) 100% of the noble gases
b) 50% of the halogens
c) 1% of the remaining fission product inventory.

The above sources, tabulated in Table 11.2-10, are assumed to be homogeneously distributed within the free volume of the reactor containment. The source intensity in Mev per second as a function of time after the accident is determined by considering decay only; no c[illegible] is taken for filtration or washdown.

The direct dose rate in the control room due [illegible]ne activity dispersed within the containment is calculated by a dig[illegible]al computer program which is based on a point kernel attenuation model. The source region is divided into a number of incremental source volumes and the associated attenuation, gamma ray buildup, and distance through regions between each source point and the control room are computed. The summation of all point source contributions gives the total direct dose rate at the control room.

In addition to the direct done, the contribution of scattered radiation was also estimated. These estimates indicate that scattered radiation levels will contribute less than 10% of the direct dose. Scattered radiation levels include both scattering from air (sky-shine) and scattering from large surfaces in the vicinity of the container. The direct and scattered dose in the control room for an exposure time of one month [illegible]llowing the accident is calculated to be less than 500 mr.

In order to determine the possible dose that an operator could receive while operating a manual backup item (e.g., valve), it is estimated rather conservatively that it will require 15 minutes to operate the valve. In addition, it is assumed that an additional 15 minutes is required to get to and from the manual equipment. The total integrated whole body dose that an operator would receive performing the above operation would be less than 8 rem. This dose is calculated for the first half hour immediately following the accident and assumes that the equipment being operated or serviced is adjacent to the containment surface. Doses in the vicinity of equipment located within the auxiliary building would be much less due to the shielding afforded by the concrete walls of the auxiliary building.

Liquid Waste Release

All liquid wastes releases will be assayed for radioactivity to comply with the limits specified (1/10 of 10 CFR 20 for unrestricted areas).

11.2.5 TESTS AND INSPECTIONS

Comp[illegible] radiation surveys will be made throughout the plant containment and auxilia[illegible] building during initial phases of plant start-up. Survey data will be taken and compared to design levels at power levels of 10%, 50% and 100%, at rated full power. Survey data will be reviewed for conformance to design levels before increasing to the next power range.

The gas and particulate effluent monitors shall be tested at each refueling shutdown with calibrated sources and normal response of each monitor shall be tested daily using a remotely operated test source and to verify the instruments response. Liquid effluent monitors shall be tested at each refueling shutdown with calibrated sources and normal response of each monitor shall be tested daily using a remotely operated test source to verify the instruments response.

TABLE 11.2-1

PLANT ZONE CLASSIFICATIONS

Zone	Condition of Occupancy	Maximum Dose Rate (1% failed fuel) M Rem/hv.
0	Continuous access	≤ 0.1
I	Normal Continuous access	≤ .75
II	Periodic access	≤ 2.0
III	Limited access	≤ 15
IV	Controlled access	> 15
V	Limited controlled access*	> 200

* Access to Zone V areas must be cleared with the reactor operators in the control room.

TABLE 11.2-2

PRIMARY SHIELD NEUTRON FLUXES AND
DESIGN PARAMETERS

Calculated Neutron Fluxes

Energy Group	Incident Fluxes (n/cm^2 - sec)	Leakage Fluxes (n/cm^2 - sec)
E < 1 Mev	7.2×10^8	2.6×10^2
5.3 Kev E ≤ 1 Mev	1.0×10^{10}	5.9×10^2
.625 ev ≤ E ≤ 5.3 Kev	5.3×10^9	1.1×10^3
E < .625 ev	1.5×10^9	8.8×10^4

Design Parameters

Core thermal power	3216 MW(t)
Active core height	144 In.
Effective core diameter	132.7 in.
Baffle wall thickness	1.125 in.
Barrel wall thickness	2.25 in.
Thermal shield wall thickness	2.75 in.
Reactor vessel I.D.	173.0 in.
Reactor vessel wall thickness	8.625 in.
Reactor coolant cold leg temperature	555°F
Reactor coolant hot leg temperature	613°F
Maximum thermal neutron flux exiting primary concrete	10^6 n/cm^2 sec.
Reactor shutdown dose exiting primary concrete	<15 mr/hr

TABLE 11.2-3

SECONDARY SHIELD DESIGN PARAMETERS

Core power density	98.5 watt/cm^3
Reactor coolant liquid volume	12,600 ft^3
Reactor coolant transit times:	
Core	0.817 sec.
Core exit to steam generator inlet	2.001 sec.
Steam generator inlet channel	0.592 sec.
Steam generator tubes	3.220 sec.
Steam generator tubes to vessel inlet	2.758 sec.
Vessel inlet to core	2.167 sec.
Total out of core	10.738 sec.
Full power dose rate outside secondary shield	<0.75 mr/hr

TABLE 11.2-4

ACCIDENT SHIELD DESIGN PARAMETERS

Core thermal power	3216 MW(t)
Minimum full power operating time	1000 days
Equivalent fraction of core melting	1.0
Fission product fractional releases:	
Noble gases	1.0
Halogens	0.5
Remaining fission product inventory	0.01
Clean-up rate following accident	0
Maximum integrated direct dose (one week exposure) in the control room	<1.5 rem
Maximum integrated direct dose (one week exposure) at the site boundary	<350 mrem

TABLE 11.2-5

REFUELING SHIELD DESIGN PARAMETERS

Total number of fuel assemblies	193
Minimum full power exposure	1000 days
Minimum time between shutdown and fuel handling	56 hours
Maximum dose rate adjacent to spent fuel pit	0.75 mr/hr
Maximum dose rate at water surface	2.0 mr/hr

TABLE 11.2-6

PRINCIPAL AUXILIARY SHIELDING

Component	Concrete Shield Thickness, Ft. – In.
Demineralizers	4 – 0
Charging pumps	2 – 6
Liquid waste holdup tanks	2 – 6
Volume control tank	3 – 6
Reactor Coolant filter	3 – 6
Gas stripper	2 – 6
Gas decay tanks	3 – 6
Gas Compressor	2 – 0
Waste evaporator	2 – 0

Design parameters for the auxiliary shielding include:

Core thermal Power	3216 MW(t)
Fraction of fuel rods containing small clad defects	0.01
Reactor coolant liquid volume	12,600 ft^3
Letdown flow (normal purification)	75 gpm
Effective cesium purification flow	7 gpm
Cut-in concentration deborating demineralizer	150 ppm
Dose rate outside auxiliary building	0.75 mr/hr
Dose rate in the building outside shield walls	0.75 mr/hr

TABLE 11.2-7

RADIATION MONITORING SYSTEM CHANNEL SENSITIVITIES

Channel	Sensitivity Range	Detected Isotopes
R-11	1.0×10^{-9} to 1.0×10^{-6}*	I^{131}, I^{133}, Cs^{134} Cs^{137}
R-12	1.0×10^{-6} to 1.0×10^{-3}*	Kr^{85}, Ar^{41}, Xe^{133} Xe^{135}
R-14	5.0×10^{-7} to 1.0×10^{-4}*	Kr^{85}, Ar^{41}, Xe^{133} Xe^{135}
R-15	1.0×10^{-6} to 1.0×10^{-3}*	Kr^{85}, Ar^{41}, Xe^{133} Xe^{135}
R-16	1.0×10^{-5} to 1.0×10^{-2}*	Co^{60}, Mixed Fission Products
R-17	1.0×10^{-5} to 1.0×10^{-2}*	Co^{60}, Mixed Fisson Products
R-18	1.0×10^{-5} to 10^{-2}*	Co^{60}, Mixed Fission Products
R-19	1.0×10^{-5} to 1.10×10^{-2}*	Co^{60}, Mixed Fission Products
R-20	1.4×10^{1} to 1.4×10^{5}***	Kr-85
R-23	1.0×10^{-5} to 1.0×10^{-2}	Co^{60}, Mixed Fission Products
R-1 through R-8	1.0×10^{-1} to $1.0 \times 10^{+4}$***	

Note: * is given in μc/cc

** is given in mr/hr

*** is curies in tank

TABLE 11.2-8

DETECTING MEDIUM CONDITIONS

Channel	Medium	Temperature Range (°F)
R-11	Air	50-122
R-12	Air	50-122
R-14	Air	40-122
R-15	Air	50-122
R-16	Water	60-160
R-17	Water	40-160
R-18	Water	60-160
R-19	Water	60-160
R-20	Air	50-122
R-23	Water	60-160
R-1	Air	50-122
R-2	Air	50-122
R-3	Air	50-122
R-4	Air	50-122
R-5	Air	50-122
R-6	Air	50-122
R-7	Air	50-122
R-8	Air	50-122

TABLE 11.2-9

PORTABLE RADIATION SURVEY INSTRUMENTS

Number	Type
4	Ionization Chamber (high range, of which at least 2 have a range of 250 rem per hr.)
4	Geiger-Mueller Survey Meter, 0 - 2 rem per hr.
1	Alpha Survey Meter
3	Neutron Survey Meter
8	Air Particulate Fixed Filter Geiger-Mueller Monitor, Beta sensitive.
5	Geiger-Mueller Survey Meter 0 - 20 mrem/hr
2	NaI gamma survey meters 0 - 1,000 rem/hr

TABLE 11.2-10

RADIATION SOURCES RELEASED TO THE CONTAINMENT
FOLLOWING EQUIVALENT CORE MELTDOWN ACCIDENT - MEV/SEC

Time After Release	Energy - Mev/γ					
	0.4	0.8	1.3	1.7	2.2	>2.5
0	2.3×10^{18}	1.3×10^{19}	3.1×10^{18}	1.4×10^{19}	1.2×10^{19}	1.2x1[illegible]
2 hr	2.5×10^{18}	7.1×10^{18}	1.6×10^{18}	9.6×10^{17}	4.9×10^{18}	2.9x1[illegible]
1 day	1.1×10^{18}	1.2×10^{18}	9.5×10^{16}	1.2×10^{17}	3.5×10^{17}	3.0x1[illegible]
1 week	4.2×10^{17}	2.1×10^{17}	5.8×10^{15}	7.6×10^{16}	1.6×10^{15}	7.0x1[illegible]
1 month	2.6×10^{17}	1.3×10^{17}	2.1×10^{15}	2.5×10^{16}	1.5×10^{15}	2.4x1[illegible]

TABLE 11.2-11 MEDICAL SUPP

AVAILABLE FOR IMMEDIATE CONTRO

Medical Supplies

2	Kidney Basins - S/S
2	Tissue Forceps 2 x 3 Teeth 4-1/2" - S/S
6	Hemostatic Forceps - Halstead - Mosquito 5" Straight - S/S
6	Hemostatic Forceps - Halstead - Mosquito 5" Curved - S/S
6	Hemostatic Forceps - Kelly 5-1/2" Straight - S/S
6	Hemostatic Forceps - Kelly 5-1/2" Curved - S/S
2 pair	Ralks Sponge Forceps - 7" Serrated - Straight
2 pair	Ralks Sponge Forceps - 7" Serrated - Curved
1	Lufkin - Steel Tape Measure 6' Graduated Meters/Ft
1	Taylor - Percussion Hammer Regular
2	Bowles - Stethescope Standard - 1-3/4" - Model B-D
1	B-P Sterilizing Tray
2	Instrument Tray and Cover - Recessed Handle Approximately 12-3/16" x 7-11/16" x 2-1/8" - S/S
2	Forcep Jars Tube Diameter - 2-5/16" - Tube Depth 4-7/16" - S/S
1	W/A Diag. Set Complete
2	Thermometer Jars - S/S
2	Thomas Splint, Leg Adult, Hinged Ring
2	Thomas Splint - Lewin
6	Finger Splint - Thumb - Lewin
3	Splints Colles Right
3	Splints Colles Left

QUIPMENT, SURGICAL SUPPLIES, AND DRUGS

MERGENCY TREATMENT OF INJURED PERSONNEL

__l Supplies (Cont'd)__

Splints Forearm Right

Fracture - Splints and Padding

Adjustable Crutches - Maple

Baumanometer - Model 300

Tycus Aneroid Blood Pressure Instrument - Hand Model Complete with Case

Bard-Parker Handles No. 7

Bard-Parker Blades Nos. 22, 20, 11 and 15

Lister Bandage Scissors - 5-1/2" - S/S

Sponge Forceps

Utility Surgical Scissors 6" Sharp/Sharp - Straight

Utility Surgical Scissors 6" Sharp/Sharp - Curved

Utility Surgical Scissors 6" Sharp/Blunt - Straight

Scissors - Mayo 6-1/2" - Straight

Scissors - Mayo 6-1/2" - Curved

Splinter Forcep, Thumb, Fine 4" - S/S

Thumb Dressing Forceps - 6" - S/S

__ent__

New London Casel Lamp

Devilbiss Irrigator

National Diagnostic set complete

__Equipment (Cont'd)__

3	Hypo Syringe 50 cc
1	Hypo Syringe 20 cc
13	Hypo Syringe 10 cc
12	Hypo Syringe 5 cc
2	Tuberculin Syringes
2	Hypo Syringe 2 cc
3	4 oz. Ear and Bladder Syringe
63	Hypo Needles, guage 22 x 1" - guage 26 x 1/2" guage 25 x 1/2"- guage 22 x 3"
1	Flashlight and Batteries
6	Catheters - French 14, 16, 18
1	Oxygen Machine
1	Wheel Chair
1	Ambu Breathing Equip.
1	Army Litter
1	Robinson Litter, Folding
1	Magnalite Lamp
1	Gooseneck Floor Lamp
1	Medical Scale
1	Castle Auto Clave
16 oz	Tincture Green Soap
14	Vaseline Guaze Dressings 6" x 13"

Equipment (Cont'd)

1	Shadow Box Twin
1	Examining Table
6	Thermometers Oral - Rectal
1 pkg.	Emergency Sutures
1	Picker Lead Apron
1 pair	Picker X-Ray Gloves

Surgical Supplies

2 vi.	Sterile Water 30 cc
1	Isotonic Sodium Chloride
2	5% Dextrose in Saline 500 cc
2	Sterile Administration Sets
800	Steri-Pads 3" x 3"
3 pair	Latex Surgeons Gloves
4 oz	Tincture Merthiolate
2 oz	Thermometer Solution
5	2" Adhesive Tape
7	1" Adhesive Tape
5	1/2" Adhesive Tape
48	1" Bandage
36	2" Bandage
18	3" Bandage
2 pkg.	Tongue Depressors

Surgical Supp

1 pkg.	Cotto
2-8oz.	Steri
5	3" Ac
5	2" Ac
3 pkg.	Iodin
1 pkg.	Burn
12	Rubbi

(CONTINUED)

nt'd)	Drugs	
Applicators	200	Tablets Potassium Iodide 325 mg
n	50	H.T. Nitroglycerin 1/150 gr
e	50	H.T. Nitroglycerin 1/100 gr
e	50	Quinidine Sulfate Tablets 3 gr
	25	Seconal Capsules 1-1/2 gr
	50	Crystodigin 0.1 mg Tablets
ol	50	Crystodigin 0.2 mg Tablets
	6	Crystodigin Amps 0.2 mg
	6	Mercuhydrin 10 cc
	6	Aminophyllin 0.25 gm amp
	100	Aromatic Ammonia Vaporoles
	100	H.T. Atropine Sulfate 1/150 gr
	10	Adrenalin Aq. 1/1000 amps
	2	Solu Cortef 1 vi
	2	Wyamine Inj. Amps
	1	Benadryl Inj. 5 cc
	1	Clinitest Complete
	1	Uristix
	1	Acitest
	1	Hematest

TABLE 11.2-12

Equipment and Supplies Available for Immediate Gross Decontamination of Injured and Other Personnel

Decontaminating equipment and containers
Plastic bags and cans for clothing
Lan-O-Kleen and pHisoHex
Skin Decontamination brushes
Face towels
Sterile saline in liter bottles
Surgical gowns and caps
Plastic aprons
Disposable plastic gloves
Plastic overshoes
Vinyl sleeve protectors

TABLE 11.2-13 QUALIFIC[...]

OF ON-SITE A[...]

Name and Function	Professional Education and Special Training
Moises B. Menache, Part-time Physician	Doctor of Medicine General knowledge of decontamination procedure
Health Physicists	Instrument and Control Engineers trained in Health Physics methods and completing the Basic Radiological Health Cour[...] administered by the U.S. Public Health Service
George Blizzard, First Aid Instructor	Experience in Red Cross First Aid Training for over 30 years, Supervisor of Medical Procedu[...]
First Aid Men (Minimum of 2 per shift)	Operating and maintenance personnel specially trained in first aid procedures by Medical Dept.
S. Charles Franco, Asst. Vice President, Medical Dept. and Safety Services, Con Edison	Doctor of Medicine Developed nuclear medical program. Fellow, American College of Physicians, Industrial Medical Association, American Academy of Occupational Medicine; Assoc. of Clincial Professor (part-time), Institute o[...] Environmental Medicine, N. Y. University Medical Center
Thomas J. Doyle, Executive Medical Director Medical Dept., Con Edison	Doctor of Medicine Certified by Training Seminar on Medical Care and Treatment of Radiation Accident, Brookhaven National Laboratory. Fellow of the Industrial Medical Association. Member, American Academy of Occupational Medicine.

...FESSIONAL EDUCATION, AND SPECIAL TRAINING

...Y AVAILABLE OFF-SITE PERSONNEL

...ity ...cation	Provisions for Meeting Cost	Hours of On-Site Duty	Off-Site Availability Location and Distance and Distance from Site	Hospital Staff Appointment
pro- ...l and ...n	... for services when on call	9:00 A.M. to 12 noon Monday and Friday	Around-the-clock 220 Tate Ave. Buchanan, N. Y. 1 mile	Attending Physician Peekskill Community Hospital
...onal	Employees of Con Edison	Around-the-clock		
semi- ...onal	Employee of Con Edison	8:00 A.M. to 4:30 P.M. (Mon. through Friday)		
semi- ...onal	Employees of Con Edison	Around-the-clock		
...e off- ...l medical	Employee of Con Edison		Around-the-clock, Con Edison 4 Irving Place N. Y. C., N. Y. (40 miles)	Assoc. Physician University Hospital
...e off- ...l	Employee of Con Edison		Around-the-clock Con Edison 4 Irving Place N. Y. C., N. Y. (40 miles)	

TABLE 11.2-13 (C

Name and Function	Professional Education and Special Training	Availa?ility Classificati
Albert C. Hamilton, Asst. Medical Director, Medical Dept., Con Edison	Doctor of Medicine Supervisor of Medical Services for Indian Point since 1962. Fellow, Industrial Medical Association AEC Seminar - June 64	Readily avai able off-sit personnel medical
Roy E. Albert, Medical Consultant	Doctor of Medicine Institute of Environmental Medicine, Sterling Forest, N. Y. Professor of Environmental Medicine, full time. N. Y. University Medical Center	Readily avai able off-sit personnel medical

...ions ...r ...g Cost	Hours of On-Site Duty	Off-Site Availability Location and Distance from Site	Hospital Staff Appointment
...ee of ...dison		Around-the-clock Con Edison 4 Irving Place N. Y. C., N. Y. (40 miles)	
...r ...es		Around-the-clock, Institute of Environmental Medicine, Sterling Forest, N. Y. (15 miles)	Attending Physician, University Hospital

12

FIGURE 11A-1 PORTABLE RADIOISOTOPE DECONTAMINATION KIT

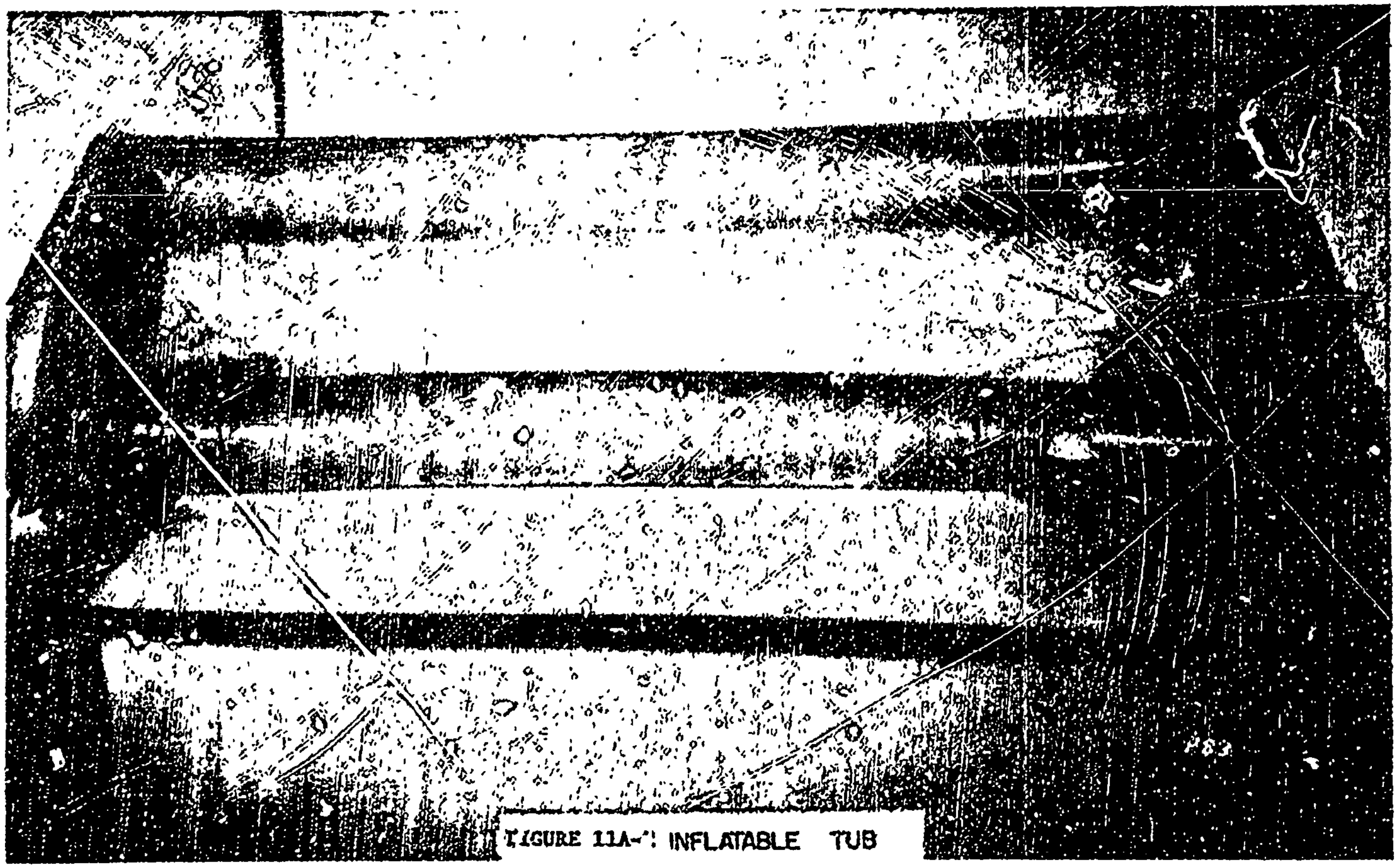
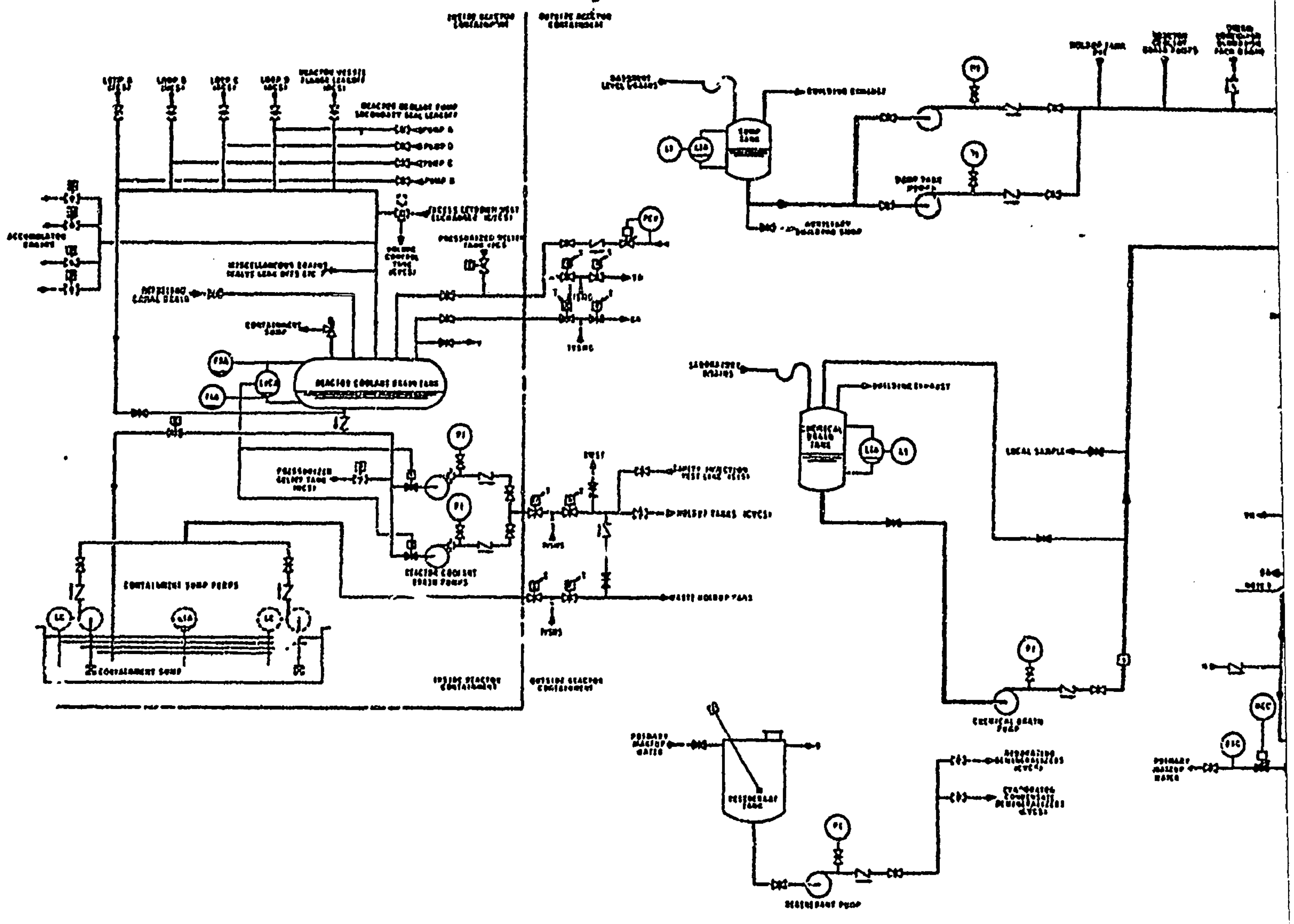
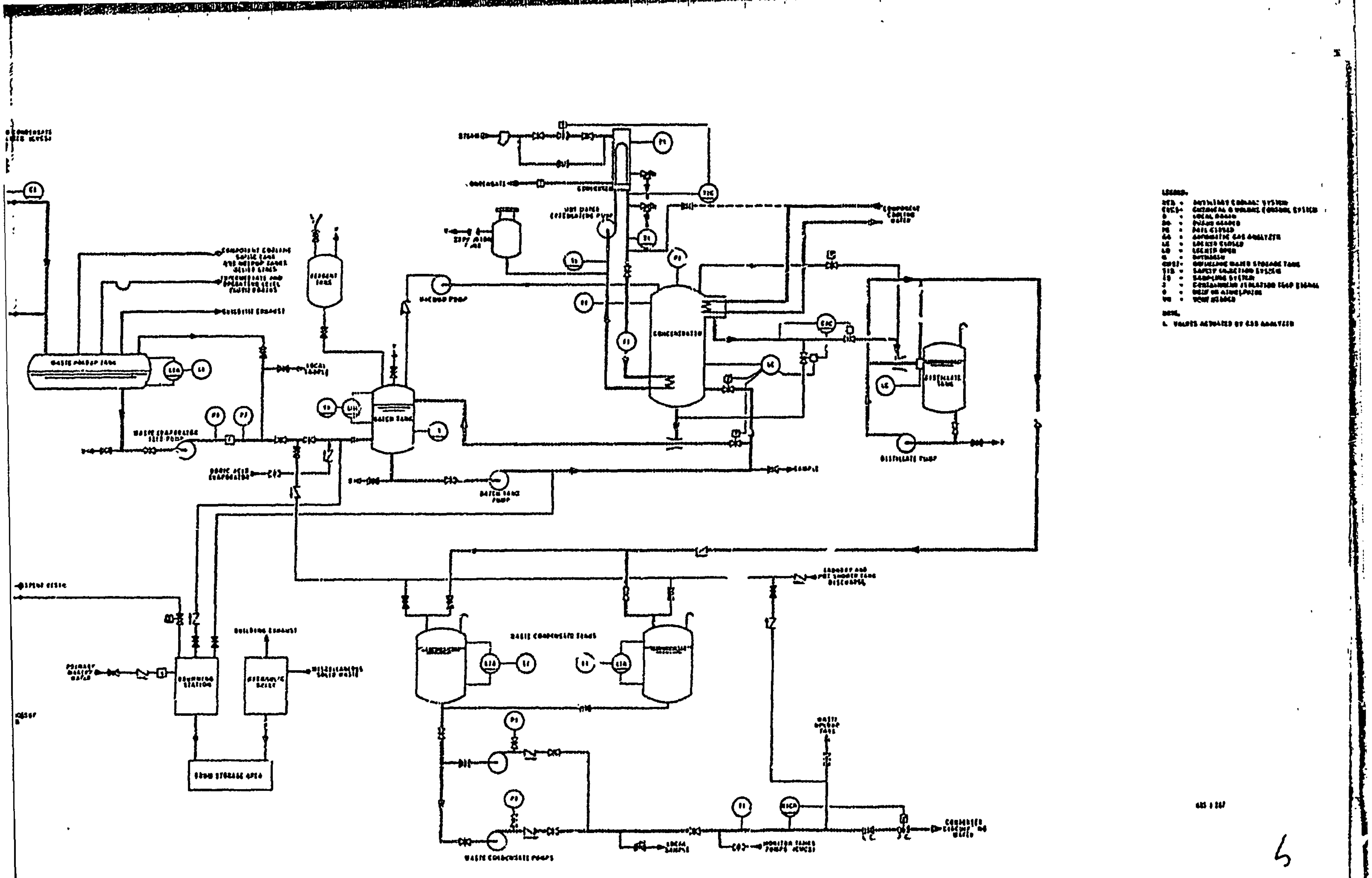
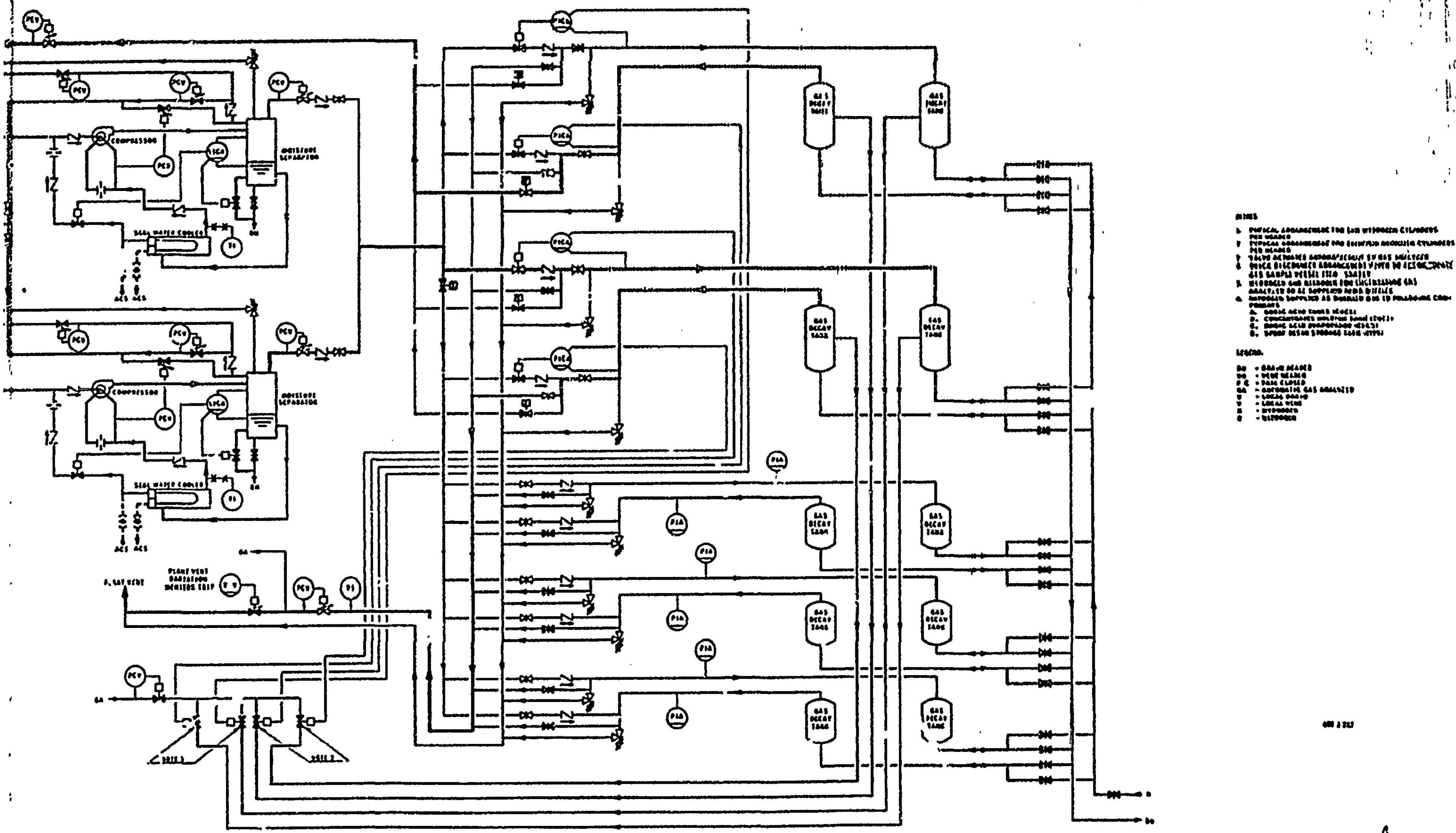


FIGURE 11A-[illegible]: INFLATABLE TUB

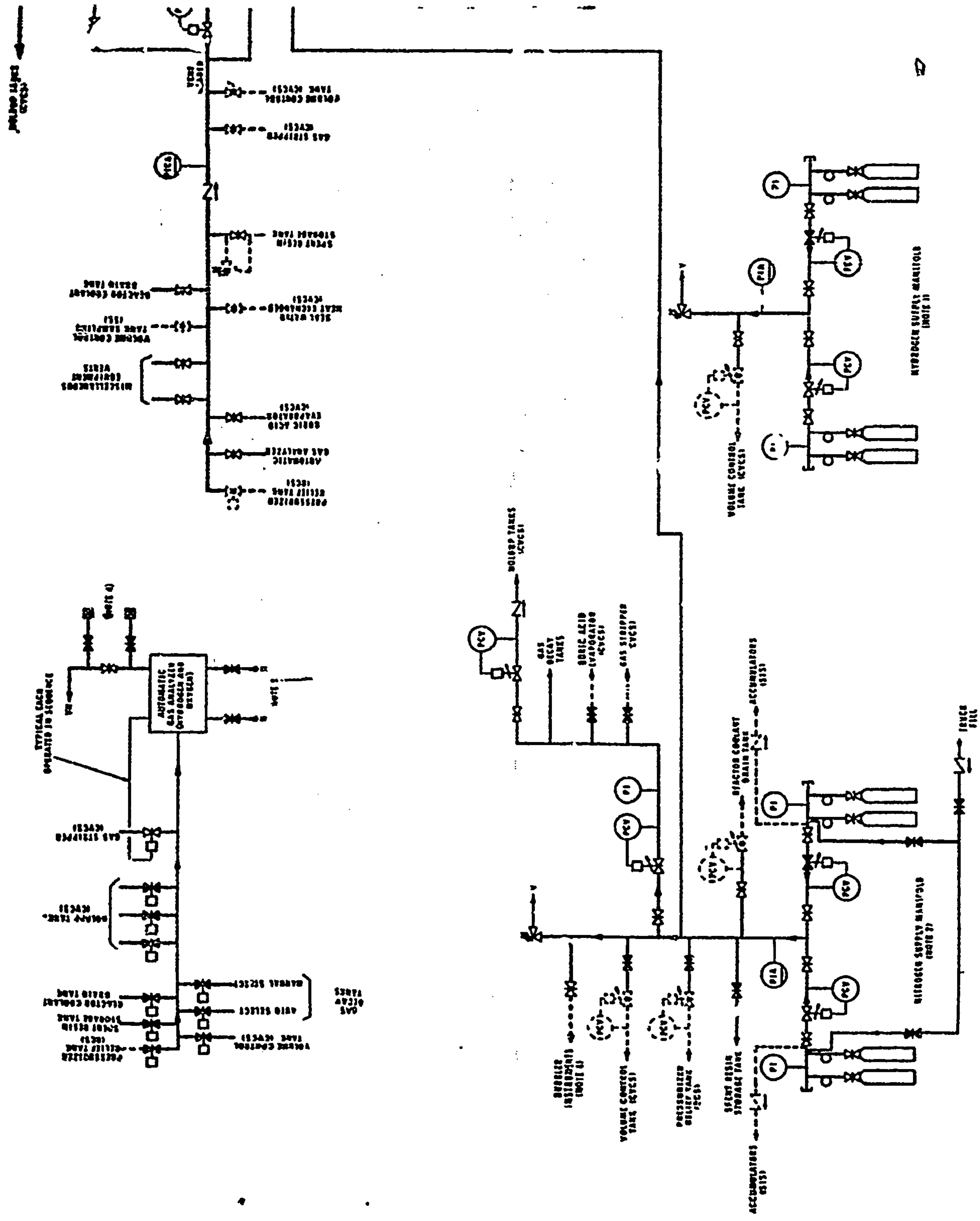


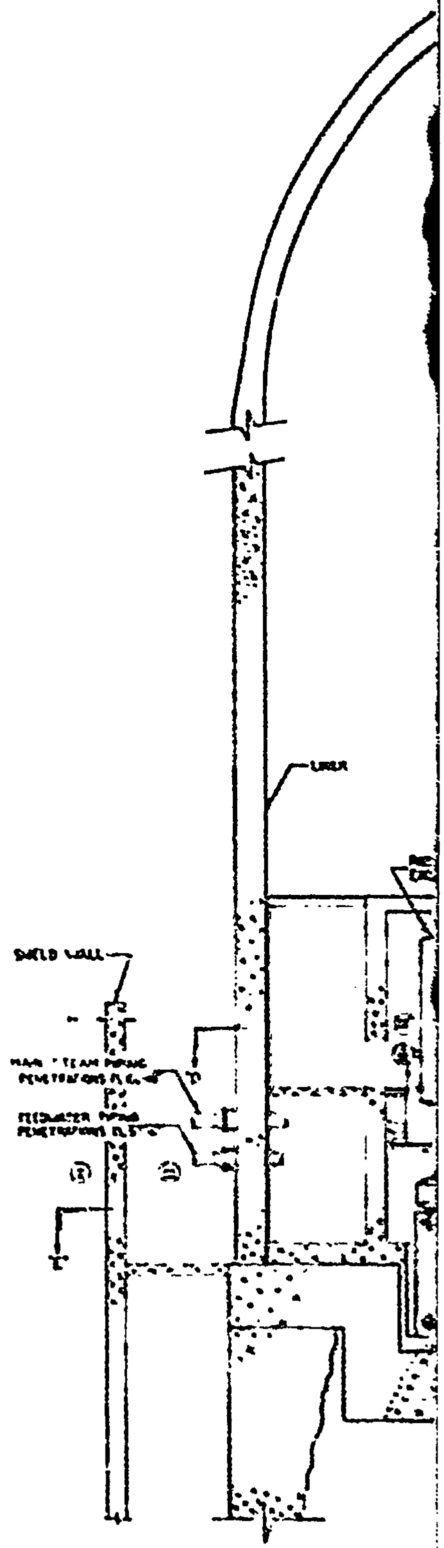


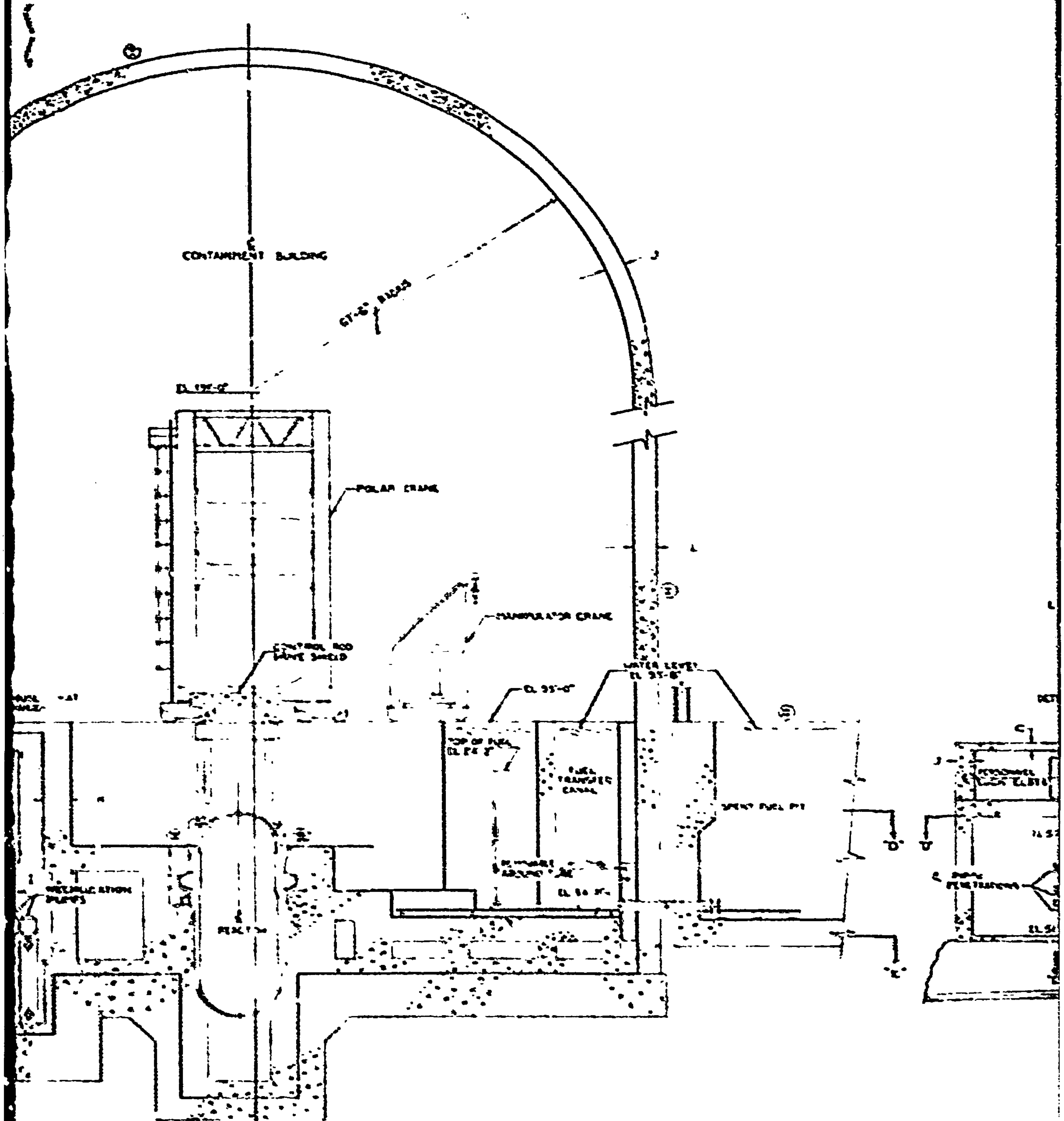
Waste Disposal System Process Flow Diagram - Sheet 1
Figure 11.1-1



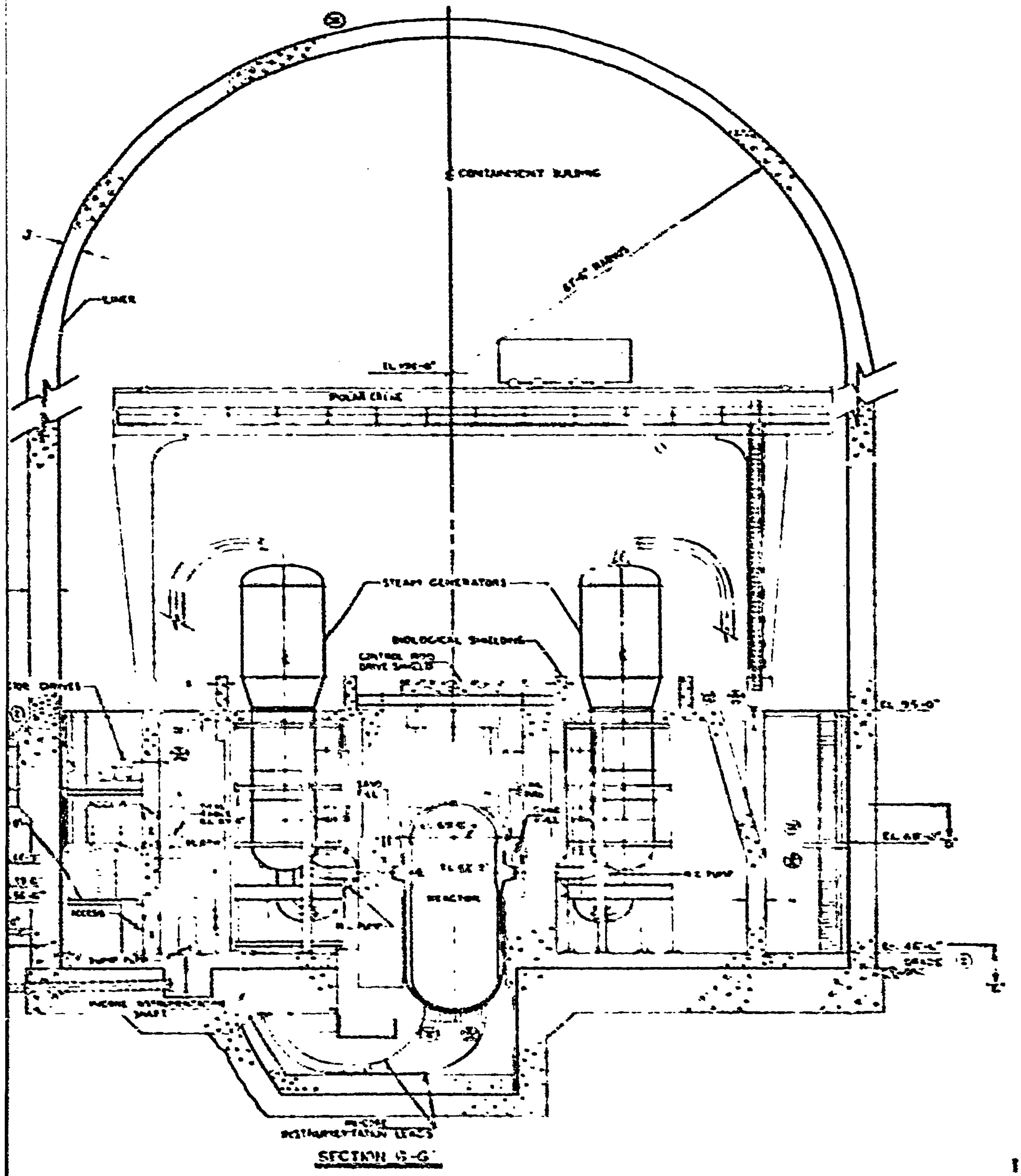
Waste Disposal System Process Flow Diagram - Sheet 2
Figure 11.1-2



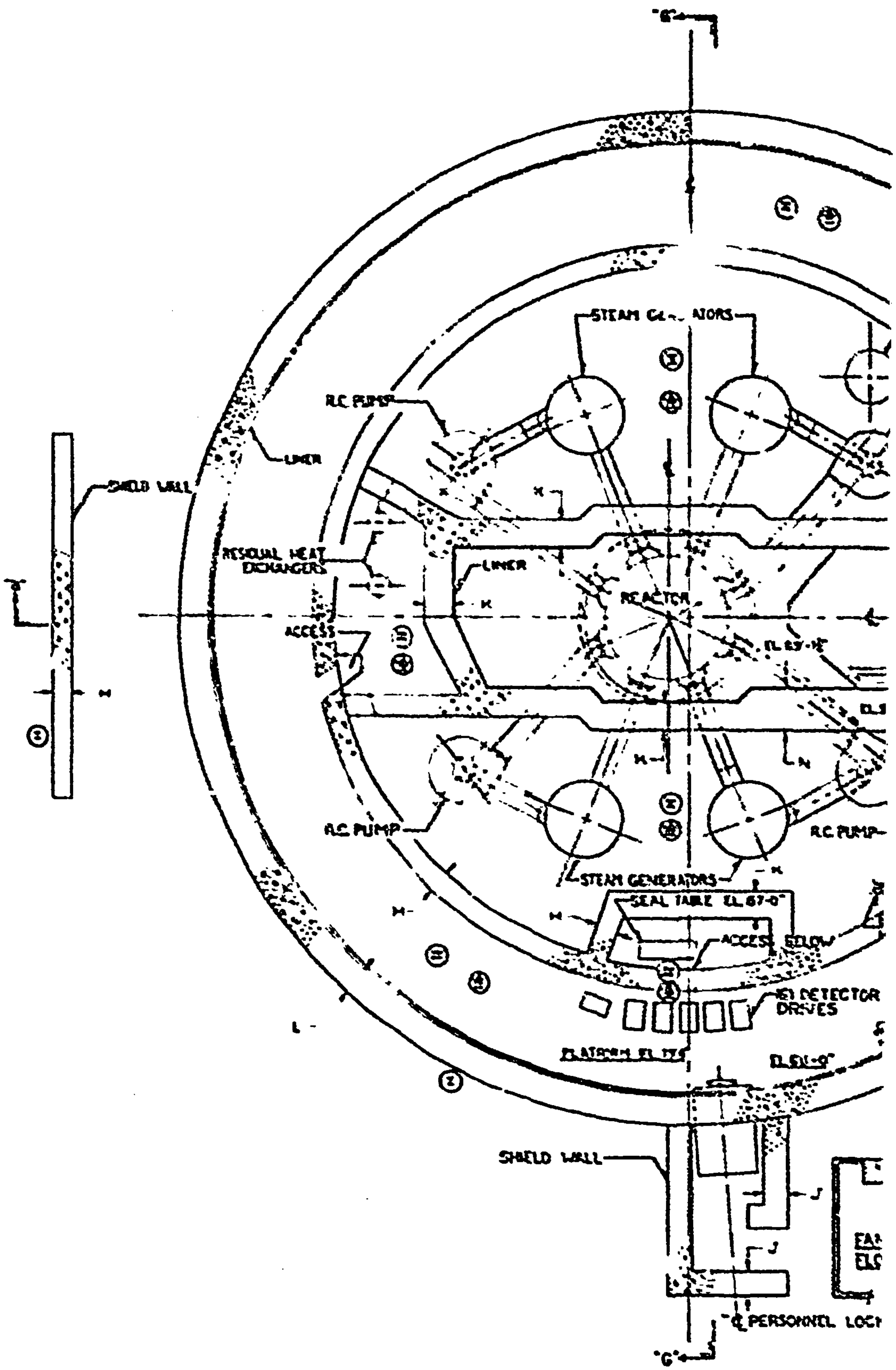

SHIELD WALL
LINER


CONTAINMENT BUILDING
POLAR CRANE
MANIPULATOR CRANE
CONTROL ROD DRIVE SHIELD
WATER LEVEL
FUEL TRANSFER CANAL
SPENT FUEL PIT
RECIRCULATION PUMPS
REACTOR

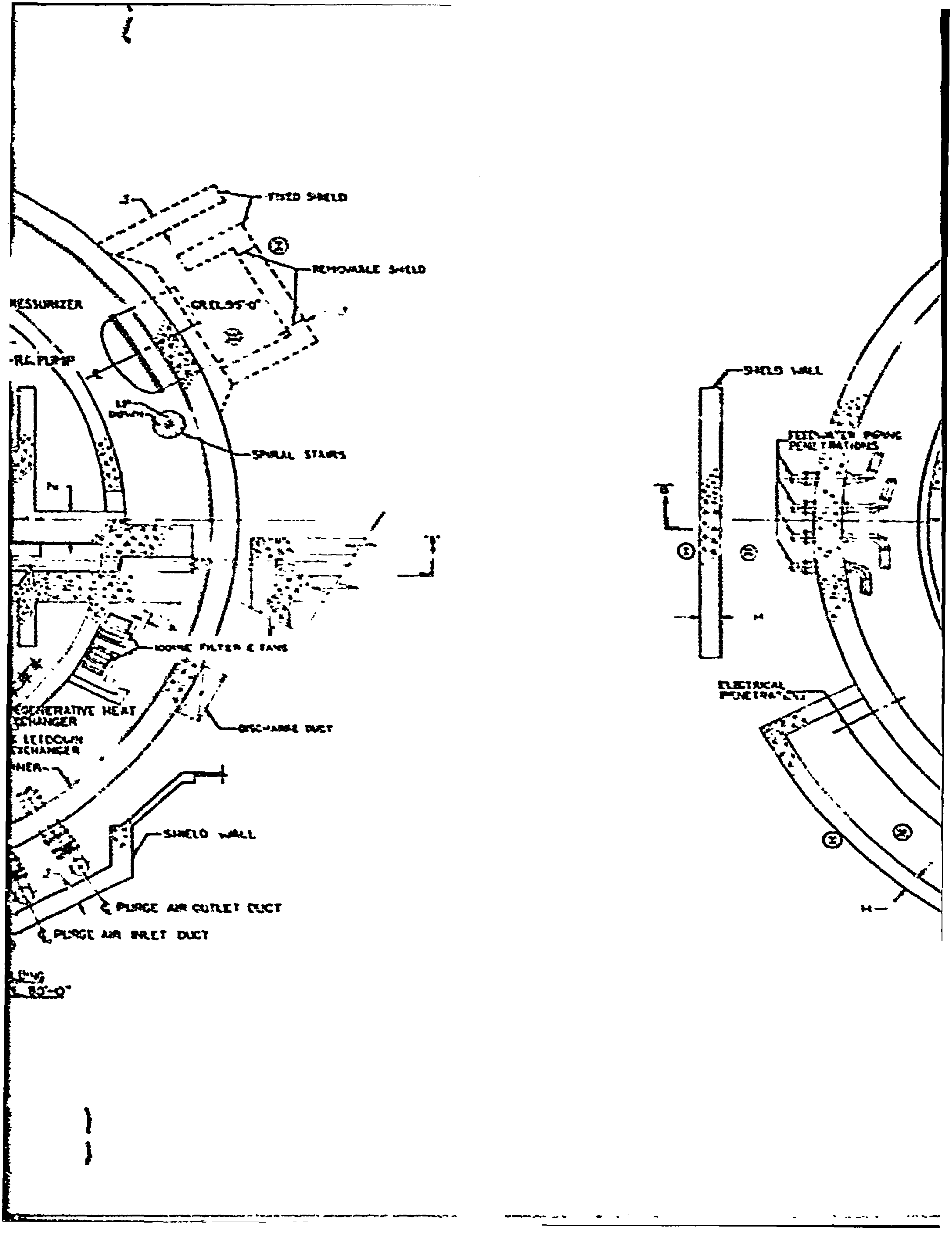
SECTION B-B



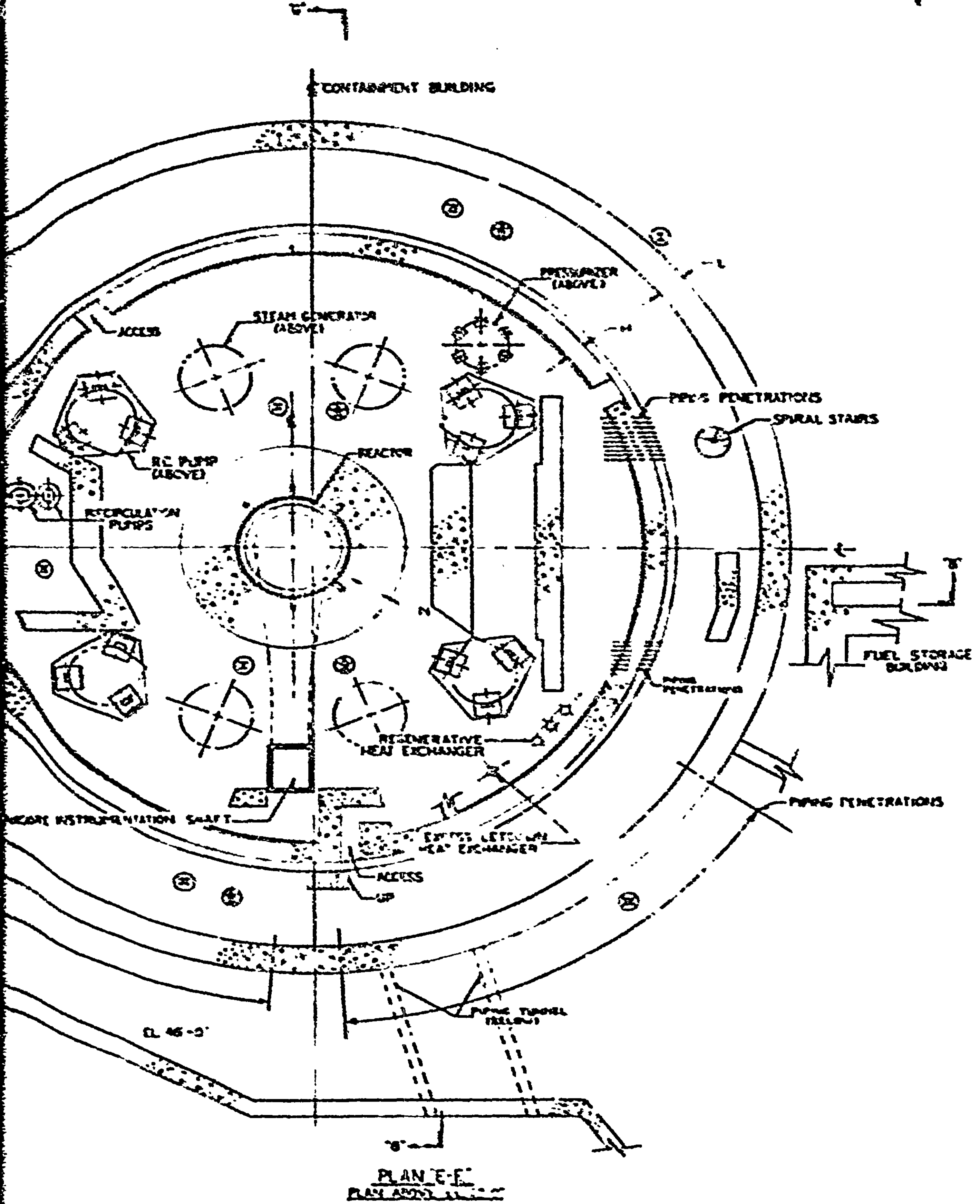
Containment Building Primary Shielding Arrangement
Figure 11.2-1



PLAN "D-D"
PLAN ABOVE EL. 69-0"



FIXED SHIELD
REMOVABLE SHIELD
RESSURIZER
SHIELD WALL
SPIRAL STAIRS
FEEDWATER PIPING PENETRATIONS
ELECTRICAL PENETRATIONS
IODINE FILTER & FANS
EGENERATIVE HEAT XCHANGER
DISCHARGE DUCT
SHIELD WALL
PURGE AIR OUTLET DUCT
PURGE AIR INLET DUCT



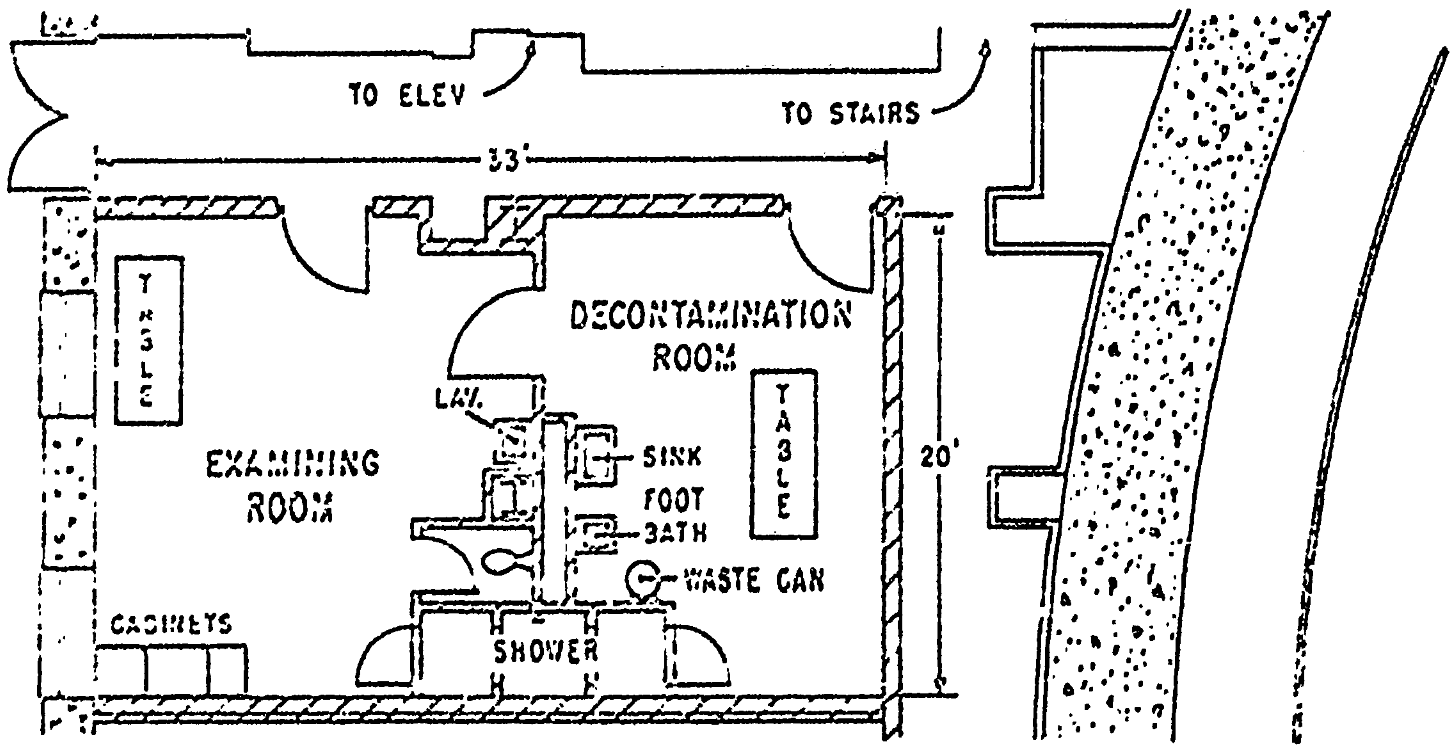
Containment Building Primary Shielding Arrangement
Figure 11.2-2

Indian Point
Final Facility Description and Safety Analysis Report
Dated: October 15, 1968

Ex 4

Figure 11.2-3, Titled "Primary Auxiliary Building Shield Arrangement Plan"

The above was **redacted** by NRC staff as **sensitive information** to allow release of this document to the public.



MEDICAL FACILITY
INDIAN PT. NUCLEAR PLANT

FIGURE 11.2-4

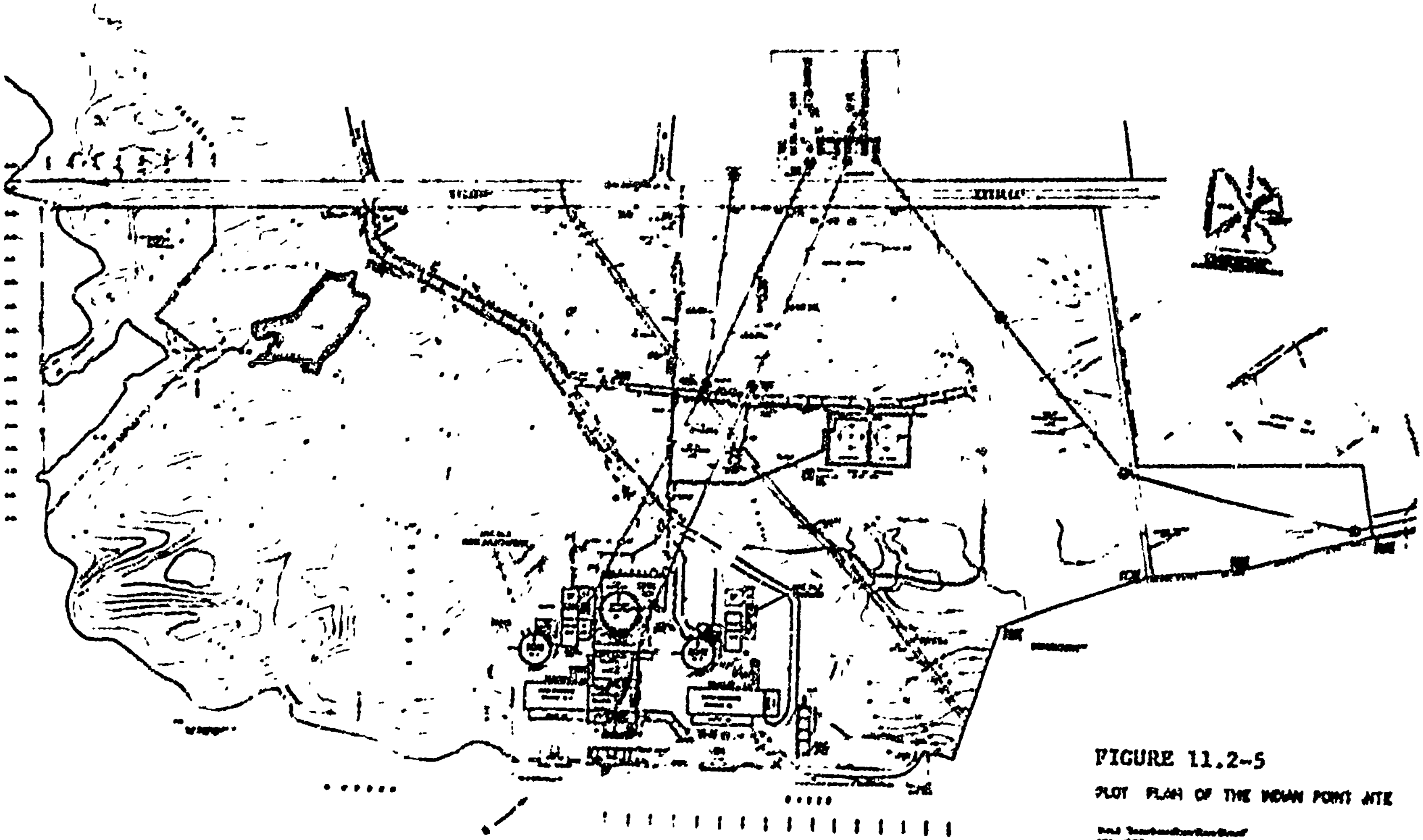


FIGURE 11.2-5

PLOT PLAN OF THE INDIAN POINT SITE

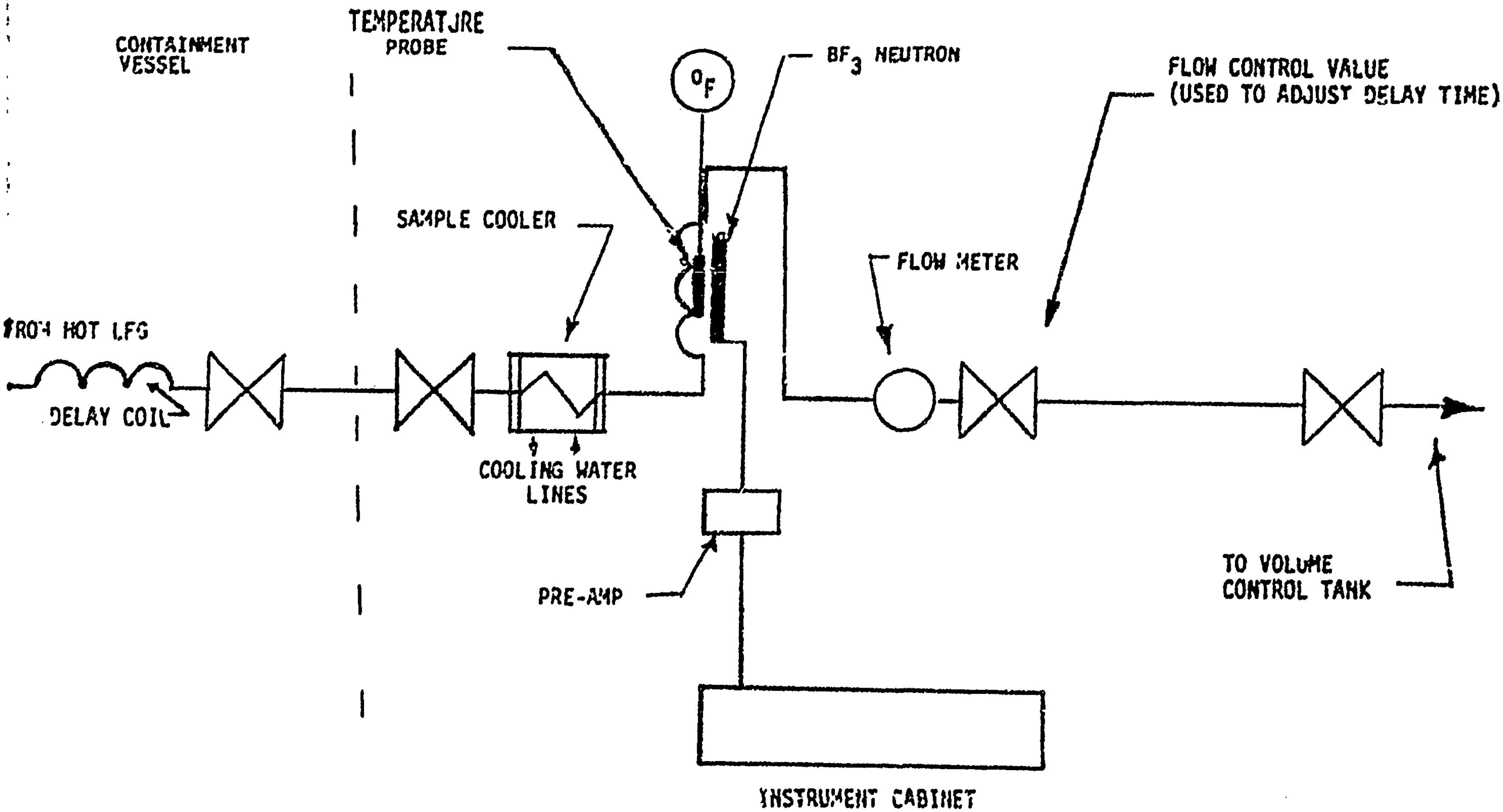


FIGURE 11.2-6 · GROSS FAILED FUEL DETECTOR FLOW DIAGRAM

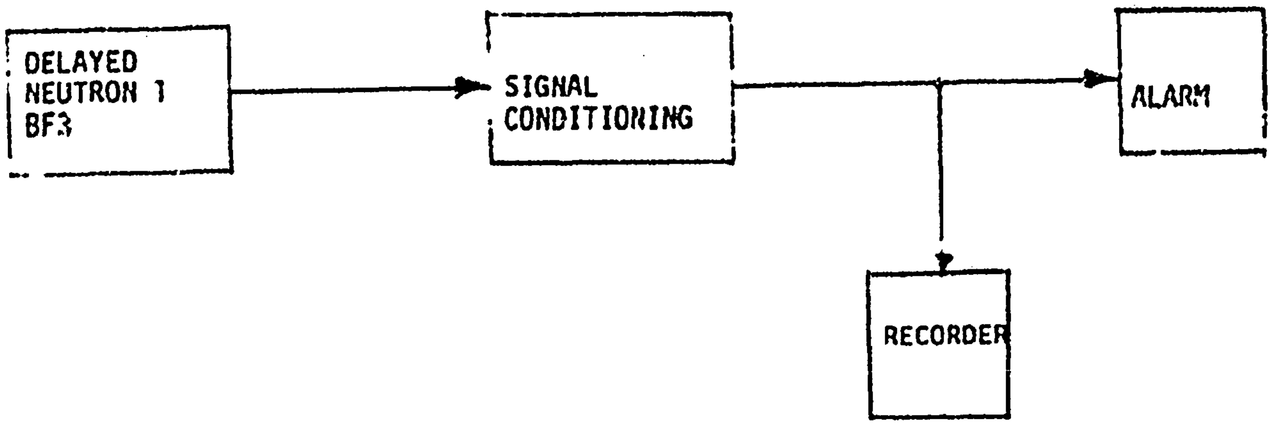


FIGURE 11.2-7 BLOCK DIAGRAM OF THE GROSS FAILED FUEL DETECTOR

APPENDIX 11A

PORTABLE RADIOISOTOPE DECONTAMINATION KIT
DECONTAMINATION INSTRUCTIONS

This kit contains the basic materials (see Table 11A-1) for decontamination of one patient who may be both incapacitated and contaminated with radioisotopes. This kit is shown in Figure 11A-1.

I. Objectives of Decontamination

1 - To reduce the patient's radiation exposure

2 - To prevent spread of isotopes

II. Decontamination Procedures

CAUTION: Life saving procedures should not be delayed in order to decontaminate the patient.

1 - Don protective clothing (cap, gown, plastic apron, gloves, and overshoes).

2 - Transfer patient to the inflatable plastic tub (Figure 11A-2) after inflating it with the hand pump. The inflatable tub should be on a table or stretcher having a slight incline to facilitate drainage of contaminated liquids.

3 - Remove patient's contaminated clothing and place in large plastic bag.

4 - Clean the contaminated skin (if uninjured) first with Lan-O-Kleen and water. The skin should be brushed gently for several minutes, and then checked with survey meter. This may be repeated several times but discontinued before the skin becomes abraded. If the skin is injured or near a wound, use the phisohex. If available, use sterile saline to flush the wound. If the skin around the wound is to be cleaned, care should be taken to avoid transferring contamination into the wound.

Decontamination of the eyes, ears and nostrils is done by saline irrigation. Contaminated hair in nostrils or elsewhere whould be clipped if necessary. If the nose is irrigated, the head should be positioned to avoid backward drainage into the pharynx and swallowing of contaminated saline.

5 - Remove contaminated water through drainage spout at end of plastic tub to storage container, or mop up the liquid with towels and store in plastic bags.

TABLE 11A-1

CONTENTS OF PORTABLE RADIOISOTOPE DECONTAMINATION KIT

I. **Protective Material for 4 Persons**

Surgical gowns and caps - 4 sets (2 large, 2 medium)

Plastic aprons - 4

Disposable plastic gloves - 25 pair

Plastic overshoes - 4 sets

Vinyl sleeve protectors - 4 pair

II. **Protective Materials for Decontamination Area**

Inflatable plastic tub and hand pump - 1

Radiation signs: Radiation Area - 2
Radioactive Materials - 2

Radiation tape - 1 roll

Radiation tags - 6

III. **Decontaminating Equipment and Containers**

3' x 2' plastic bags for clothing - 2

Lan-O-Kleen - 1 bottle

PHisoHex - 1 bottle

Skin decontamination brushes (sterile) (surgical scrub-soft bristle) - 4

Face towels (sterile) - 6

Sterile Saline - 2 liters

IV. **Instruction Sheet**

V. **Valise Storage for Contents of Decontamination Kit**

APPENDIX 11B

GUIDE FOR MANAGEMENT OF RADIOACTIVE CASUALTIES AT UNIVERSITY HOSPITAL

1. Scope of Guide

This guide deals with the health protection problems associated with the management of radiation casualties. The object of the guide is to minimize the radiation dose both to the patient and the hospital staff and to prevent serious contamination of the hospital and its equipment with radioisotopes.

2. Radiation Casualty Team

This should consist of 3 physicians and 3 radiological health physicists. At least one physician and one health physicist should be at the hospital to receive a radiation casualty. At their discretion, the AEC's Radiological Emergency Assistance Team (REAT) could be called upon for help.

The physician would assume over-all supervision of the patient's care and would mobilize and guide the necessary hospital professional and non-professional staff.

The health physicist would have the responsibility for establishing both the area and personnel radiation protection procedures. The latter includes: the issuing of film badges and pocket dosimeters and the keeping of radiation records. In addition, the health physicist would establish, immediately upon admission, the maximum allowable time for proximity of the staff to the patient.

The radiation casualty team will be provided with copies of the "Medical Aspects of Radiation Accidents A Handbook for Physicians, Health Physicists and Industrial Hygienists" edited by Eugene L. Saenger, M.D., and published by the United States Atomic Energy Commission, and other pertinent literature for their information, and guidance.

A training exercise on a simulated radiation casualty would be carried out at least twice a year.

Suggested Personnel
Physicians: Albert, Friedman, Beranbaum
Health Physicists: Van Dyke, Collica, Cleary

3. Alerting the Radiation Casualty Team (RCT)

The University Hospital operator would be provided with the names of the RCT personnel and would contact them upon notification of the pending admission of a radiation casualty. If no RCT personnel are located she would call the AEC's REAT.

4. Initial Evaluation of Patient

The patient would be evaluated, upon arrival at the hospital, by the physician and the health physicist, preferably in the ambulance to avoid possible contamination of the hospital receiving area and unnecessary exposure of hospital staff.

They will assess the nature and severity of the illness and the radiological problems and judge the relative urgency of treatment (surgical or otherwise) and decontamination. For example, if heavily contaminated (but not badly injured or severely ill) the patient would be taken first to an emergency decontamination facility in the employee health area on a special constructed decontamination stretcher and then to the appropriate location for treatment. However, if the patient's life were to be jeopardized by delay in treatment he could be transported directly to (eg.) the operating room which would then have to serve the dual function of a decontamination and treatment area.

If the patient has little surface contamination and requires minor medical or surgical treatment he could be transported on the same type of stretcher to a specially designated private room where both procedures would be carried out.

Wherever possible, the various diagnostic, treatment, and decontamination procedures should be done in the fewest possible places in the hospital.

5. Storage of Decontamination Equipment, Clothing and Radiation Monitors

This material will be kept in readily accessible form in the Department of Radiology at University Hospital.

6. Radiation Hazard Areas

The sites within the hospital where the contaminated patient is treated, should be declared a "RADIATION HAZARD AREA". This involves the establishment of barricades to prevent unauthorized access, and provision of protective clothing, monitors, floor covering, warning signs and decontamination equipment. Once the patient is removed and the contaminated material is packaged in labeled, shielded containers, the site can be declassified if the radiation levels are down to background levels.

7. Hospital Personnel Exposures

The hospital personnel should not be required to receive radiation exposures in excess of the maximum permissible occupational levels. These levels should guide the health physicist in establishing the patient proximity time limits for the staff. The only exception might involve the surgeons and operating room nurses in the event of a protracted operative procedure. Here, it should be possible, by rotation of personnel and improvised shielding to limit the radiation exposure to the quarter-year maximum.

NEW YORK UNIVERSITY MEDICAL CENTER
UNIVERSITY HOSPITAL

RADIATION CASUALTY PLAN 1970

1. TELEPHONE OPERATOR

In case of any emergency involving radiation, whether the call originates with the Medical Center or not, the operator should record the following information:

(a) Name of person calling
(b) Place of accident
(c) Number of persons involved
(d) Telephone number of person calling.

The operator must tell the person calling to stand by the phone unitl the Radiation Safety Officer contacts him.

The operator will immediately first contact the Radiation Safety Officer (or his stand-in) at extension 2628, or at his home, at any time.

Pursuing instructions from the Radiation Safety Officer the operator should be ready to call other members of the Radiation Casualty Team (see attached list).

Should the Radiation Safety Officer be unavailable the operator should contact the first physician listed and follow his instructions.

2. EMERGENCY ROOM

If a patient involved in a radiation accident is presented at the emergency room:

(a) Confine the patient to the ambulance if possible
or
(b) Restrict the patient to a separate part of the room and do not allow him to be examined.
(c) Call the first available physician on the Radiation Casualty Team, and then the first available physicist on the team.
(d) Follow the instructions of the physician.

3. RADIOLOGY RESIDENTS

If an incident is reported which involved radiation safety, determine:

(a) Name of person calling.
(b) Place of incident.
(c) Whether the incident involves potential human radiation injury.
(d) Telephone number of person calling.

If there is potential human radiation injury - proceed as in the instructions for the emergency room.

If there is no potential human injury - contact the first available physicist on the Radiation Casualty Team if necessary.

NEW YORK UNIVERSITY MEDICAL CENTER - UNIVERSITY HOSPITAL
RADIATION CASUALTY TEAM

The membership of the Radiation Casualty Team is revised annually on January 1. Contact the Radiation Safety Officer, N.Y.U. Medical Center Ext. 2628 to obtain the revised list if necessary or with reference to any other radiation safety matters.

PHYSICIANS	PHYSICISTS
1) Roy E. Albert (From June 2, 1970) Dept. Environmental Medicine OR 9-3200 x 2353, 2881 5-212 MU-5-593C	1) C. H. Marshall, Ph.D. Radiation Safety Officer OR 9-3200 x 2628, 3321
Home: 20 Tintern Lane Scarsdale, New York 914-GR 2-0696	Home: 3 Washington Sq. Village New York, New York 10012 212-473-7613

PHYSICIANS	PHYSICISTS
2) Manfred Blum Dept. Medicine OR 9-3200 x 3144, 2629 U.H. Ans. Serv. x 3535	2) Colin G. Orton, Ph. D. Senior Physicist OR 9-3200 x 2628
Home: 330 East 33 Street New York, New York 10016 212-686-3777	Home: 20 Lakewood Drive Lake Arrowhead Denville, N. J. 201-627-2077
3) Marcus Rothschild Dept. Medicine VA Hospital 686-7500 Ext 405	3) Ramesh Chandra, Ph. D. OR 9-3200 x 2629
Home: 1070 Park Avenue New York, New York 10028 212-348-1566	Home: 300 East Prospect Avenue Mt. Vernon, New York 914-699-7518

Should none of the Radiation Casualty Team members be available radiation emergency assistance may be obtained from the following sources:

(a) If the radiation incident occurred in the Medical Center or New York City, call---

1. daytime, New York City Office of Radiation Control
   Tele. No. 566-7754

2. night and weekends, New York City Dept. of Health
   Tele. No. 566-8020

(b) If the radiation incident occurred outside New York City, call---

1. anytime, U. S. Atomic Energy Commission, Radiological Emergency Assistance Team - Tele. No. YUkon 9-1000

In any case, the operator should tell the party that she has a radiation emergency call and to give that party the information in item 1.

Exhaust all possibilities of contacting the Radiation Safety Officer of the Medical Center or any member of the Casualty Team in order of listing.

APPENDIX 11C

EMERGENCY DECONTAMINATION APPARATUS
FOR
STANDARD HOSPITAL STRETCHER

Components

1. Emergency Decontamination Frame for Standard Hospital Stretcher (Figure 11C-1)
2. Emergency Decontamination Tray for Standard Hospital Stretcher (Figure 11C-2)
3. Flexible Plastic Liner for Frame.

Description

The purpose of this apparatus is to provide a storage rig which when set on a standard hospital stretcher can be used in the hospital both to transport and decontaminate patients.

The apparatus consists of a wooden frame (Figure 11C-1), 84" long 26" wide and 12" deep with struts on the lower face to support the frame on a hospital stretcher.

The frame is lined with a flexible plastic insert containing a drainage tube on its lower surface.

In operation the patient is first placed on a perforated (or webbed) steel tray (Figure 11C-2). The tray is then set into the frame and finally the frame is then lifted onto a standard hospital stretcher.

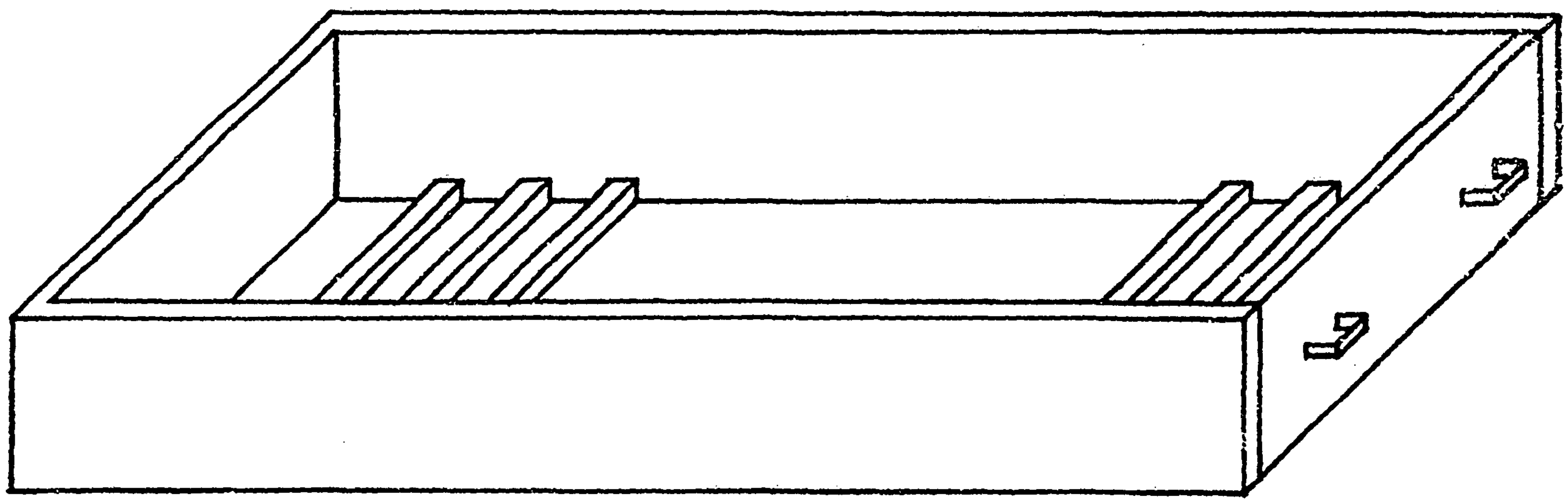


FIGURE 11C-1 EMERGENCY DECONTAMINATION FRAME FOR STANDARD HOSPITAL STRETCHER

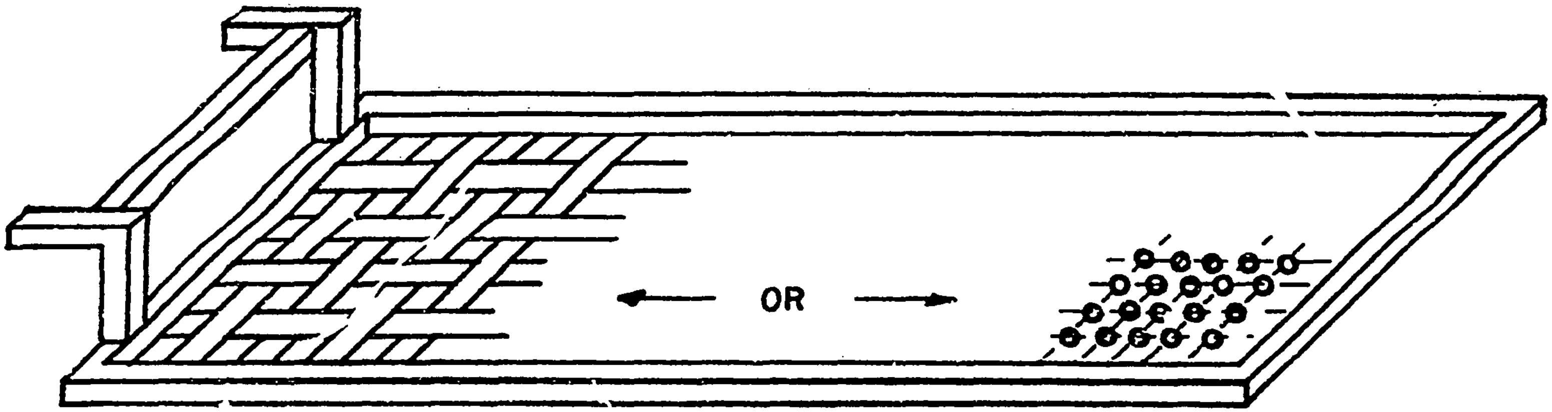


FIGURE 11C-2 EMERGENCY DECONTAMINATION TRAY FOR STANDARD HOSPITAL STRETCHER

REFERENCES - SECTION 11

1. Radiation Accidents and Emergencies in Medicine, Research, and Industry. Edited by Lanzl, L. H., Pingel, J. H., Rust, J. H. page 100, © 1965, by Charles C. Thomas, Publisher.

2. AEC Appendix 0524, Standards for Radiation Protection, III. Guidance For Emergency Exposure During Rescue and Recovery Activities. Approved May 12, 1964.

3. Medical Aspects of Radiation Accidents. Saenger, E. L., Editor. United States Atomic Energy Commission, Washington, D. C., 1963.

4. Thoma, G. E., Jr, and Wald, N. The Diagnosis and Management of Accidental Radiation Injury, J. Occup, Med. 1:421-447, Aug. 1959.

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UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	Docket No. 50-247
Consolidated Edison Company of	)	
New York, Inc.	)	

Amendment No. 9

to

A[illegible]

[illegible] Inc.,

[illegible] captioned proceeding, hereby files Amendment No. 9 to its Application for Licenses for the purpose of transmitting its Final Facility Description and Safety Analysis Report, consisting of four volumes.

WHEREFORE, Applicant prays as in its original Application for Licenses.

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

By W. Donham Crawford

W. Donham Crawford
Administrative Vice President

Dated: October 11, 1968

Subscribed and sworn to before me
this 11th day of October, 1968.

Frances E. Flynn

Notary Public

My Commission Expires March 30, 1969.

FRANCES E. FLYNN
Notary Public, State of New York
24-1253285
Qualified in Kings County
Cert. Filed in New York County
Commission Expires March 30, 1969

CONTENTS OF REPORT

A detailed table of contents, list of tables, and list of figures are included at the front of each major section as listed below. The tables and figures are grouped at the end of each subsection.

Chap 12

TABLE OF CONTENTS

ADDITIONAL INFORMATION SUPPLIED
BY QUESTION RESPONSES IN
VOLUMES 5 AND 6

Section	Title	Question
12.1	Staffing, training and experience - shift and plant organization, responsibilities of facility personnel, training program for facility personnel, resumes for facility personnel	12.1
	Security measures to guard against and to detect unauthorized access to the Reactor Site, the control building and other principal buildings of the facility, the containment building and the fuel storage building.	12.6
	Extent of access to the facility for members of the general public, off-site Con Edison employees, off-site non-Con Edison technicians and engineers, and procedures for admitting the above groups.	12.7
	Resumes for Superintendent performance, Supervisor Engineering (health physics) Assistant Superintendent (maintenance), Assistant Supervisor Engineering (nuclear plant instrumentation), Health, Physics and Conventional Plant instrumentation) and the remaining General Watch Foreman.	12.8
	Relative to Figure 1, Section 12, Supplement No. 2 - The number of individuals under the following job titles: Maintenance Mechanics, Technical Assistants (Chemist), Senior Production Technicians (Shift Chemist), Production Technicians (Chemist) Senior Production Technicians, Production Technicians (Performance, Technicians (Nuclear Plant instrumentation), Technicians (Shift Health Physics) and Technicians (Conventional Plant instrumentation.	12.9
	Relative to Figures 1 & 2, Section 12, Supplement No. 2, all positions for which the intent is to license personnel on Unit No. 2, whether the licenses are SRO or RO, and whether these persons will be "cold" or "hot' licensed.	12.10

Section	Title	Question
	Expansion of the staff or Superintendent Performance and/or staff of the Supervisor Engineering (Health Physics) for Indian Point Unit No. 2 operation, and training received by new personnel.	
	Training received by Superintendence Performance, Assistant Superintendent (Maintenance) and the Supervisor Engineering (Health Physics).	12.11
12.3	Written procedures - safety-related operations for which written procedures are required, review and approval procedure for written procedures including the means for assuring these procedures have been distributed, read and understood by the appropriate operating personnel.	12.2
	Contingency Plans - Earthquake, Fire, Tornado and Radiation	12.5
12.4	Records - abnormal occurrences, principal maintenance activities, fuel inventories and transfers	12.4
12.5	Nuclear Facilities Safety Committee - Quorum, Criteria for membership, minimum meeting frequency, specific functions and responsibilities, recording and reporting requirements	12.3

12. CONDUCT OF OPERATIONS

12.1 ORGANIZATION AND RESPONSIBILITY

At the time of Consolidated Edison's application for a facility operating license for its Unit No. 1 at Indian Point, a description of its proposed operating organization for that facility was submitted as Exhibit O-1A, Docket 50-3. The basic organization and lines of responsibility as described therein are substantially in effect at present. It is planned that the existing organization will be expanded to provide the administrative and technical needs of both Unit No. 1 and the proposed Unit No. 2. Since the Unit No. 2 Radiation Area, Central Control Room, and conventional plant buildings will be physically contiguous to those of Unit No. 1, an expansion of the existing organization, with appropriate personnel additions, is expected to serve fully the needs of both units.

Specifically, and as stated in Consolidated Edison's application to the Commission for a Construction Permit to build its proposed Unit No. 2, the General Superintendent responsible for the overall operation of its Unit No. 1 will also be responsible for Unit No. 2. Reporting to him is the level of Superintendent, which is presently divided into the areas of operation, maintenance, and performance. No increase is contemplated in the numbers of persons at this level in the areas of maintenance and performance, where the existing Superintendent level of supervision is considered adequate to accept the additional responsibilities of a second unit and will therefore not be increased in number. However, there will be additions in maintenance, the number of additions to be established as the needs develop, with the actual additions accomplished as maintenance needs and consequent manpower requirements dictate.

In the area of operations, Exhibit O-1A describes the function and qualifications of the Reactor Engineer, for the facility. At the time of that submittal, that position included responsibility for not only reactor surveillance in the somewhat specialized matters of core physics, physics testing. nuclear instrumentation response, calibration, etc., but also the routine directing of nuclear plant activities. In anticipation of the added responsibilities associated with a second unit at Indian Point, this function was recently divided so as to provide for the startup of Unit No. 2, a Reactor Engineer, whose responsibilities are basically staff in nature and include all of the duties associated with reactor surveillance, and also a Nuclear Plant Superintendent, where responsibilities deal primarily with the personnel and hardware aspects of nuclear plant activities. In these two cases, and in the case of the Conventional Plant Superintendent, it is intended to increase the scope of responsibility of these persons to include Unit No. 2. Two persons have been added at this level, to provide support in the areas of specialized nuclear plant activities such as refueling and electrical operations and training for both units.

Below the level of Superintendent, the presently Senior Operator licensed General Watch Foreman will broaden to assume responsibility at the working shift level for both units. Whereas he now has an additional supervisor, a Watch Foreman, responsible to him for Unit No. 1., he will then have responsible to him a second Watch Foreman licensed as a Senior Reactor Operator for Unit No. 2. Below this level, there will be a Control Operator "A" licensed as a Reactor Operator solely responsible for the Operation of Unit No. 2. He will be in addition to and will share the common Central Control Room with the existing Control Operator "A" responsible for Unit No. 1 operations. Supporting them both will be a Control Operator "B" who will function generally as an assistant to both routinely, and to either in the event of specific need. The operating posts outside the Central Control Room will be manned by persons in the title of Operating Mechanic "A", and are anticipated to be two per shift in number, one in the nuclear plant portion of the facility and one in the "conventional" portion.

12.2 TRAINING

Training of operating personnel for Unit No. 2 is simplified somewhat by the availability of Unit No. 1 as a reactor plant training facility. As a prerequisite for Unit No. 2 startup Watch Foreman candidacy, Consolidated Edison has established that the person must hold a Unit No. 1 Senior Reactor Operator license. Similarly, as a prerequisite for startup Control Operator "A" responsibilities on Unit No. 2, the person must hold a Reactor Operator's license on Unit No. 1. In this manner, personnel with the best possible background qualifications will be provided for Unit No. 2 operating responsibilities. Further, it is anticipated that the other operating posts, i.e., those outside the Central Control Room, will be filled by persons presently holding counterpart responsibilities on Unit No. 1, with their places being filled by persons presently in lesser titles who will be trained over the coming months.

In Spring of 1968, the first phase of training specifically for Unit No. 2 was commenced when certain key staff personnel and those who will assume Unit No. 2 Watch Foreman responsibilities attended a six-week orientation course presented by the Westinghouse Electric Corporation at their Atomic Power Department headquarters near Pittsburgh. The purpose of the course was to provide formal classroom indoctrination on the specific design, operating, and safety feature aspects of Unit No. 2. Staff personnel who attended the course, and all of those assigned to Watch Foreman responsibilities on Unit No. 2 will participate in the writing of startup operating and test procedures. As construction progresses, this group will also participate in the various pre-operational testing efforts necessary for system acceptance.

Upon the return of this group from the Westinghouse course, the material presented to the group is in turn being presented to the Unit No. 2 Control Operator "A" group by one of those who attended the course originally.

Additional on-the-job training efforts are also planned, not only for those who will have Reactor Operator and Senior Reactor Operator responsibilities for Unit No. 2, but for those with plant area responsibilities as well. This training will be provided by Consolidated Edison personnel, Westinghouse personnel, and by representatives of certain manufacturers whose equipment is of such a specialized nature as to make on the job orientation sessions beneficial.

The job of replacement training will be handled in the same manner as is currently being followed on Unit No. 1. Training in the form of extended classroom sessions on the various disciplines required for Reactor and Senior Reactor Operator licensing are given on an almost continuous basis to those for whom licenses are to be sought. In the cases of those seeking Reactor Operator licences, this is coupled with extensive training tailored to the "internal" qualification stage, i.e., the displaying, under a formal examination procedure of those areas of knowledge and technique associated with the synchronizing of the turbine - generator, combustion control for the house service boilers and other "conventional" responsibilities. In the cases of those seeking Senior Reactor licenses, this includes that field training necessary to demonstrate the plant-wide knowledge needed for overall unit responsibilities.

Consolidated Edison employees assigned to Indian Point Station are thoroughly trained in the areas of fire prevention and protection.

General rules and regulations pertaining to the prevention of fires at any of its electric generating facilities are enumerated in a handbook that each employee receives upon his assignment to a station. Supervisory personnel are responsible for assuring that all personnel under their jurisdiction comply with these general rules as well as any others that are applicable to a particular job location. The Safety Services Bureau of the Company maintains a "fire-school" where qualified instructors are available to train and periodically retrain station employees in fire protection techniques and matters. Invaluable experience is gained at this school in as much as actual fires of the type encountered in generating stations are controlled and extinguished by the trainees.

The detail fire-emergency procedure presently in effect at Indian Point Station will be revised to broaden its scope to include Unit No. 2. It is not anticipated that the procedural actions will be substantially changed, however, as the same general plan of fire control will be applicable to both units.

Adequate familiarization with the fire-emergency procedures by all members of the operating staff is demonstrated periodically through the use of "fire drills." To provide a true measure of the degree of readiness, these "fire drills" are generally initiated without prior announcement.

12.3 WRITTEN PROCEDURES

As is the case with Unit No. 1, operating procedures will be developed in advance of startup of the facility. Also as with Unit No. 1, those procedures will be maintained in the possession of all licensed Reactor and Senior Reactor Operators, and at the various key operating locations within the facility.

From time to time, it is expected that specific operating instructions will be needed to accomplish certain operating needs. Such instructions, as with Unit No. 1, will be considered as supplemental to the basic operating procedures, and will be issued to the operating personnel after review and approval by the General Superintendent and appropriate members of his operating staff. Deviations from or modifications to the basic operating procedures, however, will only be made following review and approval by the Station General Superintendent and Consolidated Edison's Nuclear Facility Safety Committee.

Procedures to be followed in the event of an unscheduled radioactive release to the environment in excess of regulation limits will closely parallel those procedures presently in effect for Unit No. 1, and will involve off-site surveys by Consolidated Edison personnel, as well as a close cooperative effort with outside agencies such as the Atomic Energy Commission Division of Compliance, the A.E.C. New York Operations Office Radiological Emergency Assistance Team, the New York State Department of Health, the New York State Department of Police, and the United States Coast Guard. The extent of involvement with these agencies will, as with Unit No. 1, depend on the magnitude of the release, how that release relates to the requirements of the facility Technical Specifications, and the consequent degree of need for cooperative efforts of such agencies.

12.4 RECORDS

Records concerning facility operations will be maintained in the form of log books, charts, and such other internal reports as may be needed to document pertinent operating conditions. The principal logs to be maintained will be those in the Central Control Room, the General Watch Foreman's Office, the shift chemist, and the shift health physics technician. These logs will include descriptions of the operating conditions which exist at the time, descriptions of significant operational efforts accomplished during the shift, and such operating events or circumstances as are deemed pertinent to maintain proper continuity of knowledge and understanding of such matters as responsibility in those areas is passed on from shift to shift.

A record of radiation safety conditions, internal and environmental, is maintained in the form of appropriate log entries, and continuous recording chart information in those functional systems and areas provided with radiation survey instruments. In addition, Radiation Work Permit survey information provides the necessary record of radiation exposure conditions prior to job commencement, and actual personnel radiation exposure information is maintained in the form of film badge and dosimeter records. Records of controlled radiation releases to the environment will be maintained by the station chemical and health physics groups, and all necessary information describing specific radioactivity concentrations, total volumes to be released, along with any dilution requirements, will be entered on the Radioactive Waste Release Permit prepared for each release.

12.5 REVIEW AND AUDIT OF OPERATIONS

In matters such as design changes to the facility involving unreviewed safety questions, changes to operating procedures, or changes to the technical specifications, a review of the question by the Nuclear Facility Safety Committee will be requested by the station General Superintendent. If the Committee concludes that such a change is acceptable from the standpoint of safety, the change will be approved by the Committee, or, if approval by the Atomic Energy Commission is required, a change request for permission to make the change will be initiated by the Committee.

A continuing review of operations is performed by the station operating staff, the Production Department administrative staff, and the executive level for those departments with operating, design and safety responsibility for the facility. In addition, there is periodic review of facility operations by the Nuclear Facility Safety Committee. Frequent communications, both written and oral, between the station General Superintendent and the Chairman of the Nuclear Facility Safety Committee assure the degree of awareness by that Committee of facility operations necessary for it to meet its audit responsibilities. There are frequent meetings of the Committee, at which time it reviews shutdowns of the facility and the reasons therefore, unusual operating conditions, releases to the environment, and proposed changes to the facility and its operating procedures. Periodically, the meetings of the Committee are held at Indian Point so as to allow the entire Committee an opportunity to tour the facility and scrutinize its operations. Approximately once a month, a different member visits the station for a personal audit of facility activities.

Nuclear Facilities Safety Committee

A Nuclear Facilities Safety Committee was established on April 23, 1962 for Indian Point Station, Unit No. 1. This Committee will also perform the same function for Unit No. 2.

The purpose of the Nuclear Faciltles Safety Committee will be to review the operation of the facility, the operating organization, the procedures for operation, changes in the fac[illegible] and the conduct of tests or experiments therein:

a. Membership

The Committee shall have a membership of at least 12 persons of which a majority are independent of the Nuclear Power Generation Department and shall include technically competent persons from all departments of Consolidated Edison having a direct interest in nuclear plant design, operation or in nuclear safety. The Chairman and Vice Chairman will be Senior Officials of the Company experienced in the field of nuclear energy.

The Committee shall consist of:

The Chairman who shall be appointed by the Chairman of the Board or the President of the Company.

The Vice Chairman who shall be appointed by the Chairman of the Board or the President of the Company.

The Secretary who shall be appointed by the Chairman of the Committee.

The following Committee Member shall be designated by the Vice President of the Company who is responsible for the functioning of the department or position stated below with the approval of the Chairman:

The Radiation Safety Officer of the Company.

A medical doctor from the Medical Department having experience in nuclear medicine.

A representative from the Mechanical Engineering Department having experience in nuclear engineering with special emphasis on reactor physics.

A representative from the Nuclear Power Generation Department having experience in nuclear chemistry.

An engineer from the Fuel Department having experience with nuclear fuel.

An engineer from the Electrical Engineering Department having experience in electrical engineering related to nuclear power plants with special emphasis on instrumentation and control.

An engineer from the Mechanical Engineering Department having experience in mechanical engineering related to nuclear power plants with special emphasis on heat transfer.

A representative from the Civil Engineering Department having experience in environmental engineering.

A lawyer from the Law Department who shall be familiar with legal matters affecting nuclear power plants.

The Manager of the Nuclear Power Generation Department.

The Manager of the System Operation Department.

The Reactor Engineer at the Indian Point Station.

Outside consultants, as required, appointed by the Chairman without the right to vote.

Each member will designate a permanent alternate to serve in his absence. The name of the alternates will be filed with the Chairman. Only the permanent member, however, will have the right to vote.

b. Minimum Meeting Frequency

The Committee shall meet not less frequently than quarterly, and at more frequent intervals at the call of the Chairman or in his absence the Vice Chairman, as required.

c. Quorum

A majority of the full committee members which shall include the Chairman or the Vice Chairman and of which a minority are from the Nuclear Power Generation Department shall constitute a quorum for meetings of the full committee.

d. Responsibilities

The Committee will:

Not less than once each year audit and report the adequacy of all procedures used in the operation, maintenance and environmental monitoring of each nuclear power plant. The audits will include on-site inspections and verifications that procedures are adhering to the Operating Licenses and Technical Specifications.

Review and report upon each emergency or infrequent condition relating to nuclear safety including as a minimum those abnormal occurrence defined in the facilities Technical Specifications.

Review and report upon the adequacy of all proposed changes in plant facilities or procedures pertaining to the operation, maintenance and environmental monitoring having safety significance, or which may constitute an "unreviewed safety question" as defined in Part 50, Title 10, Code of Federal Regulations.

Conduct not less than quarterly unannounced spot inspections of plant and monitoring operations and report the results thereof.

Review and report upon any activity, the occurrence or lack of which may affect the safe operation of the nuclear plants.

Review and report upon all proposed changes to the Technical Specifications of licenses.

At the request of the Nuclear Power Generation Manager or a Nuclear Plant General Superintendent the Committee will be promptly convened to review and act upon those nuclear safety matters deemed essential to the safe operation of the facility.

e. Authority

A Nuclear Facilities Safety Committee is constituted to advise the Executive Vice President, Central Operations, concerning the safety aspects of the operation of the nuclear power facilities. The Committee shall report to the Executive Vice President, Central Operations.

The Executive Vice President, Central Operations is responsible for the design, construction, operation and maintenance of nuclear power generation plants. The Vice President, Power Supply and, under him, the Assistant Vice President in charge of Power Generation, the Nuclear Power Generation Manager and Nuclear Plant General Superintendent are responsible for the day-by-day operation and maintenance of the plant. The Nuclear Facilities Safety Committee herein established will advise the Executive Vice President and through him the President and the Chairman of the Board concerning the safety aspects of the nuclear plant operation. The Safety Committee is to be kept fully and currently informed by the Nuclear Power Generation Manager and Nuclear Plant General Superintendent of all matters bearing on the safe operation of the plant. The Chairman may establish subcommittees and designate members of the full committee for assignment to the subcommittees.

The Chairman, Vice Chairman and Secretary of the Nuclear Facilities Safety Committee are ex-officio members of all subcommittees.

f. Records

Draft minutes of all meetings will be distributed promptly to each committee member for comment and any corrections. Amended and corrected minutes will be circulated to committee members for final approval. Copies of approved minutes will be promptly distributed to each committee member and to the Assistant Vice President, Power Generation, the Vice President, Power Supply, the Executive Vice President, Central Operations, the President, the Chairman of the Board, and the Corporation Secretary.

Special reports shall be in writing and will be distributed to all members of the Committee, the Assistant Vice President, Power Generation, the Vice President, Power Supply, the Executive Vice President, Central Operations, the President, the Chairman of the Board, the Corporation Secretary.

g. Charter

A written charter delineating the establishment, composition, mission, and the dissemination of minutes and reports shall be maintained which may be amended as required.

Chap 13

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ADDITIONAL INFORMATION SUPPLIED
BY QUESTION RESPONSES IN
VOLUMES 5 AND 6

Section	Title	Question
13.1	Primary Coolant System; Vibration tests - number, type and location of instruments and flow conditions for the tests. Availability of instrumentation during plant operation.	13.1
	For the startup organization - chains of responsibility and authority for all groups participating, including Westinghouse support groups, Westinghouse personnel, Con Edison personnel; identification of who will analyze and approve test results.	13.4
13.3	[illegible]-operational and in-service tests and monitoring for X-Y xenon stability.	13.3
13.4	In-service monitoring for vibration and loose parts in the reactor vessel and other portions of the Primary Systems.	13.2

13. INITIAL TESTS AND OPERATION

The initial testing and start-up operation of the unit systems prior to full power operation of the unit includes tests prior to reactor fueling, core loading, precritical tests, zero power tests and power level escalation.

The purpose of this program is to test and operate the reactor and its various systems (1) to make certain that the equipment has been installed and will operate in accordance with the design requirements (2) to provide safe procedures for initial fuel loading and to determine zero power values of core parameters significant to the design and operation and (3) to bring the unit to its rated capacity in a safe and orderly fashion.

Procedures stating the test purpose, conditions, precautions, and limitations will be prepared for each test. The procedures will include a delineation of administrative procedures and test responsibility, equipment clearance procedures, and an overall sequence of startup operations.

The test program described in the following sections is based upon the reference plant design and experience gained during startup of other units. Detailed procedures specify the sequence of tests to be conducted and the conditions under which each is to be performed to ensure the relevance and consistency of the results obtained. This includes expected values and acceptance criteria which demonstrate the degree to which the facility does meet design criteria.

13.1 TESTS PRIOR TO REACTOR FUELING

The following tabulation is the sequence of major start-up tests and operations performed to place all equipment in the specified system in service. The systems and items to be tested are listed in approximately chronological order. Consolidated Edison Company of New York, Inc. in cooperation with Westinghouse Electric Corporation will prepare detailed test procedures prior to scheduled initial testing of systems and determination of reactor physics parameters.

The tests conducted on the engineered safety systems are included under the Containment System, Safety Injection System, the Containment Spray System, and the Containment Air Recirculation Cooling and Filtration System.

The test objectives incorporate testing of redundant equipment where it is involved.

Abnormal plant conditions may be simulated during testing when such conditions do not endanger personnel, equipment or contaminate clean systems. Where predicted emergency or abnormal conditions are involved in the testing program the detailed operation will be provided in the test procedure.

Acceptance Criterion for all components and systems is that the test results are acceptable when the test objectives are met within the design specification limits and within the applicable Technical Specification.

1. Switchgear System
2. Voice Communications Systems
3. Service Water System
4. Fire Protection System
5. Instrument and Service Air Systems
6. Nitrogen Storage System
7. Reactor Coolant System Cleaning
8. Reactor Containment Air Recirculation and Filtration System
9. Feedwater and Condensate Circulation Systems
10. Auxiliary Coolant System
11. Chemical Feed System
12. Chemical & Volume Control System
13. Safety Injection System
14. Fuel Handling System
15. Containment Isolation and Isolation Valve Seal Water Systems
16. Containment Penetration and Weld Channel Pressurization System
17. Reactor Containment High Pressure Test
18. Cold Hydrostatic Tests

19. Radiation Monitoring System

20. Nuclear Instrumentation System

21. Radioactive Waste Disposal System

22. Sampling System

23. Instrumentation Calibration

24. Hot Functional Test

    Reactor Coolant System
    Chemical & Volume Control System
    Sampling System
    Safety Injection System
    Waste Disposal System

25. Primary and Secondary Systems Safety Valves Tests

26. Turbine Steam Seal & Blowdown Systems

27. Emergency Diesel-Electric System

General procedures and responsibilities for the initial tests and operations of Unit No. 2 are discussed in Appendix 13A. 15

TABLE 13.1-1

OBJECTIVES OF TESTS PRIOR TO REACTOR FUELING

SYSTEM OR TEST	TEST OBJECTIVE
1. Switchgear System (Electrical Tests)	To ensure continuity, circuit integrity, and the correct and reliable functioning of electrical apparatus. Electrical tests will be performed on transformers, switchgear, turbine-generator motors, cables, control circuits, excitation switchgear, D-C System, annunciator system, lighting distribution switchboard, communication system and miscellaneous equipment. Special attention will be directed to the following tests: a. 480 v. switchgear breaker interlock test. b. Station loss of voltage auto-transfer test. c. Critical power transfer test. d. Tests of protective devices. e. Equipment automatic start tests. f. Check exciter for proper voltage build up.
2. Voice Communication Systems	To verify proper communication between all intra plant stations, for interconnection to commercial phone service and to balance and adjust amplifiers and speakers.
3. Service Water System	To verify, prior to critical operations, the design head-capacity characteristics of the service water pumps, that the system will supply design flow rate through all heat exchangers, and will meet the specified requirements when operated in the safeguards mode.

TABLE 13.1-1 (Continued)

SYSTEM OR TEST		TEST OBJECTIVE
4.	Fire Protection System	To verify proper operation of the system by ensuring the design specifications are met for the fire service booster pump and fire service pumps, checking that automatic start functions operate as designed, and that level and pressure controls meet specifications.
5.	Instrument and Service Air Systems	To verify the operation of all compressors to design specifications, the manual and automatic operation of controls at design setpoints, design air-dryer cycle time and moisture content of discharge air, and proper air pressure to each instrument served by the system.
6.	Nitrogen Storage System	To verify system integrity, valve operability, regulating and reducing station performance and the ability to supply nitrogen to interconnecting systems as required.
7.	Reactor Coolant System Cleaning	To flush and clean the reactor coolant and related primary systems to obtain the degree of cleanliness required for the intended service. Provisions to maintain cleanliness integrity and protection from contamination sources will be made after system cleaning and acceptance.

TABLE 13.1-1 (Continued)

SYSTEM OR TEST		TEST OBJECTIVE
		The system, component, or section of a system shall be considered clean when the flush cloth shows no grindings, filings or insoluble particulate matter larger than 40 microns (lower limit of naked eye visibility). After systems have been flushed clean of particulate matter within the limit specified, the cleanliness integrity of the system will be maintained filled with water which meets the system cold chemistry requirement. After fill and pressurization and prior to hot operation, cold chemistry requirements will be maintained. Oxygen will be analyzed prior to exceeding 200°F and brought into specification prior to exceeding 200°F.
8.	Reactor Containment Air Recirculation and Filtration System	To verify, prior to critical operation, the fan capacities; and the remote and automatic operation of system louvers and valves in accordance with the design specifications.
9.	Feedwater and Condensate Circulation Systems	To verify proper operation of feedwater and circulating water pumps according to specifications, valve and control operability and set points, flushing and hydro as applicable, inspection for completeness and integrity. Functional testing will be performed when a steam supply is available.

TABLE 13.1-1 (Continued)

SYSTEM OR TEST	TEST OBJECTIVE
10. Auxiliary Coolant System	To verify component cooling flow to all components, and to verify proper operation of instrumentation, controllers and alarms. Specifically each of the three loops, i.e. component cooling loop, residual heat removal loop and spent fuel pit cooling loop will be tested to ensure: a. All manual and remotely operated valves are operable manually and/or remotely. b. All pumps perform according to manufacturers specifications. c. All temperature, flow, level and pressure controllers function to control at the required setpoint when supplied with appropriate signals. d. All temperature, flow, level and pressure alarms provide alarms at the required locations when the alarm setpoint is reached and clear when the reset point is reached. e. Design flow rates established through heat exchangers.

TABLE 13.1-1 (Continued)

SYSTEM OR TEST	TEST OBJECTIVE
11. Chemical Feed System	To verify valve and control operability and setpoints, flushing and hydro as applicable, inspection for completeness and integrity. Functional testing will be performed when a steam supply is available.
12. Chemical and Volume Control System (CVCS)	To verify, prior to critical operation, that the CVCS functions as specified in the system description and appropriate technical manuals. More specifically that: a. All manual and remotely operated valves are operable manually and/or remotely b. All pumps perform to manufacturers specifications c. All temperature, flow, level and pressure controllers function to control at the required setpoint when supplied with appropriate signal(s) d. All temperature, flow, level, and pressure alarms provide alarms at the required locations when the alarm setpoint is reached and clear when the reset point is reached. e. The reactor make-up control controls blending, dilution, and boration as designed. f. The design seal water flow rates are attainable to each reactor coolant pump g. The boric acid evaporator package functions as specified in the manufacturers technical manual.

12a	Containment Spray	To verify performance of the containment spray pumps.
13.	Safety Injection System (SIS)	To verify prior to critical operation, response to control signals and sequencing of the pumps, valves, and controllers of this system as specified in the system description and the manufacturers technical manuals; and check the time required to actuate the system after a safety injection signal is received. More specifically that:

a. All manual and remotely operated valves are operable manually and/or remotely.

b. All pumps perform their design functions satisfactorily.

c. For each pair of valves to redundant flow paths, disabling one of the valves does not impair remote operation of the other.

d. The proper sequencing of valves and pumps occurs on initiation of a safety injection signal.

e. The fail position on loss of power for each remotely operated valve is as specified.

f. Valves requiring coincidence signals of safety injection and high containment pressure operate when supplied with these signals.

TABLE 13.1-1 (Continued)

SYSTEM OR TEST		TEST OBJECTIVE TEST
13.	Continued	g. All level and pressure units are set at the specified points and provide alarms at the required location(s); and re t at the specified point. h. The time required to actuate the system is within the deisgn specifications.
14.	Fuel Handling System	To show that the system design is capable of providing a safe and effective means of transporting and handling fuel from the time it reaches the plant until it leaves the plant. In particular the tests will be designed to verify that: a. The major structures required for refueling such as the reactor cavity, refueling canal, spent fuel storage and decontamination facilities are in accordance with the design specifications. b. The major equipment required for refueling such as the manipulator crane, spent fuel pit bridge and fuel transfer system, operate in accordance with the design specifications. c. All auxiliary equipment and instrumentation function properly.
15.	Containment Isolation and Isolation Valve Seal Water Systems	To verify the capability for reliable operation and demonstrate the manual and automatic operation of the system. Demonstrate the operation and proper sequence of isolation

TABLE 13.1-1 (Continued)

SYSTEM OR TEST	TEST OBJECTIVE
15. Continued	valve closure and seal water addition. Demonstrate function of Isolation Valve Seal Water System independent of other systems. Demonstrate the operation and system response time induced by as isolation signal. Manual valves will be manipulated to assure proper operation of the seal gas injection portion of the system.
16. Containment Penetration and Weld Channel Pressurization System	To verify air system and nitrogen backup system integrity, operate valves, check flowmeters and pressure gauges as required to ensure pressure differential meets design specifications.
17. Reactor Containment High Pressure Test	To verify prior to critical operation, the structural integrity and leak tightness of the containment.
18. Cold Hydrostatic Tests	To verify the integrity and leak tightness of the Reactor Coolant System and related primary systems with the performance of a hydrostatic test at the specified test pressure with no visible leakage, nor distortion.
19. Radiation Monitoring System	To verify the calibration, operability, and alarm setpoints of all radiation level monitors, air particulate monitors, gas monitors and liquid monitors which are included in the Operational Radiation Monitoring System and the Area Radiation Monitoring System.

TABLE 13.1-1 (Continued)

SYSTEM OR TEST	TEST OBJECTIVE
20. Nuclear Instrumentation System	To ensure that the instrumentation system is capable of monitoring the reactor leakage neutron flux from source range through 120% of full power and that protective functions are operating properly. In particular the tests will be designed to verify that: a. All system equipment, cabling, and interconnections have been properly installed. b. The source range detector and associated instrumentation respond to neutron level changes and that the source range protection (high flux level reactor trip) as well as alarm features and audible count rate operate properly. c. The intermediate range instrumentation reactor protective and control features high level reactor trip and high level rod stop signals operate properly and that permissive signals for blocking source range trip and source range high voltage off operate properly. d. The power range instrumentation operates properly and that the protective features such as the overpower trips, permissive and dropped-rod functions operate with the required redundancy and separation through the associated logic matrices, and nuclear power signals to other systems are available and operating properly.

TABLE 13.1-1 (Continued)

SYSTEM OR TEST	TEST OBJECTIVE
	e. All Auxiliary Equipment such as the comparator and startup rate channel, recorders, and indicators operate as specified. f. All instruments are properly calibrated and all set points and alarms are properly set.
21. Radioactive Waste Disposal System	To verify satisfactory flow characteristics through the equipment; to demonstrate satisfactory performance of pumps and instruments; to check for leak-tightness of piping and equipment, and to verify proper operation of alarms, instrumentation and controls. More specifically that: a. All piping and components properly installed as per design specifications b. All manual and automatic valves are operable c. All instrument controllers operate to control process at requried values d. All process alarms operable at required locations e. All pumps perform to manufacturers specifications

TABLE 13.1-1 (Continued)

SYSTEM OR TEST	TEST OBJECTIVE
	f. All pumps Indication & Controls are operable at designated stations
	g. The Waste Gas Compressors packages operate as specified in manufacturers technical manual
	h. The Gas Analyzer operates as specified in manufacturers technical manual
	i. The Waste Boiler operates as specified in manufacturers technical manual
	j. The Hydrogen & Nitrogen supply packages sufficient for all modes of operation
22. Sampling System	To verify that a specified quantity of representative fluid can be obtained safely and at design conditions from each sampling point. In particular the tests will be designed to verify that:
	a. All system piping and components are properly installed
	b. All remotely and manually operated valving operates in accordance with the design specifications
	c. All sample containers and quick-disconnect couplings function properly, and as specified

TABLE 13.1-1 (Continued)

SYSTEM OR TEST	TEST OBJECTIVE
23. Instrument Calibration	Instrumentation and control devices will be checked to assure their accuracy. Primary sensing elements, transducers, transmitters, receivers, recorders and indicators will be thoroughly inspected and adjusted for accuracy of their set point characteristics. Interconnecting piping and wiring will be checked for continuity and functional requirements. Each device will then be tested in accordance with established test procedures. Limit switches used for initiating indicating lights, alarms and inter-lock functions will be checked under actual or simulated operating conditions. Control devices will be exercised to assure proper operation with the required accuracy and response characteristics. Set points for devices will be checked and adjusted to their specified values. Each individual circuit of the reactor and turbine protection systems will be tested to verify that appropriate signals initiate reactor and turbine trips. As a signal level corresponding to the particular condition is reached, trip or cutback functions will annunciate as provided for the particular channel under test.

TABLE 13.1-1 (Continued)

SYSTEM OR TEST	TEST OBJECTIVE
24. Hot Functional Tests	The Reactor Coolant System will be tested to check heatup (using pump heat) and cooldown procedures; to demonstrate satisfactory performance of components prior to installation of the core; to verify proper operation of instrumentation, controllers and alarms; and to provide operating conditions for checkout of auxiliary systems. The Chemical and Volume Control System will be tested to determine that water can be charged at rated flow against normal Reactor Coolant System pressure; to check letdown flow against design rate for each pressure reduction station; to determine the response of the system to changes in pressurizer level; to check procedures and components used in boric acid batching and transfer operations; to check operation of the reactor makeup control; to check operation of the excess letdown and seal water flowpath; and to verify proper operation of instrumentation, controllers and alarms. The Sampling System will be tested to determine that a specified quantity of representative fluid can be obtained safely and at design conditions from each sampling point.

TABLE 13.1-1 (Continued)

SYSTEM OR TEST	TEST OBJECTIVE
	The Auxiliary Coolant System will be tested to evaluate its ability to remove heat from reactor coolant; to verify component cooling flow to all components; and to verify proper operation of instrumentation, controllers and alarms.
	The Safety Injection System will be tested to check the time required to actuate the system after a safety injection signal is received; to check that pumps and motor operated valves are properly sequenced; and to verify proper operation of instrumentation, controllers and alarms.
	The Radioactive Waste Disposal System will be tested to verify satisfactory flow characteristics through the equipment; to demonstrate satisfactory performance of pumps and instruments; to check for leak-tightness of piping and equipment; and to verify proper operation of alarms.
	The Ventilation System will be tested to adjust proper flow characteristics of ducts and equipment; to demonstrate satisfactory performance of fans, filters, and coolers; and to verify proper operation of instruments and alarms.

TABLE 13.1-1 (Continued)

SYSTEM OR TEST	TEST OBJECTIVE
25. Primary and Secondary Systems Safety Valves Tests	To test pressurizer and boiler safety and relief valves to ensure each valve is operable.
26. Turbine Steam Seal and Blowdown Systems	To verify valve and control operability and setpoints, flushing and hydro as applicable, inspection for completeness and integrity. Functional testing will be performed when a steam supply is available.
27. Emergency Diesel Electric System	To demonstrate that the system is capable of providing power for operation of vital equipment under power failure conditions. In particular the tests will be designed to verify that: a. All system components have been properly installed. b. The emergency diesels function according to the design specification under emergency conditions. c. The emergency units are capable of supplying the required power to vital equipment under emergency conditions. d. All redundant features of the system function according to the design specifications.

13.2 FINAL PLANT PREPARATION

13.2.1 CORE LOADING

The as-loaded core configuration is specified as part of the fuel core design studies conducted well in advance of plant startup and as such is not normally subject to change at plant startup. In the relatively unlikely event that mechanical damage is sustained during core loading operations by a fuel assembly of a type for which no spare is available on-site, a previously examined alternate core loading scheme whose characteristics closely approximate those of the initially prescribed pattern may be invoked.

The core is assembled in the reactor vessel in water containing enough dissolved boric acid (usually at least 2000 ppm) to maintain the core multiplication constant at 0.90 or lower and is not subsequently distributed or changed until the end of the core cycle. Core moderator chemistry conditions (particularly boron concentration) are prescribed in the core loading procedure document and are verified by chemical analysis of moderator samples every eight hours during core loading operations.

The Reactor Coolant System will be isolated and applicable tagging and administrative procedures will be used to prevent unauthorized change in the boron concentration. The boric acid tank will be filled with concentrated boric acid solution and the residual heat removal system will be in service and available to provide moderator mixing and temperature control, if required. A detailed preloading checkoff list will be followed to insure that all systems, equipment and conditions affecting the loading operation are met. Periodically, the checkoff list will be reviewed to insure that systems and equipment continue to meet requirements of the core loading operation.

The core loading sequence will follow a step procedure to insure at each loading step that:

a) Fuel assemblies of the correct enrichments are installed in the proper locations.

b) Rod cluster control assemblies are inserted into the proper fuel assemblies prior to loading the assemblies into the core.

c) Neutron sources and neutron detectors are properly located in the core during fueling. Continuous radiation monitoring will be provided at the core loading stations during fuel handling and core loading operations.

Core loading instrumentation will consist of two permanently installed plant source range (pulse-type) nuclear channels and two temporary incore source range channels plus a third temporary channel to be used as a spare. The permanent channels are monitored in the control room by licensed plant operators; the temporary channels are installed in the vapor container and are monitored by technical specialists of the Westinghouse Electric Corporation and by licensed Senior Reactor Operators of Consolidated Edison Co. of New York, Inc. At least one plant channel and one temporary channel are equipped with audible count range indicators. Both plant channels and both regular temporary channels display neutron count rate on count rate meters and strip chart recorders. Minimum count rates of two counts per second, attributable to core neutrons, are required on at least two of the four available nuclear channels at all times during core loading operations. Two artificial neutron sources, each rated at approximately 200 curies of Po^{210} - alpha activity, are introduced into the core at appropriate specified points in the core loading program to ensure a neutron population large enough for adequate monitoring of the core.

Fuel assemblies together with inserted control components (RCC units or burnable poison inserts) are added to the core one at a time according to a previously established and approved sequence which has been developed to provide reliable core monitoring with minimum possibility of core mechanical damage. The core loading procedure documents include a detailed tabular check sheet which prescribes and verifies the successive movements of each fuel assembly and its specified inserts from its initial position in the storage racks to its final positions in the core. Multiple checks are made of component serial numbers and types at successive transfer points to guard against possible inadvertent exchanges or substitutions of components.

An initial nucleus of eight fuel assemblies, the first of which bears an activated neutron source, has been determined to be the minimum source-fuel nucleus which will permit subsequent meaningful inverse count rate monitoring. This initial nculeus is known by calculation and previous experience to be markedly subcritical (k_{eff} <0.90) under the required conditions of loading.

Subsequent fuel additions are made, one assembly at a time with detailed inverse count rate ratio monitoring after each addition. The results of each loading step must be evaluated by both Westinghouse Electric Corporation technical specialists and licensed Consolidated Edison Co. of New York, Inc. operations personnel and concurrent approval to proceed must be granted before the next prescribed step can be started.

Criteria for safe loading require that loading operations stop immediately if:

a) The neutron count rates on all responding nuclear channels double during any single loading step.

b) The neutron count rate on any individual nuclear channel increases by a factor of five during any single loading step.

A sphere evacuation alarm is coupled to the plant source range channels with a set point at five times the current count rate to provide automatic indication of high counter rate during fuel addition.

In the event that an unacceptable increase in count rate is observed on any or all responding nuclear channels, special procedures involving fuel withdrawal from the core, detector relocation and charging of additional boric acid into the moderator are invoked by Westinghouse Electric Corporation technical specialists with the approval of licensed operational personnel of the Consolidated Edison Company of New York, Inc.

Core loading procedures specify alignment of fluid systems to prevent inadvertent dilution of the reactor coolant, restrict the movement of fuel to minimize the possibility of mechanical damage, prescribe the conditions under which loading may proceed, identify chains of responsibility and authority and provide for continuous and complete fuel and core component accountability.

13.2.2 PRECRITICAL TESTS

Upon completion of core loading and installation of the reactor upper internals and the reactor vessel head, certain mechanical and electrical tests are performed prior to initial criticality. The electrical wiring for the rod drive circuits, the rod position indicators, primary and secondary trip circuits, and the in-core thermocouples are tested. Final operational tests are repeated on these electrical items.

Mechanical and electrical tests are performed on the RCC unit drive mechanisms. Tests include a complete operational checkout of the mechanisms. Checks are made to ensure that the rod positon indicator coil stacks are connected to their proper position indicators. Similar checks are made on the RCC unit drive coils.

After filling and venting is completed, the final hydro tests are conducted.

Tests are performed on the reactor trip circuits to test manual trip operation. Actual RCC unit drop times are measured for each rod control cluster at operating temperature, pressure and flow. By use of dummy signals, the various plant abnormalities that require tripping are simulated and accurate trip delay times are measured for the control and protection system circuitry.

A complete electrical and mechanical check is made on the in-core nuclear flux mapping system at the operating temperature and pressure.

The in-core thermocouple tests check circuit continuity and compare the thermocouple readings for their relative errors (offsets) in the isothermal condition.

13.3 INITIAL TESTING IN THE OPERATING REACTOR

Tests which will be performed from the initial core loading to rated power are summarized in Table 13.3-1.

13.3.1 INITIAL CRITICALITY

Initial criticality is established by withdrawing the shutdown and control banks of RCC units from the core, leaving the last-withdrawn control bank inserted far enough to provide effective control when criticality is achieved, and then slowly and continuously diluting the heavily borated reactor coolant until the chain reaction is self-sustaining.

Successive stages of RCC bank withdrawal and of boron concentration reduction are monitored by observing change in neutron count rate as indicated by the regular plant source range nuclear instrumentation as functions of RCC bank position and, subsequently, of primary water addition to the reactor coolant system during dilution.

Primary safety reliance is based on inverse count rate ratio monitoring as an indication of the nearness and rate of approach of cirticality of the core during RCC bank withdrawal and during reactor coolant boron dilution. The rate of approach toward criticality is reduced as the reactor approaches extrapolated criticality to ensure that effective control is maintained at all times.

Relevant procedures specify alignment of fluid systems to allow controlled start and stop and adjustment of the rate of which the approach to criticality may proceed, indicate values of core conditions under which criticality is expected and identify chains of responsibility and authority during reactor operations.

13.3.2 ZERO POWER TESTING

Upon establishment of criticality a prescribed program of reactor physics measurements is undertaken to verify that the basic statics and kinetic characteristics of the core are as expected and that the values of kinetics coefficients assumed in the safeguards analysis are indeed conservative.

Measurements made at zero power and primarily at or near operating temperature and pressure include verification of calculated values of RCC group and unit worths, of isothermal temperature coefficient under various core conditions, of differential boron concentration worth and of critical boron concentrations as function of RCC control group configuration. Preliminary checks on relative power distribution are made in normal and abnormal RCC unit configurations.

Concurrent tests are conducted on the plant instrumentation including the source and intermediate range nuclear channels. RCC unit operation and the behavior of the associated control and indicating circuits are demonstrated.

Detailed procedures specify the sequence of tests and measurements to be conducted and the conditions under which each is to be performed to ensure the relevancy and consistency of the results obtained. These tests will cover a series of prescribed control rod configurations with intervening measurements of differential control rod worths and boron worth during boron dilution or boron injection. As the successive configurations are established, the measurement techniques to be used will be:

1) <u>Dynamic Temperature Coefficient Measurement</u> - Differential moderator coefficient measurement will be made by continuously increasing or decreasing the moderator average temperature and observing the resultant change in core reactivity.

2) <u>Dynamic Control Rod Worth Measurements</u> - Control rod differential worth measurements will be made by monotonically withdrawing or inserting selected control rods or groups of rods and part length rods and observing the resultant change in core reactivity.

3) <u>Dynamic Boron Worth Measurements</u> - Differential boron worth measurements will be made by monotonically increasing or decreasing main coolant boron concentration and observing the resultant change in core reactivity.

13.3.3 POWER LEVEL ESCALATION

In order to ensure that operation of the core is as expected in all respects, and that achievement of rated power is under carefully controlled conditions, a Power Escalation Test Program will be established to carry the plant from completion of zero power physics testing through full power operation. The Power Escalation Test Program provides for stepwise achievement of full power, with careful review of significant core parameters at each step, to ensure that fuel and control rod mechanical performance, flux distribution, temperature distribution hot channel factors and reactivity control worths are acceptable, before additional escalation is undertaken.

The Power Escalation Test Program provides for measurements to be made at convenient power levels in the vicinity of minimum self sustaining power, discrete levels approaching 100%, and at rated power. In each case, progress to higher levels is contingent upon acceptable core performance.

Preparation for Power Escalation

In order to monitor performance, the following analytical results must be on hand before power escalation is undertaken:

1. Expected values for local power ratios in each of the in-core flux detector thimbles.

2. Expected values for relative power in each fuel assembly and in individual fuel rods of interest in various control group configurations.

3. Expected values of nuclear peaking factors.

4. Combined power and programmed temperature reactivity defect as a function of primary power level at expected boron concentrations.

5. Equilibrium xenon reactivity defect as a function of primary power level.

6. Identification and integral reactivity worth of the most significant single RCC assemblies in the control group, when fully withdrawn, with various operating control rod configurations, for both full and part length rods.

7. Identification and integral reactivity worth of the most significant single RCC assemblies among all groups, for both full and part-length rods.

Other conditions that must be met before commencement of the Power Escalation Test Program are as follows:

1. The following plant conditions are established:

   a. The Zero Power Reactor Physics Test Program has been successfully completed as prescribed. Experimental values of zero power reactivity parameters have been reduced and are available for guidance in the elevated power program.

   b. Discrepancies between analytically predicted and experimentally measured values of reactivity parameters have been identified and appropriate revisions have been made in the values of expected primary coolant boron concentrations and RCC group positions listed in the Power Escalation Test Sequence.

   c. The Reactor Coolant System and all required components of the Secondary Coolant System are fully assembled, mechanically tested and ready for service as required.

   d. All control, protection and safety systems are fully installed; all required pre-operational tests are satisfactorily completed and all components are ready for service as required.

   e. The reactor coolant is at required temperature, pressure, lithium and boron concentration.

f. Demineralized water is available in adequate quantity for extensive boron dilution.

g. Concentrated boric acid solution is available in sufficient quantity to permit increases in main coolant boron concentration as required.

h. All special equipment and instrumentation required for the Power Escalation Test Program is installed and calibrated and is available for service as specified.

i. Thermocouple correction constants derived from the hot, isothermal calibrations.

j. Reactor coolant flow coastdown measured and found acceptable.

2. A pre-test check-off list indicating the required status of all systems and auxiliary equipment affecting the power Escalation Test Program is available. The pre-test check-off list shall include, but shall not be limited to, provisions for verification and certification of all items specified in Condition 1, above.

3. Experimental procedures suitable for executing the Power Escalation Test Sequence, are available for distribution to all personnel concerned with the Power Escalation Test Program.

4. The procedure, schedule and personnel assignments and responsibilities are thoroughly discussed with and are understood by the operational and experimental personnel.

The following tests are to be conducted during the power escalation test program.

Electrical Trip Testing

Electrical tripping relays that are initiated by plant on-power malfunctions will be retested and the consequent trip sequence rechecked under operating conditions for correct operation and sequence.

Turbine Trip Testing

The turbine protection system will be checked to confirm that the appropriate initiation will either trip the turbine through the main trip solenoid or will mechanically trip the turbine. As the various set-points or status conditions are reached, the trip or runback functions will be verified.

Elevated Power Reactivity Coefficient Evaluation

During the approach to full power and during initial operation at power a sequence of reactor physics measurements will be carried out to experimentally determine power and temperature coefficients and power defects at various power levels, differential (full and part length) control rod worth and boron worths during boron dilutions, and xenon worth during initial operation. Measurements techniques are:

1) Dynamic Differential Power Coefficient - Differential power coefficient measurements are to be made at elevated power over a limited range in power level by initiating a small power level change. The change in core reactivity associated with the compensating control rod motion, is to be related to the net change in power level.

2) Dynamic Power Defect Measurements - The change in reactivity defect associated with a relatively large change in power level is to be measured by adjusting control rod positions during a ramp change in power level to maintain moderator average temperature at the prescribed value and by observing the compensating change in core reactivity due to control rod movement as indicated by the reactivity computer.

3) Dynamic Control Rod Worth Measurements - Control rod differential worth measurements are to be made at elevated power and by initiating a transient change in boron concentration in the coolant by adjusting control rod position during the transient to maintain moderator average temperature and power level essentially constant, and by observing the compensating change in core reactivity due to control rod movement as indicated by the reactivity computer.

4) Dynamic Boron Worth Measurements - Differential boron worth measurements are to be made at elevated power by monotonically increasing or decreasing main coolant boron concentration. Compensation for the reactivity effect or the boron concentration change will be made by withdrawing or inserting, respectively, control rods to maintain moderator average temperature and power level constant and observing the resultant accumulated change in core reactivity corresponding to successive rod motion steps.

5) Dynamic Xenon Transient Worth Measurements - Integral xenon worth transient measurements are to be made at elevated power, after a change in power level, by adjusting control rod position to maintain moderator average temperature and power level constant during the reactivity transient associated with the transient change in effective xenon concentration and observing the resultant accumulated change in core reactivity corresponding to successive compensating rod motion steps.

6) Elevated Power Transient Response Evaluation - As the power level is increased during the initial power escalation, a series of transient response measurements will be made to determine plant response to load changes. The test technique in each case will consist of establishing the transient change in plant conditions and closely monitoring system response during and after the transient period. The responses of system components are measured for 10% loss of load and recovery, loss of load with steam dump, turbine trip, loss of reactor coolant flow and trip of single RCC units, reactor coolant coastdown is also measured.

7) Elevated Power Determination of Power Distribution - At successive power levels and in prescribed control rod configurations (full and part-length), measurements of flux and power distributions within the core will be made and nuclear hot channel factors will be evaluated. Use will be made of the miniature in-core flux detector system, and of the in-core temperature sensors, to determine the nuclear power and thermal and hydraulic conditions within the core. Ex-core nuclear instrumentation will be calibrated to indicate actual in-core axial power distribution.

8) Determination of Primary Coolant Flow Rate - Primary coolant flow rate will be evaluated by measuring primary coolant pump power and elbow tap pressure differential.

9) Verification of Remote Control Stations - After the plant has been certified to operate at elevated power levels the capability for manually taking the plant to hot shutdown from stations remote from the control room will be verified. This test will demonstrate that controls and information available in the local control stations are functioning properly and are sufficient to permit the operators to trip the plant, control heat removal, and borate in an orderly manner to reach and maintain the reactor in a hot shutdown status should the control room ever become uninhabitable.

Test	Conditions
RCC Unit Drop Tests	a) Cold, Shutdown b) Hot, Shutdown
Thermocouple/RTD Intercalibration	Various temperatures during system heatup at zero power
Nuclear Design Check Tests	All two dimensional RCC control group configurations at hot, zero power
RCC Control Group Calibration	All RCC control groups at hot, zero power
Power Coefficient Measurement	0% to 100% of full power
Automatic Control System Checkout	Approximately 20%
Power Range Instrumentation Calibration	During static and/or transient conditions at: 30% 70% 90% 100%

TABLE 13.3-1 (Page 1 of 4)

INITIAL TESTING SUMMARY

Objectives	Acceptance Criteria
To measure the scram time of RCC units under full flow and no flow conditions	Droptime less than value assumed in Safety Analysis
To determine in-place isothermal correction constants for all core exit thermocouples and reactor coolant RTD's	RTD's verify that RTD system meets setpoint requirements of technical specifications
To verify that nuclear design predictions for endpoint boron concentrations, isothermal temperature coefficient & power distributions are valid	FFD and SAR limiting values for $\delta\rho/\delta T$, $F_{\Delta H}$
To verify that nuclear design predictions for control group differential worths with and without partial length RCC units are valid	FFD and SAR limiting values for $\delta\rho/\delta h$, [illegible] h
To verify that nuclear design predictions for differential power coefficient are valid	FFD and SAR limiting values for $\delta\rho/\delta q$
To verify that control system response characteristics for the: a) Steam generator level control system b) RCC automatic control system c) turbine control system	No safety criteria applicable
To verify that all power range instrumentation consisting of: power range nuclear channels in-core flux mapping system core exit thermocouple system reactor coolant RTD's are responsive to changes in reactor power level and power distribution and to intercalibrate the several systems	Verify that setpoints cited in Technical Specifications are met

TAB

Test	Conditions	Ob
Load Swing Test	$\pm$ 10% steps at: $\sim$40% to 50% $\sim$100%	To sys
Plant Trip	Full load rejection from: $\sim$50% $\sim$100%	To
Pressurizer Effectiveness Test	Hot, shutdown	To can by
Minimum shutdown Veritifcation	Hot, zero power	To dic bor RCC
Psuedo Ejection Test	Hot, zero power	To of pow RCC gro
Pseudo Ejection Test	30% of rated power	To of dis uni
Loss of Flow Test	Hot shutdown	Mea fol
Power Redistribution Follow	70% of rated power	To men in tra

13.3-1 (Page 2 of 4)

...tives	Acceptance Criteria
...rify that reactor control ...n performance	No safety criteria applicable
...rify reactor control performance	Proper operation of steam dump and feedwater overrides.
...rify that pressurizer pressure ...e reduced at the required rate ...essurizer spray actuation	No safety criteria applicable
...rify the nuclear design pre- ...on of the minimum shutdown ...concentration with one "stuck" ...nit	Verify stuck rod shutdown criteria
...rify nuclear design predictions ...fects on core reactivity and ...distribution of ejection of one ...nit from a fully inserted control	FFD and SAR limiting values for $F_{\Delta H}$, reactivity insertion
...rify nuclear design predictions ...fects on core reactivity and power ...bution of ejection of one RCC ...from typical operating configuration.	FFD and SAR limiting values for $F_{\Delta H}$, reactivity insertion
...e reactor coolant flow coastdown ...ing trip of reactor coolant pumps.	Flow coastdown no faster than FFD and SAR curves.
...rify that excore nuclear instru- ...tion adequately monitors changes ...e power distribution under ...ent xenon conditions.	FFD and SAR symmetric offset F_{q} correlation

TABLE 13.3

Test	Conditions	Objectives
Static RCC Drop Test	~50% of rated power	To verify t serted full control ba nuclear ins thermocoupl conditions adjustment points
RCC Insertion Test	~50% of rated power	To determin fully inser reactivity under typic bases for s limits
Dynamic RCC Drop Test	70% of rated power	To verify a dropped rod matic rod s
Load Reduction Test	50% reduction from 70% 50% reduction from 100%	To verify t
P/L Group Operational Maneuvering	90%	To verify t maneuvering containing xenon trans
Load Cycle Test	40% to 85%	To verify t capable of ations with operational weekly cycl
Turbo-Generator Startup Tests	Pre- and Post-Synchronization	To verify t unit and as trips are i ready for s

	Acceptance Criteria
hat a single RCC unit in- y or part way below the k can be detected by excore trumentation and core exit es under typical operating and to provide bases for of protection system set	Inserted rod detectable with instrumentation
e the effect of a single ted RCC unit on core and core power distribution al operating conditions as etting turbine runback	See next step
utomatic detection of , and subsequent auto- top and turbine cutback	Required power reduction and rod withdrawal block accomplished
eactor control system	No safety criteria applicable
hat the part-length RCC scheme is effective in and suppressing spatial ients	FFD and SAR limiting values for F_q $F_{\Delta H}$
hat all plant systems are sustaining load follow oper- out encountering unacceptable limits through a typical e.	FFD and SAR limiting values for F_q, $F_{\Delta H}$, shutdown margin.
hat the turbo-generator sociated controls and n good working order and ervice.	Successful completion of all mechanical and electrical and control functional checks

Test	Conditions
Turbo-Generator	Power level sufficient for turbine auxiliaries to be operating
Control Valve Tests	~70% of rated power
Acceptance Run	100 hours at rated full power

Objective	Acceptance Criteria
o verify normal trouble free perfor- ance of the turbo-generator at low ower	Performance within manufacturers limitations
o verify capability of exercising ontrol valves at significant load nd evaluate function of valves and ontrols	Normal trouble free operation
o verify reliable steady state full ower capability	100 hours reliable equilibrium plant operation at full power

13.4 OPERATING RESTRICTIONS

13.4.1 SAFETY PRECAUTIONS

Measurements and test operations during zero power and power escalation phase are always performed under several active trip functions. Any verification program will be concluded by several trip functions if the program attempts to violate any of the criteria of the protective circuitry. Furthermore, to insure that transients are concluded early in the life of the transient, several of the setpoints of the trip functions are reduced, as referenced in Section 7.

Measurements are made at various points in the power escalation program as power level is increased. Considerations are made of the instrument accuracy and extrapolations are made for these parameters before proceeding in the program, including both instrument inaccuracies and uncertainties. A continuing verification is then made that the reactor parameters are no more limiting then those assumed in the accident analysis, which are the most limiting values.

Each power step is relatively small, so that a high degree of certainty is associated with the prediction of plant parameters. The accuracy of the prediction obtained for each power level is a major factor in determining further power escalation.

The reactor protection system ensures that the public safety is further protected, as stated above.

13.4.2 INITIAL OPERATION RESPONSIBILITIES

Ultimate responsibility of the facility rests with the holder of the Operating License. During the transition from a construction oriented job to a commercial power producing plant, equipment and systems are tested to prove their capability in accordance with design criteria.

Test procedures are written and approved by both Westinghouse Electric Corporation and The Consolidated Edison Company of New York, Inc., prior to plant testing. Post core load test procedures are prepared by Westinghouse and are reviewed prior to performance by Consolidated Edison through the Nuclear Facility Safety Committee. Pertinent safety comments from the committee will be factored into the procedures prior to performance. Westinghouse Electric Corporation will provide technical direction for these tests; however, all tests and test procedures shall be under the control of the General Superintendent of the plant to ensure that proper emphasis is placed on safety by all during these acceptance tests.

He shall ensure that each test has been reviewed by all responsible parties, that initial plant conditions and prerequisites the test have been met, and that proper personnel are available and understand the test procedures and precautions.

As part of the precautions, all licensed Senior Reactor Operators and manufacturer's representatives whose equipment is being tested will be instructed to stop a test or a portion of a test if the test is not being performed safely or in accordance with the written test procedures. The test shall be promptly continued if only minor modification to the test procedure is required and is approved by the General Superintendent or his representative and the Westinghouse Electric Corporation representative. If substantial revision is required, however, the General Superintendent shall review the change with the same approach as that taken with a new test procedure before the test can be continued.

Detailed discussion of the startup organization and responsibilities are given in the response to Question 13.4. in Volume 5 of the FSAR.

APPENDIX 13A

GENERAL PROCEDURES AND RESPONSIBILITIES FOR INITIAL TESTS AND OPERATION

Prior to full power operation of Indian Point Unit No. 2, the plant will undergo a thorough, systematic testing program which successively demonstrates the capability and safety of the plant to proceed to each following stage of testing until full power is achieved and maintained. WEDCO, a wholly owned subsidiary of Westinghouse, has the overall responsibility for engineering, construction management, and initial startup testing. The initial startup tests are subdivided into several stages, each to be completed before the next stage is undertaken. Following the initial startup and testing program, periodic system and plant performance tests will be performed as described in the Technical Specifications.

A. Tests prior to initial reactor fueling

The first stage of the initial tests is a comprehensive testing program which ensures that equipment and systems perform in accordance with design criteria prior to fuel loading. As the installation of individual components and systems is completed, they are tested and evaluated according to predetermined and approved written testing techniques, procedures, or check-off lists. Field and engineering analyses of test results are made to verify that systems and components are performing satisfactorily and to recommend corrective action, if necessary.

The program includes tests, adjustments, calibrations, and system operations necessary to assure that initial fuel loading and subsequent power operation can be safely undertaken. In general, the types of tests are classified as installation, flushing, hydrostatic, hot functional, and preoperational tests. These tests are aimed at verifying that the system or equipment is capable of performing the function for which it is designed.

Where practical, preoperational tests involve actual operation of the system and equipment under design or simulated design conditions. In addition, the reactor protection and safeguards instrumentation system will be performance tested prior to core loading.

The reactor coolant system vibration testing program will overlap the plant testing program. Data for this particular program will be taken during cold hydro and hot functional testing prior to fuel loading and also during the low power physics tests which follow fuel loading.

B. <u>Core loading</u>

Fuel loading does not begin until the prerequisite system tests and operations as defined in the detailed core loading procedures are satisfactorily completed and the facility operating license is obtained. Upon completion of fuel loading, the reactor upper internals and pressure vessel head are installed and additional mechanical and electrical tests are performed. The purpose of these activities is to prepare the system for nuclear operation and to establish that all design requirements necessary for operation are achieved.

The overall responsibility and direction for initial core loading is exercised by the General Superintendent assisted by the Unit No. 2 Acting Superintendent. During the initial core loading operation the WEDCO Refueling Manager is in charge of the Westinghouse activities. The process of initial core loading is, in general, directed from the operating floor of the containment structure. Standard procedures for the control of personnel and the maintenance of containment security are established prior to fuel loading. The core configuration is specified as part of the core design studies conducted well in advance of station startup and as such is not subject to change at startup. The core is assembled in the reactor vessel, submerged in water containing sufficient quantities of boric acid to maintain the fully loaded core substantially subcritical.

The core loading procedure documents include a detailed tubular check sheet which prescribe and verify the successive movements of each fuel assembly and its specified inserts from its initial position in the storage racks to its final position in the core. Multiple checks are made of component serial numbers and types at successive transfer points to guard against possible inadvertent exchanges or substitutions of components. The results of each loading step are evaluated by the applicant's licensed senior reactor operator and the WEDCO Refueling Manager before the next prescribed step is started.

Core loading procedures prevent inadvertent dilution of the boron in the reactor coolant, restrict the movement of fuel to preclude the possibility of mechanical damage, prescribe the conditions under which loading can proceed, identify chains of responsibility and authority and provide for continuous and complete fuel and core component accountability.

C. Post core loading tests

Upon completion of core loading, the reactor upper internals and the pressure head are installed. An operational leak test is conducted after filling of the reactor coolant system and final reactor vessel head stud tensioning is completed.

Mechanical and electrical tests are performed on the control rod drive mechanisms. These tests include a complete operational checkout of the mechanisms. In addition, tests are performed on the reactor trip circuits to test manual trip operation and, by use of dummy signals, the reactor control and protection system is made to produce trip signals for the various unit abnormalities that require tripping. Finally, a complete functional electrical and mechanical check is made of the movable nuclear detector system (cold shutdown).

D. Initial testing in the operating reactor

After satisfactory completion of fuel loading and final precritical tests, nuclear operation of the reactor is initiated. This final stage of startup and testing includes initial criticality, low power testing and power level e ation. The purpose of these tests is to establish the operational characteristics of the unit and core, to verify design prediction, to demonstrate that license requirements are being met, and to ensure that the next prescribed step in the test sequence can be safely undertaken. Reactor Control set point verification will also be performed during this stage of startup testing.

(1) Initial criticality

Initial criticality is established by sequentially withdrawing the shutdown and control groups of control rod assemblies from the core and then slowly diluting the heavily borated reactor coolant until the chain reaction is self-sustaining.

Written procedures specify alignment of fluid systems to allow controlled start and stop and adjustment of the rate at which the approach to criticality can proceed, indicate values of core conditions under which criticality is expected, specify allowed deviations in expected values, and identify reactor operation responsibilities.

(2) Low power testing

A prescribed program of reactor physics measurements is undertaken to verify that the basic static and kinetic characteristics of the core are as expected and that the values of the kinetic coefficients assumed in the safeguards analysis are conservative.

The measurements are made at low power and primarily at or near operating temperature and pressure. Measurements include verificatic of control rod assembly group reactivity worths, of isothermal temperature coefficient under various core conditions, of differential boron concentration reactivity worth and of critical boron concentrations as functions of control rod assembly group configuration are made. In addition measurements of the relative power distributions are made. Concurrent tests are conducted on the instrumentation including the source and intermediate range nuclear channels.

Detailed procedures are prepared to specify the sequence of tests and measurements to be conducted and conditions under which each is to be conducted and conditions which each is to be performed to ensure both safety of operation and the relevancy and consistency of the results obtained. If significant deviations from design predictions exist, unacceptable behavior is revealed, or apparent anomalies develop, the testing is suspended and the situation reviewed by the Applicant and the WEDCO/Westinghouse technical advisors to determine whether a question of safety is involved, and what corrective action must be taken, prior to resumption of testing. The ultimate responsibility for these determinations rests with Con Edison.

(3) Power level escalation

When the operating characteristics of the reactor and unit are verified by the preliminary zero power tests, a program of power level escalation in successive stages brings the unit to its full-rated power level. Both reactor and unit operational characteristics are closely examined at each stage and the safeguards analysis assumptions verified before escalation to the next programmed level is effected.

Additional reactor physics measurements are made and the ability of the reactor control and protection system to respond effectively to signals from primary and secondary instrumentation under a variety of conditions encountered in normal operations is verified. At prescribed power levels the dynamic response characteristics of the reactor coolant and the steam systems are evaluated.

The sequence of tests, measurements and intervening operations is prescribed in the power escalation procedures together with specific details relating to the conduct of the several tests and measurements. The measurement and test operations during power escalation are similar to those during normal operation.

E. Initial operation responsibilities

A Joint Test Group consisting of responsible WEDCO and Con Edison personnel review and concur in release of test procedures for implementation. Technical responsibility for each individual phase of actual startup resides with the functional group most directly concerned with the results of the test. WEDCO and Westinghouse have on-site representatives of supporting functional groups to provide technical advice, recommendations and assistance in planning and executing the respective stages of unit startup. Specific responsibilities during each stage of testing are discussed in preceding respective sections.

All system operations in the testing program are performed by station operators in accordance with the approved written procedures. These procedures include such items as delineation of administrative procedures and test responsibilities, equipment clearance procedures, test purpose, conditions, precautions, limitations, and sequence of operations. Procedural changes are made only in accordance with an approved standard operating procedure that requires review and approval of the changes by experienced supervisory personnel.

Test procedures stating the test purpose, conditions, precautions, limitations and criteria for acceptance are preapred for each test by WEDCO and/or Westinghouse technical advisors. All such procedures are reviewed and concurred in by the Joint Test Group in accordance with approved standard operating procedures prior to implementation.

All test results will receive a preliminary review and evaluation by Con Edison site personnel. Cognizant WEDCO/Westinghouse startup engineers and technical advisors will determine the adequacy of test data for verification of design objectives. Detailed analyses of test results and issuance of final test reports will be performed by WEDCO site startup and/or Westinghouse engineering and design personnel with input from Con Edison where appropriate. Con Edison will review all final test results to determine that design objectives and criteria have been met and will give final approval as to the acceptability of plant components, systems and operating characteristics of the facility.

Chap 14

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14.1	Analysis of expected transients with failure to scram. The transients analyzed are: (1) loss-of-load (2) excessive loading increase (3) loss of one feedwater pump (4) loss of one primary coolant pump (5) rod withdrawal accident (6) startup accident (7) station blackout (8) startup of inactive primary coolant pump (9) accidental depressurization of the reactor coolant	14.7
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	Plots of time-to-signal versus break size for the following three safety injections: Coincidence of low pressurizer pressure and level, high containment pressure and high-high containment pressure.	
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14 SAFETY ANALYSIS

This section evaluates the safety aspects of the plant and demonstrates that the plant can be operated safely and that exposures from credible accidents do not exceed the guide lines of 10 CFR 100.

This section is divided into three subsections, each dealing with a different behavior category:

Core and Coolant Boundary Protection Analysis, Section 14.1

The incidents presented in Section 14.1 have no off-site radiation consequences.

Standby Safeguards Analysis, Section 14.2

The accidents presented in Section 14.2 are more severe and may cause release of radioactive material to the environment.

Rupture of a Reactor Coolant Pipe, Section 14.3

The accident presented in Section 14.3, the rupture of a reactor coolant pipe, is the worst case accident and is the primary basis for the design of engineered safety features. It is shown that even this accident meets by a wide margin the guide lines of 10 CFR 100.

Parameters and assumptions that are common to various accident analysis are described below to avoid repetition in subsequent sections.

Steady State Errors

For accident evaluation, the initial conditions are obtained by adding maximum steady state errors to rated values. The following steady state errors are considered:

Power	2% of full power (2758 MWt) which is 55.2 for calorimetric error
Core Inlet Temperature	4°F for deadband and measurement error
Primary Pressure	30 psi for steady state fluctuations and measurement error

Initial values for power, primary pressure, and core inlet temperature are selected to minimize the initial W-3 DNB ratio. A maximum design inlet temperature of 543°F is assumed for all accidents initiated from full power. This adequately covers operation at lower inlet temperatures dependent upon steam generator heat transfer performance with a lower limit set by maintaining a non-positive moderator coefficient of reactivity.

Hot Channel Factors

Unless otherwise stated in the sections describing specific accidents, the hot channel factors used are:

F_q (heat flux hot channel factor) = 3.23
$F_{\Delta H}$ (enthalpy rise hot channel factor) = 1.77

The in-core instrumentation system will be employed to verify that actual hot channel factors are, in fact, no higher than those used in the accident analyses.

Reactor Trip

A reactor trip signal acts to open the two series trip breakers feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the control rods, which then fall by gravity into the core. There are various instrumentation delays associated with each tripping function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. The time delay assumed for each tripping function is as follows:

Tripping Function	Time Delay (Seconds)
Overpower (nuclear)	0.5
Overtemperature ΔT	2.30
Overpower Temperature ΔT	2.30
High pressurizer pressure	1.0
Low pressurizer pressure	1.0
High pressurizer pressure	1.0
Low reactor Coolant flow (from loop flow (detectors)	0.6
Turbine Trip	1.0
Low-Low Feedwater Flow (SG Level)	1.0

The negative reactivity insertion following a reactor trip is a function of the acceleration of the control rods and the variation in rod worth as a function of rod position. Control rod positions after trip have been determined experimentally as a function of time using an actual prototype assembly under simulated flow conditions. The resulting rod positions were combined with rod worths to define the negative reactivity insertion as a function of time, according to Figure 14-1.

The trip levels used in the following analysis are maximum values including the trip setpoint and the error allowance. The trip setpoints are established in Section 15, Technical Specifications.

The maximum nuclear overpower trip point assumed for the analysis is 118%. The trips will be calibrated at power such that the calibration error is the calorimetric error of 2%. The design allowance for non-repeatable errors is 6%. Non-repeatable errors include both instrument drift and errors due to process changes such as control rod motion since both are observable as an error between the indicated signal and the known power from calorimetric measurements. In summary the trip setpoints are less than the trip value assumed in the analysis to ensure that trip occurs within the assumed value when including the design error allowance.

REACTIVITY INSERTION VERSUS TIME FOR REACTOR TRIP, HOT ZERO POWER CONDITIONS, 105 PERCENT REACTOR COOLANT FLOW

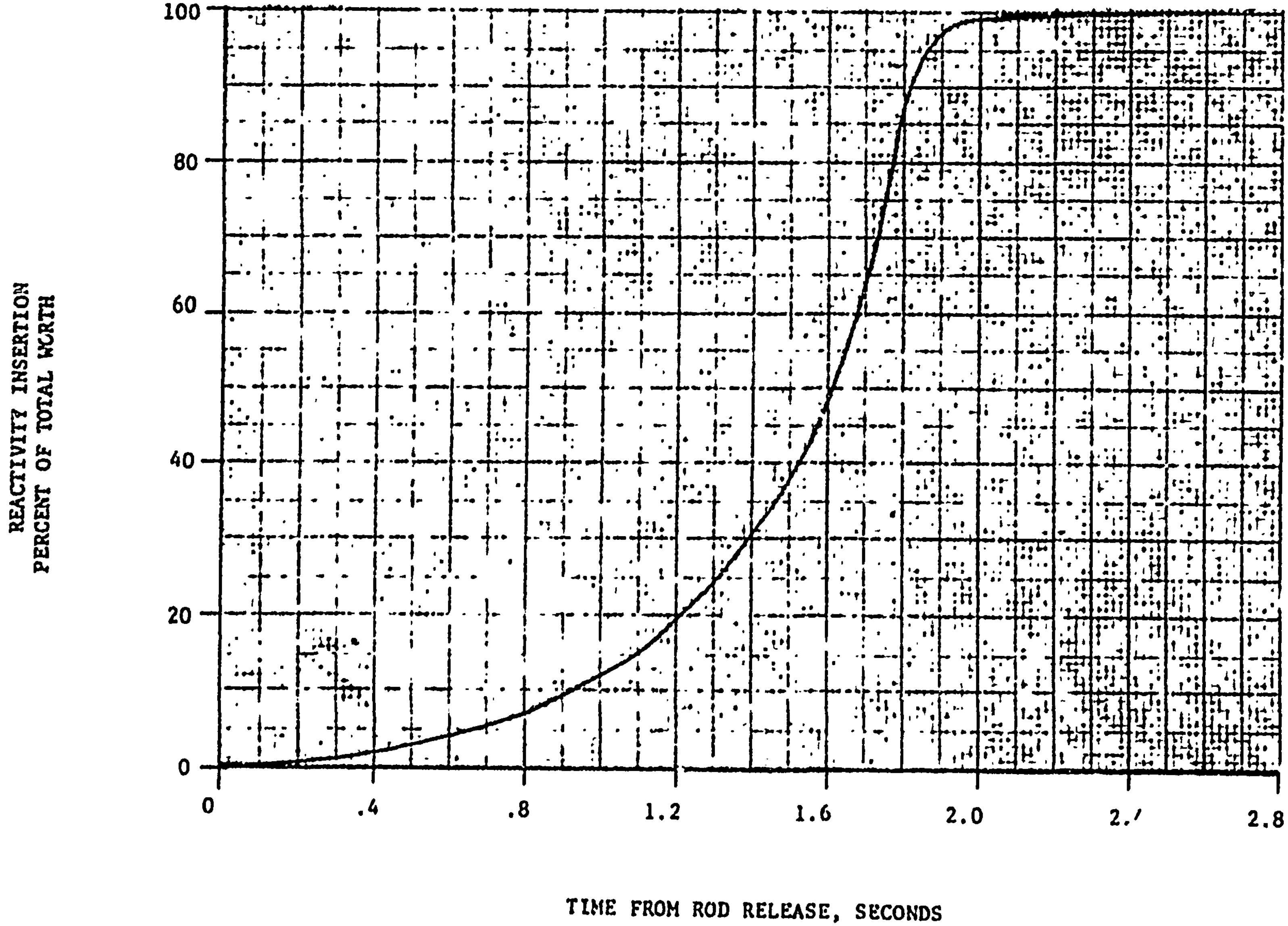


FIGURE 14-1

14.1 CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

For the following plant abnormalities and transients, the reactor control and protection system is relied upon to protect the core and reactor coolant boundary from damage:

a) Uncontrolled RCCA withdrawal from a Subcritical Condition
b) Uncontrolled RCCA withdrawal, at power
c) Malpositioning of the Parth Length Rods
d) Rod Cluster Control Assembly (RCCA) Drop
e) Chemical and Volume Control System (CVCS) Malfunction
f) Loss of Reactor Coolant Flow
g) Startup of an Inactive Reactor Coolant Loop
h) Loss of External Electrical Load
i) Loss of Normal Feedwater
j) Excess Feedwater Incident
k) Excessive Load Increase Incident
l) Loss of All Normal A-C Power to the Station Auxiliaries
m) Likelihood and Consequences of Turbine Generator Overspeed

' reactor protection criteria are met presupposing the most reactive RCC assembly is in its fully withdrawn position with the part length rods in their lowest permissible worth position. Trip is defined for analytical purposes as the insertion of all full length RCC assemblies except the most reactive assembly which is assumed to remain in the fully withdrawn position. This is to provide margin in shutdown capability against the remote possibility of a stuck RCC assembly condition existing at a time when shutdown is required.

Instrumentation is provided for continuously monitoring all individual RCC assemblies together with their respective group position. This is in the form of a deviation alarm system. If the rod should deviate from its intended position, the reactor would then be shut down in an orderly manner and the condition corrected. Such occurrences are expected to be extremely rare based on operation and test experience to date.

In summary, reactor protection is designed to prevent cladding damage in all transients and abnormalities listed above. The most probable modes of failure in each protection channel result in a signal calling for the protective trip. Coincidence of two out of three (or two out of four) signals is required where single channel malfunction could cause spurious trips while at power. A single component or channel failure in the protection system itself coincident with one stuck RCCA is always permissible as a contingent failure and does not cause violation of the protection criteria. The reactor protection systems are designed in accordance with the IEEE "Standard for Nuclear Plant Protection Systems."

14.1.1 UNCONTROLLED RCCA WITHDRAWAL FROM A SUBCRITICAL CONDITION

An RCCA withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of rod cluster control assemblies resulting in a power excursion. While the probability of a transient of this type is extremely low, such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This could occur with the reactor either subcritical or at power, as discussed in Section 14.1.2.

Reactivity is added at a prescribed and controlled rate in bringing the reactor from a shutdown condition to a low power level during startup by RCCA withdrawal. Although the initial startup procedure uses the method of boron dilution, the normal startup is with RCCA withdrawal. RCCA motion can cause much faster changes in reactivity than can be made by changing boron concentration.

The rod cluster drive mechanisms are wired into preselected groups, and these group configurations are not altered during core life. The rods are therefore physically prevented from withdrawing in other than their respective groups. Power supplied to the rod groups is controlled such that no more than two groups can be withdrawn at any time. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming the simultaneous withdrawal of the combination of the two rod groups with the maximum combined worth at maximum speed, which is well within the capability of the protection system to prevent core damage.

Should a continuous RCCA withdrawal be initiated and assuming the source and intermediate range alarms and indications are ignored, the transient will be terminated by the following automatic Safety Features.

a. Source range flux level trip - actuated when either of two independent source range channels indicates a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed when either intermediate range flux channel indicates a flux level above the source range cutoff power level. It is automatically reinstated when both intermediate range channels indicate a flux level below the source range cutoff power level.

b. Intermediate range rod stop - actuated when either of two independent intermediate range channels indicates a flux level above a preselected, manually adjustable value. This rod stop may be manually bypassed when two out of the four power range channels indicate a power level above approximately ten per cent power. It is automatically reinstated when three of the four power range channels are below this value.

c. Intermediate range flux level trip - actuated when either of two independent intermediate range channels indicates a flux level above a preselected, manually adjustable value. This trip function is manually bypassed when two of the four power range channels are reading above approximately ten per cent power and is automatically reinstated when three of the four channels indicate a power level below this value.

d. Power range flux level trip (low setting) - actuated when two out of the four power range channels indicate a power level above approximately 25 per cent. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately ten per cent power and is automatically reinstated when three of the four channels indicate a power level below this value.

e. Power range flux level trip (high setting) - actuated when two out of the four power range channels indicate a power level above a preset setpoint. This trip function is always active.

The nuclear power response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative fuel temperature coefficient. This self-limitation of the initial power burst results from a sizable prompt negative fuel temperature coefficient (Doppler effect) and is of prime importance during a startup accident since it limits the power to a tolerable level prior to external control action. After the initial power burst, the nuclear power is momentarily reduced and then if the accident is not terminated by a reactor trip, the nuclear power increases again, but at a much slower rate.

Termination of the startup accident by the above protection channels prevents core damage. In addition, the high pressure reactor trip serves as a backup to terminate the accident before an overpressure condition could occur.

Method of Analysis

Analysis of this transient is performed by digital computation incorporating the neutron kinetics, including six delayed groups, and the core thermal and hydraulic equations. In addition to the nuclear flux response, the average fuel, clad and water temperature, and also the heat flux response, are computed.

In order to give conservative results for a startup accident, the following additional assumptions are made concerning the initial reactor conditions;

a. Since the magnitude of the nuclear power peak reached during the initial part of the transient, for any given rate of reactivity insertion, is strongly dependent on the fuel temperature reactivity coefficient, the least negative design value is used for the startup accident.

b. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the nuclear flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. Accordingly, a conservatively large positive value is used, since this yeilds the maximum rate of power increase.

c) The reactor is assumed to be at hot zero power. This is more conservative than cold zero power since the higher initial temperature causes a higher overall heat transfer coefficient; a smaller (less negative) Doppler coefficient; and an increased thermal capacity of the fuel. Initial multiplication (k_o) is assumed to be 1.0 since this results in the maximum nuclear power peak. The values of the temperature reactivity coefficients are shown on Figure 14.1.1.2. The fuel heat capacity was 0.07 btu/lb-°F. The total fuel to water heat transfer coefficient was the full power value, 2020 btu/sec-°F. Rated coolant flow was assumed.

d. The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and rod release, are taken into account. Also, the rate of negative reactivity insertion corresponding to the trip action is based on the assumption that the highest worth rod is stuck in its fully withdrawn position.

Results

Figure 14.1.1-1 shows the sensitivity of peak heat flux to initial power level for various reactivity insertion rates. It shows that peak heat flux initially decreases with increasing initial power level and then, depending on the rate, it increases again being asymptotic to 25 per cent (reactor trip is assumed to be initiated at the value). However, for the faster insertion rates, which result in the greatest energy addition, the flux peak is greatest for the lowest initial power level.

Figures 14.1.1-2 through 14.1.1-4 show the transient behavior for a reactivity insertion rate of 8 x 10^{-4} δk/sec with the accident terminated by reactor trip at 25 per cent power. This insertion rate is greater than that for the two highest worth banks both assumed to be in their highest incremental worth region. Figure 14.1.1-2 shows the nuclear power increase. The power is seen to increase to the trip point in 7.6 seconds.

The nuclear power overshoots to approximately 970 per cent, but this occurs for only a very short time period. Hence, the energy release and the fuel temperature increases are small. The thermal flux response, of interest for DNB considerations, is shown on Figure 14.1.1-3. The beneficial effect of the inherent thermal lag of the fuel is evidenced by a peak heat flux of only 44.8 per cent of the nominal value. There is a large margin to DNB during the transient since the rod surface heat flux remains below the design value, and there is a high degree of subcooling at all times in the core. Figure 14.1.1-4 shows the response of the average fuel, cladding and coolant temperatures. The fuel temperature increases to 981°F. The average coolant temperature increases only to 559°F.

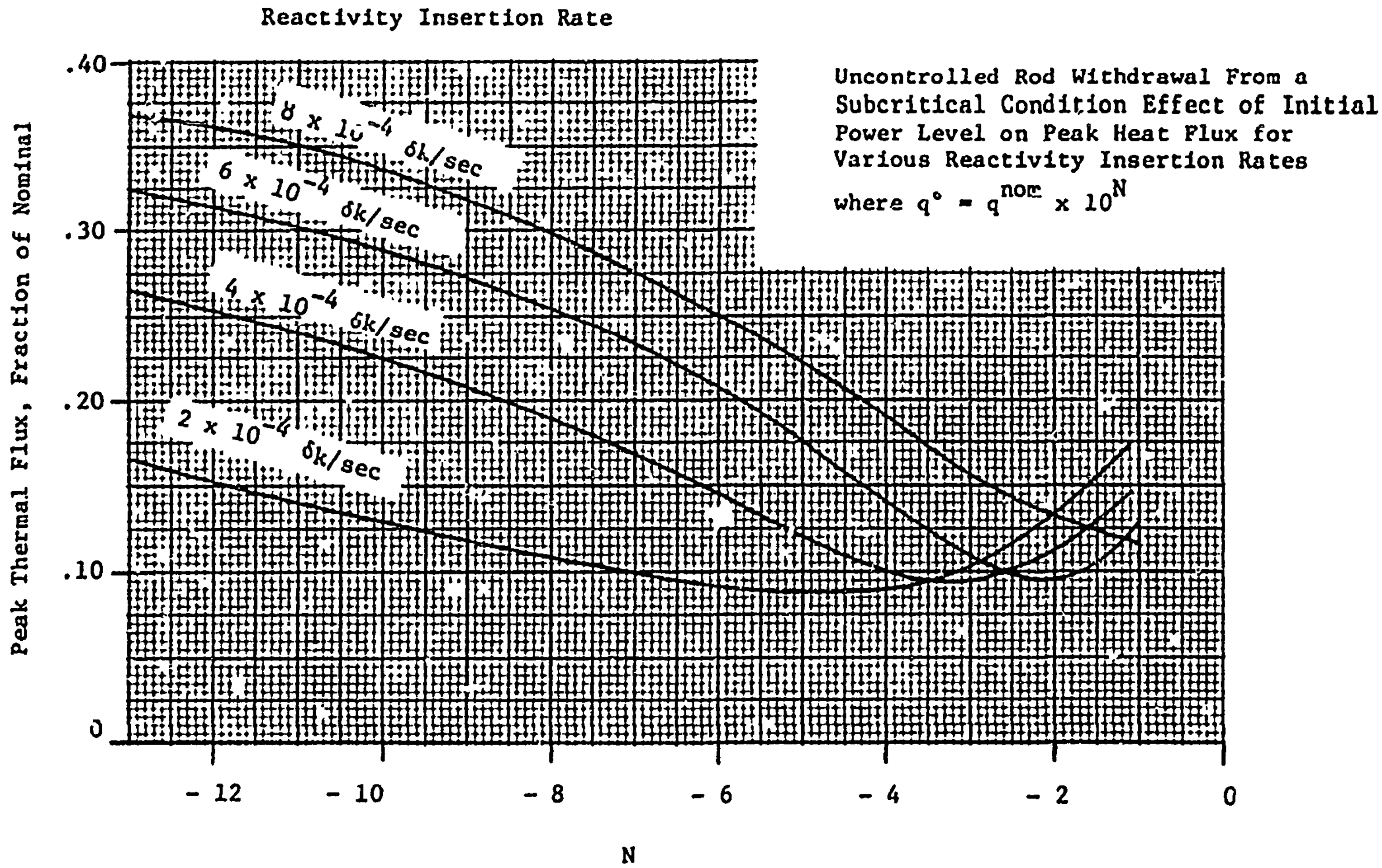


FIGURE 14.1.1-1

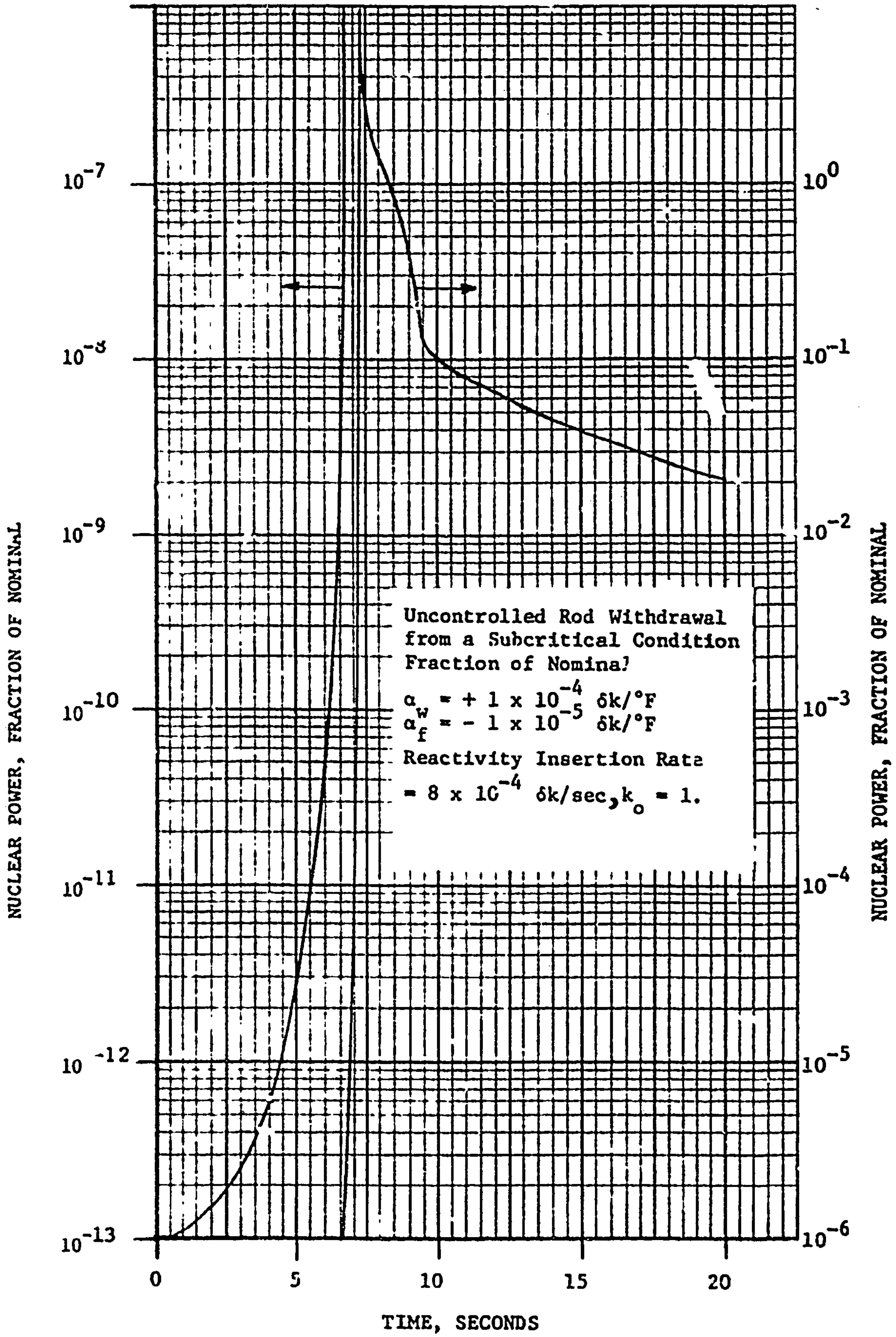


FIGURE 14.1.1-2

Uncontrolled Rod Withdrawal
From a Subcritical Condition
Fraction of Thermal Flux

$\alpha_w = + 1 \times 10^{-4}$ δk/°F
$\alpha_f = - 1 \times 10^{-5}$ δk/°F
Reactivity Insertion Rate = 8×10^{-4}
δk/sec, k_o = 1.

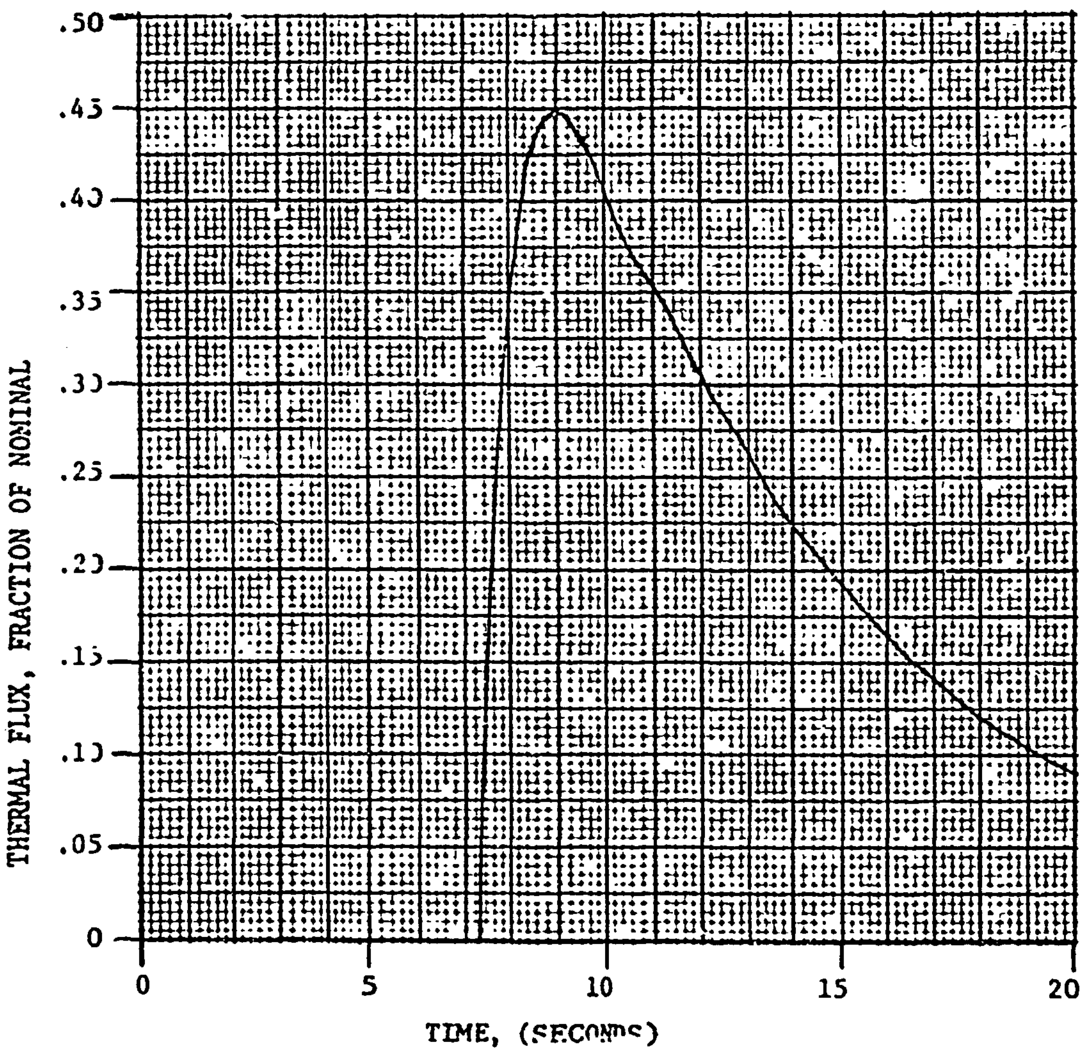


FIGURE 14.1.1-3

Uncontrolled Rod Withdrawal
From a Subcritical Condition
Temperatures

$\alpha_w = +1 \times 10^{-4}\ \delta k/°F$
$\alpha_f = -1 \times 10^{-5}\ \delta k/°F$

Reactivity Insertion Rate = 8×10^{-4} $\delta k/sec.$

$k_o = 1.$

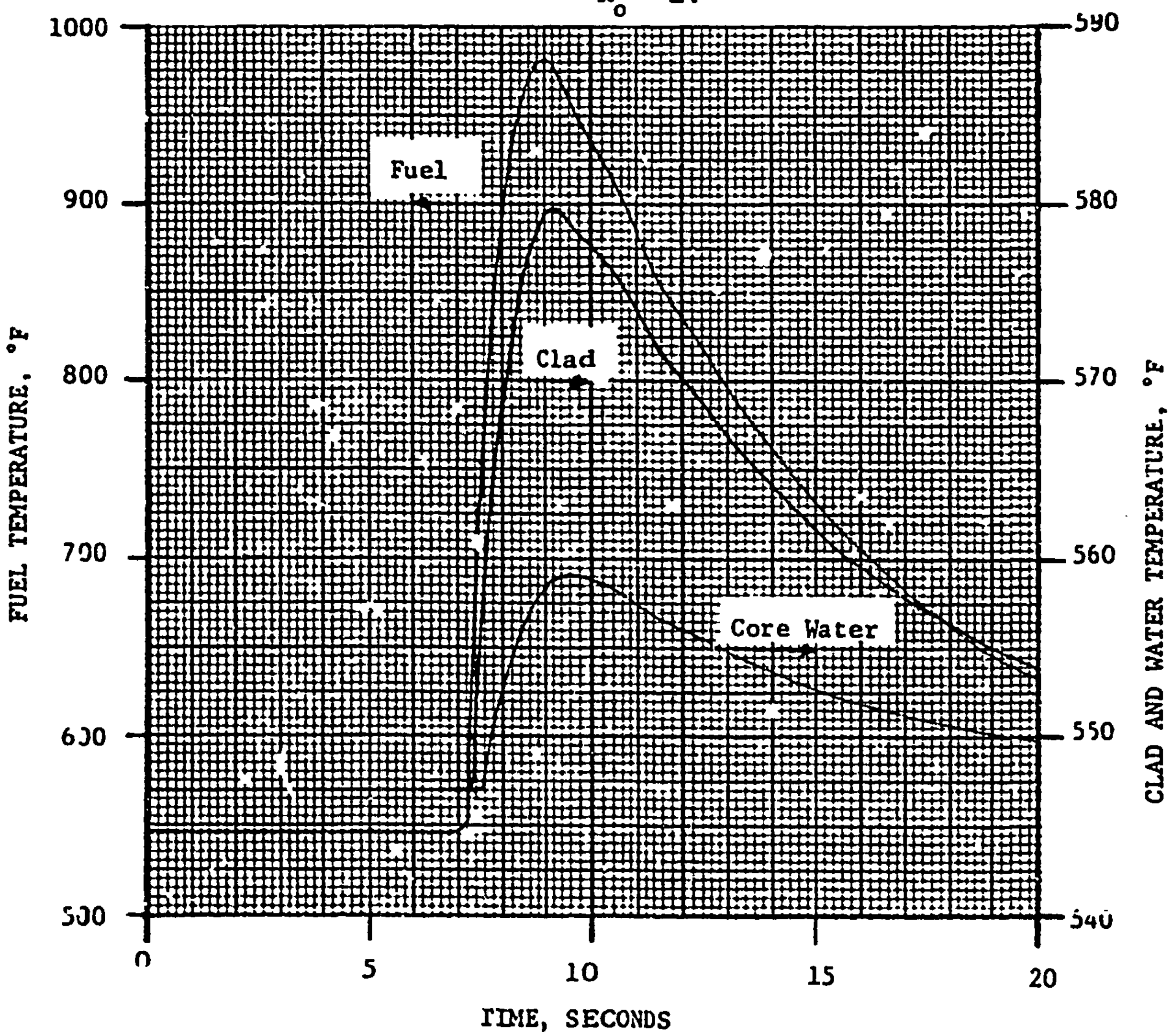


FIGURE 14.1.1-4

14.1.2 UNCONTROLLED RCCA WITHDRAWAL AT POWER

An uncontrolled RCCA withdrawal at power results in an increase in core heat flux. Since the heat extraction from the steam generator remains constant, there is a net increase in reactor coolant temperature. Unless terminated by manual or automatic action, this power mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, to prevent the possibility of damage to the cladding, the Reactor Protection System is designed to terminate any such transient with an adequate margin to DNB.

The automatic features of the Reactor Protection System which prevent core damage in a rod withdrawal accident at power include the following:

a. Nuclear power range instrumentation actuates a reactor trip if two out of the four channels exceed an overpower setpoint.

b. Reactor trip is actuated if any two out of four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with power distribution, temperature and pressure to protect against DNB.

c. Reactor trip is actuated if any two out of four ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with power distribution to ensure that the allowable fuel power rating is not exceeded.

d. A high pressure reactor trip, actuated from any two out of three pressure channels, is set at a fixed point. This set pressure will be less than the set pressure for the pressurizer safety valves.

e. A high pressurizer water level reactor trip, actuated from any two out of three level channels, is actuated at a fixed setpoint. This affords additional protection for RCCA withdrawal accidents.

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of reactivity insertion rates is illustrated in Section 7. Figure 7.2-11 represents the possible conditions of reactor vessel average temperature and ΔT with the design power distribution in a two-dimensional plot. The boundaries of operation defined by the overpower ΔT and the overtemperature ΔT trip are represented as "protection lines" on this diagram - the over ΔT trip as a horizontal line at the maximum value for an overpower ΔT trip and the overtemperature ΔT trip as a family of sloping lines whose temperature intercepts increase with increasing pressure. The protection lines are drawn to include all adverse instrumentation and setpoint errors, so that under nominal conditions trip would occur well within the area bounded by these lines. The nominal full power operating condition for the reactor is shown as point + on the figure.

The utility of the diagram just described is in the fact that the operating limit imposed by a given DNB ratio can be represented as a line on this coordinate system. The DNB lines represent the locus of conditions for which the DNBR equals 1.3. All points below and to the left of this line have a DNB ratio greater than this value. The diagram shows that DNB is prevented for all cases if the area enclosed within the maximum protection lines is not traversed by the applicable DNB ratio line at any point.

Method of Analysis

The region of permissible operation (power, pressure and temperature) is completely bounded by the combination of reactor trips: nuclear overpower (fixed setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints). These trips are designed to preclude a DNB ratio of less than 1.30.

The purpose of this analysis is to demonstrate the manner in which these protective systems function for various reactivity insertion rates from different initial conditions. Reactivity coefficients, initial conditions and effects of control functions govern which protective function occurs first.

Analysis is performed using several digital computer techniques. First, the actual core limits are determined employing the W-3 DNB correlation described in Section 3. These limits are shown on Figure 7.2-3. Protection lines, illustrated in Figure 7.2-11, are then selected and incorporated in a transient analysis by a detailed digital simulation of the unit. The detailed digital simulation consists of neutron kinetics, core thermal and hydraulic equations, primary loop hydraulic equations, including pressurizer, and a detailed representation of the steam generator, both primary and secondary, including the effect of heat transfer between the two regions.

In the analysis, the effect of the RCCA movement on core power distribution is considered in its effect of causing a decrease in overtemperature ΔT and overpower ΔT trip setpoints proportionate to the decrease in margin to DNB. This has the effect of causing a reactor trip sooner in the transient.

Results

Figures 14.1.2-1 and 14.1.2-2 show the response of nuclear power, average coolant temperature, pressure, and DNB ratio to a rapid rod withdrawal accident starting from full load. Initial conditions assumed maximum power and temperature errors. Nominal reactivity coefficients for beginning of core life were assumed because there is less moderator feedback for this coefficient. Reactor trip on high nuclear power conservatively assumed to be actuated at 118% of nominal occurs in approximately 3 seconds. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result. A large margin to DNB is maintained, the minimum DNB ratio being approximately 1.72.

The response of nuclear power, average coolant temperature, pressure, and DNB ratio for a slow rod withdrawal from full power is shown in Figures 14.1.2-3 and 14.1.2-4. Reactor trip or overtemperature ΔT trip occurs at approximately 150 seconds. The rise in temperature is quite large. Pressure rises to the relief valve setpoint, but relief valve capacity is not exceeded, even if only one of the two valves open. The minimum DNB ratio is 1.34.

Starting from full power, there is no reactivity insertion rate which will exceed the capacity of the two power-operated relief valves. Hence, peak pressure is limited by the relief valve set pressure. Figure 14.1.2-5. shows the minimum DNB ratio as a function of reactivity insertion rate from initial full power operation. Figure 7.2-11 implied a DNB ratio of 1.30 at all points at which the overtemperature ΔT trip setpoint is reached. However, the transient calculation shows that the minimum DNB ratio is asymptotic to a value of 1.30 and will reach that value only for very small reactivity insertion rates. This is due to the thermal lag between nuclear power and heat flux out of the fuel rod. Note that for reactivity insertion rates smaller than approximately 4.5×10^{-6} δk/sec., the high pressurizer water level trip will be actuated before the overtemperature ΔT trip causing an increase in the minimum DNB ratio.

Figure 14.1.2-6 shows the minimum DNB ratio as a function of reactivity insertion rate for rod withdrawal accidents starting from 80 per cent power. It is interesting to note that the overtemperature ΔT trip will be actuated only for a small range of reactivity insertion rates. Fast rates will actuate the overpower trip first. While the high pressurizer water level reactor trip will be the first one actuated for slow rates of reactivity insertion.

Figure 14.1.2-7 shows the minimum DNB ratio as a function of reactivity insertion rate for rod withdrawal accidents starting from 60 per cent power. Note that the high pressurizer level trip encompasses a larger range than at 80 per cent power. The net result is that for rod withdrawal accidents occurring below 60 per cent power, the range of reactivity insertion rates encompassed by the high pressurizer level trip will increase until the overtemperature ΔT trip will not be actuated and the minimum design DNB ratio of 1.3 will not be reached.

Conclusions

In the unlikely event of a control rod withdrawal incident, whether it be from subcritical condition, from full power operation, or at any other power level between these two extremes, the core and reactor coolant system are not adversely affected. Protection is provided by the nuclear overpower reactor trips, and the overtemperature ΔT trip, as well as by the overpower ΔT trip, the fixed high and low pressure trips and high pressurizer level trips. The preceding subsections have described the effectiveness of this protection.

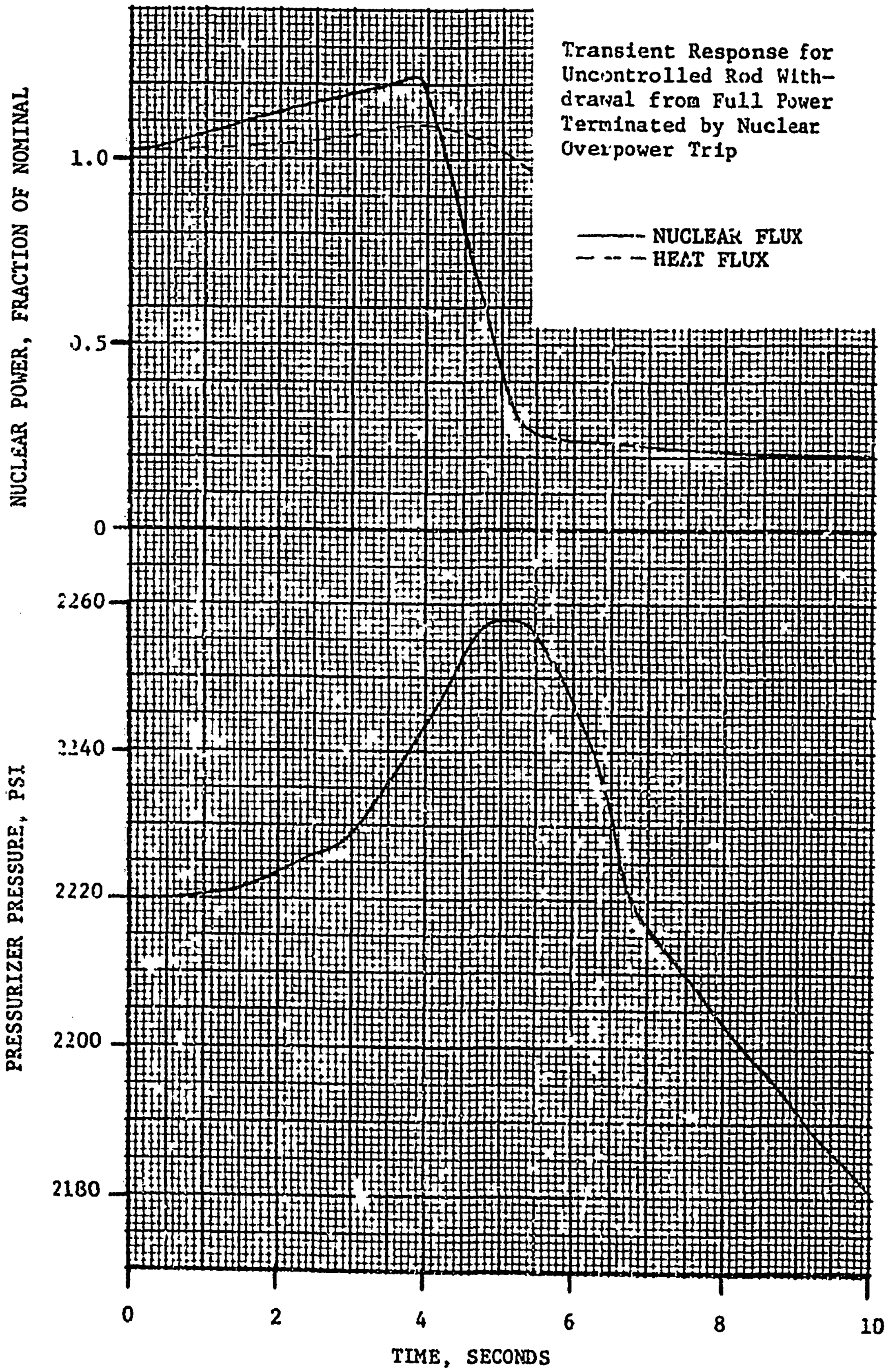


FIGURE 14.1.2-1

Transient Response for Uncontrolled
Rod Withdrawal from Full Power Terminated
by Nuclear Overpower Trip

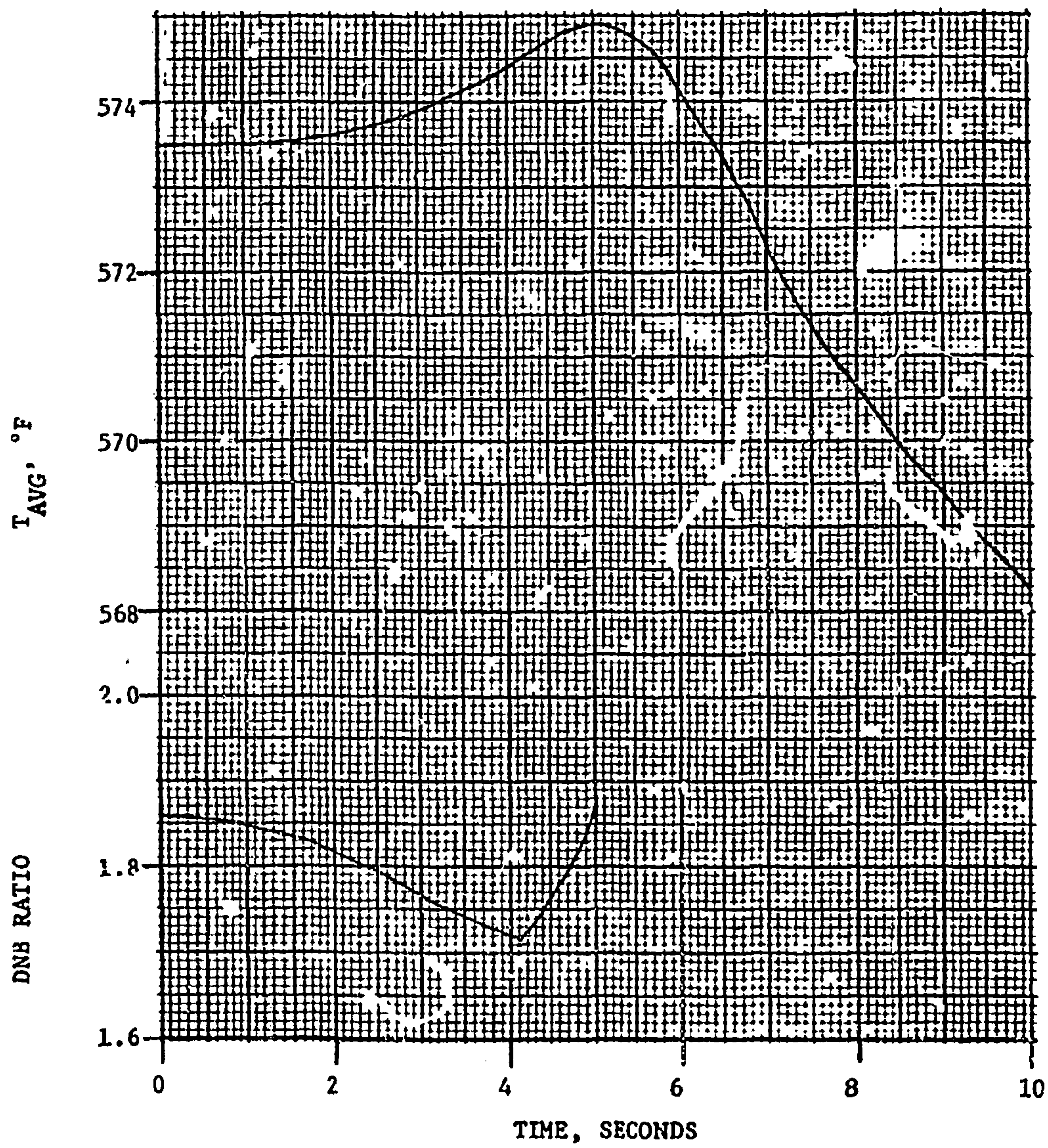
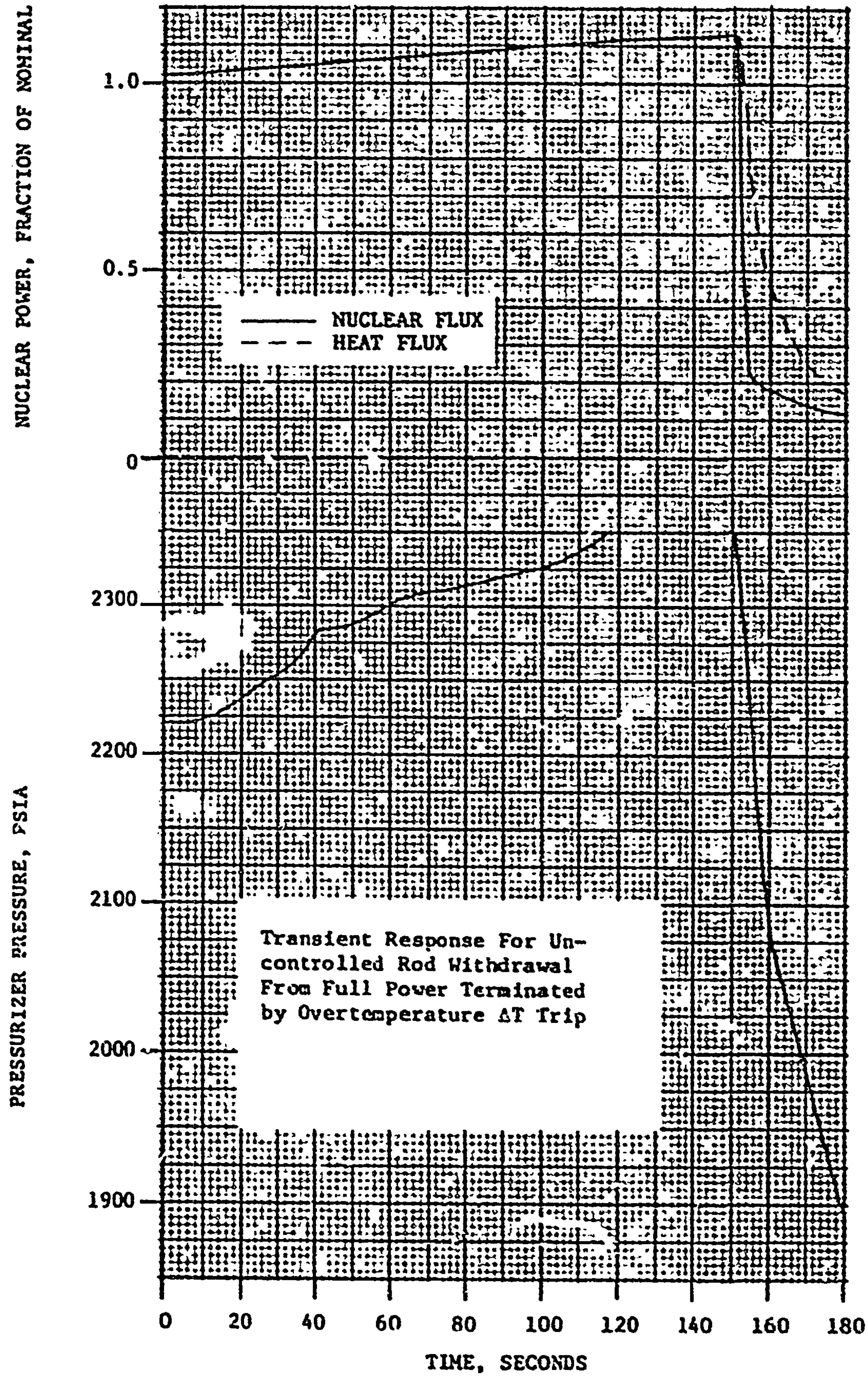


FIGURE 14.1.2-2



Transient Response For Uncontrolled Rod Withdrawal From Full Power Terminated by Overtemperature ΔT Trip

FIGURE 14.1.2-3

Transient Response For Uncontrolled
Rod Withdrawal From Full Power
Terminated by Overtemperature ΔT Trip

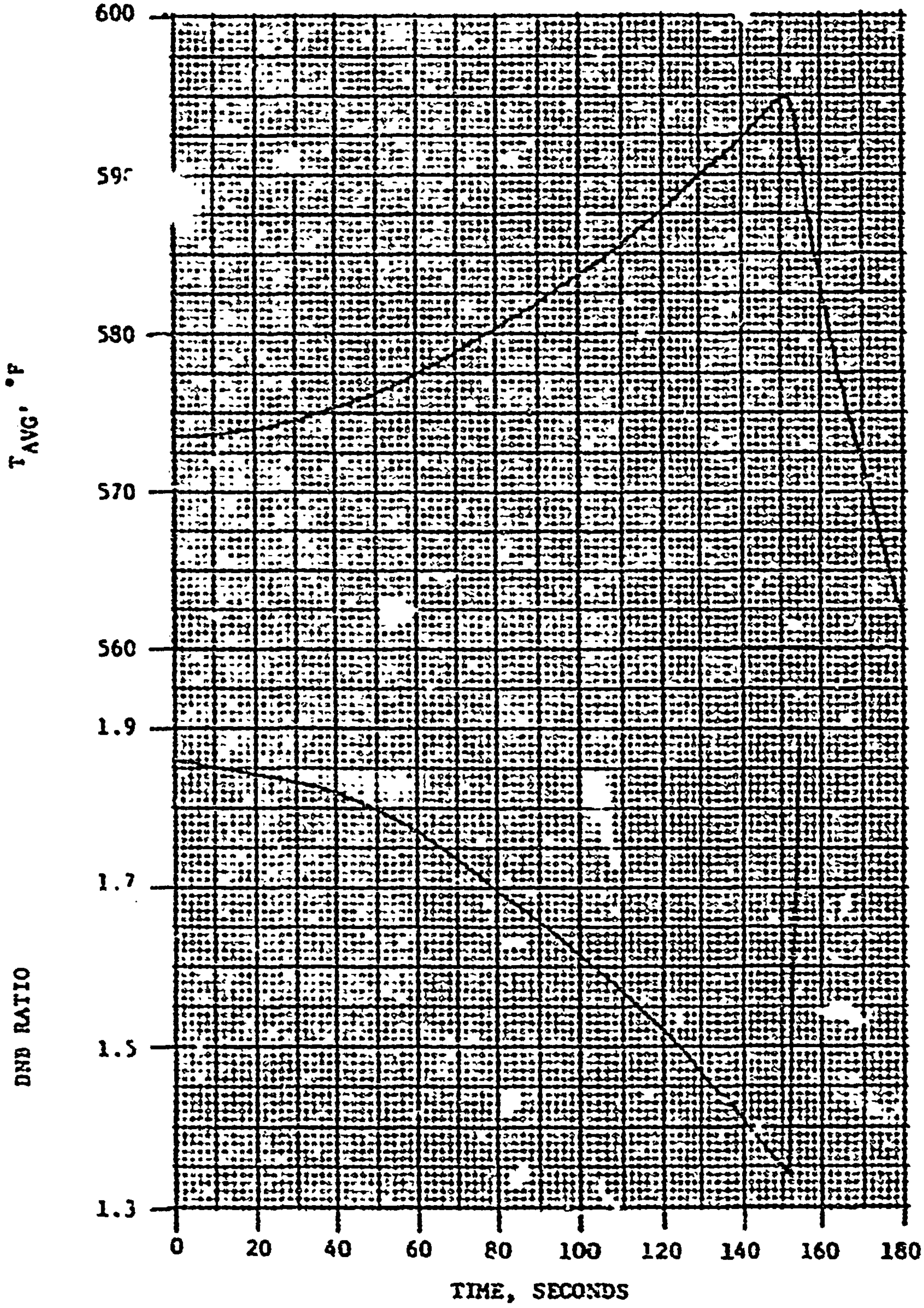


FIGURE 14.1.2-4

Effect of Reactivity Insertion Rate on Minimum DNB Ratio for A Rod Withdrawal Accident From Full Power Conditions

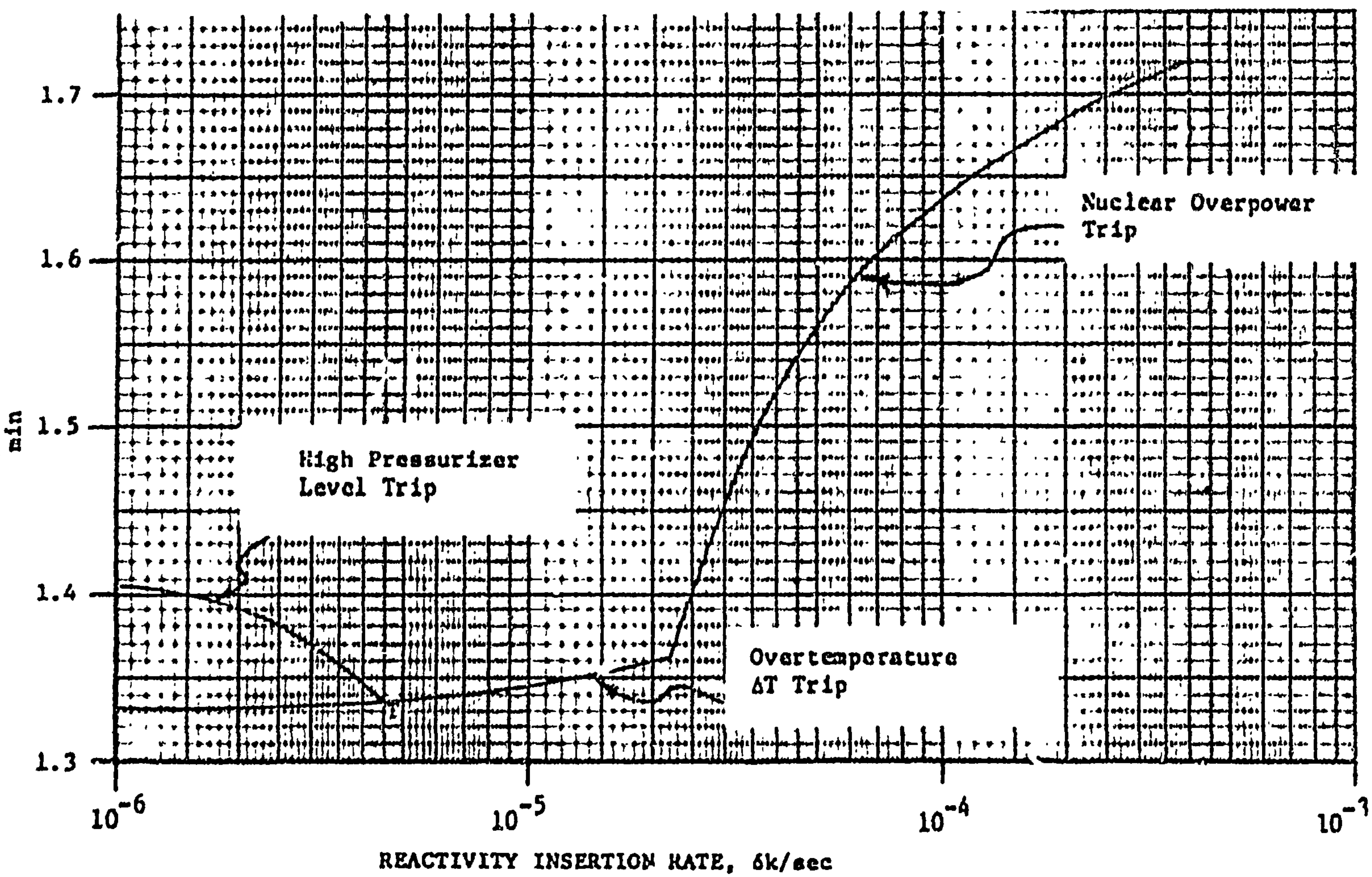


FIGURE 14.1.2-5

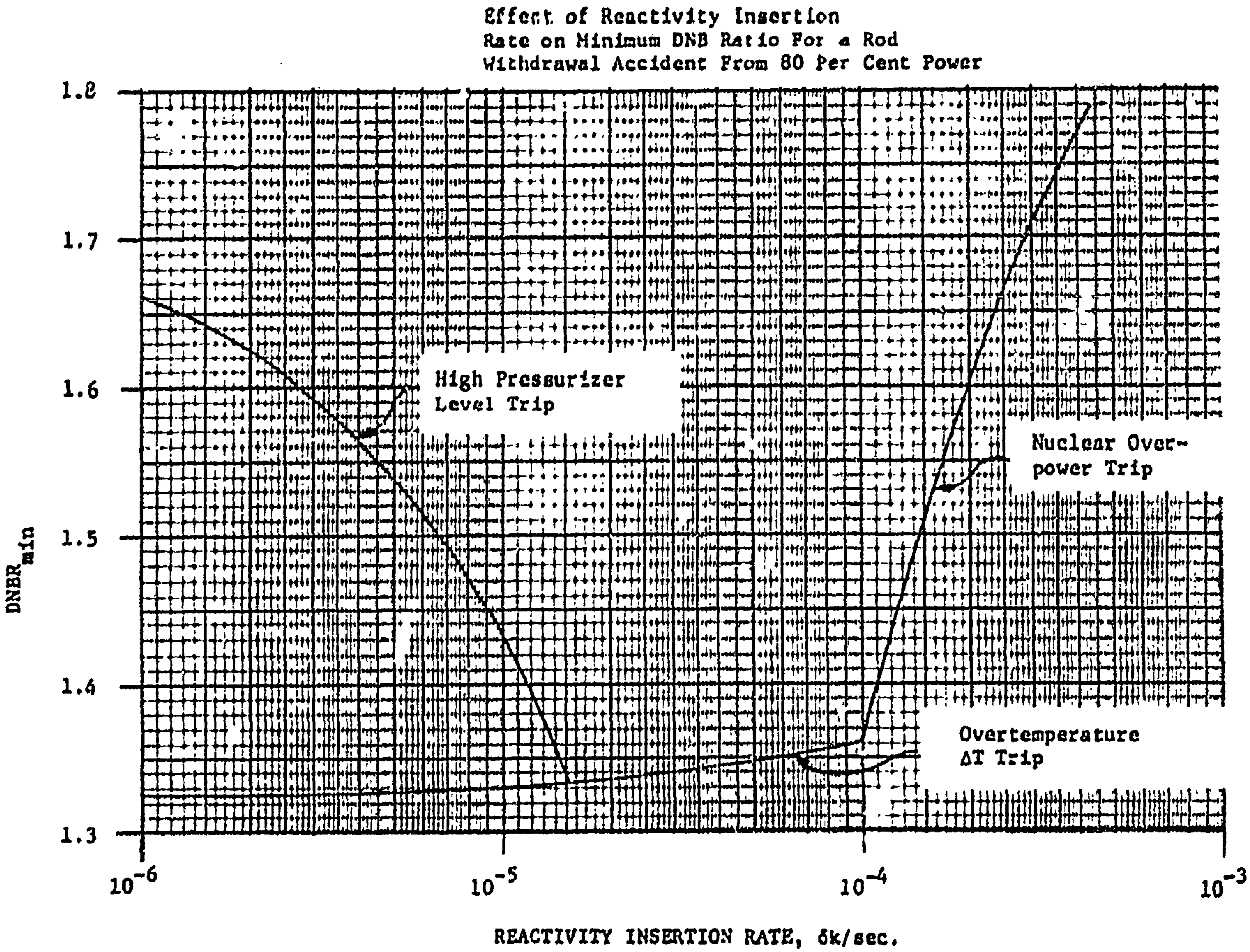


FIGURE 14.1.2-6

Effect of Reactivity Insertion Rate on Minimum
DNB Ratio for a Rod Withdrawal Accident from
60 Per Cent Power

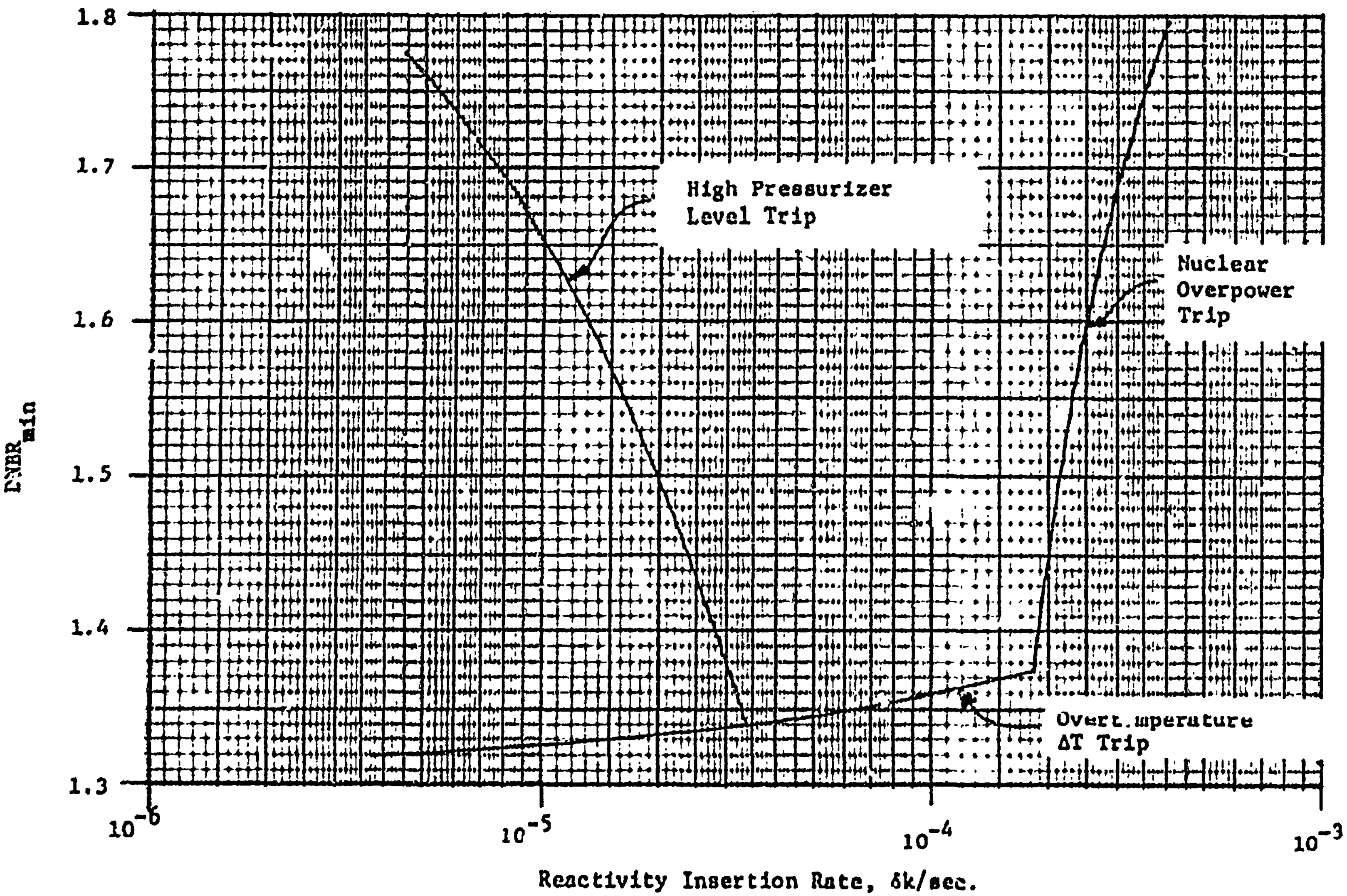


FIGURE 14.1.2-7

14.1.3 MALPOSITIONING OF THE PART LENGTH RODS

As discussed in the previous sections, part length rods are employed to improve the axial power distributions as well as to control potential axial xenon oscillation.

When the plant is operating at steady state, motion of the part length rods is only necessary at intervals of several hours. When major changes in load are made some administrative control is performed to maintain the part length rods within a prescribed allowable region of travel.

The instrumentation system provides adequate information for the manual control of the part length rods. Certain administrative procedures are established, including:

a. The part length rods are not moved into the upper part of the core above an administratively imposed upper travel limit. This upper limit is a function of the control bank location.

b. The part length rods are moved periodically (every 3-5 hours) to damp out xenon oscillations. A time period of the order of 24 hours without control would be required for axial oscillations to become serious.

The axial distribution is continuously monitored by the upper and lower sections of out-of-core ion chambers. Out-of-limit signals are generated should the relative readings differ by a preset amount. A reduction in power capability might be necessary until the normal power distribution is restored. The reactor protection system automatically resets the overpower and overtemperature trip at a lower level consistent with the existing power distribution. At the same time, if warranted, the control system automatically initiates a turbine load cutback to prevent unnecessary trips.

14.1.4 ROD CLUSTER CONTROL ASSEMBLY (RCCA) DROP

Full Length RCCA

Dropping of an RCCA could occur from de-energizing a drive mechanism. It would result in a power reduction and a possible increase in the hot channel factor. If no protective action occurred, the Reactor Control System would restore the power to the level which existed before the incident occurred. This would lead to a reduced safety margin or possibly DNB, depending upon the magnitude of the hot channel factor.

If a RCCA should drop into the core during power operation, this would be detected by the rod bottom signal device, which provides an individual position indication signal for each RCCA. Initiation of this signal is independent of lattice location, reactivity worth or power distribution changes inherent with the dropped RCCA. Further indication of an RCCA drop would be obtained by an independent means, using the out of core power range channel signals. This power range rod drop detection circuit is actuated upon sensing a rapid decrease in local flux such as could occur from depression of flux in one region by a dropped RCCA. This detection circuit is designed such that normal load variation does not cause it to be actuated.

A rod drop signal from any rod position indication channel, or from one or more of the four power range channels, initiates protective action by reducing turbine load by a preset adjustable amount and blocking of further automatic rod withdrawal. Either action individually prevents core damage. The turbine runback is redundantly obtained by acting upon the turbine load limit and on the turbine governor control system. The rod stop is also redundantly actuated. Rod drop protection is discussed in Section 7.2

Method of Analysis

The transient following a dropped RCCA accident is determined by a detailed digital simulation of the plant. The dropped rod is assumed to cause a step decrease in reactivity and the core power generation is determined using a point neutron kinetics model. The overall plant response is calculated by simulating the turbine load runback and blocking of automatic rod withdrawal. The analysis is performed for the case in which the load cutback nearly matches the power decrease from the negative reactivity for a dropped rod (-2.3×10^{-3} δk), and also for the case in which the load cutback is greater than that required to match the worth of the dropped rod (-1.2×10^{-3} δk). In both cases the load is assumed to be cut back from 100 to 75 per cent of full load at a conservatively slow rate of one per cent per second. The actual amount of load cutback to be used will be determined during initial startup experiments and will be set to match the power reduction caused by the highest worth dropped rod.

The most negative values of moderator and Doppler temperature coefficients of reactivity are used in this analysis resulting in the highest heat flux during the transient. These are a moderator temperature coefficient of -3.5×10^{-4} δk/°F and a Doppler coefficient of -1.65×10^{-5} δk/°F. A control group worth of 6×10^{-5} δk/in is assumed as equilibrium conditions are restored.

Results

Figures 14.1.4-1 and 14.1.4-2 illustrate the transient response following a dropped rod of -2.3×10^{-3} δk. The coolant average temperature decreases initially, then increases under the influence of the negative reactivity effect of the moderator and Doppler temperature coefficients coupled with the load cutback. Later the equilibrium temperature is restored under automatic control rod insertion. Equilibrium temperature is achieved in about six minutes. The peak heat flux following the initial response to the dropped rod is 92.5 per cent of nominal. At the same time the core inlet temperature has dropped to 545.7°F and the pressure to 2188 psia.

Figures 14.1.4-3 and 14.1.4-4 illustrate the transient response following a dropped rod of -1.2×10^{-3} δk. Again the coolant average temperature decreases initially, and then increases because of the negative reactivity feedback and the load cutback. The equilibrium temperature will again be achieved in about six minutes. For this case the peak heat flux following the initial response to the dropped rod is 96.6 per cent of nominal. At the same time the core inlet temperature has increased to 546.7°F and the pressure has decreased to 2207 psia.

An analysis has been made of the amount of flux tilt that can be tolerated without core damage for the maximum full power operating conditions (102 per cent power; core water inlet temperature of 547°F; primary pressure of 2220 psia), a more conservative condition than those mentioned above. The effect of the flux tilt was represented by an increase in the radial heat flux hot channel factor. It was found that this factor could be increased by 23 per cent before reaching a DNB ratio of 1.30. During initial startup experiments, it will be verified that the flux tilt caused by the worst dropped rod, coupled with the thermal flux, coolant temperature, and primary system pressure responses, will not result in a condition of DNB.

Conclusion

Protection for a dropped RCCA is provided by automatic turbine power cutback and blocking of automatic rod withdrawal. The magnitude of the power cutback is to be determined during the initial startup tests. As the analyses presented show, the protection system, in conjunction with the load cutback, protects the core from DNB for a power tilt of 23 per cent at maximum full power conditions, which is a greater tilt than expected for the plant. At the reduced power condition following the rod drop, this allowable tilt will be even greater.

The power tilt will be experimentally determined and the protection system set to maintain a DNBR greater than 1.30.

Part Length RCCA

The part length rod drive mechanisms are designed so that power is required to move the rods. Hence on loss of power to the mechanisms the rods remain stationary and no reactivity transient would result. These mechanisms are described in Section 3.2.3.

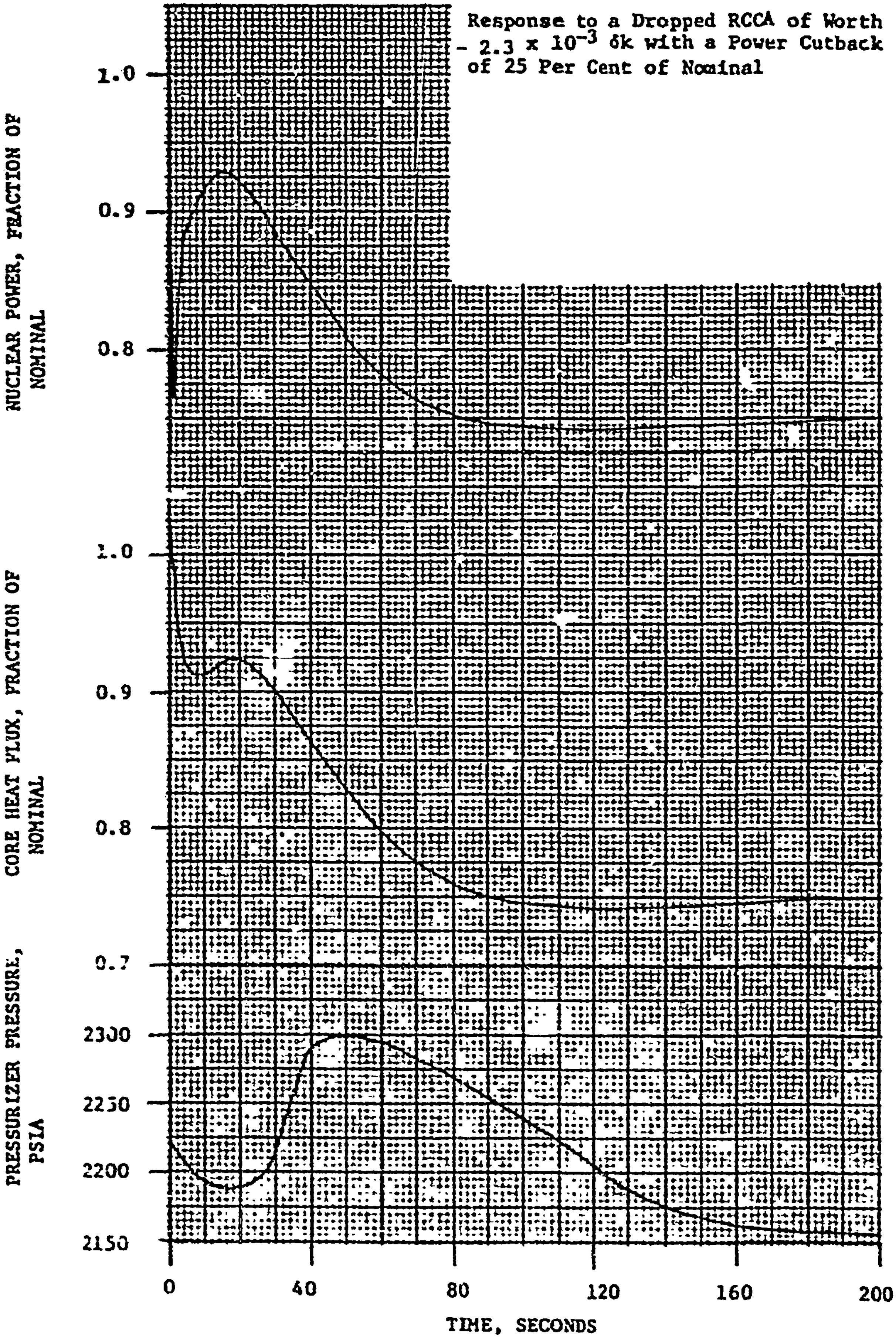


FIGURE 14.1.4-1

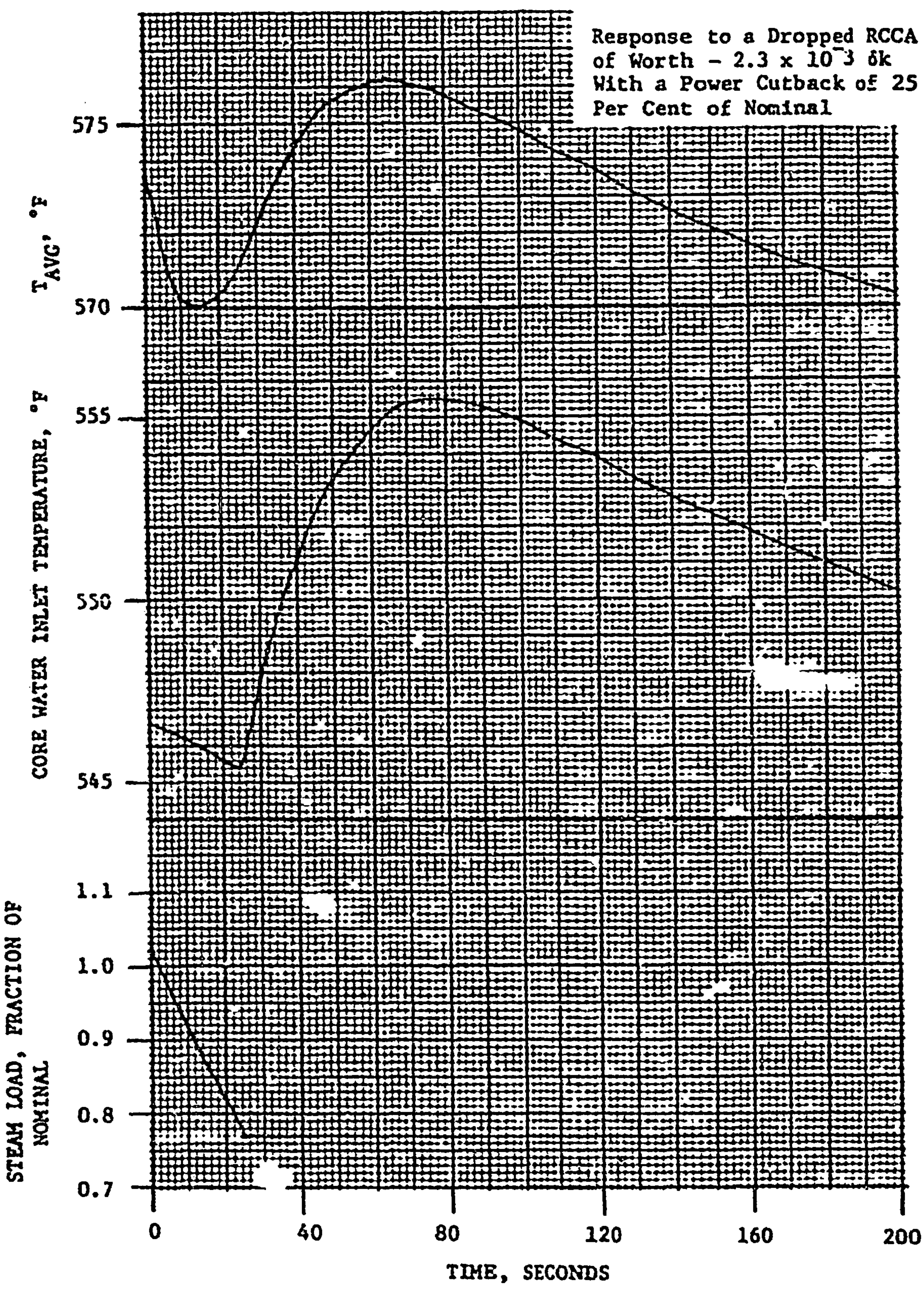


FIGURE 14.1.4-2

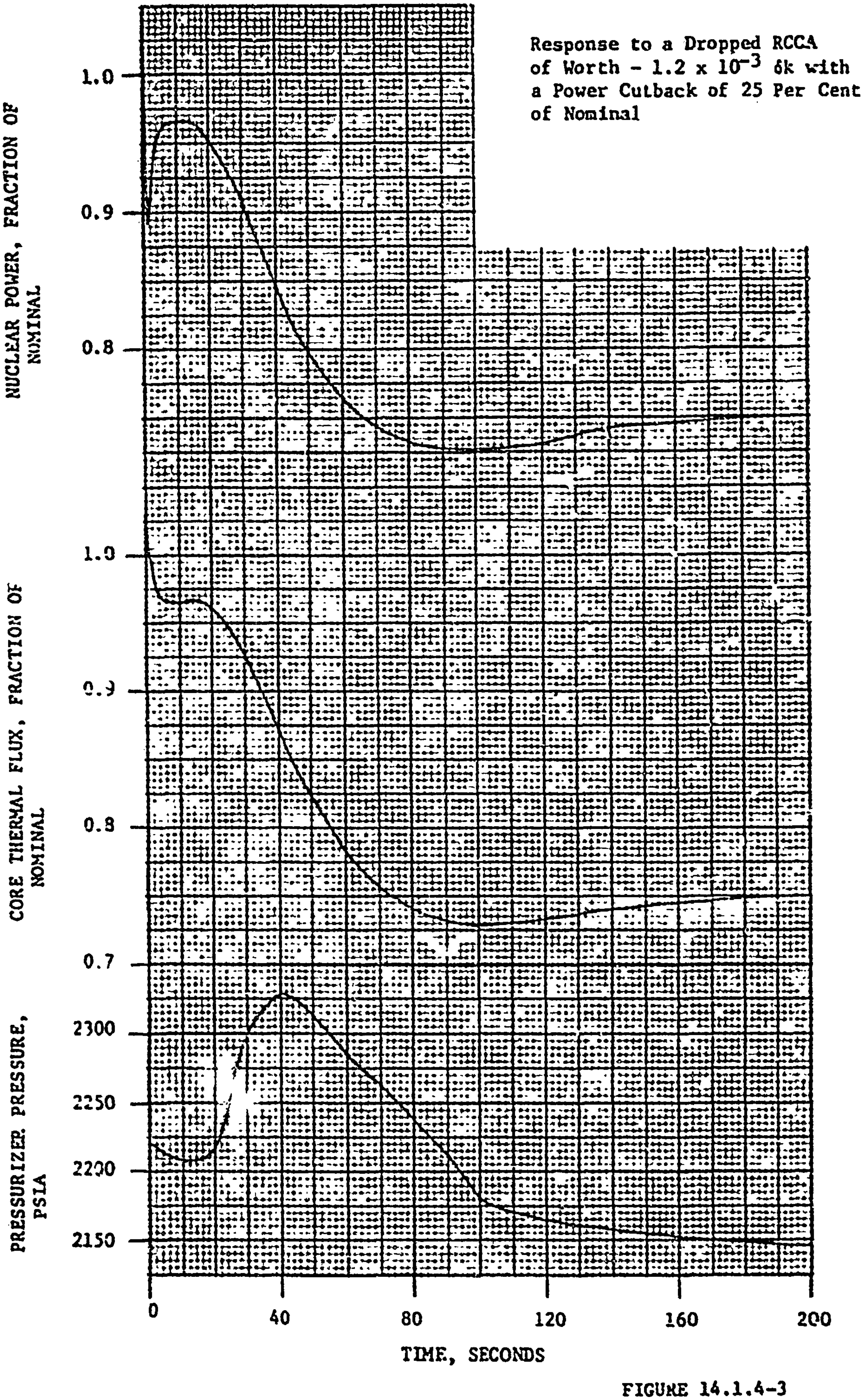


FIGURE 14.1.4-3

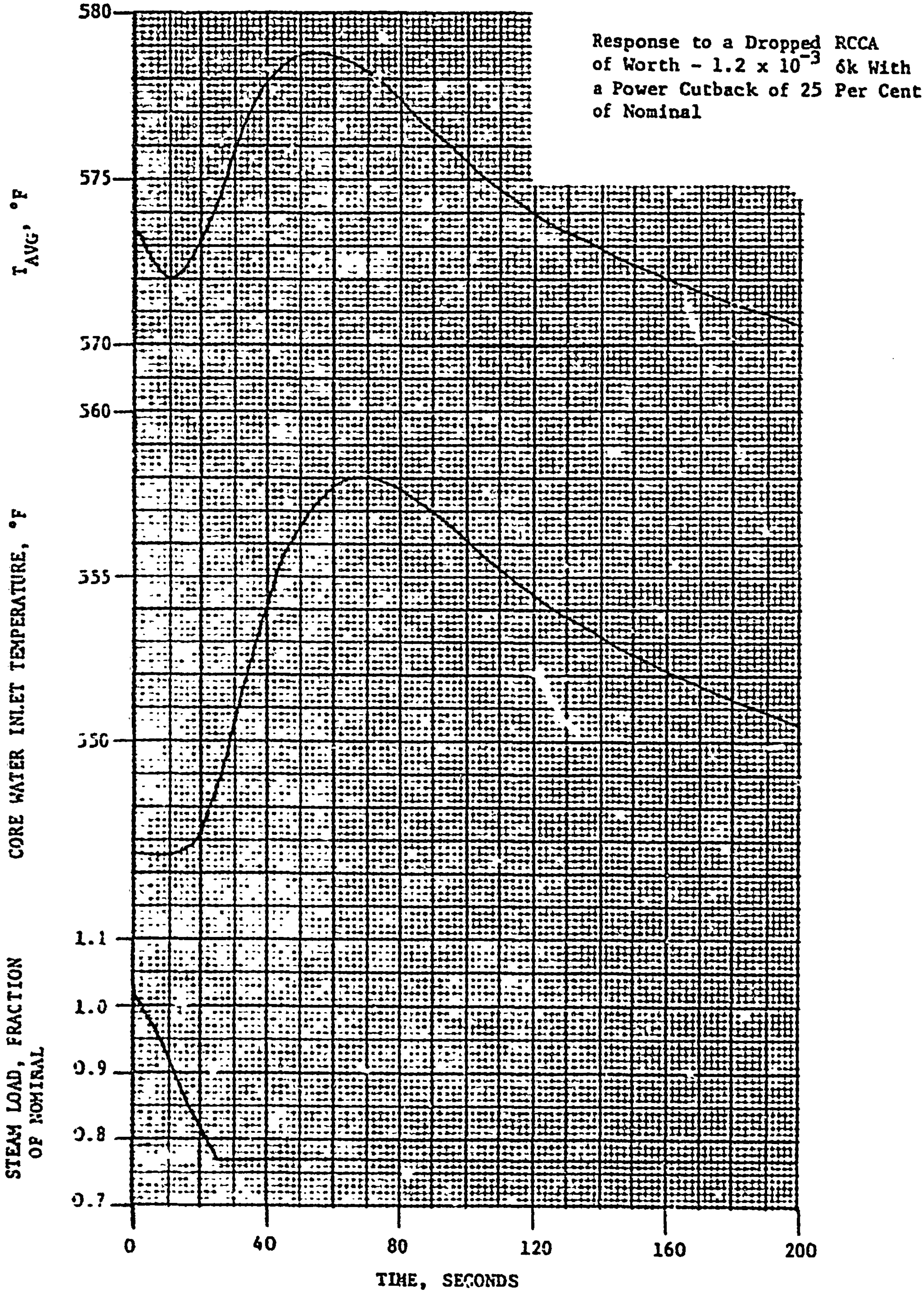


FIGURE 14.1.4-4

14.1.5 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION

Reactivity can be added to the core with the Chemical and Volume Control System by feeding reactor makeup water into the Reactor Coolant System via the reactor makeup control system. The normal dilution procedures call for a limit on the rate and magnitude for any individual dilution, under strict administrative controls. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the concentration of reactor coolant makeup water to that existing in the coolant at the time. The Chemical and Volume Control System is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

There is only a single, common source of reactor makeup water to the Reactor Coolant System from the reactor makeup water system, and inadvertent dilution can be readily terminated by isolating this single source. The operation of the reactor makeup water pumps which take suction from this tank provides the only supply of makeup water to the Reactor Coolant System. In order for makeup water to be added to the Reactor Coolant System the charging pumps must be running in addition to the reactor makeup water pumps.

The rate of addition of unborated water makeup to the Reactor Coolant System is limited to the capacity of the charging pumps. This limiting addition rate is 300 gpm for all three charging pumps. This is the maximum delivery rate based on a pressure drop calculation comparing the pump curve with the system resistance curve. Normally only one charging pump is operating while the others are on standby.

The boric acid from the boric acid tank is blended with the reactor makeup water in the blender and the composition is determined by the preset flow rates of boric acid and reactor makeup water on the Reactor Makeup Control. Two separate operations are required. First, the operator must switch from the automatic makeup mode to the dilute mode. Second, the start button must be depressed. Omitting either step would prevent dilution. This makes the possibility of inadvertent dilution very small.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of pumps in the Chemical and Volume Control System. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

To cover all phases of plant operation, boron dilution during refueling, startup, and power operation are considered in this analysis.

Method of Analysis and Results

Dilution During Refueling

During refueling the following conditions exist:

a. One residual heat removal pump is running to ensure continuous mixing in the reactor vessel,

b. The valve in the seal water header to the reactor coolant pumps is closed,

c. The valves on the suction side of the charging pumps are adjusted for addition of concentrated boric acid solution,

d. The boron concentration of the refueling water is 2500 ppm, corresponding to a shutdown of 18 per cent Δk with all control rods in; periodic sampling ensures that this concentration is maintained, and

e. Neutron sources are installed in the core and BF_3 detecters connected to instrumentation giving audible count rates are installed within the reactor vessel to provide direct monitoring of the core.

A minimum water volume in the Reactor Coolant System of 5509 ft^3 is considered. This corresponds to the volume necessary to fill the reactor vessel above the nozzles to ensure mixing via the residual heat removal loop. The maximum dilution flow of 300 gpm and uniform mixing are also considered. The flow rate of the residual heat removal loop is 3000 gpm.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High count rate is alarmed in the reactor containment and the main control room. The count rate increase is proportional to the inverse multiolication factor. At 1560 ppm, for example, the core is 4.5 per cent shutdown and the count rate is increased by a factor of four over the count rate at 2500 ppm.

The boron concentration must be reduced from 2500 ppm to approximately 1150 ppm before the reactor will go critical. This would take at least 1.6 hours. This is ample time for the operator to recognize the audible high count rate signal and isolate the reactor makeup water source by closing valves and stopping the reactor makeup water pumps.

Dilution During Startup

Prior to refueling, the Reactor Coolant System is filled with borated (2500 ppm) water from the refueling water storage tank by the charging pumos. Core monitoring is by external BF_3 detectors. Mixing of reactor coolant is maintained by operation of the reactor coolant pumos. Again the maximum dilution flow (300 gpm) is considered. The volume of reactor coolant is approximately 10,390 ft^3 which is the volume of the Reactor Coolant System excluding the pressurizer. High source level and all reactor trip alarms are effective.

The minimum time required to reduce the reactor coolant boron concentration to 1150 ppm, where the reactor could go critical with all rods in, is about 3 hours. Once again, this should be more than adequate time for operator action to the high count rate signal, and termination of dilution flow.

In any case, if continued dilution occurs, the reactivity insertion rate and consequences thereof are considerably less severe than those associated with the uncontrolled rod withdrawal analyzed in Section 14.1.1, Uncontrolled RCCA Withdrawal from a Subcritical Condition.

Dilution at Power

The effective reactivity addition rate for a boron dilution flow of 300 gpm at 569.5°F is shown as a function of reactor coolant boron concentration on Figure 14.1.5-1. This figure includes the effect of increasing boron worth with dilution. The reactivity addition rate used in this evaluation is 1.2×10^{-5} δk/sec, a conservatively high value for the expected at power boron concentration.

With the reactor in automatic control, at full power, the power and temperature increase from the boron dilution results in control group insertion and a decrease in shutdown margin. A continuation of the dilution and rod insertion would cause the rods to reach the minimum rod insertion limit in approximately six minutes. Before reaching this point, however, two alarms would be actuated to warn the operator of the accident condition. The first of these, the LO (rod position) alarm, alerts the operator to initiate normal boration. The other, LO-LO alarm, alerts the operator to follow emergency boration procedures. The LO alarm is set well above the LO-LO alarm to provide for sufficient normal boration without the need for emergency procedures.

With no boration, it takes 14 minutes before the total shutdown margin (one per cent) is lost due to dilution. Therefore, plenty of time is available following the alarms for the operator to determine the cause, isolate the reactor water makeup source, and initiate reboration.

If the reactor is in manual control, and the operator takes no action, power and temperature rise to the overtemperature ΔT trip setpoint in approximately two minutes. Prior to this the high temperature alarm would be actuated. In any case, there are approximately 14 minutes available for the operator to terminate dilution before the reactor can return to criticality following the trip.

Conclusions

Because of the procedures involved in the dilution process, an erroneous dilution is considered incredible. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and take corrective action before excessive shutdown margin is lost.

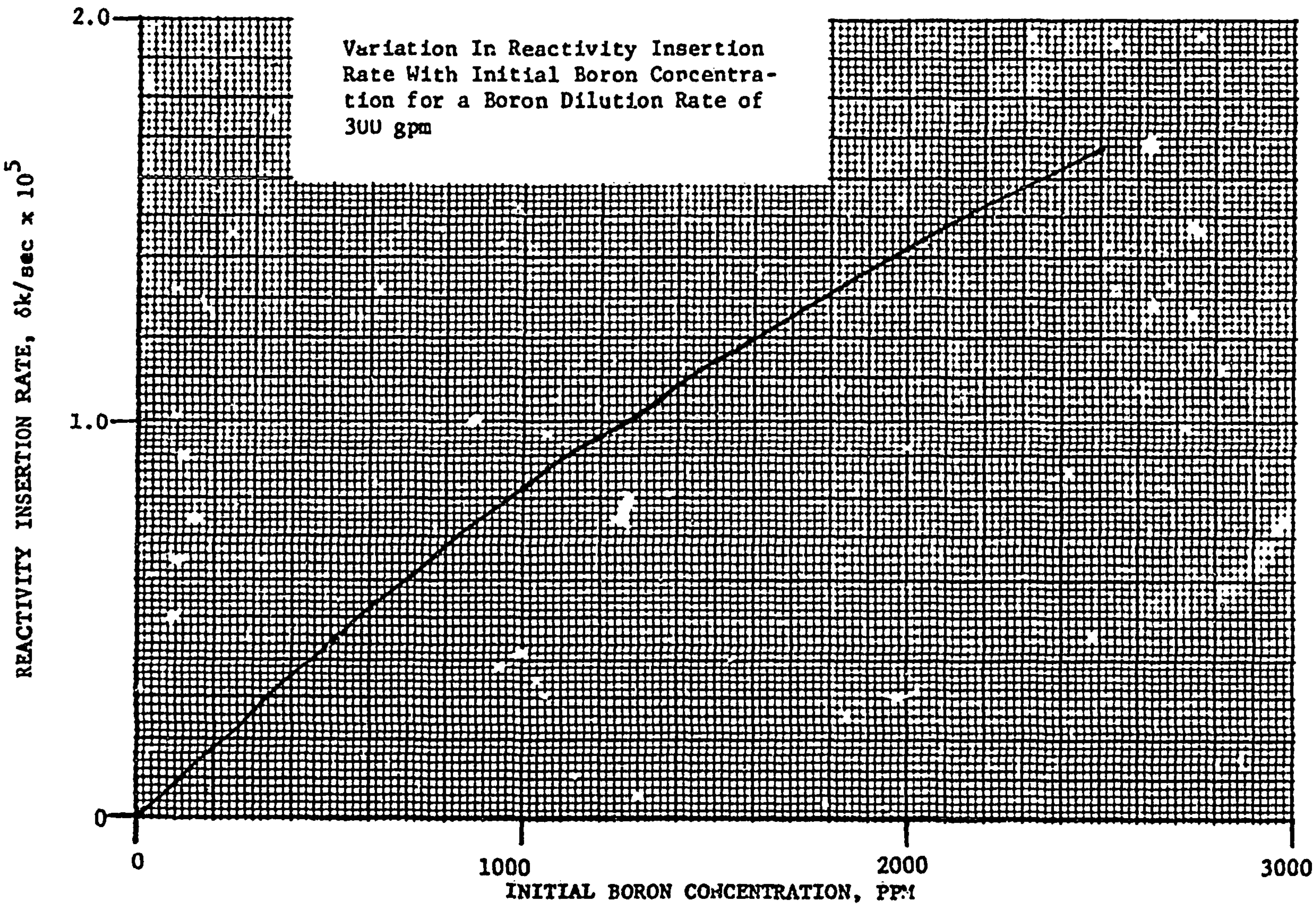


FIGURE 14.1.5-1

14.1.6 LOSS OF REACTOR COOLANT FLOW

A loss of coolant flow incident may result from a mechanical or electrical failure in one or more reactor coolant pumps, or from a fault in the power supply to these pumps. If the reactor is at power at the time of the incident, the immediate effect of loss of coolant flow is a rapid increase in coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly. The following trip circuits provide the necessary protection against a loss of coolant flow incident and are actuated by:

1) Low voltage or low frequency on pump power supply bus
2) Pump circuit breaker opening
3) Low reactor coolant flow

These trip circuits and their redundancy are further described in Section 7.2, Reactor Control and Protection System.

Simultaneous loss of electrical power to all reactor coolant pumps when the reactor is operating at full power represents the most severe credible loss-of-coolant flow condition. For this condition reactor trip together with flow sustained by the inertia of the coolant and rotating pump parts will be sufficient to prevent DNB. Therefore, the fuel will not be damaged as a result of the most severe credible loss of coolant flow accident.

Method of Analysis

The following loss of flow cases are analyzed:

1) Loss of four pumps from 2,758 MWt during four loop operation.
2) Loss of one pump from 2,758 MWt during four loop operation.
3) Loss of three pumps from 2,068 MWt during three loop operation.
4) Loss of one pump from 2,068 MWt during three loop operation.

The normal power supplies for the pumps are the four buses connected to the generator, each of which supplies power to one of the four pumps. When a turbine trip occurs, the pumps are automatically transferred to the buses supplied from an external power line, and the pumps will continue to supply coolant flow to the core. The simultaneous loss of power to the four reactor coolant pumps is a highly unlikely event. Since the pumps are on separate buses, a single bus fault would also result in the loss of only one pump.

A full plant simulation is used in the incident analysis to compute the core average and hot spot heat flux transient responses. The model includes flow coastdown, temperature, reactivity, and control rod insertion effects. Results of the plant simulation are then used in a detailed thermal-hydraulic computation to obtain the DNB margin. This computation solves the continuity, momentum, and energy equations of fluid flow together with the W-3 DNB correlation discussed in Section 3.2.2. The following assumptions are made in the calculations:

Initial Operating Conditions

The initial operating conditions, which are assumed to be most adverse with respect to the margin to DNB, are maximum steady state power level, minimum steady state pressure, and maximum steady state inlet temperature:

2,758 MWt - 4 loop operation:

Power	(1.02) (2,758 MWt) = 2813 MWt
Pressure	2250 - 30 = 2220 psia
Inlet Temperature	543 + 4 = 547°F

2,068 MWt - 3 loop operation:

Power	(1.02) (2,068 MWt) = 2,110 MWt
Pressure	2250 - 30 = 2220 psia
Inlet Temperature	544 + 4 = 548°F

Reactivity Coefficients

A conservatively high absolute value of the Doppler (-1.6×10^{-5} F^{-1}) and a sero moderator (0.0×10^{-4} F^{-1}) temperature coefficients were assumed since these result in the maximum hot spot heat flux and therefore the minimum DNB ratio during the transient.

Reactor Trip

For the one pump loss of flow incidents, the reactor trip is assumed to be actuated by the redundant flow monitoring channel (2/3) since this results in the largest delay to reactor trip. For the four and three pump loss of flow incidents, the reactor trip was assumed to be actuated by redundant bus undervoltage or breaker trip (1/4 or 1/3).

The low flow trip setting is 90 per cent of full flow; the trip signal is assumed to be initiated at 87 per cent of full flow, allowing 3 per cent for flow instrumentation errors. Upon reactor trip it is assumed that the most reactive RCCA is stuck in its fully withdrawn position, hence resulting in a minimum insertion of negative reactivity. The negative reactivity insertion upon trip is conservatively based on a 1 per cent shutdown margin at no load conditions.

Heat Transfer Coefficient

The overall heat conductance between the fuel and water regions varies considerably during the transient mostly as a result of the change of fuel gap conductance. The overall heat conductance was conservatively evaluated during the transient by a detailed calculation of fuel rod heat transfer. The hot spot heat transfer coefficient was increased 10% above the design value to obtain a conservatively high heat flux response.

Residual Heat

A conservative residual heat rate equal to 8 per cent of full power is assumed. That is, due to the very short time to reach the minimum DNB ratio, no credit was taken fro the reduction of residual heat after trip.

Flow Coastdown

Reactor coolant flow coastdown curves are shown on Figures 14.1.6-1 to 14.1.6-4. These curves are based on high estimates of loop pressure losses and include the effect of inertia from the pump flywheels.

Results

Figure 14.1.6-5 shows the neutron flux, the average heat flux, and the hot spot heat flux response for the four pump loss from 2,758 MWt during four loop operation. Figure 14.1.6-6 shows the DNB ratio as a function of time for this case. A minimum W-3 DNB ratio value of 1.42 is reached about 2.0 seconds after initiation of the incident.

Figure 14.1.6-7 shows the transients for loss of one pump from 2,758 MWt during four loop operation. Figure 14.1.6-8 shows the DNB ratio as a function of time for this case. A minimum W-3 DNB ratio value of 1.52 is reached about 3.0 seconds after initiation of the incident.

The transient for loss of three pumps from 2,068 MWt during three loop operation is shown on Figure 14.1.6-9. The minimum DNB ratio of 1.57 occurs about 2.0 seconds after initiation of the transient, as shown on Figure 14.1.6-10.

The transient for loss of one pump from 2,068 MWt during three loop operation is shown on Figure 14.1.6-11. The minimum DNB ratio of 1.81 occurs about 3.0 seconds after initiation of the transient, as shown on Figure 14.1.6-12.

Conclusions

Since DNB does not occur in any loss of coolant flow incident, there is no cladding damage and no release of fission products into the reactor coolant. Therefore, once the fault is corrected, the plant can be returned to service in the normal manner. The absence of fuel failures would, of course, be verified by analysis of reactor coolant samples as the plant is brought to power.

Locked Rotor Accident

A transient analysis is performed for the postulated instantaneous seizure of a reactor coolant pump rotor. Flow through the reactor coolant system is rapidly reduced, leading to a reactor trip on a low-flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer to the secondary system causes an insurge into the pressurizer and a pressure increase throughout the Reactor Coolant System. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and eventually opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect is not included in the analysis.

The locked rotor analysis was performed for both four loop and three loop operation.

Method of Analysis

Initial Conditions

At the beginning of the postulated locked rotor accident, i.e., at the time the shaft in one of the reactor coolant pumps is assumed to seize, the plant is assumed to be in operation under the most severe steady state operating conditions. These initial conditions are the same as the one described for the Loss of Coolant Flow Accident on page 14.1.6-2.

Evaluation of the Pressure Transient

A detailed digital code was used to determine the peak pressure in the Reactor Coolant System under the postulated accident conditions and to obtain the nuclear power as a function of time which is used elsewhere in the analysis. For four pump operation, the coolant flow through the core is conservatively assumed to be reduced from full power to 70% of its initial value at the time of the pump seizure. For three pump operation, it is reduced to 60% of its initial value (which is 71% of nominal flow).

After pump seizure, nuclear power is rapidly reduced because of the control rod insertion upon plant trip and void shutdown due to bulk boiling.

No credit was taken for the pressure-reducing effect of the pressurizer relief valves, steam dump and controlled feedwater flow after plant trip. Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The safety valves start operating at 2500 psia and their combined capacity for steam and water relief is, respectively, 42 and 14 ft^3/sec.

Evaluation of DNB in the Core During the Accident

A calculation of the extent of DNB in the core during the accident was performed using a multichannel THINC-III model. Heat flux distribution in the core is an input to the THINC-III code and was evaluated with a detailed digital model for the heat transfer in the fuel. The model is similar to the model incorporated in the LOCTA code but features a larger number of lumps in the fuel in order to obtain a detailed temperature distribution.

The THINC-III model consisted of 9 concentric channels. The axial power distribution was a chopped cosine with an axial hot channel factor of 1.79. The radial rod power distribution is shown in Figure 14.1.6-13.

In order to estimate the severity of the accident in the core as far as the integrity of the fuel rods are concerned, the thermal behavior of the fuel located at the hot spot was investigated after DNB. This study was made for four pump operation which is the most pessimistic case since the high initial flux at full power produces a high stored heat in the fuel at the hot spot. Results obtained from an analysis of this "hot spot" condition represent the upper limit with respect to clad temperature, clad melting and zirconium-steam reaction.

Film Boiling Coefficient

The following empirical equation is included in the digital program to calculated the film boiling coefficient:

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

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

and the steam properties are evaluated at film temperature (avg. between wall and bulk temperatures). The units and reference for this correlation are given in Section 3.2.2.
The program calculates the film coefficients at every time step based upon the actual clad temperature at this time.

The system pressure, bulk density and mass flow rate were an input of the program as a function of time. For conservatism, DNB was assumed to start at the beginning of the accident and the heat transfer coefficient between clad and water was reduced suddenly from its steady-state value to the film boiling value at time = 0, without any period of transition boiling.

Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad has a pronounced influence on the thermal results. The larger the value of this coefficient, the more heat is transferred between pellet and clad. For initial times after the accident, a high gap coefficient produces higher clad temperatures as the heat stored and generated in the fuel pellet tries to redistribute itself in the cooler clad. This effect of the gap coefficient, however, is reversed when the clad temperature exceeds the pellet temperature in cases when zirconium-steam reaction is present. A conservatively high value of 3000 Btu/hr-ft^2-°F was used in the present study.

In order to estimate the effect of the film boiling heat transfer coefficient, runs were made with h_{film} = 100, 150 and 200 Btu/hr-ft^2-°F (cases 1, 2 and 3) and the maximum clad temperatures were found to be 2125°F, 1912°F, and 1761°F.

Since it was found that in the worst cases examined, the clad temperature exceeded 1800°F, it was necessary to consider the possibility of a zirconium-steam reaction which can become significant above this temperature. In order to take this phenomen into account, the following correlation which defines the rate of the zirconium-steam reaction has been introduced into the model:

$$\frac{d}{dt}(w^2) = 33.3 \times 10^6 \, e^{\frac{-45,500}{1.986T}}$$

w = amount reacted, mg/cm^2

t = time, sec

T = temperature, °K

The reaction heat is about 1510 cal/gm.

The three runs with constant h_{film} = 100, 150, 200 were repeated with Zr-steam reaction and the maximum clad temperature was 2185°F, 1917°F, and 1762°F, respectively (cases 4, 5 .d 5). It can be seen that although the zirconium-steam reaction can be detected at 1800°, it produces only a moderate increase in the maximum (60°F) for clad temperature as high as 2100°F.

Two more cases were run with a variable film boiling coefficient calculated from the correlation mentioned above; they include also the zirconium-steam reaction.

Case 7 assumes the pessimistic gap coefficient of 3000 and in case 8, a more realistic calculation of the gap coefficient was used based on an elastic clad. Cases 7 and 8 gave 1899°F and 1802°F respectively.

For Case 7, the extent of zirconium-steam reaction is less than 1% of the clad at the hot spot, and less than 0.001% for the total core.

Results

The primary coolant pressure versus time is shown in Figury 14.1.6-14. The peak pressure is 2416 psia, reached at 1.6 second for 4 pump operation and 2508 psia, reached at 2.3 second for 3 pump operation. The minimum DNB ratio in the hot channel is shown in Figure 14.1.6-15. The DNB ratio in the other channels remains at all times above 1.3.

Table 14.1.6-1 and Figure 14.1.6-16 summarizes the maximum clad temperature at the hot spot during the accident.

Conclusions

a) The peak pressure of 2508 psia for the worst case ensures that the integrity of the primary coolant system is not endangered and can be considered as an upper limit, considering the conservative assumptions used in the study:

1) Credit was not taken for the negative moderator coefficient

2) It was assumed that the pressurizer relief valves were inoperative

3) The coolant flow in the core was assumed to drop suddenly (0.1 sec or less) from its initial valve to its final valve.

4) No DNB was assumed in the core which gives the highest average heat flux during the accident.

b) Only the hot channel exhibited a DNB ratio of less than 1.3 which represents only very small fraction of the rods (less than 0.1%). Hence, any fuel damage would be limited to less than 0.1% of the rods.

c) The peak clad surface temperature of 1899°F, calculated for the hot spot includes the effect of the zirconium-steam reaction (which is still quite small at that temperature). It can be considered an upper limit since:

1) The hot spot was assumed to be in DNB from time zero

2) A high gap coefficient ([illegible]00 Btu/hr-ft^2-°F was used

3) No credit was taken for transition boiling. The heat transfer coefficient for fully developed film boiling was used from time zero

4) The nuclear heat released in the fuel at the hot spot was based on a zero moderator coefficient.

A more realistic value for the maximum cladding temperature would be 1802°F, which was based on an elastic cladding resulting in a variable gap coefficient.

TABLE 14.1.6-1

Case n	h_{film}	h_{gap}	Including Zr-Steam Reaction Calculation	Maximum Clad Surface Temp.
1	100	3000	No	2125
2	150	3000	No	1912
3	200	3000	No	1761
4	100	3000	Yes	2185
5	150	3000	Yes	1917
6	200	3000	Yes	1762
7	Variable	3000	Yes	1899
8	Variable	Variable	Yes	1802

FOUR PUMP LOSS OF FLOW INCIDENT

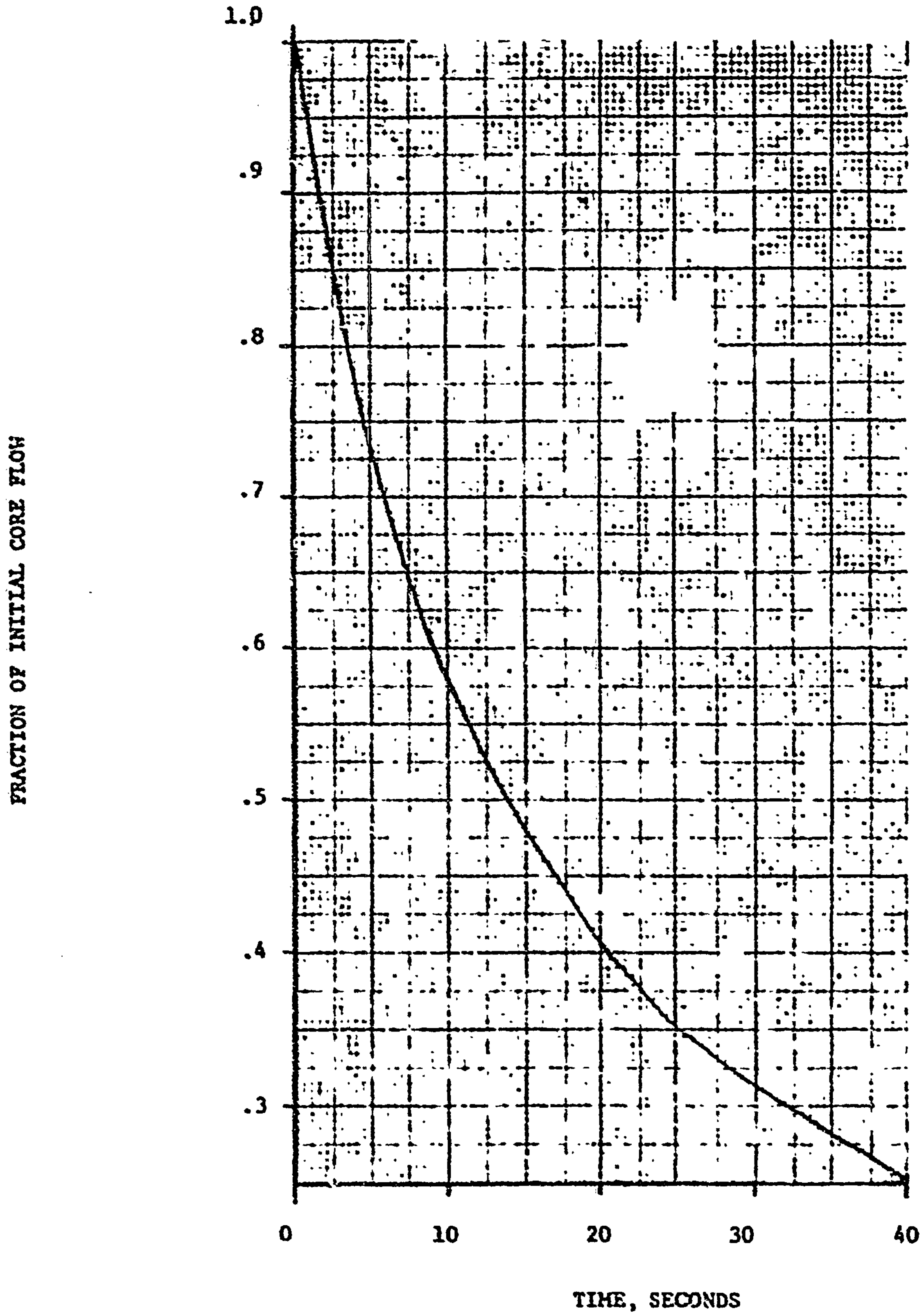


FIGURE 14.1.6-1

LOSS OF ONE PUMP OUT OF FOUR

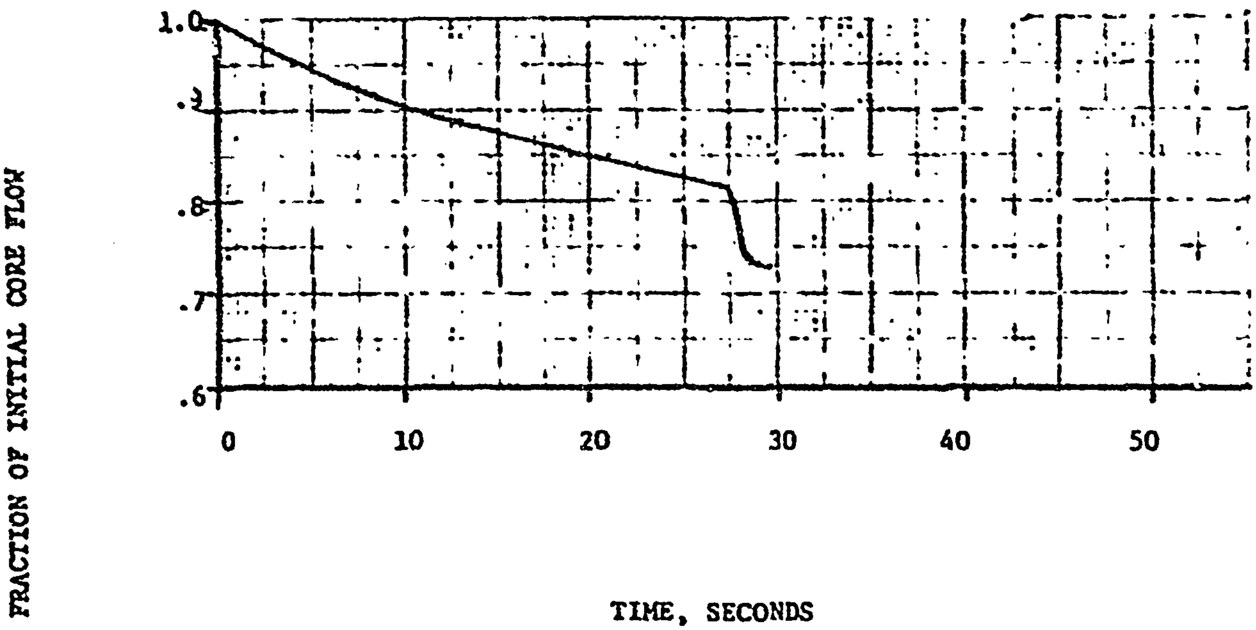


FIGURE 14.1.6-2

LOSS OF THREE PUMPS OUT OF THREE

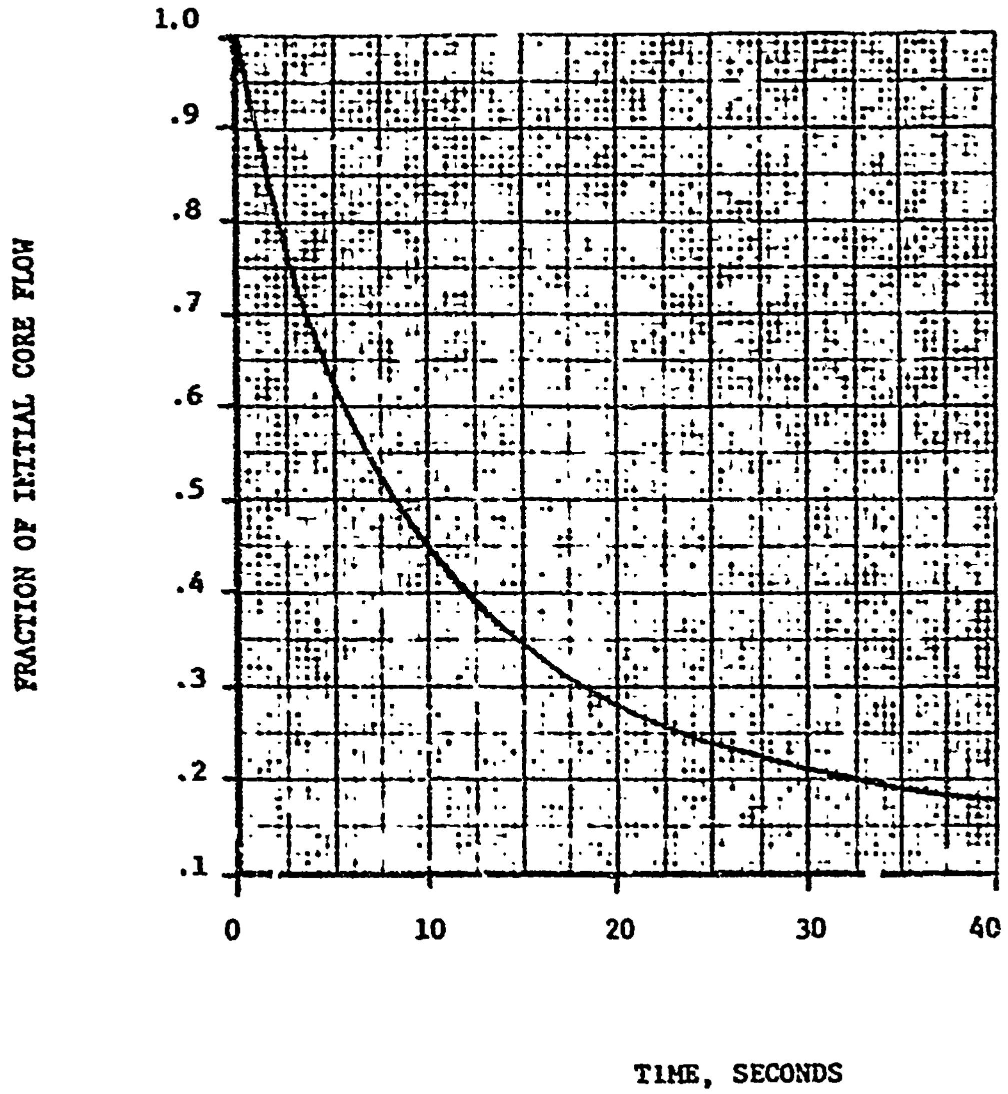


FIGURE 14.1.6-3

LOSS OF ONE PUMP OUT OF THREE

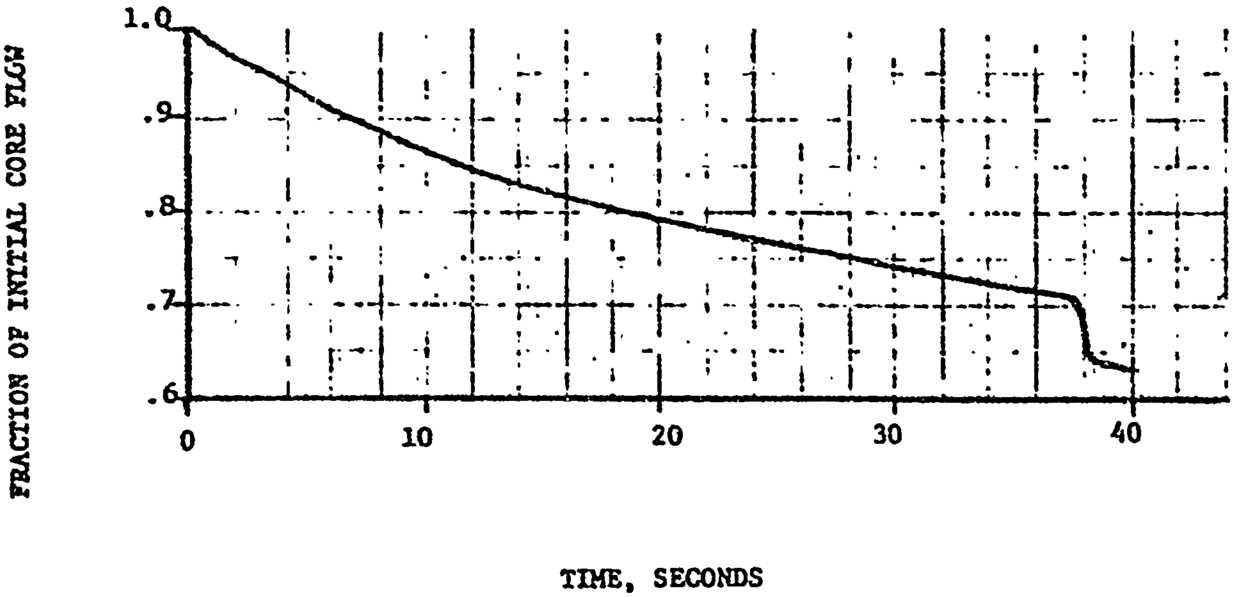


FIGURE 14.1.6-4

LOSS OF FOUR PUMPS OUT OF FOUR

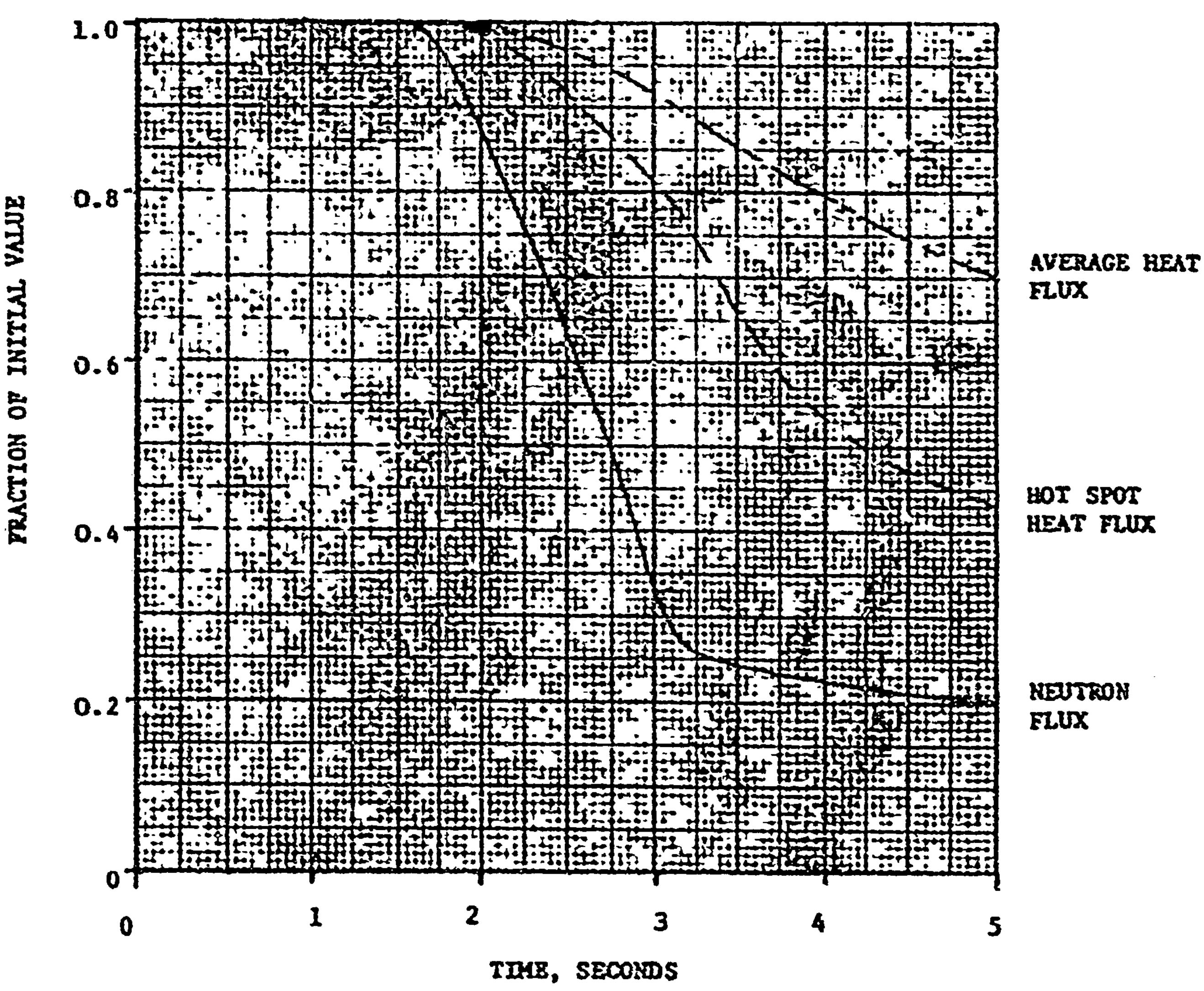


FIGURE 14.1.6-5

LOSS OF FOUR PUMPS OUT OF FOUR

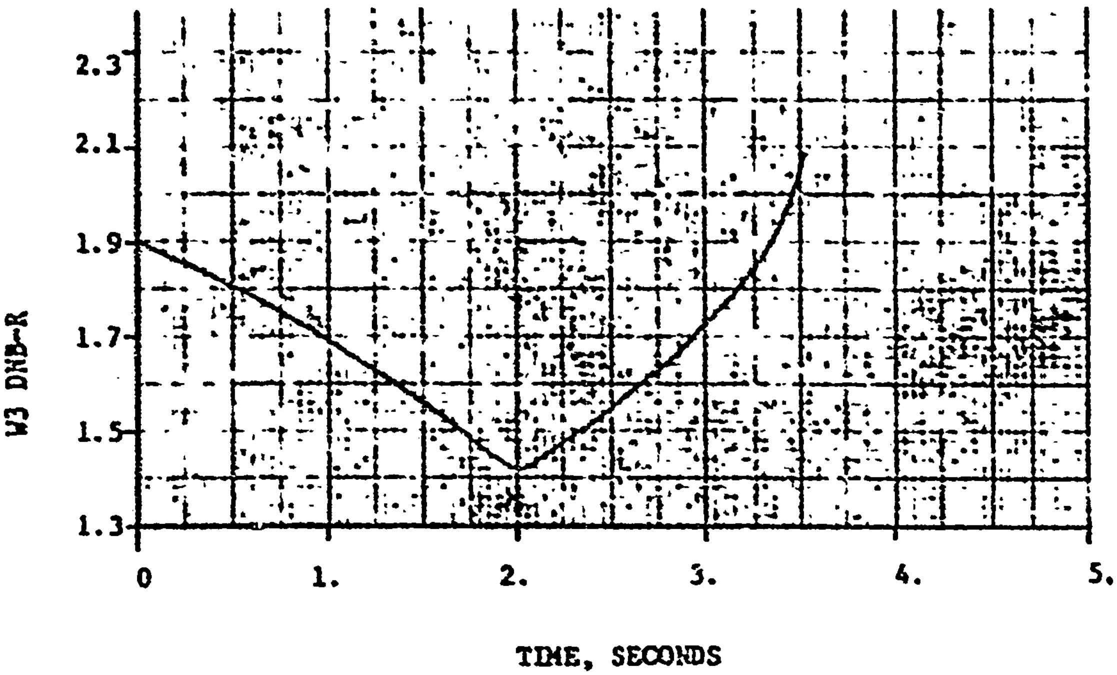


FIGURE 14.1.6-6

LOSS OF ONE PUMP OUT OF FOUR

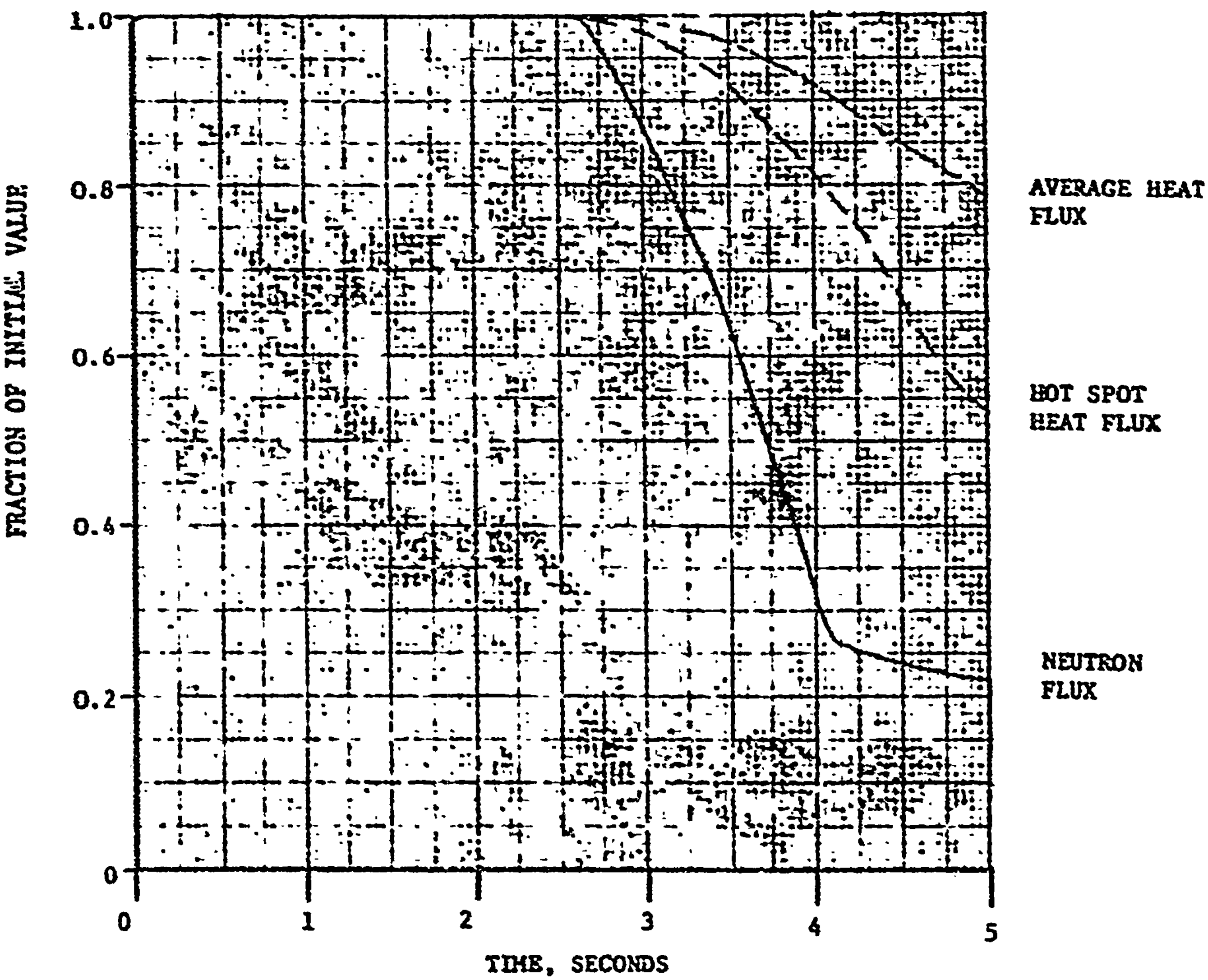


FIGURE 14.1.6-7

LOSS OF ONE PUMP OUT OF FOUR

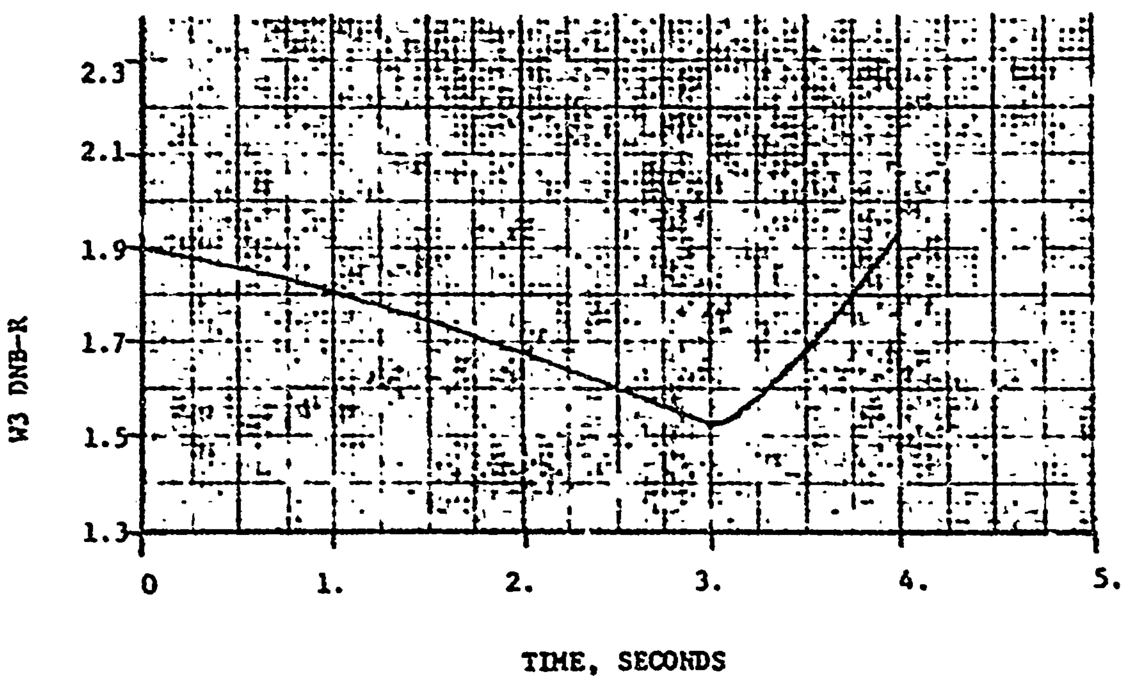


FIGURE 14.1.6-8

LOSS OF THREE PUMPS OUT OF THREE

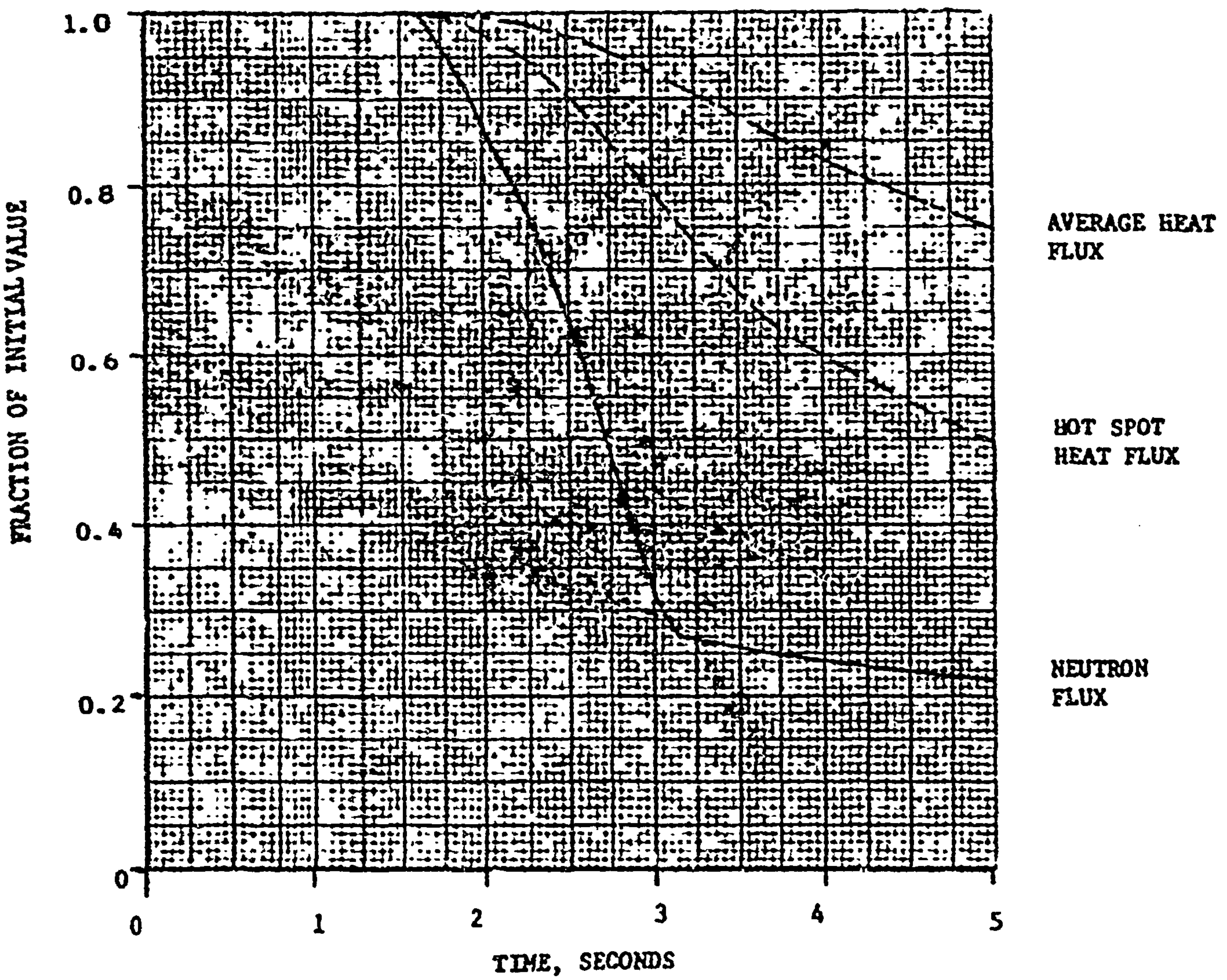


FIGURE 14.1.6-9

LOSS OF THREE PUMPS OUT OF THREE

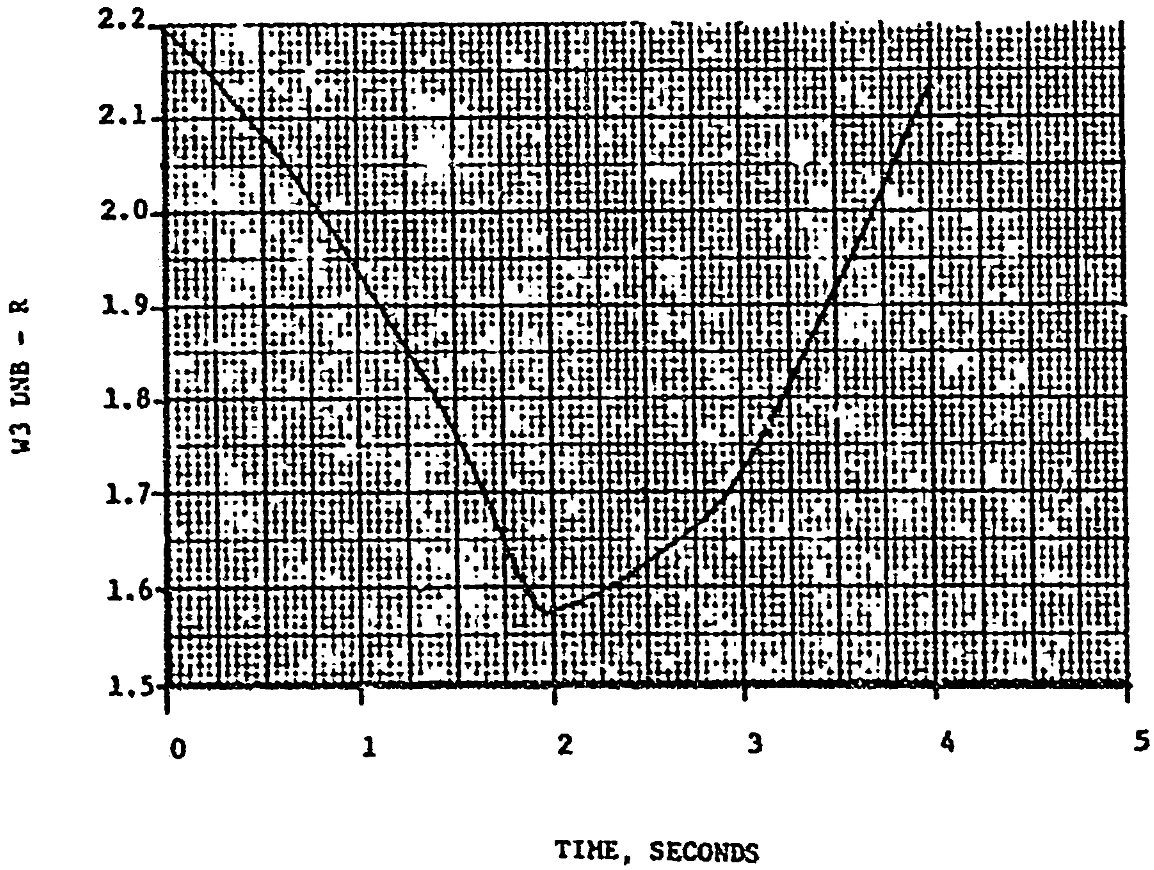


FIGURE 14.1.6-10

LOSS OF ONE PUMP OUT OF THREE

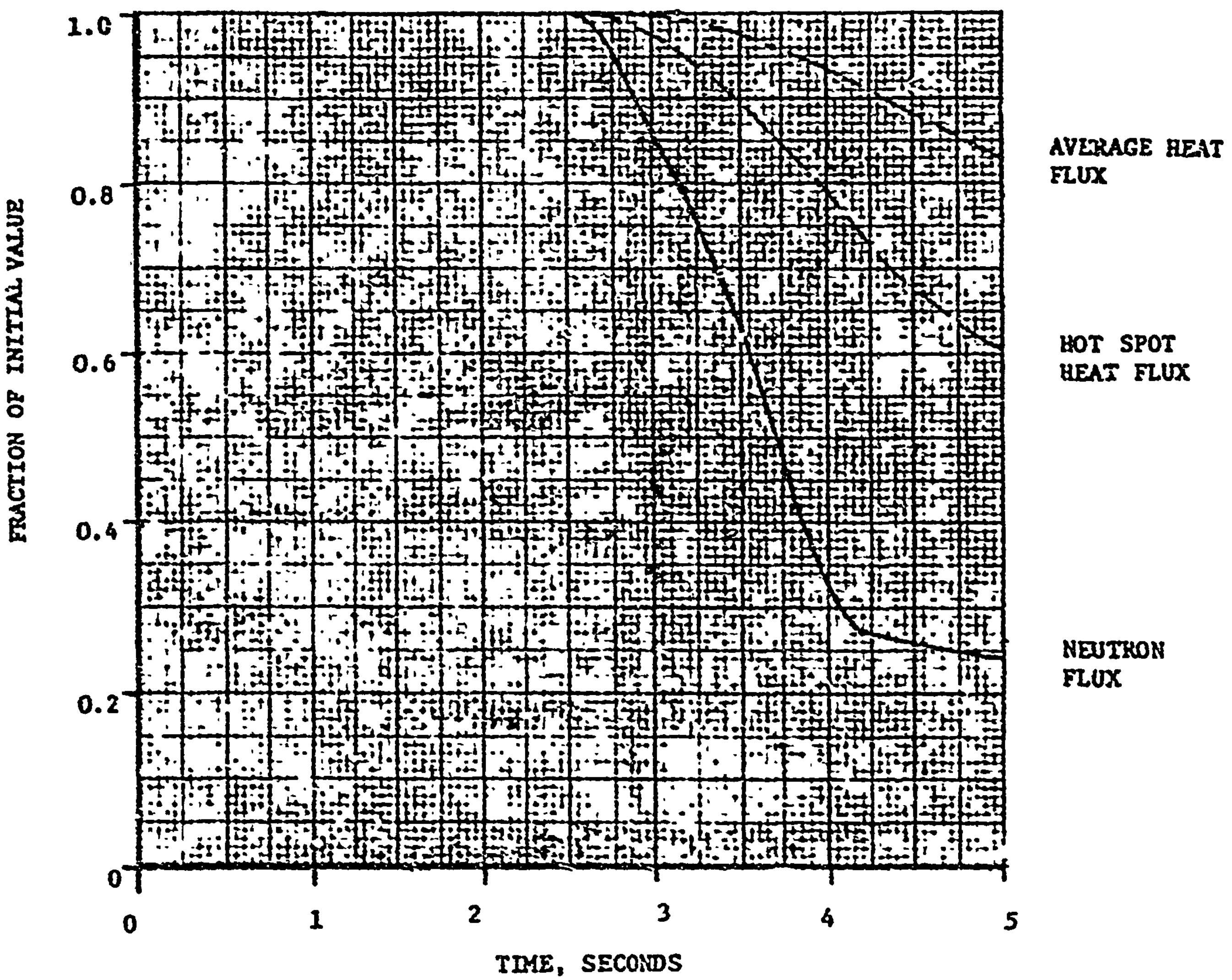


FIGURE 14.1.6-11

LOSS OF ONE PUMP OUT OF THREE

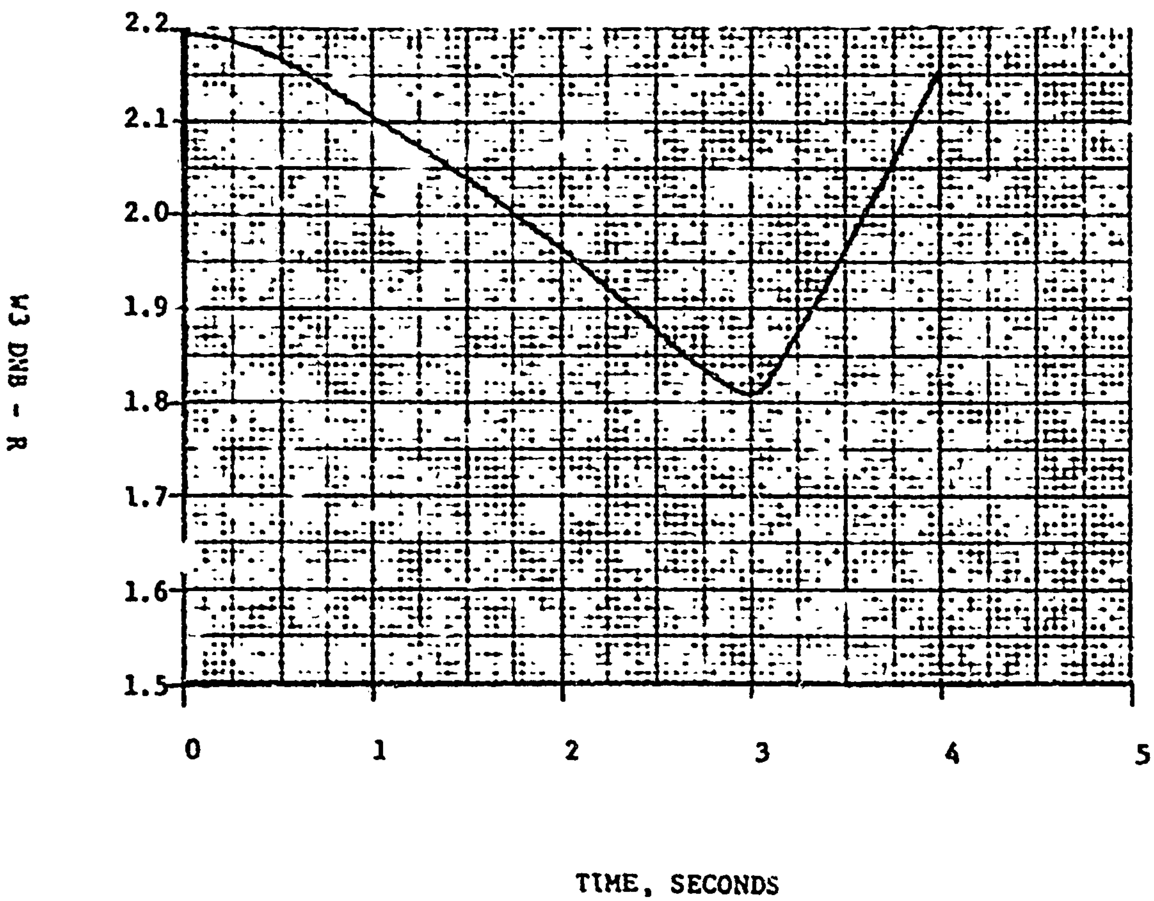


FIGURE 14.1.6-12

RADIAL ROD POWER DISTRIBUTION

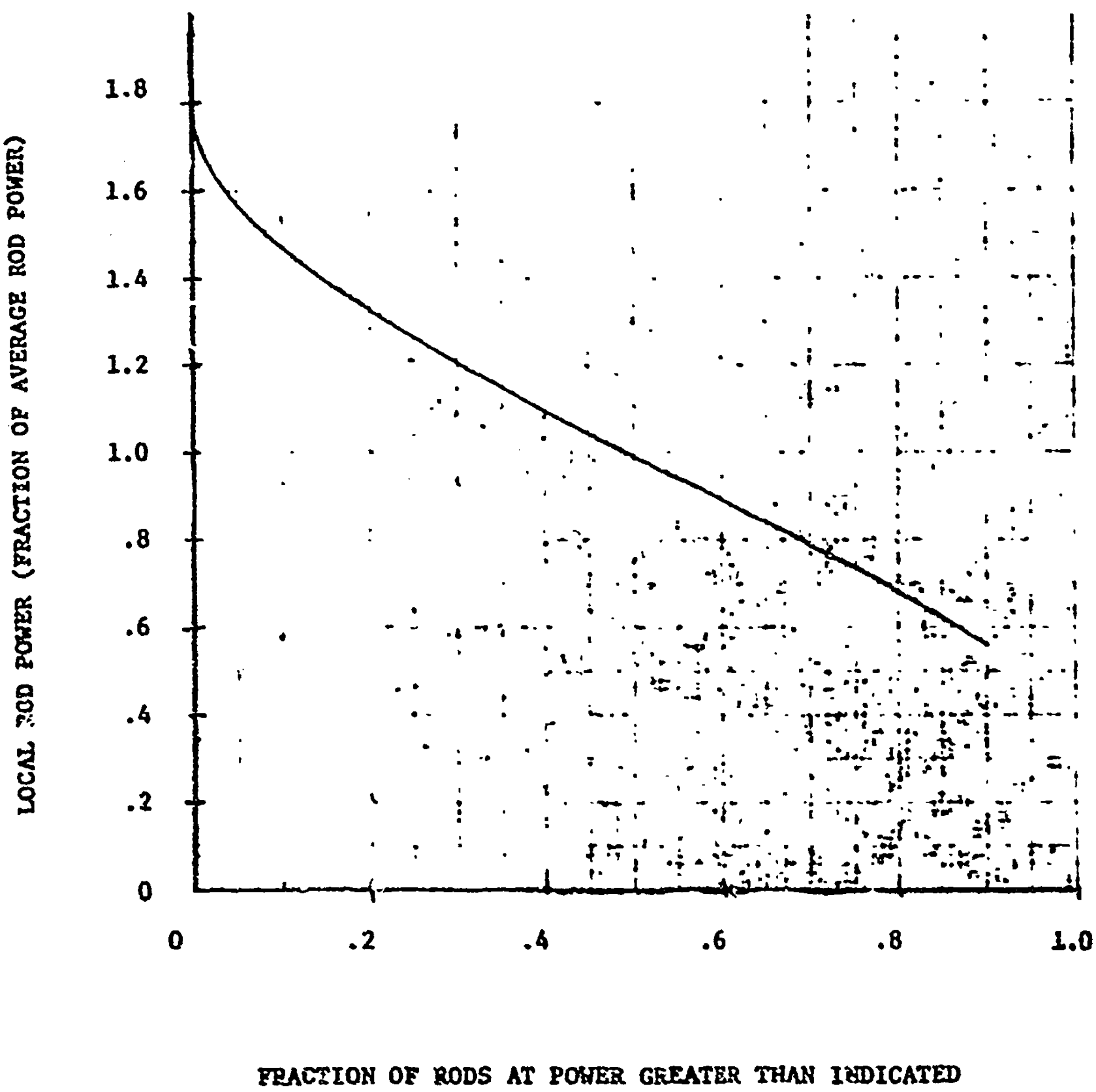


FIGURE 14.1.6-13

PRIMARY COOLANT PRESSURE VS TIME

LOCKED ROTOR

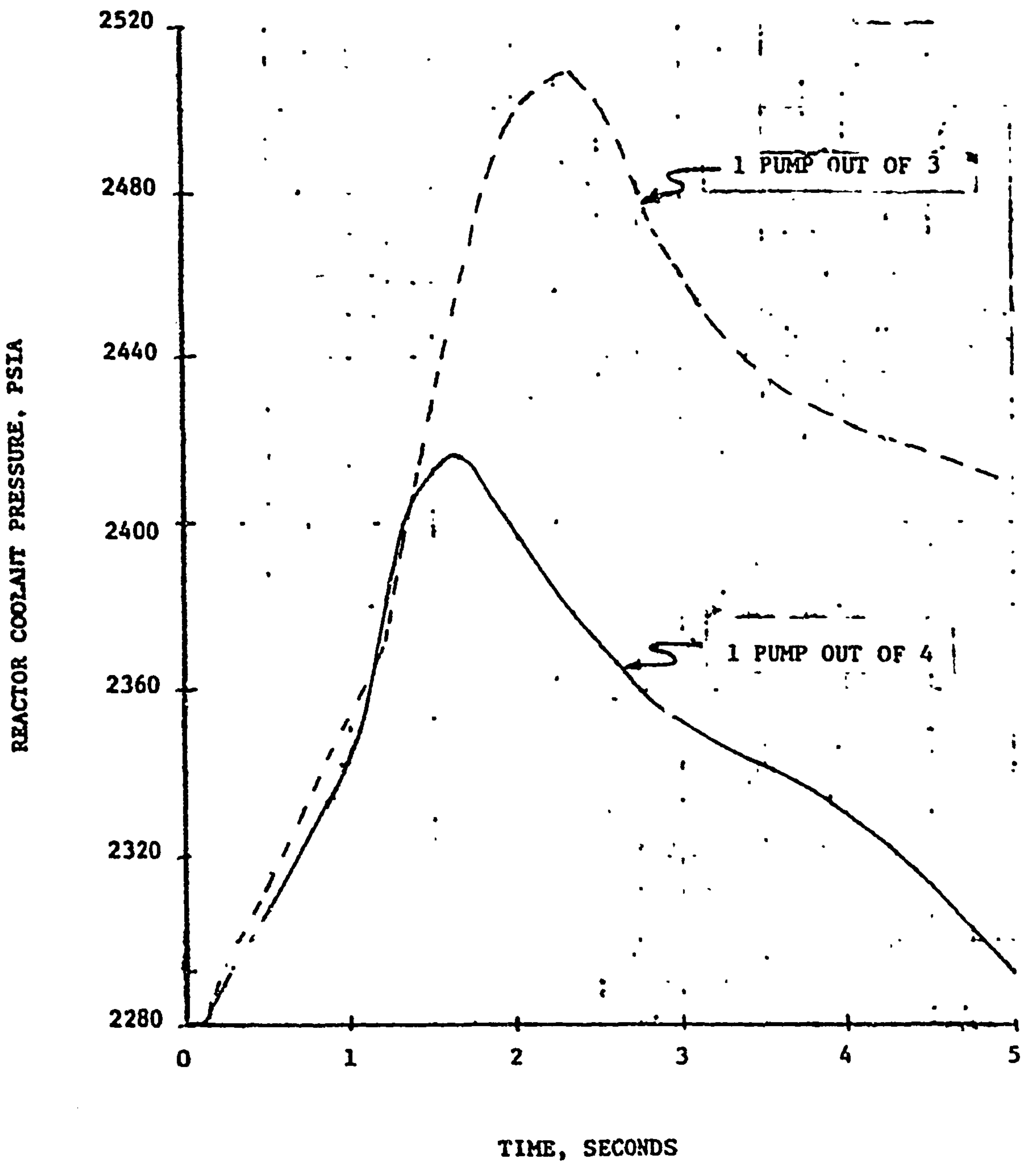


FIGURE 14.1.6-14

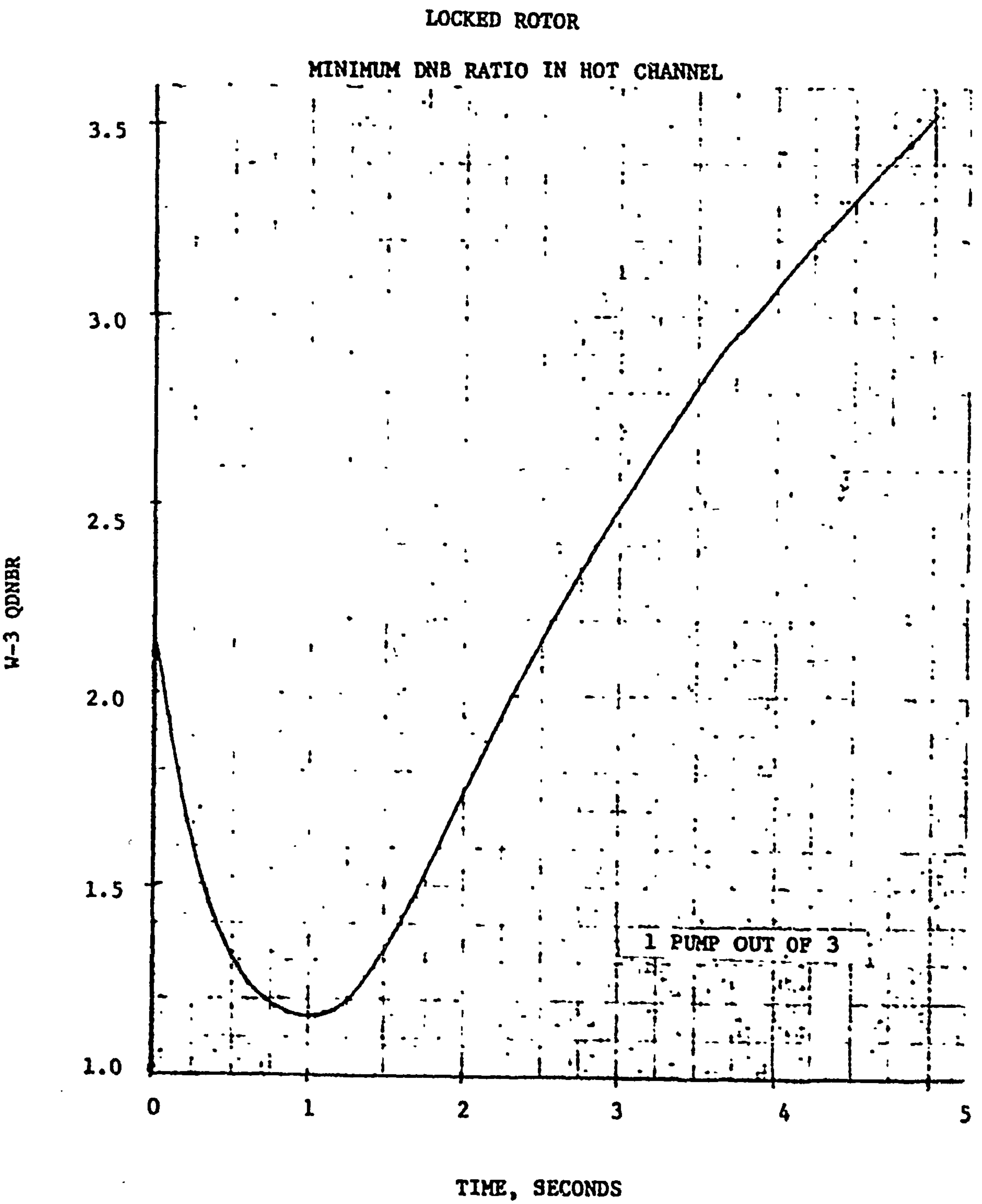


FIGURE 14.1.6-15A

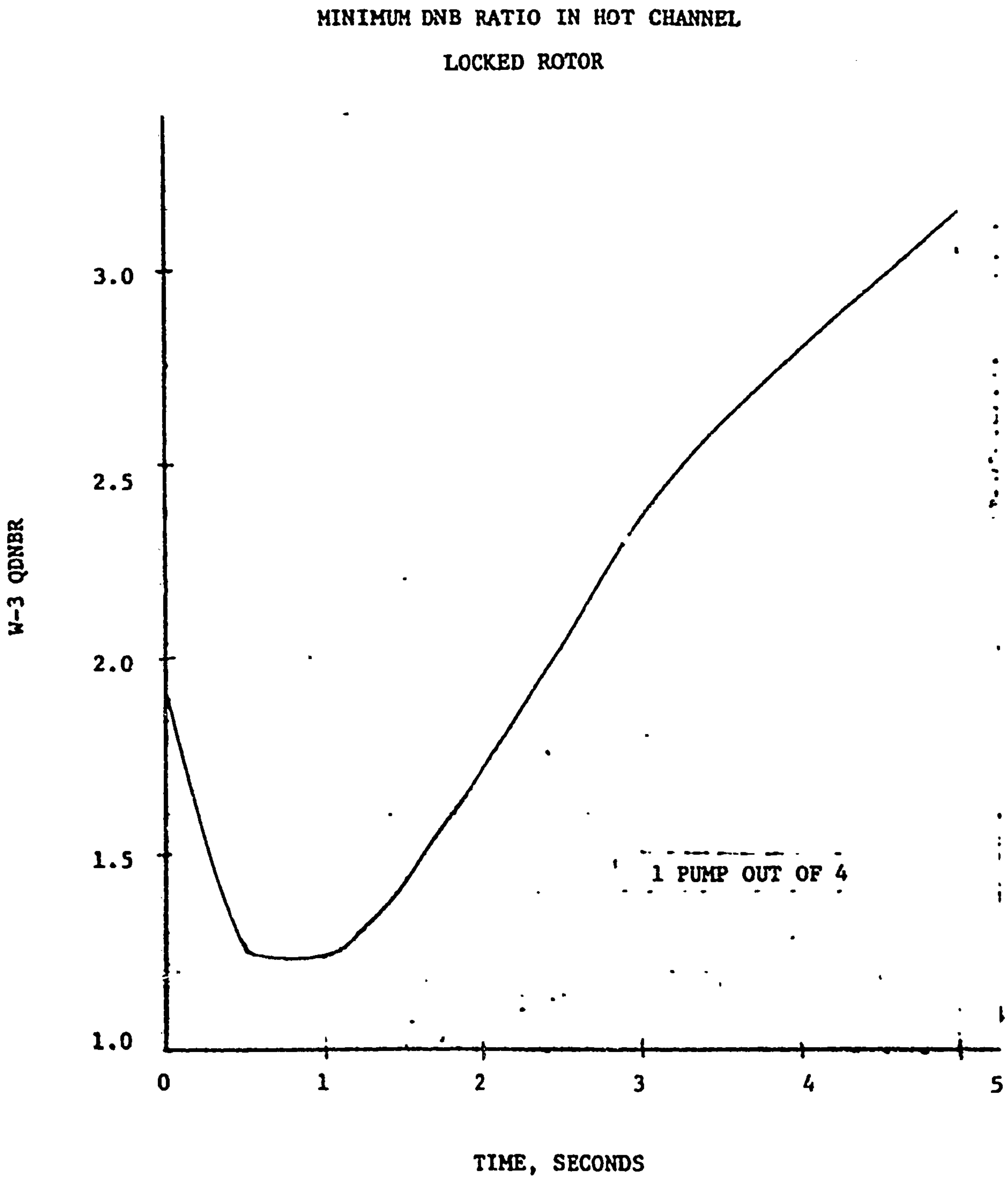


FIGURE 14.1.6-15B

MAXIMUM CLAD TEMPERATURE AT HOT SPOT

LOCKED ROTOR

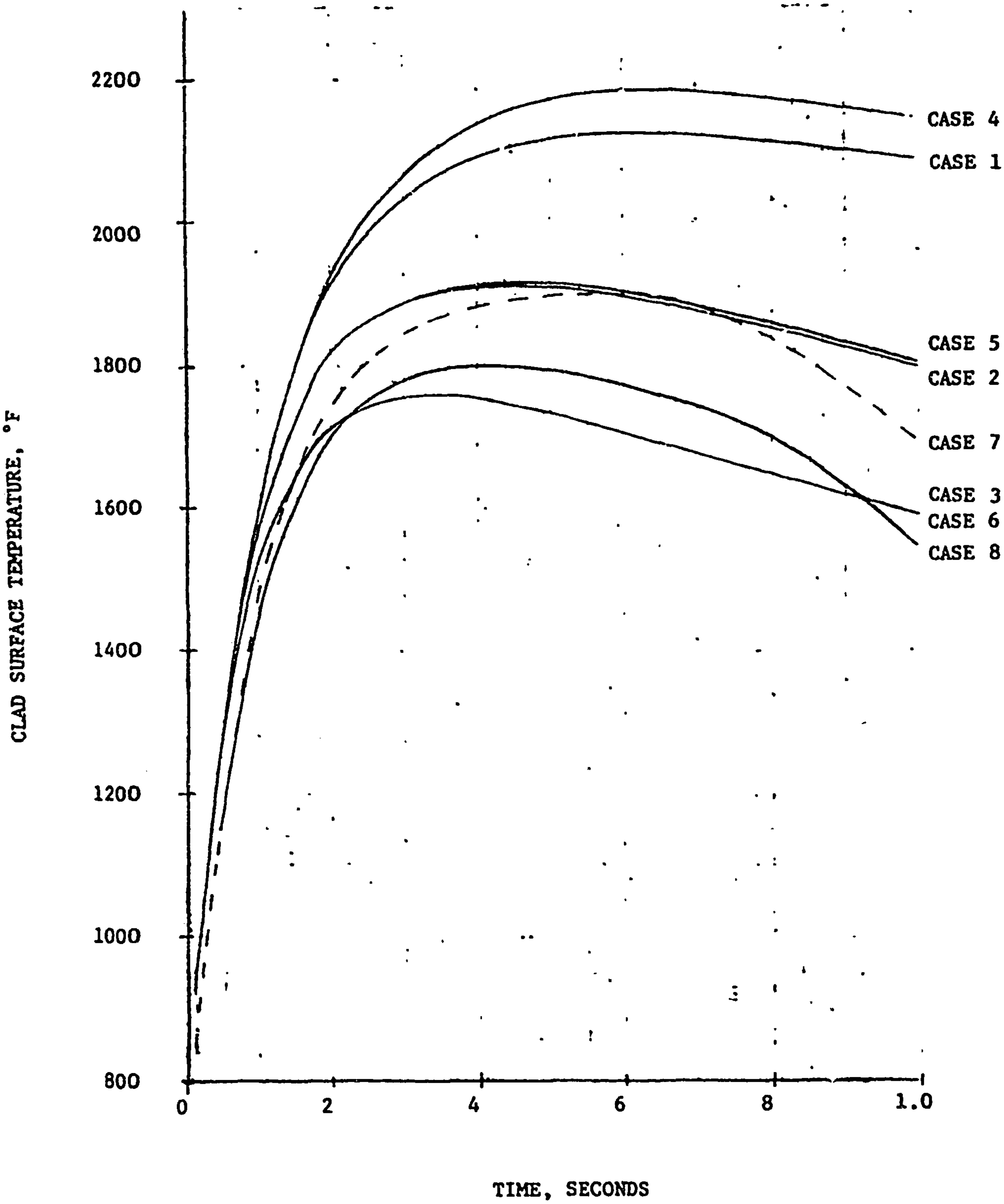


FIGURE 14.1.6-16

14.1.7 START-UP OF AN INACTIVE REACTOR COOLANT LOOP

Operation of the plant with an inactive loop causes reversed flow through the inactive loop because there are no isolation valves or check valves in the reactor coolant loops.

If the reactor is operated at power in this condition, there is a decrease in the coolant temperature in that loop in comparison with the other loops. The subsequent re-start of the idle reactor coolant pump, without bringing the loop temperature closer to the average temperature would result in the injection of cold water into the core. This cooler water causes a rapid reactivity increase.

Assumptions and Method of Analysis

The following assumptions are made:

1. The idle pump on starting accelerates to full flow instantaneously, i.e., no slip and the time to accelerate the pump and coolant are zero.

2. A conservative maximum negative moderator coefficient of -3.5×10^{-4} $\delta k/°F$ is assumed.

3. A low Doppler coefficient of -1.0×10^{-5} $\delta k/°F$ is taken.

4. A high heat transfer coefficient between the primary and secondary system is assumed for the inactive loop. This implies that the temperature of the water in that part of the inactive loop from the steam generator plenum to the reactor exit plenum is at a temperature equal to the saturation temperature on the secondary side.

5. The core power to flow ratio is taken to be constant at the normal loop operational value.

6. The secondary pressure is taken to be the value corresponding to the above core power.

Account is taken of the delay in the cold water reaching the core from the time the pump is started.

An analog simulation of the complete plant was used to study the ensuing plant transients.

Results

The analog study of the transient behavior of the plant was made at 75% load (2069 MWt). A conservatively high value of the temperature difference of -30°F is taken as the input disturbance. The cold water entering the reactor plenum chamber is assumed to mix with the water coming from the active loops. The cold water slug is taken to last for 15 seconds. The temperature coming fro. all steam generators is assumed to be the same. The delay before the cold slug reaches the inlet to the reactor core is taken to be 4.0 seconds.

The results are shown in Figures 14.1.7-1 through 14.1.7-4. The thermal power build up is slow and this leads to the cooling of the primary circuits by the temperature slug. This accounts for the drop in pressurizer pressure and the average temperature.

Conclusions

The results show that for the -30°F change in core inlet temperature, the nuclear power rises to 104%, this does not cause a reactor trip. During the transient, both the primary pressure and the core inlet temperature fall so that there is no significant decrease in DNBR from these effects.

It is expected that the actual transients effects will be less severe, because of alleviating factors which have not been taken into account, e.g., the time constant of the pump is likely to be about 10 seconds. This means that the change in temperature will occur more gradually and that the transient will be less severe.

The conclusion is that the minimum DNBR reached is well in excess of the 1.3 limit value and therefore DNB will not occur during this transient.

START UP OF AN INACTIVE LOOP

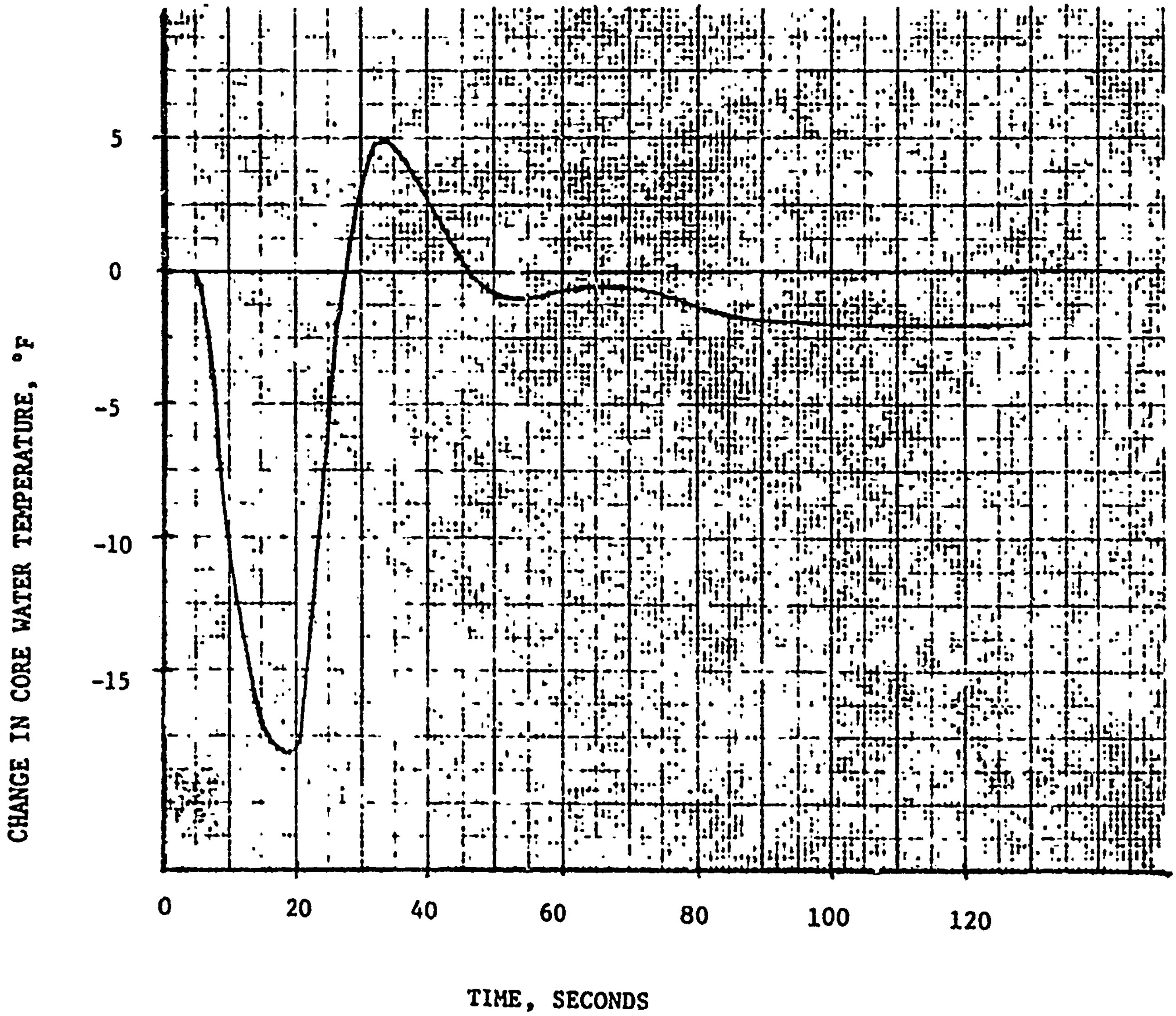


FIGURE 14.1.7-1

START UP OF AN INACTIVE LOOP

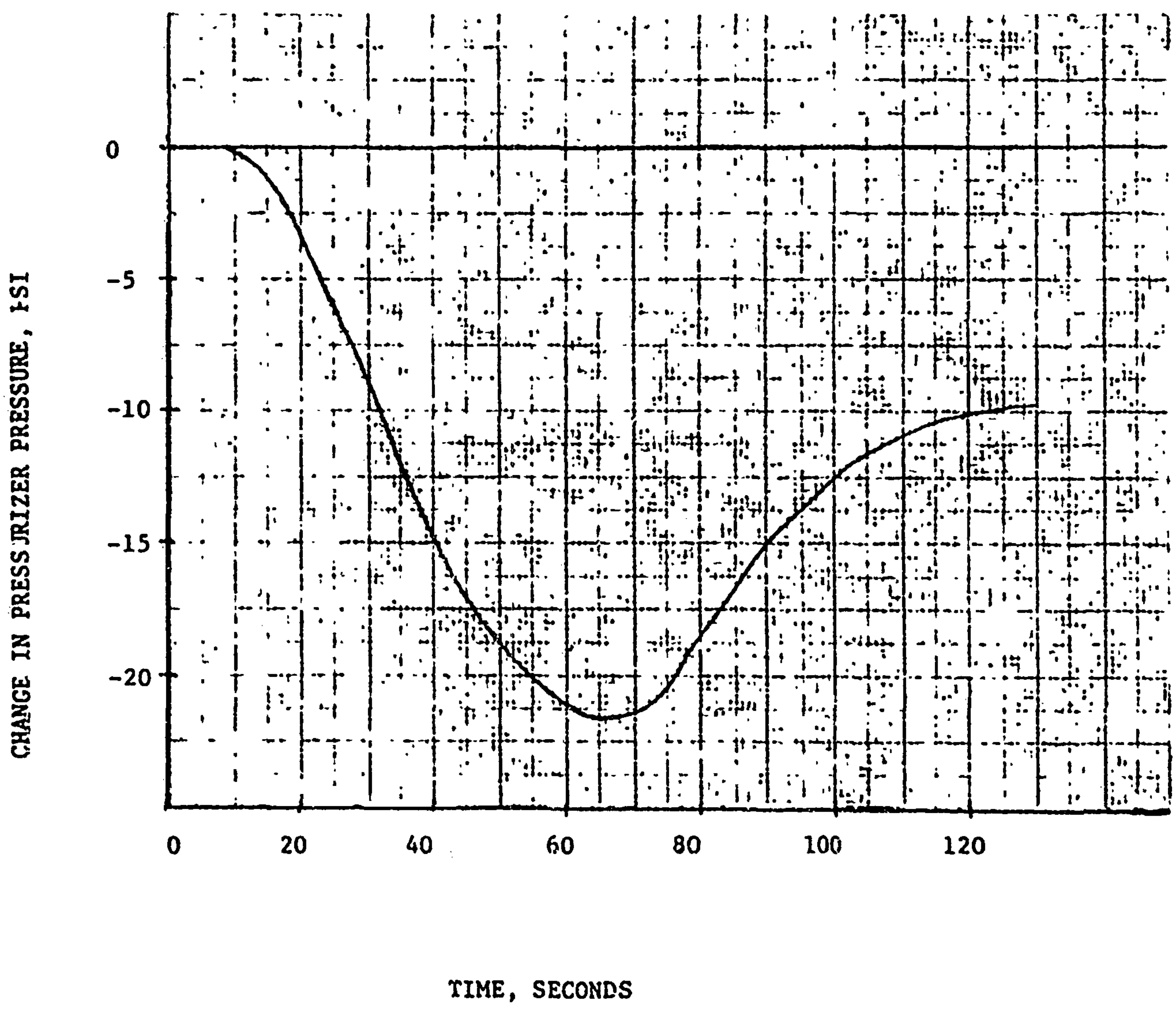


FIGURE 14.1.7-2

START UP OF AN INACTIVE LOOP

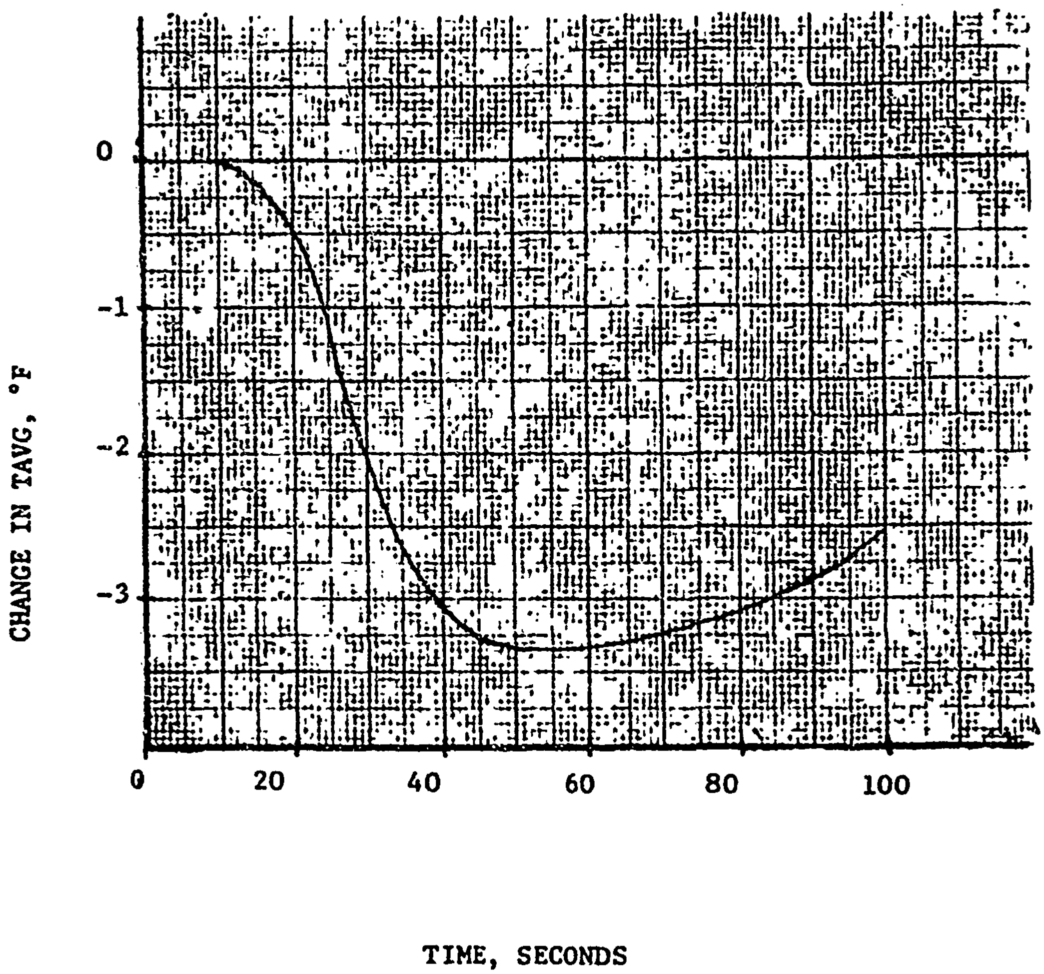


FIGURE 14.1.7-3

START UP OF AN INACTIVE LOOP

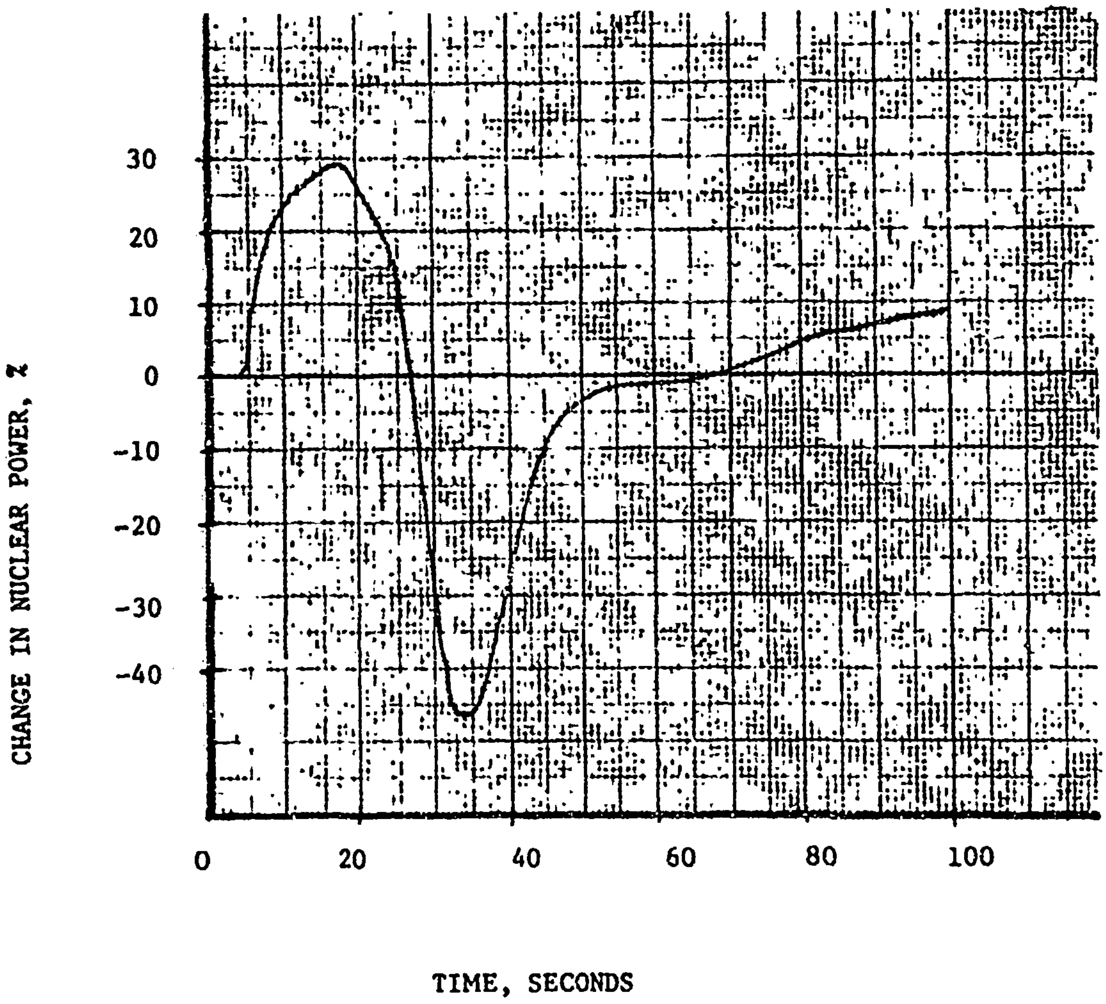


FIGURE 14.1.7-4

14.1.8 LOSS OF EXTERNAL ELECTRICAL LOAD

The loss of external electrical load may result from an abnormal increase in network frequency, or an accidental opening of the main breaker from the generator which fails to cause a turbine trip but causes a rapid large load reduction by the turbine governor control.

The plant is designed to accept a 50 percent step loss of load without actuating a reactor trip. The automatic steam bypass system with 40 percent dump capacity is able to accommodate this abnormal load rejection by reducing the transient imposed upon the reactor coolant system. The reactor power is reduced to the new equilibrium power level at a rate consistent with the capability of the rod control system. The pressurizer relief valves may be actuated, but the pressurizer safety valves and the steam generator safety valves are not actuated in this case.

In the event the steam bypass valves fail to open following a large load loss, the steam generator safety valves are actuated and the reactor may be tripped by the high pressurizer pressure signal or the high pressurizer level signal. The steam generator shell side pressure and reactor coolant temperatures increase rapidly. The pressurizer safety valves are sized to protect the reactor coolant system against overpressure without taking credit for the steam bypass system.

The most likely source of a complete loss of load on the Nuclear Steam Supply System is a trip of the turbine-generator. In this case there is a direct reactor trip signal derived from turbine autostop oil pressure (a two out of three signal). Reactor coolant temperatures and pressure do not increase if the steam bypass system and pressurizer pressure control system are functioning properly. However, the plant behavior is also evaluated for a complete loss of load from full power without a direct reactor trip, primarily to show the adequacy of the pressure relieving devices as well as to show that no core damage occurs. The Reactor Coolant System and Steam System pressure relieving capacities are designed to ensure the safety of the plant without requiring the automatic rod control, pressurizer pressure control, and/or steam bypass control systems.

Method of Analysis

The total loss of load transients are analyzed by employing a detailed digital computer program. This code describes the neutron kinetics, decay heat, Reactor Coolant System with pressurizer, steam generators, and the associated steam bypass system and rod control system.

The objectives of this analysis are to determine margins to core protection limits and to establish pressure relieving requirements for the Reactor Coolant and Steam Systems.

Initial Operating Conditions

The initial reactor power, coolant temperatures and pressure are all assumed at extreme values consisting with steady state, full power operation, including allowances for calibration and instrument errors. This results in the maximum power difference for the load loss, and the minimum margin to core protection limits at the initiation of the total loss of load accident.

Moderator and Doppler Coefficients of Reactivity

The total loss of load is analyzed for both beginning-of-life and end-of-life conditions.

At beginning-of-life the least negative value of moderator coefficient is used with the least negative value of Doppler coefficient. This results in a maximum nuclear power increase following the loss of load. At end-of-life the most negative value of moderator coefficient is used with the most negative value of Doppler coefficient. This results in a least shutdown margin following reactor trip. See Section 3.2.2.

Reactor Control

Two cases are analyzed:

1. The reactor is assumed to be in normal automatic control with the control rods in the minimum incremental worth region.

2. The reactor is assumed to be in manual control. There is no control rod insertion following the accident.

Steam Release

Also analyzed for two cases:

1. The steam dump system is assumed to be in normal automatic control.

2. No credit is taken for steam dump. The steam generator pressures rise toward the safety valve set point where steam release through safety valves limits secondary steam pressure at the set point.

Pressurizer Spray and Power Operated Relief Valves

Full credit is taken in evaluating margins to DNB for the effect of pressurizer spray and relief valves in reducing or limiting coolant pressure since this may prolong the high pressure reactor trip. A second case is analyzed where no credit is taken for pressure control, and pressureizer safety valves may be actuated during the transient.

Results

The transient responses for a total loss of load from full power operation are shown for four cases, two cases for beginning of core life and two cases for end of core life. The sensitivity of the minimum DNB ratio to heat transfer between fuel and core water is also demonstrated.

Figures 14.1.8-1 and 14.1.8-2 show the transient responses for the loss of load accident at beginning of life with zero moderator coefficient. In this transient 40 percent steam bypass capacity was assumed. Full credit is taken for the effect of pressurizer spray and relief valves in reducing or limiting coolant pressure. Credit is also taken for the effect of control rods insertion in reducing the nuclear power to prolong the time to a high pressure trip. It can be seen from the transients that the power operated relief valve capacity is large enough to limit the pressurizer pressure at 2350 psia and prevent a high pressure trip. The peak increase in coolant average temperature is about 33°F. The high level trip will be actuated at about 35 seconds following the accident. The high ΔT trip will be actuated at about 50 seconds with a minimum DNB ratio of 1.69 which is well above the 1.3 design value.

Figures 14.1.8-3 and 14.1.8-4 show the responses for total loss of load at end of life with the most negative moderator coefficient (-2.5×10^{-4} δk/°F). The rest of the plant operating conditions are the same as the case shown before. The pressurizer pressure increases to 2330 psia initially. The spray valves are fully open to limit the pressure at this value. The pressure decreases rapidly after about 30 seconds resulting from the large reduction in nuclear power. The increase in coolant average temperature is about 16°F. No reactor trip will be actuated for this case. However, the DNB ratio is higher than at the beginning-of-life condition because of the large nuclear power reduction which results from the large negative moderator coefficient and the control rod insertion.

Figure 14.1.8-5 demonstrates the sensitivity of DNB ratio to heat transfer coefficient between fuel and core water. Beginning of life conditions are used for this study which gives the most conservative DNB ratio. The heat transfer coefficient is arbitrarily increased and decreased by a factor of 2. It can be seen that change in minimum DNB ratio is in the order of 0.1.

The total loss of load accident was also studied assuming the plant is operating at full power with manual control. There is no control rod insertion following the accident. Pressurizer spray, relief valves and steam bypass valves are all ignored. The reactor is tripped on the high pressure signal which is set at 2400 psia. Figure 14.1.8-6 shows the beginning-of-life transients with zero moderator coefficient. The nuclear power remains at constant full power before the reactor is tripped. The peak pressure is 2450 psia and the maximum surge rate is about 23 ft^3/sec. This is compared to a pressurizer safety valve capacity of approximately 42 ft^3/sec. Note, however, that the trip prevents any safety valve actuation. Figure 14.1.8-7 is the transient at end of life. The nuclear power decreases before the reactor trips as a result of the large negative moderator coefficient. The peak pressurizer pressure is 2440 psia and the maximum surge rate is about 20 ft^3/sec.

Conclusions

The analysis indicates that a total loss of load without a direct or immediate reactor trip presents no hazard to the integrity of the Reactor Coolant System and the Steam System. Pressure relieving devices incorporate in the two systems are more than adequate to limit the maximum pressures. The integrity of the core is maintained by the high pressurizer pressure reactor trip. The minimum DNB ratio is 1.69 for the beginning-of-life case which is well above the 1.3 design value. At end-of-life the DNB ratio during the total loss of load transient is even better than that for the steady state, full power operating condition.

LOSS OF LOAD WITH 40 PER CENT
STEAM DUMP. BOL, ZERO MODERATOR

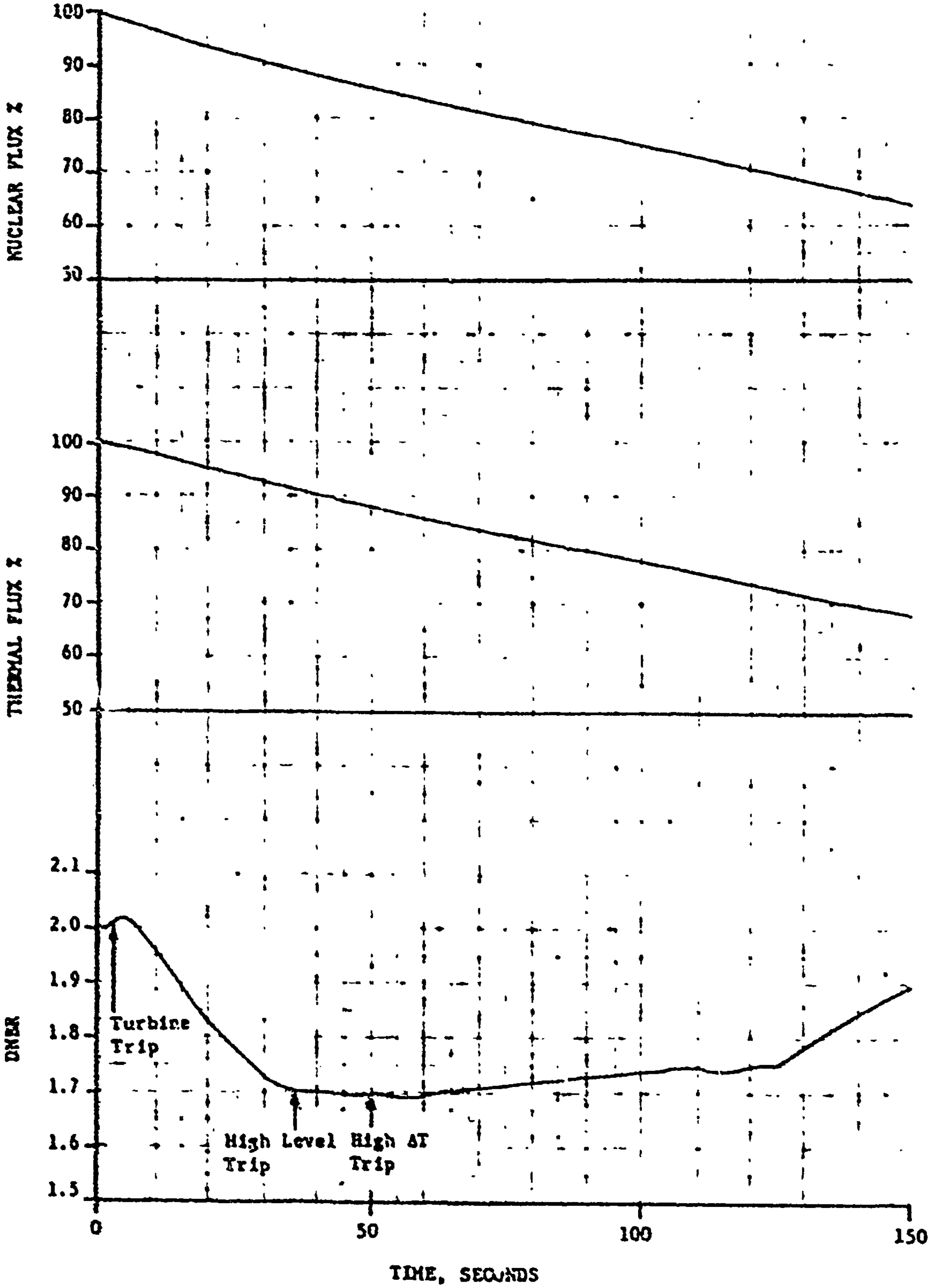


FIGURE 14.1.8-1

LOSS OF LOAD WITH 40 PER CENT
STEAM DUMP, BOL, ZERO MODERATOR

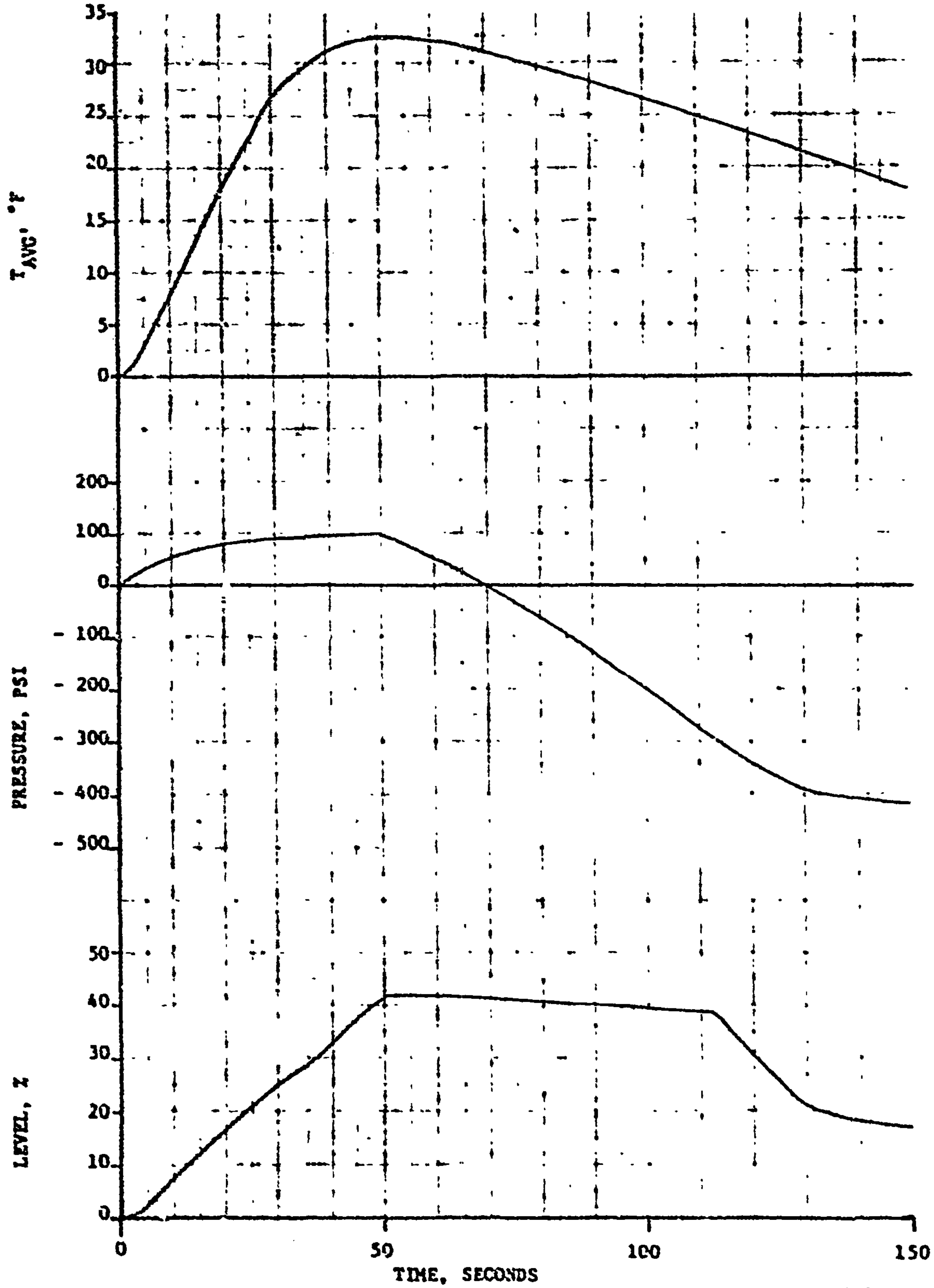


FIGURE 14.1.8-2

LOSS OF LOAD WITH 40 PER CENT
STEAM DUMP, EOL

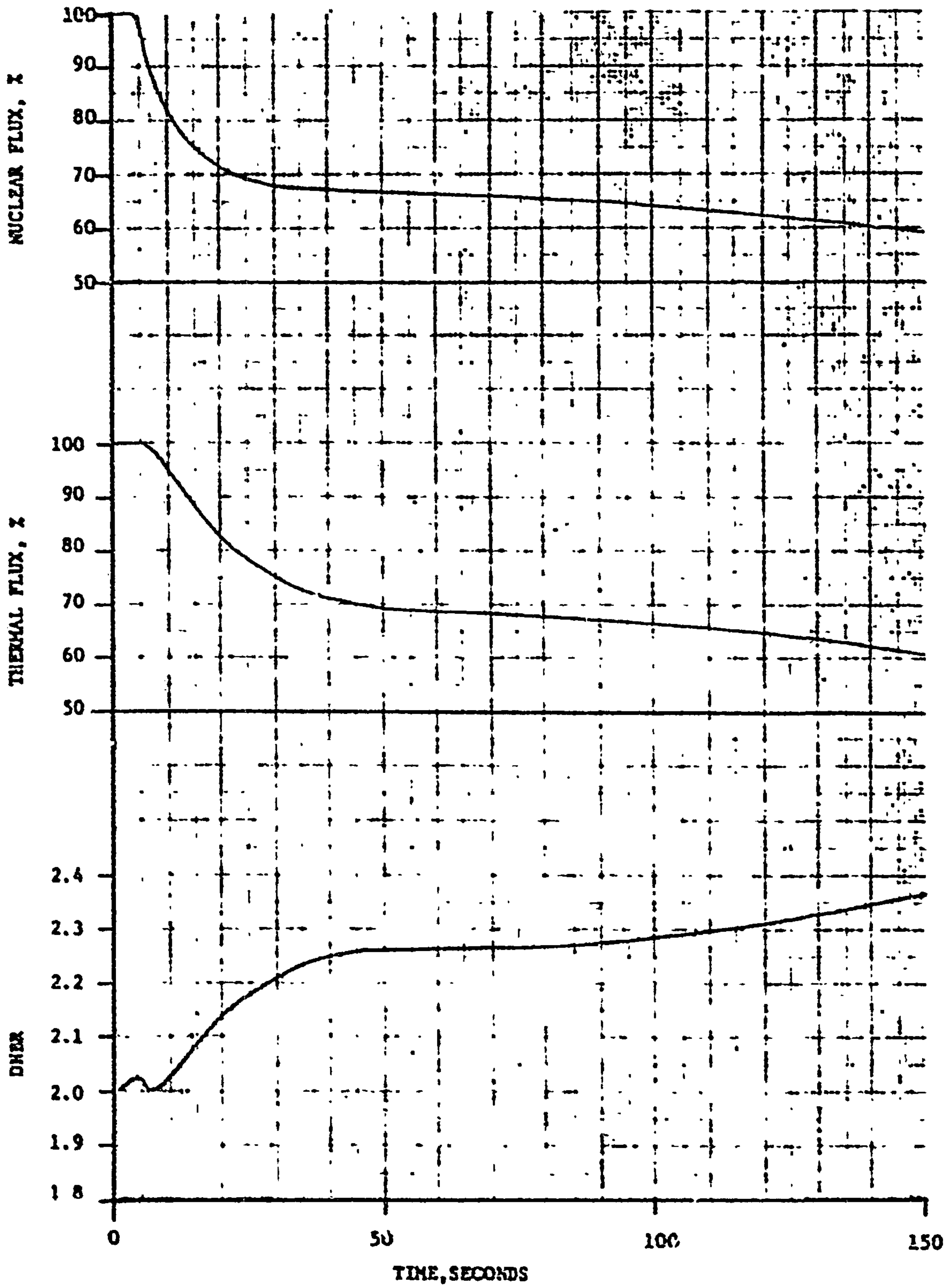


FIGURE 14.1.8-3

LOSS OF LOAD WITH 40 PER CENT
STEAM DUMP, EOL

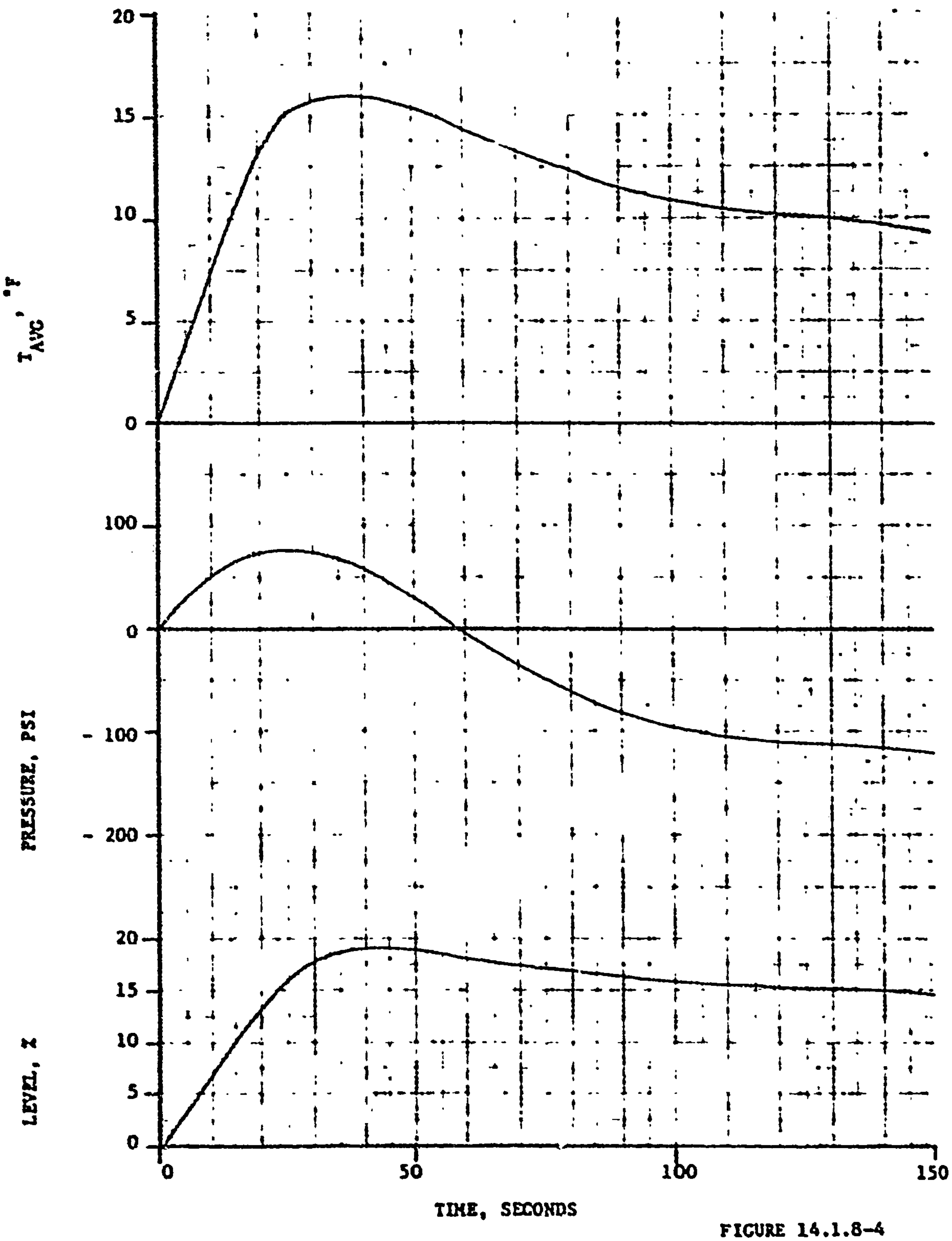


FIGURE 14.1.8-4

LOSS OF LOAD WITH 40 PER CENT STEAM DUMP
BOL, ZERO MODERATOR
SENSITIVITY TO FUEL-TO-WATER HEAT TRANSFER COEFFICIENT

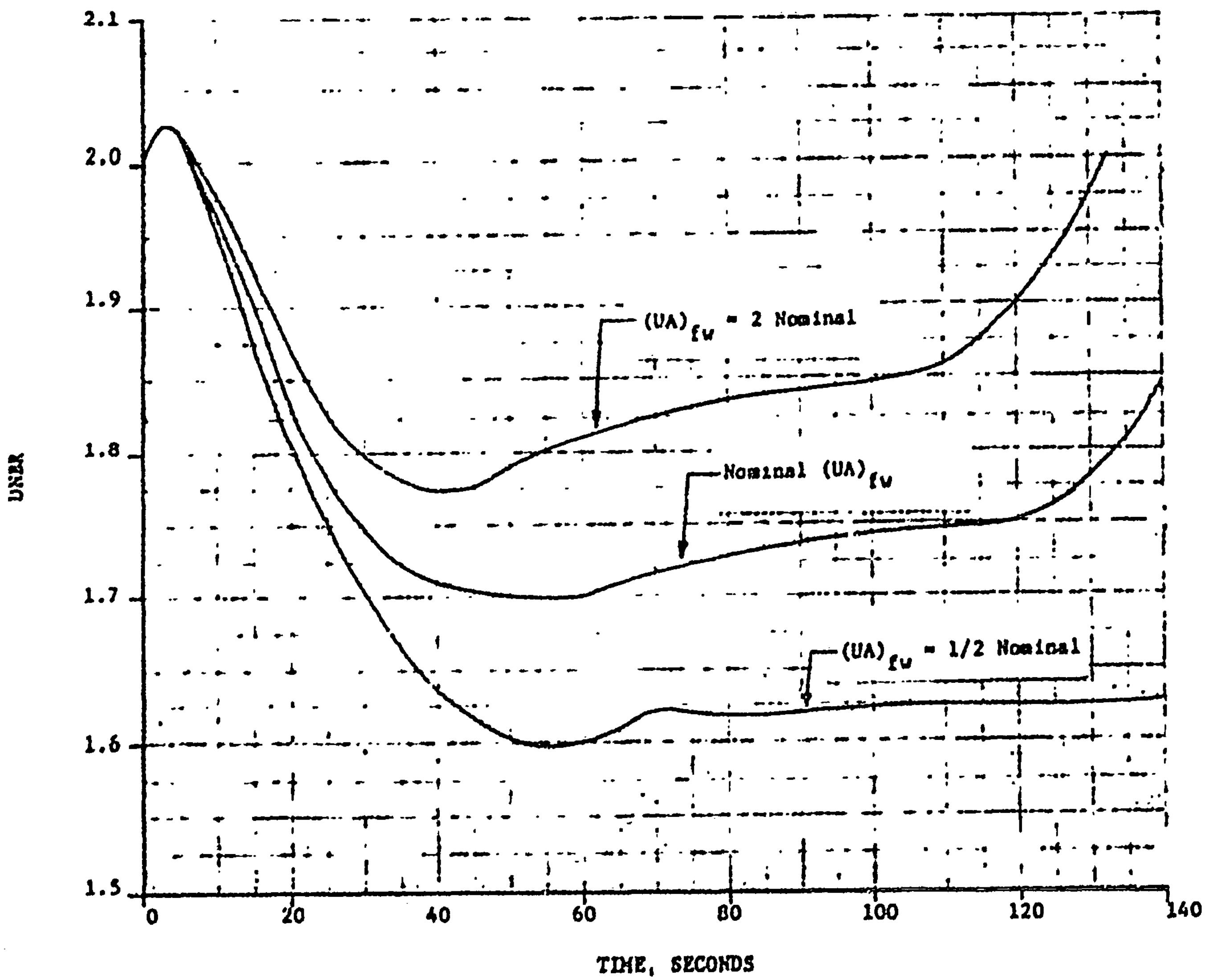


FIGURE 14.1.8-5

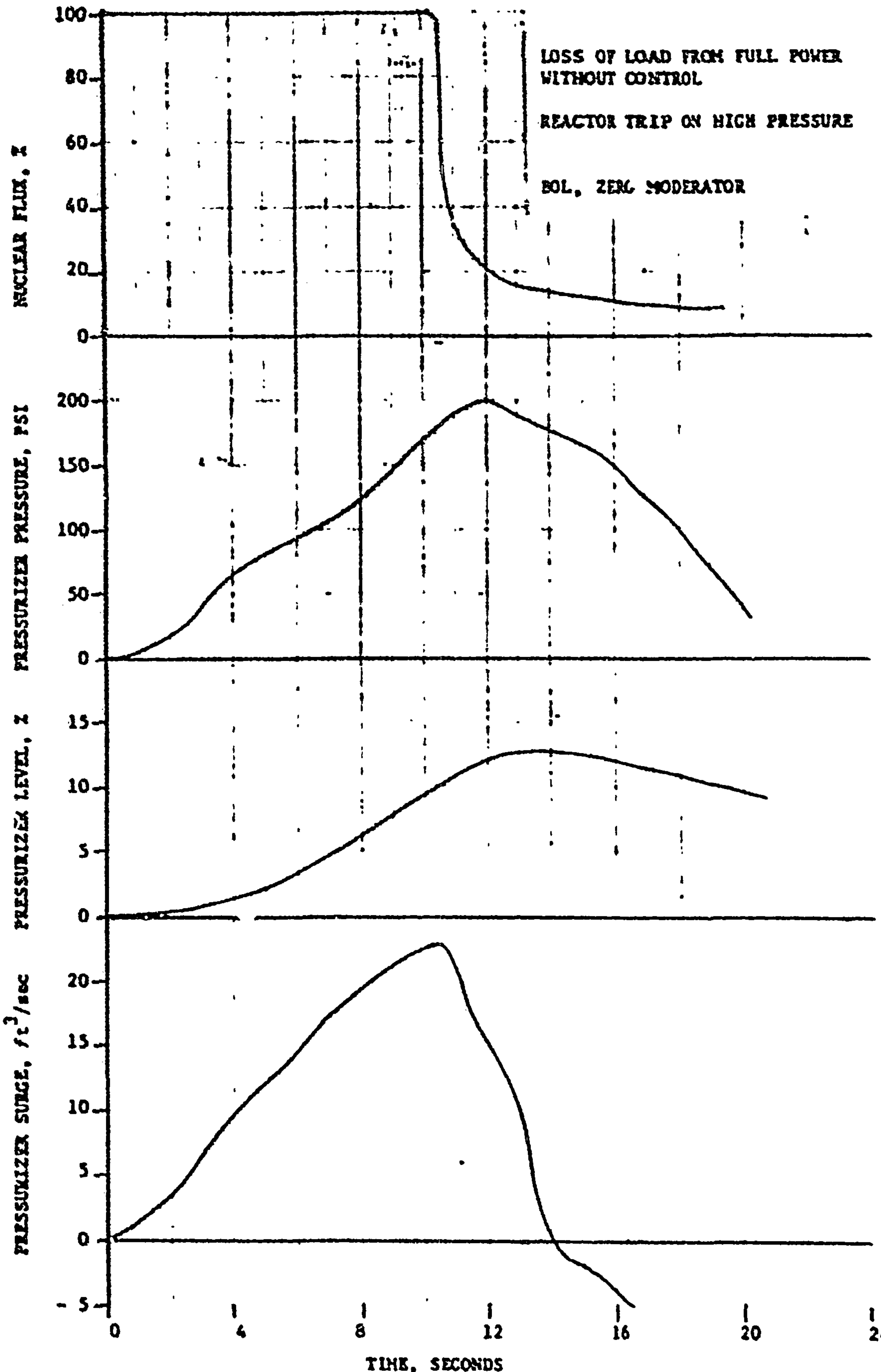


FIGURE 14.1.8-6

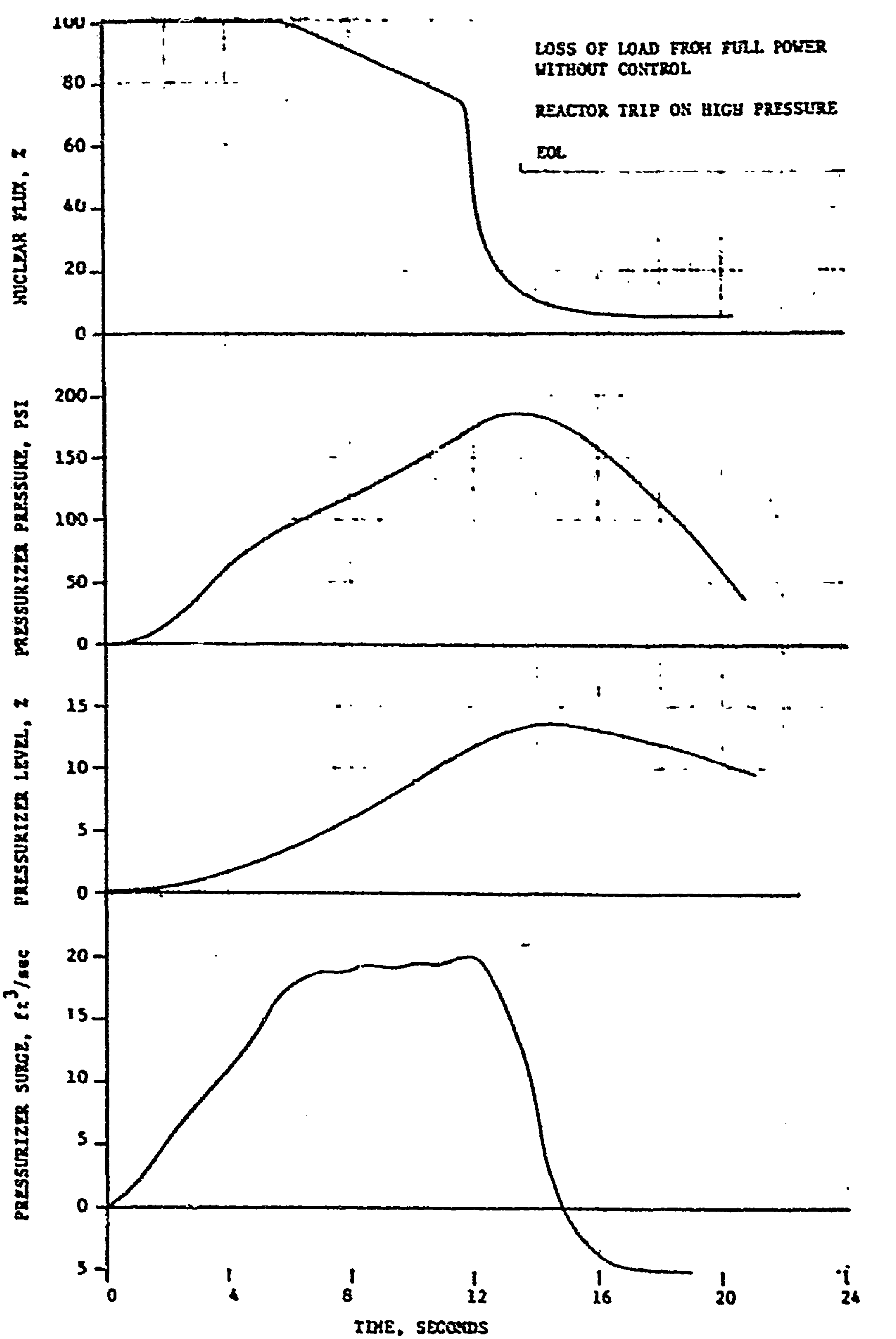


FIGURE 14.1.8-7

14.1.9 LOSS OF NORMAL FEEDWATER

A loss of normal feedwater (from a pipe break, pump failures, or valve malfunctions) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident primary plant damage could possibly occur from a sudden loss of heat sink. If an alternate supply of feedwater were not supplied, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer occurs. Loss of significant water from the Reactor Coolant System could conceivably lead to core damage. Since the plant is tripped well before the steam generator heat transfer capacity would be reduced, the primary system variables never approach a DNB condition.

The following provides the necessary protection against a loss of normal feedwater.

1. Reactor trip on very low water level in any steam generator

2. Reactor trip on steam flow-feedwater flow mismatch in coincidence with low water level in any steam generator

3. Two motor driven auxiliary feedwater pumps (400 gpm each) which are started on

   a. Low-low level in any steam generator
   b. Trip of any main feed pump turbine
   c. Any Safety Injection signal
   d. Manually
   e. Loss of outside power

4. One turbine driven pump (800 gpm) which is started on

   a. Low-low level in any two steam generators
   b. Loss of outside power
   c. Manually

The motor driven auxiliary feedwater pumps are supplied by the diesels if a loss of outside power occurs and the turbine-driven pump utilizes steam from the secondary system. Both type pumps start within one minute. The turbine exhausts the secondary the secondary steam to the atmosphere. The auxiliary pumps take suction from the condensate storage tank for delivery to the steam generators.

The above units provide considerable backup in equipment and control logic to ensure that reactor trip and automatic auxiliary feedwater flow will occur following any loss of normal feedwater including that followed by loss of outside power.

Method of Analysis

The analysis has been performed to show that following a loss of normal feedwater, the auxiliary feedwater system is adequate to remove stored and residual heat to prevent water relief through the pressurizer relief valves.

The following assumptions were made:

1. The initial steam generator water level (in all steam generators) at the tim[illegible] eactor trip occurs is at the lowest level which will result in reactor trip and automatic initiation of auxiliary feedwater flow.

2. The plant is [illegible]lly operating at 102% of 3216.5 MWt (the maximum calculated turbine rating)

3. A conservative core residual hea. generation based upon long term operation at the initial power level preceding the trip

4. Only one motor driven auxiliary feedwater pump is available at one minute after the accident

5. A conservatively low heat transfer coefficient in the steam generator assuming reactor coolant system natural circulation

6. Secondary system steam relief through the self actuated safety valves (steam relief) will, in fact, be through the power operated relief valves or condenser dump valves for most cases of loss of normal feedwater. However, these were not assumed available in the analysis.

Results

Figure 14.1.9-1 shows the plant parameters from 1500 seconds to 3500 seconds following a loss of normal feedwater accident with the assumptions listed above. Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow continues to dissipate the stored and generated heat. One minute following the beginning of the accident the auxiliary feedwater pump is automatically started reducing the rate of water level decrease. The capacity of the auxiliary feedwater pump is such that the water level in the steam generators being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the primary system relief or safety valves.

From Figure 14.1.9-1 it can be seen that at no time is there water relief from the pressurizer. The assumption of more auxiliary feed capacity than that of one motor driven pump, a lower reactor power (2758 MWt) or one steam generator water level initially above the low-low level trip will of course result in increased margin to the point at which reactor coolant water relief occurs.

TRANSIENT RESPONSE FOLLOWING A LOSS OF NORMAL
FEEDWATER WITH ONE 400 GPM AUXILIARY FEED PUMP
DELIVERING TO TWO STEAM GENERATORS BEGINNING
AT 1 MINUTE

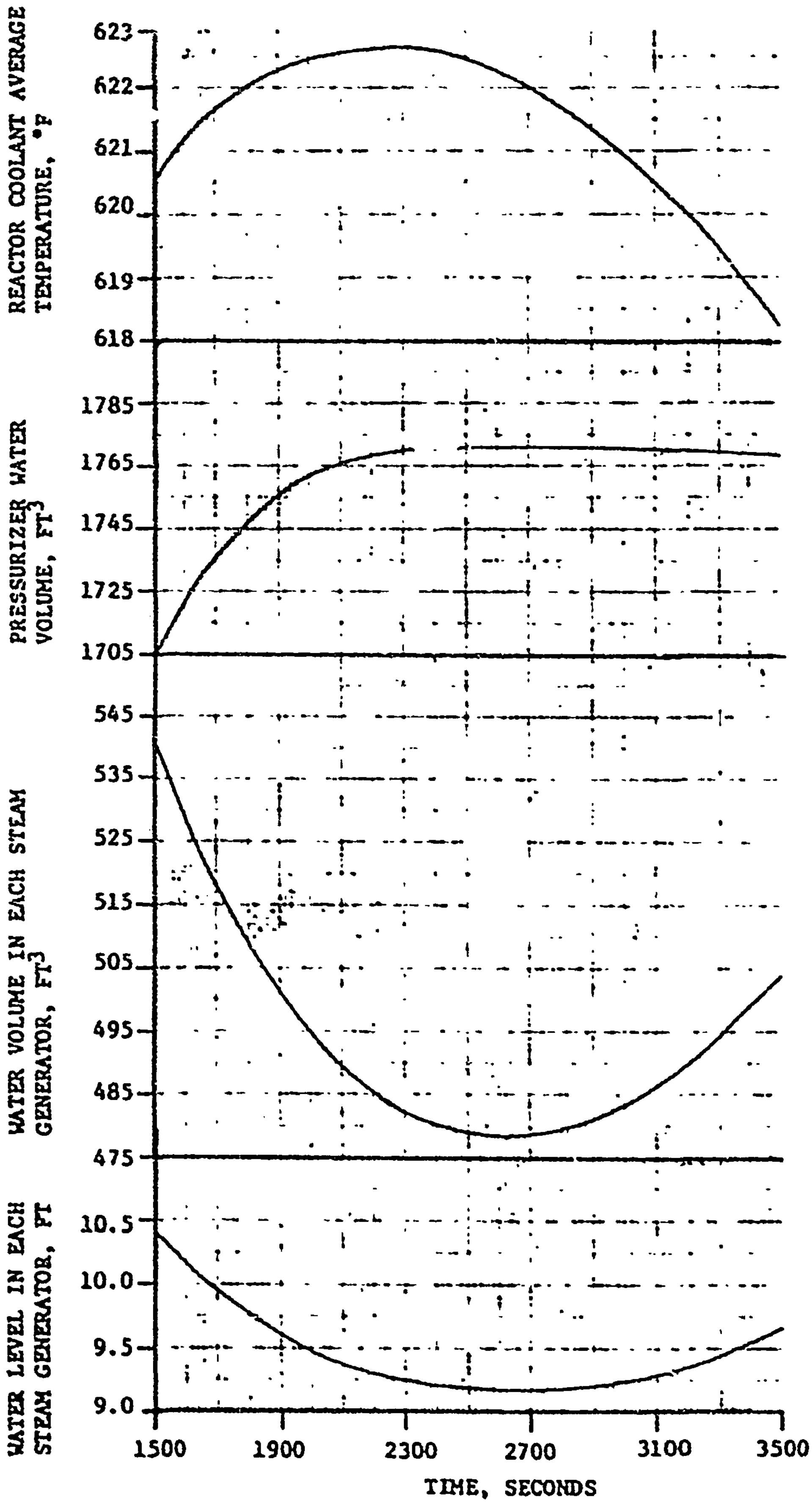


FIGURE 14.1.9-1

14.1.10 REDUCTION IN FEEDWATER ENTHALPY INCIDENT

The reduction in feedwater enthalpy is another means of increasing core power above full power. Such increases are attenuated by the thermal capacity in the secondary plant and in the Reactor Coolant System. The overpower-overtemperature protection (nuclear overpower and ΔT trips) prevents any power increase which could lead to a DNBR less than 1.30.

An extreme example of excess heat removal by the feedwater system is the transient associated with the accidental opening of the feedwater bypass valve which diverts flow around the low pressure feedwater heaters. The function of this valve is to maintain net positive suction head on the main feedwater pump in the event that the heater drain pump flow is lost, e.g., during a large load decrease.

In the event of accidental opening there is a sudden reduction in inlet feedwater temperature to the steam generators. The increased subcooling will create a greater load demand on the primary system which can lead to a reactor trip. The 3-element feedwater control system operates to regulate the feedwater flow and maintain a water level approximately constant in the steam generator.

Method of Analysis

Two cases have been analyzed to demonstrate the plant behavior in the event of a sudden feedwater temperature reduction resulting from accidental opening of the bypass valve. The first case was for an uncontrolled reactor with a zero moderator coefficient, since this represents a condition, where the plant has the least inherent transient capability. The second case was for a controlled reactor with a large negative coefficient. These results were obtained by means of a detailed digital simulation of the plant including core kinetics, Reactor Coolant System and the Steam System. Both transients were assumed to occur from full power.

Results

Figure 14.1.10-1 shows the transient without automatic control. As expected the reactor coolant average temperature and pressurizer pressure show a fairly rapid decrease as the secondary heat extraction exceeds the core power generation. The core power level remains essentially constant at full load. The fixed low pressure trip would occur at about 160 seconds. There is a considerable margin to DNB because of the accompanying large reduction in average temperature. The DNBR at the time of trip is approximately 1.8. There is a small increase in ΔT as the heat transfer increases through the steam generator.

Figures 14.1 10-2 and 14.1.10-3 illustrate the transient assuming automatic reactor control is functioning. A large negative moderator coefficient is assumed, which acts to increase power. The core power is increasing thus reducing the rate of decrease in coolant average temperature and pressurizer pressure. The results are shown with no trip actuation and steady state conditions are reached with a minimum DNBR greater than 1.5. The plant would actually be tripped from the overpower protection. A greater increase in T_{avg} would lead to an over-temperature trip as well.

Conclusions

Representative transient results for excessive load increases due to cold feedwater addition have been shown which indicate the general behavior, i.e., that a core power increase is accompanied by an average temperature decrease and without a power increase there is a larger reduction in coolant average temperature. This has the effect of maintaining considerable margin to a limiting DNBR of 1.30. Core protection for slow increases in plant output in excess of full power is provided by the combination of the overpower-overtemperature protection described in conjunction with the rod withdrawal accident. There is no radioactive release and thus no public hazard in the event of an excessive load increase. An evaluation of an accidental full opening of both feedwater control valves has been performed. The consequences of this incident have been found less severe than those resulting from the opening of the feedwater bypass valve described above.

TRANSIENT RESPONSE TO
OPENING OF FEEDWATER BYPASS VALVE
NO CONTROL, $\alpha_{mod} = 0$

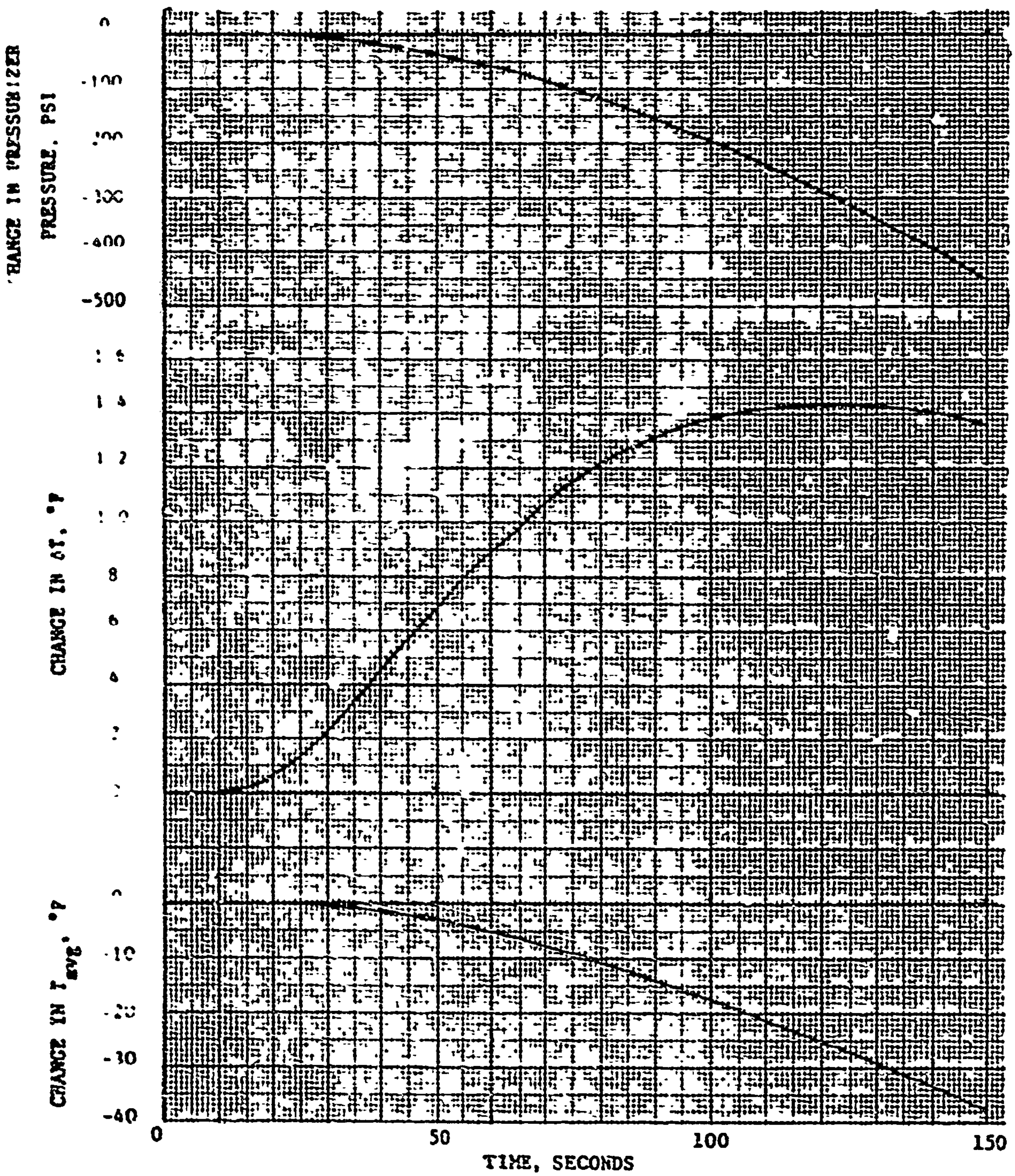


FIGURE 14.1.10-1

TRANSIENT RESPONSE TO

OPENING OF FEEDWATER BYPASS VALVE

$\alpha_{mod} = -3.5 \times 10^{-4}$

WITH CONTROL

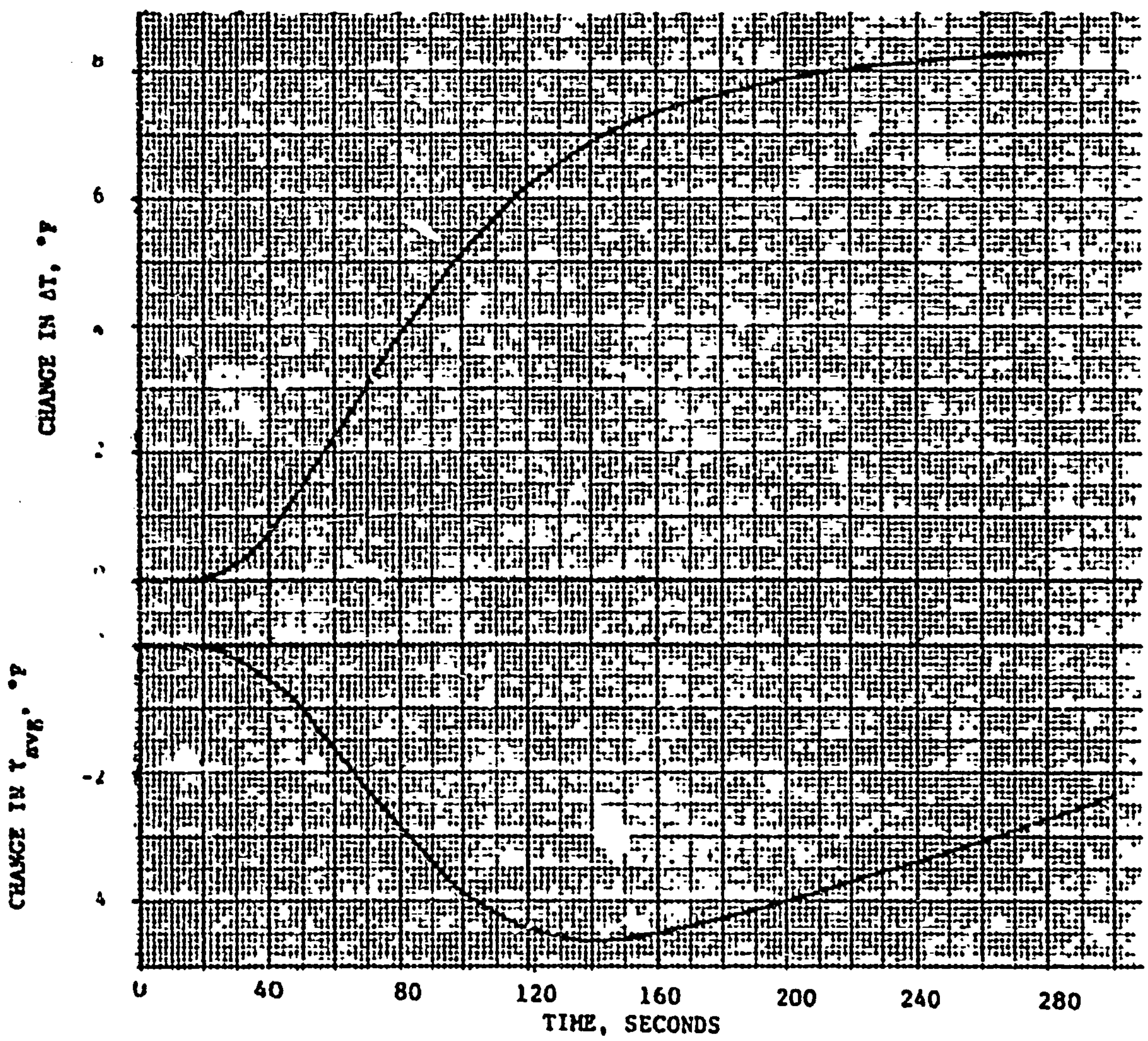


FIGURE 14.1.10-2

TRANSIENT RESPONSE TO

OPENING OF FEEDWATER BYPASS VALVE

$\alpha_{mod} = -3.5 \times 10^{-4}$

WITH CONTROL

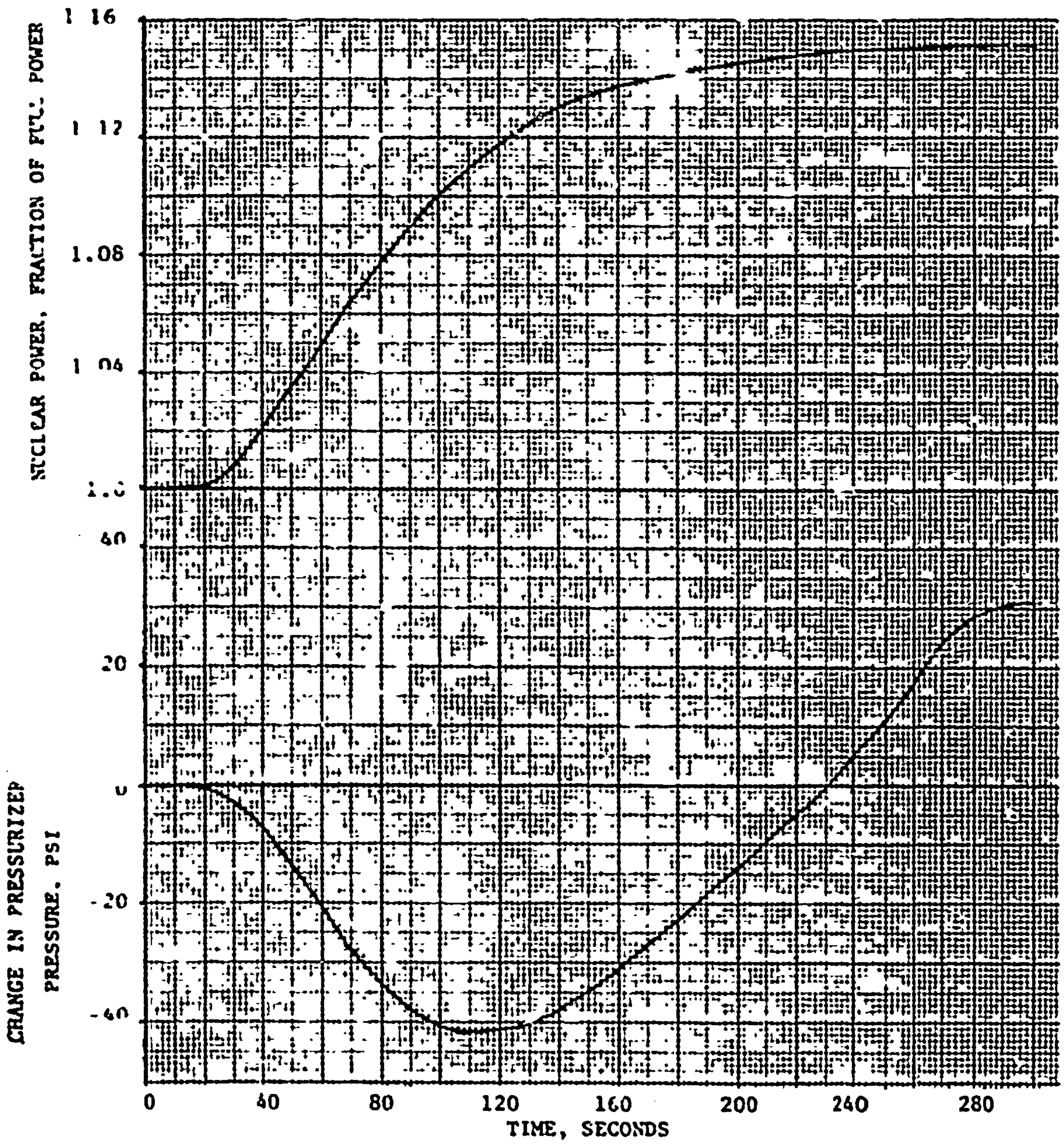


FIGURE 14.1.10-3

14.1.11 EXCESSIVE LOAD INCREASE INCIDENT

An excessive load increase incident is defined as a rapid increase in steam generator steam flow causing a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10 per cent step load increase and a 5 per cent per minute ramp load increase without a reactor trip in the range of 15 to 100 per cent full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system. If the load increase exceeds the capability of the reactor control system, the transient is terminated in sufficient time to prevent the DNBR from going below 1.3 since the core is protected by the combination of the nuclear overpower trip and the overpower-overtemperature trips as discussed in Section 7. An excessive load increase incident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction such as steam bypass control or turbine speed control.

In case of excessive loading by the operator or by system demand, the turbine load limiter limits maximum turbine load to 100% rated load. The turbine controls are discussed in Section 10.2.3.

During power operation, steam bypass to the condenser is controlled by signals of reactor coolant conditions, i.e., abnormally high reactor coolant temperature indicates a need for steam bypass. A single controller malfunction does not cause steam bypass because an interlock is provided which blocks the control signal to the valves unless a large turbine load decrease has occurred.

Increases in steam load to more than rated load are analyzed as steam line ruptures in Section 14.2.5. However, the Reactor Protection System will trip the reactor in time to prevent DNBR less than 1.30, regardless of the magnitude or rate of load increase.

Method of Analysis

Two cases have been analyzed to demonstrate the plant behavior in the event of excessive load increases. These results were obtained by means of a detailed digital simulation of the plant including core kinetics, Reactor Coolant System and the Steam System. Both transients were assumed to occur from full power where the margins to core limits are the smallest*. A zero moderator coefficient of reactivity was assumed as this represents the condition where the plant has the least inherent (uncontrolled) transient capability. The results of a 10% step increase in turbine load are presented with and without automatic control.

Results

Figure 14.1.11-1 shows the transient without automatic control. As expected, the reactor coolant average temperature and pressurizer pressure show a fairly rapid decrease as the secondary heat extraction exceeds the core power generation. The fixed low pressure trip would occur at about 150 seconds. There is a considerable margin to DNB because of the accompanying large reduction in average temperature. The DNBR at the time of trip is approximately 1.8. There is a small increase in ΔT as the heat transfer increases through the steam generator. The core power level remains essentially constant at full power.

Figure 14.1.11-2 and 14.1.11-3 illustrate the transient assuming automatic reactor control is functioning. The core power is increasing thus reducing the rate of decrease in coolant average temperature and pressurizer pressure. With no trip actuation steady state conditions are reached with a minimum DNBR greater than 1.66. This margin occurs because there is only a nominal increase in coolant temperature. For greater power increases and/or larger temperature increases, protection is provided by the combination overpower-overtemperature protection described in Section 7.

Conclusions

Representative transient results for excessive load increases have been shown which indicate the general behavior, i.e., that a core power increase is accompanied by an average temperature decrease and without a power increase there is a larger reduction in coolant average temperature. This has the effect of maintaining considerable margin to a limiting DNBR of 1.30. Core protection for slow increases in plant output in excess of full power is provided by the combination of the overpower-overtemperature protection described in conjunction with the rod withdrawal accident. There is no radioactive release and thus no public hazard in the event of an excessive load increase.

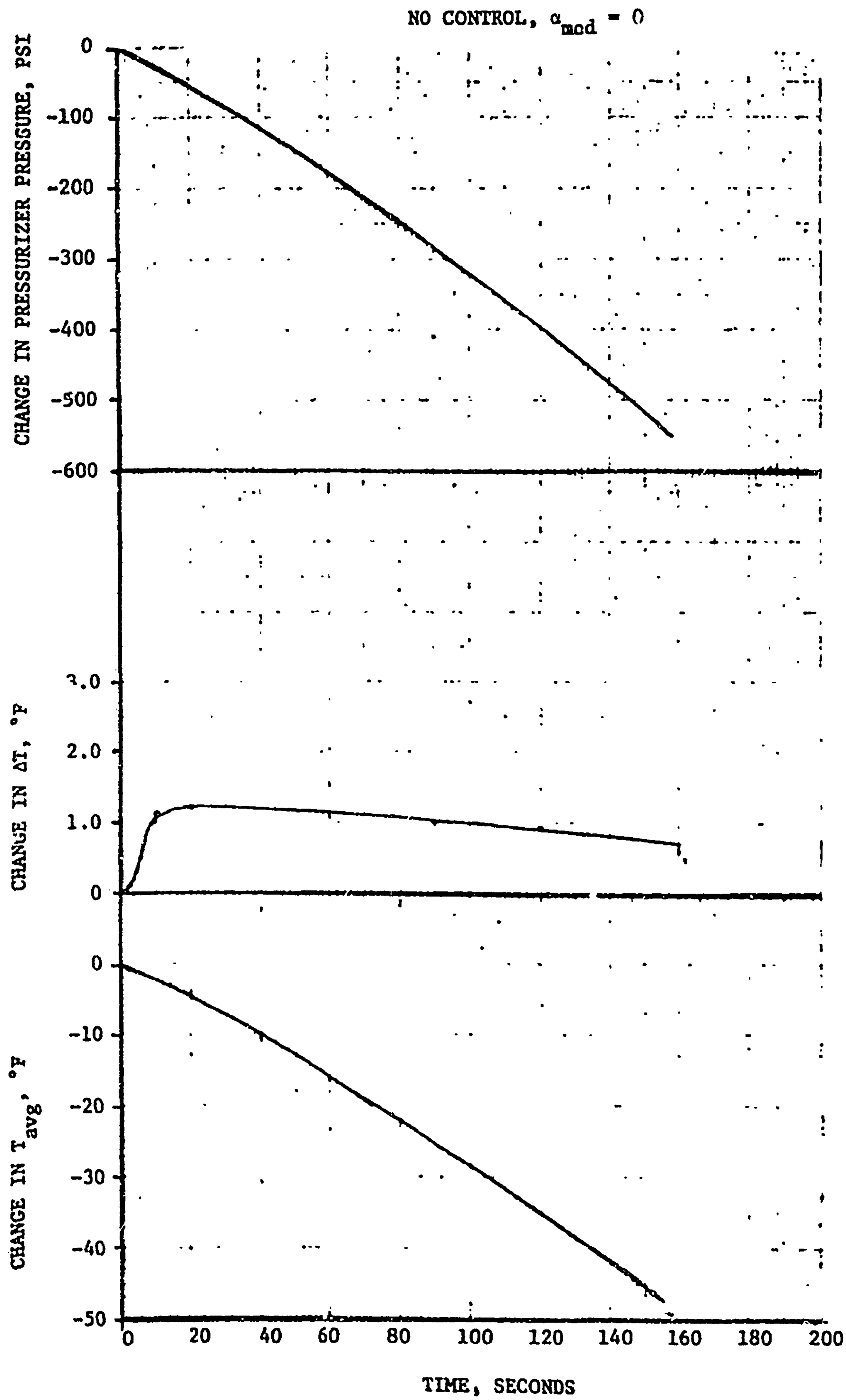


FIGURE 14.1.11-1

TRANSIENT RESPONSE TO
10% STEP LOAD INCREASE

WITH CONTROL

$\alpha_{mod} = 0$

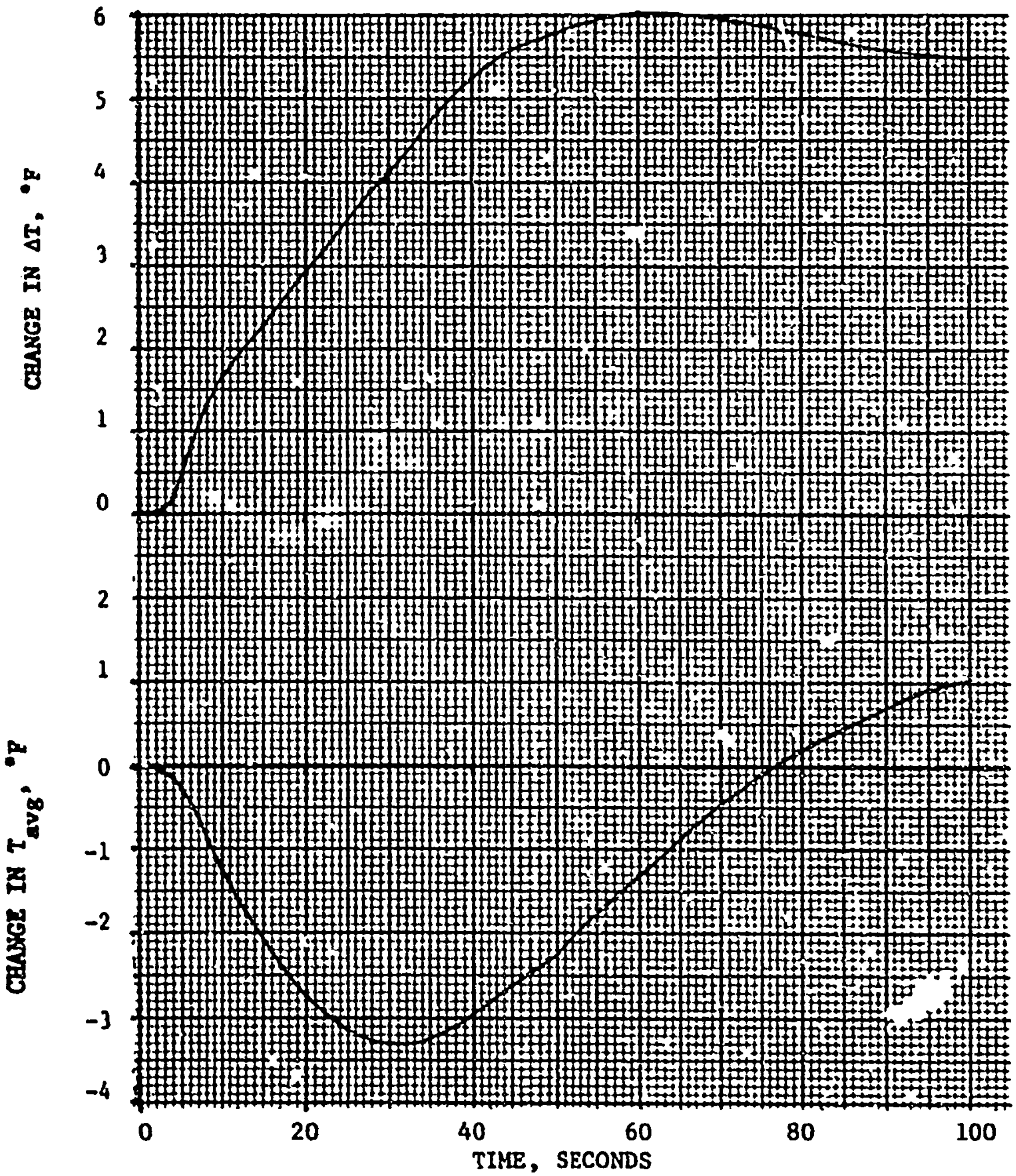


FIGURE 14.1.11-2

WITH CONTROL $\alpha_{mod} = 0$

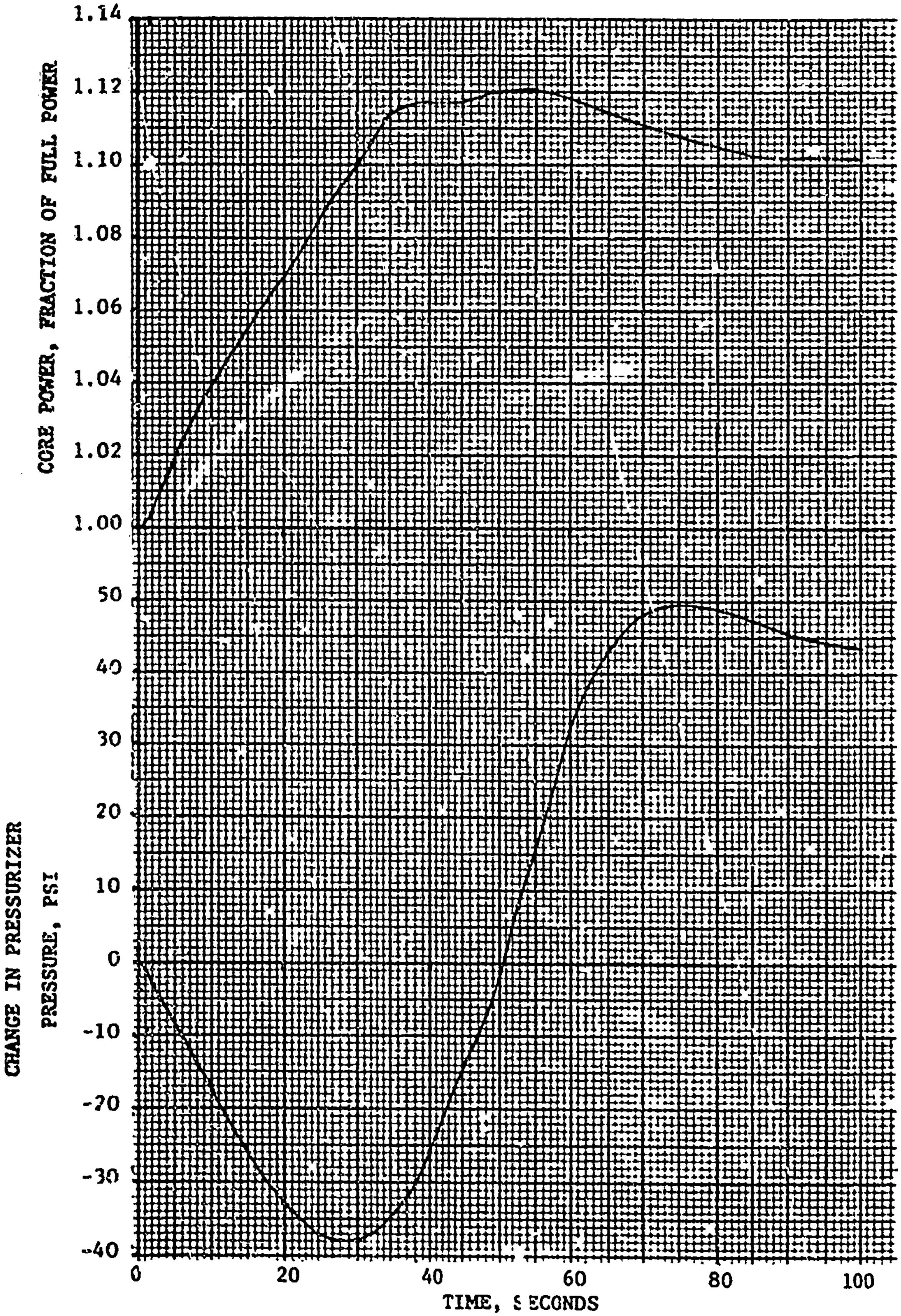


FIGURE 14.1.11-3

14.1.12 LOSS OF ALL A.C. POWER TO THE STATION AUXILIARIES

In the unlikely event of a complete loss of all off-site a-c power and turbine trip while the reactor plant is at power, the reactor is tripped. The first few seconds of the transient would be almost identical to the four pump loss of flow case presented in Section 14.1.6, that is, the pump coastdown inertia and reactor trip would result in a DNBR $\geq$ 1.3. After the trip, decay heat will be accommodated by the emergency feedwater system. This portion of the transient would be similar to that presented in Section 14.1.9 loss of normal feedwater.

a. Plant vital instruments are supplied by the emergency power sources. (See Section 8)

b. As the steam system pressure subsequently increases, the steam system power relief valves are automatically opened to the atmosphere. Steam bypass to the condenser is not available because of loss of the circulating water pumps.

c. As the steam flow rate through the power relief valves may not be sufficient, the steam generator self-actuated safety valves may temporarily lift to augment the steam flow until the rate of heat dissipation is sufficient to carry away the sensible heat of the fuel and coolant above no-load temperature plus the residual heat produced in the reactor.

d. As the no-load temperature is reached, the steam system power relief valves are used to dissipate the residual heat and to maintain the plant at the hot shutdown condition.

The loss of normal feedwater supply signals the start of the auxiliary feedwater pumps. The turbine driven pump utilizes steam from the secondary system to drive the feedwater pump to deliver makeup water to the steam generators. The turbine driver exhausts the secondary steam to the atmosphere. The electric motor driven auxiliary feedwater pumps are supplied power by the diesel generators. The pumps take suction from the condensate storage tank for delivery to the steam generators.

Following the turbine trip, there is a rapid reduction of steam generator water level. This is due to the reduction of steam generator void fraction on the secondary side and because steam flow continues after normal feedwater stops. By one minute, flow is established from at least one auxiliary feedwater pump and further reduction of water level is slow. The capacity of the auxiliary feedwater pump is selected to prevent the water level in the steam generators being fed from receding below the lowest level within the indicator range during the transient. This prevents the tube sheet from becoming uncovered at any time during the transient. The reactor operator in the control room monitors the steam generator water level and controls the feedwater addition with remote operated auxiliary feedwater control valves.

The steam driven feedwater pump can be tested at any time by admitting steam to the turbine driver. The electrically driven auxiliary feedwater pumps also can be tested at any time. The auxiliary feedwater control valves and power relief valves can be operationally tested whenever the plant is at hot shutdown and the remaining valves in the system are operationally tested when the turbine driver and pump are tested.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. The natural circulation capability for the unit has been calculated for the conditions of equilibrium flow and maximum loop flow impedence. The analytical model used to calculate the natural circulation flow has given results within 15% of the measured flow values obtained during natural circulation tests conducted at the Yankee-Rowe plant and has also been confirmed at San Onofre and Connecticut Yankee. The natural circulation flow ratio as a function of reactor power is given in Table 14.1.12-1.

The average temperature, pressurizer water volume, steam generator water volume, and steam generator level assuming the most conservative initial plant conditions and equipment availability are shown in Figure 14.1.9-1 of "Loss of Normal Feedwater Accident." It is shown in that discussion that a loss of all off-site a-c power to the station auxiliaries and a loss of normal feedwater does not result in water relief from the pressurizer relief or safety valves.

TABLE 14.1.12-1

NATURAL CIRCULATION REACTOR COOLANT
FLOW VS REACTOR POWER

Reactor Power %	Reactor Coolant Flow % Nominal
0.5	2.3
1.0	2.9
1.5	3.3
2.0	3.7
5.0	5.0
10.0	6.2

14.1.13 LIKELIHOOD AND CONSEQUENCES OF TURBINE-GENERATOR UNIT OVERSPEED

General

The likelihood of a turbine-generator unit overspeeding is very remote because of the reliability and redundancy of the turbine control and protection system.

Should all the main steam admission valves fail to close on a full load rejection, a unit runaway would take place, causing some damage of the rotating parts. However, no missiles are anticipated to leave the unit, for analysis of the potential mode of failure shows that all parts would be contained within the unit casings.

This system is completely hydraulic. There are two low pressure oil control systems, i.e., auxiliary governor system and emergency trip system.

These two systems and the 300 psi system are interconnected through orifices.

The control and protection system is fail-safe: any loss of oil pressure causes closure of the steam valves.

The main governor normally controls the unit. Should an overspeed take place, the auxiliary governor system will be actuated first, the auxiliary governor dome valve will open, the 300 psi pressure oil will drain, and the control valve will close.

Should the unit overspeed reach the mechanical overspeed trip set point, the overspeed trip valve will open, the 300 psi pressure oil will drain, and the throttle valves will close. At the same time, a second drain path will be provided for the 300 psi oil system that controls the first set of valves, so that the control valves will trip too, in case they did not trip.

Assuming, for the purpose of analysis, that all the above mentioned valves fail to close, a turbine runaway occurs. The first disks to fail are the low-pressure turbine disks closest to the steam admission As these disks burst, the unit will be decelerated because the steam flow between blades of the remaining disks will decrease significantly. The analysis of the energy required to violate the integrity of the low pressure turbine casings shows that there is a large margin between the kinetic energy of the broken parts of the burst disks at their bursting speed and the energy required to penetrate the low pressure turbine cylinders.

The stress analysis of the remaining disks shows that their bursting speed is at most 15% higher than the bursting speed of the first disks. The assumption is made that even these disks might fail at the bursting speed of the first disks. A plastic analysis of the energy required to penetrate the low pressure turbine casings shows that even the broken parts of these disks will not have enough energy to be ejected outside the casing.

The stress analysis of the high pressure turbine spindle shows that its bursting speed is at least 50% higher than the maximum speed at which the turbine can rotate. This large margin keeps the probability of a high pressure turbine spindle bursting to practically zero.

It is worthwhile to point out that due to conservative design, very careful rotor forging procurement and rigid inspection, Westinghouse turbine-generator units have never experienced such a massive failure.

A survey of the available literature on turbine-generator unit failure shows that the last massive failure of a turbine-generator unit occurred about eight years ago. The cuases of failure were identified at that time, and provisions were adopted to prevent the recurrence of massive failures. The record since that time demonstrates the soundness of these provisions and correct design.

The no-failure record of Westinghouse turbine generator units, plus the experience gained from the referenced incidents, together with the improvement in the design and inspection techniques in the past eight years indicates that the likelihood of massive turbine-generator failure is extremely remote.

With regard to design and inspection techniques, it is worthwhile to mention that a technical committee of forging suppliers and equipment manufacturers was formed about ten years ago under ASTM to study turbine and generator rotor failures. This group developed the high-toughness NiCrMoV material, now used in all turbine rotors and disks. This Task Force[(1)] has been very active in making additional improvements in quality and soundness of large forgings and is still in force.

The survey of the literature on massive turbine failures in the last 20 years indicates that all of them occurred between 1953 and 1958.

This survey has pointed out that the rare events of a catastrophic failure of turbines fell into one of two categories:

1) Failure by overstressing arising from accidental and excessive overspeed, and
2) Failure, due to defects in the material, occurring at about normal speed

No failure falling in the first category occurred in the USA. The only two documented examples occurred in the United Kingdom. Both incidents were caused by the main steam admission valves sticking in the open position after full load rejection, because of impurities in the turbine control and lubrication oil. The probability of this occurrence in this plant is very remote as previously pointed out.

Besides the provisions in the design of the turbine control and protection system during plant operation, valves will be exercised on a periodic basis, to further preclude the possibility of a valve stem sticking. Analysis of oil samples will be performed as required.

The turbine is periodically overspeeded to check the tripping speed. The remaining tripping devices are periodically checked.

The causes of the failures that fall in the second category, i.e., failures due to defect in the material occurring about normal speed, were completely identified and, if the ultrasonic test were used as one of the bases for rejection or acceptance of forging, many of them would not have occurred. Further, the stress concentration points that initiated failure in some units are strictly correlated to the peculiar design characteristics of those units. These discontinuities are not present in Westinghouse units.

Westinghouse specifies the quality and method of manufacturing of the purchased forgings. Written specifications cover the manufacturing process, the chemical and mechanical properties, the test to be performed, etc. Specifically, the tests performed are both destructive and non-destructive in nature. The destructive tests include tension tests, impact tests, and transition temperature measurement tests. The tension specimens are taken in a radial and/or longitudinal direction. The tensile properties are determined in accordance with ASTM A-370 on a Standard Round 1/2 inch Diameter 2 Inch Gage Length Test specimen. The yield strength is taken as the load per unit of original cross section at which the material exhibits an offset of 0.2 per cent of the original length. The Charpy impact specimens are taken in a radial direction, and the minimum impact strength at room temperature measured. The transition temperature is determined from 6 specimens are taken in a radial direction and machined in such a manner that the V-notch is parallel to the forging axis. Two specimens are machined from each test bar. All specimens are taken following all heat treatment. Curves of impact strength and per cent brittle failure versus test temperature are drawn.

The non-destructive tests include bore inspection, sulfur printing, magnetic particle test, thermal stability test, and ultrasonic test.

The bores are visually inspected and the walls of the finished bores shall be free from cracks, pipe shrinkage, gas cavities, non-metallic inclusions, injurious scratches, tool marks and similar defects.

A magnetic particle test is made on each forging to demonstrate the freedom from surface discontinuities. The end faces of the main body and down over and beyond the fillets joining the main body to the shaft portions are magnetic particle tested. The bore is also magnetic particle tested at a high sensitivity level in accordance with ASTM A-275. These inspecttions are done by Westinghouse inspectors prior to Westinghouse accepting these forgings. After final machining by Westinghouse, rotors are again magnetic particle inspected on the external surfaces by Westinghouse.

The face of the test prolongations at each end of the rotor body or an area on the end faces of the rotor body equivalent to the test prolongations in sulfur printed to determine the freedom from undue into corners segragation and excessive sulfide inclusions.

A thermal stability test is performed on the forging at the place of manufacture after all heat treatment has been completed.

The forgings are ultrasonically inspected at the place of manufacture by Westinghouse inspectors.

Based on conservative design, reliable turbine control system, careful rotor forging procurement and rigid inspection, the probability of a combination of excessive overspeed, new-born large forging defects, and operating temperature below the transition temperature is considered practically zero, as confirmed by years of major-failure-free operation of the many Westinghouse units.

Analysis and Results

As stated above, the possibility of a turbine overspeed is extremely remote and that if an overspeed condition is postulated and a failure of the turbine is also postulated, the associated missiles would be contained within the casing. Following is a detailed analysis of this situation.

The present status of the manufacturing technology of rotor forging and inspection techniques guarantee practically defect-free turbine rotors. Further, the Westinghouse design is conservative and eliminates any harmful stress-concentration points. This has been confirmed historically, since no failures of Westinghouse turbine-generators rotors have occurred.

The only envisaged operating condition that might lead to turbine-generator failure is an excessive overspeed of the rotating parts. Due to the redundancy and reliability of the turbine control and protection systems, and of the steam system, the possible occurrence of a unit significantly overspeeding above the design value, i.e., 115% is very remote.

The consequences of the turbine-generator runaway, caused by all the steam admission valves stuck fully open upon a full load rejection, have been evaluated, for purpose of analysis. As it will be shown later, missiles will be generated only from the rotating parts of the low-pressure turbine, but they will not have sufficient energy to penetrate through the turbine casings.

As stated above, no critical structure or component will be put in jeopardy by a turbine-generator unit failure and thus no special provision has to be taken to ensure the "no loss of function" of the above mentioned critical structures and components.

General Description of the Turbine Units

High Pressure Turbine

The high pressure turbine element, shown in Figure 14.1.13-1 is of a double flow design; therefore, it is inherently thrust balanced. Steam from the four control valves enters at the center of the turbine element through four inlet pipes, two in the base and two in the cover. These pipes feed four double flow nozzle chambers flexibly connected to the turbine casing. Each nozzle chamber is free to expand and contract relative the adjacent chambers.

Steam leaving the nozzle chambers passes through the rateau control stages and then flows through the reaction blading. The reaction blading is mounted in blade rings shown in Figure 14.1.13-2, which in turn are mounted in the turbine casing. The blade rings are centerline supported to insure center alignment while allowing for differential expansion between the blade ring and the casing The design reduces casing thermal distortion and thus, seal clearances are more readily maintained.

Steam exhausts from the high pressure turbine base, through cross-under piping, to the two combined moisutre separator live steam reheater assemblies.

The high-pressure rotor is made of NiCrMoV alloy steel. The specified minimum mechanical properties are given in Table 14.1.13-1.

TABLE 14.1.13-1

MINIMUM MECHANICAL PROPERTIES - HIGH PRESSURE ROTOR

Tensile Strength, psi, min.	100,000
Yield Strength, psi, min. (0.2% offset)	80,000
Elongation in 2 inches, per cent, min.	18
Reduction of Area, per cent, min.	45
Impact Strength, Charpy V-Notch, ft-lb (min. at room temperature)	60
50% Fracture Appearance Transition Temperature, °F, Max.	50

The main body of the rotor weight is approximately 100,000 lb. The approximate values of the transverse centerline diameter, the maximum diameter, and the main body length are 36", 66" and 138" respectively.

The blade rings and the casing cover and base are made of carbon steel casings. The specified minimum mechanical properties are given in Table 14.1.13-2.

TABLE 14.1.13-2

MINIMUM MECHANICAL PROPERTIES - CASINGS

Tensile Strength, psi, min.	70.000
Yield Strength, psi, min.	36,000
Elongation in 2", per cent, min.	22
Reduction of Area, per cent, min.	35

The bend test specimen shall be capable of being bent cold through an angle of 90 degrees and around a pin one inch in diameter without cracking on the outside of the bent portion.

The approximate weight of the four blade rings, the casing cover, and the casing base is 80,000 lb., 115,000 lb., and 115,000 lb., respectively.

The casing cover and base are tied together by means of more than 100 studs. The stud material is an alloy steel having the mechanical properties given in Table 14.1.13-3.

TABLE 14.1.13-3

MECHANICAL PROPERTIES - STUD MATERIAL

	Size, Inches		
	2-1/2 and less	Over 2-1/2 to 4 inch	Over 4 to 7 inch
Tensile Strength, psi, min.	125,000	115,000	110,000
Yield Strength, psi, min(0.2% offset)	105,000	95,000	85,000
Elongation in 2 inches, per cent, min.	16	16	16
Reduction of Area, per cent, min.	50	50	50

The studs have length ranging from 17 to 66 inches and diameter ranging from 2.75" to 4.5". About 90% of them have diameter ranging between 2.5 and 4 inches. The total stud cross-sectional area is about 900 in^2 and the total stud free-length volume is about 36,000 in^3.

Low Pressure Turbine

The double flow low pressure turbine, shown in Figure 14.1.13-3, incorporates high efficiency blading diffuser type exhaust and liberal exhaust hood design. The low pressure turbine cylinders are fabricated from steel plate to provide uniform wall thickness, thus reducing thermal distortion to a minimum. The entire outer casing is subjected to low temperature exhaust steam.

The temperature drop from the cross-under steam temperature to the exhaust steam temperature is taken across three walls; an inner cylinder number 1, a thermal shield, and an inner cylinder number 2. This precludes a large temperature drop across any one wall except the thermal shield which is not a structural element, thereby virtually eliminating thermal distortion. The fabricated inner cylinder number 2 is supported by the outer casing at the horizontal centerline and is fixed transversely at the top and bottom and axially at the centerline of the steam inlet, thus allowing freedom of expansion independent of the outer casing. Inner cylinder number 1 is, in turn, supported by inner cylinder number 2 at the horizontal centerline and fixed transversely at the top and bottom and axially at the centerline of the steam inlets, thus allowing freedom of expansion independent of inner cylinder number 2. Inner cylinder number 1 is surrounded by the thermal shield.

The steam leaving the last row of blades flows into the diffuser where the velocity energy is converted to pressure energy, thus improving efficiency and reducing the excitation forces on the last rotating row of blades.

The low pressure rotors are made of NiCrMoV alloy steel. The specified minimum mechanical properties are given in Table 14.1.13-5.

TABLE 14.1.13-4

MINIMUM MECHANICAL PROPERTIES - LOW PRESSURE ROTORS

Tensile Strength, psi, min.	115,000
Yield Strength, psi, min. (0.2% offset)	100,000
Elongation in 2 inches, per cent, min.	16
Reduction of Area, per cent, min.	40
Impact Strength, Charpy V-Notch, ft-lb. min. at room temp.	40
50% Fracture Appearance Transition Temperature, °F, max.	80

The shrunk-on disks are made of NiCrMoV alloy steel. There are twelve disks shrunk on the shaft with six per flow. These disks experience different degrees of stress when in operation. The present design shows that disk No. 3, starting from the transverse centerline, experiences the highest stress, while disk No. 6 experiences the lowest. The minimum specified mechanical properties for the disks are given in Table 14.1.13-5.

TABLE 14.1.13-5

MINIMUM MECHANICAL PROPERTIES - DISKS

	Disk No. 3	Disk No. 1,2,4,5 and 6
Tensile Strength psi, min.	130,000	120,000
Yield Strength psi, (0.2% offset)	120,000-135,000	110,000-125,000
Elongation in 2" (Disk Hub), per cent, min.	14	15
Elongation in 2" (Disk Rim), per cent, min.	16	17

TABLE 14.1.13-5

MINIMUM MECHANICAL PROPERTIES - DISKS (CONTINUED)

	Disk No. 3	Disk No. 1,2,4,5 and 6
Reduction of Area (Disk Hub), per cent, min.	35	38
Reduction of Area (Disk Rim), per cent, min.	40	43
Impact Strength, (Hub and Rim), Charpy V-notch, ft-lb, min, at room temp.	50	50
50% Fracture Appearance Transition Temperature (Disk Hub and Rim) °F, max.	0	0

The outer cylinder and the two inner cylinders are mainly made of ASTM A-285 Grade C material. The minimum specified properties are given in Table 14.1.13-6.

TABLE 14.1.13-6

MINIMUM MECHANICAL PROPERTIES - CYLINDERS

Tensile Strength, psi, min.	55,000
Yield Strength, psi, min	30,000
Elongation in 8", per cent, min.	24
Elongation in 2", per cent, min.	28

Whenever plates of thickness >2" are employed, they are made of ASTM A-212 Grade A.

Consequences of Turbine-Generator Unit Overspeeding

Low Pressure Turbine

Experience and test have shown that the mode of failure of a disk, should it occur, is mainly rupture in two or four parts. The broken parts would then be ejected normally to the rotation axis. Hence, the potential missiles considered for purposes of analysis are:

a) Half disk

b) A quarter of disk

There are twelve disks shrunk on each low-pressure turbine rotor, with six disks per flow. Numbering the disks from the steam admission, disks No. 1, 2 and 3 are contained within the inner cylinder No. 1, the inner cylinder No. 2, and the outer cylinder (reference is made to Figure 3 and 4). Therefore, if one of these disks breaks, it has to go through the corresponding stationary blade ring, the inner cylinder No. 1, the inner cylinder No. 2, and the outer cylinder. Disks No. 4 and 5 are contained within the inner cylinder No. 2 and the outer cylinder. Hence, if one of these fails, it has to pass through the directly opposite blade ring, the inner cylinder No. 2 and partially within the diffuser and within the outer cylinder. If parts of this disk come loose, they have to go through the directly opposite blade ring and the outer cylinder.

The thickness of the back plate of the three cylinders is given in Table 14.1.13-7.

TABLE 14.1.13-7
THICKNESS - BACK PLATE OF CYLINDERS

Inner cylinder No. 1	2 inches
Inner cylinder No. 2	1.25 inches
Outer cylinder	1.25 inches

The bursting speed of each disk has been calculated with a stress analysis based on a tensile strength 20 percent higher than the minimum specified tensile strength. The 20 percent increase conservatively account for the actual value of the tensile strength, usually observed to be higher than the minimum specified. The values of the minimum and maximum bursting speeds of each disk are listed in the Table 14.1.13-8.

TABLE 14.1.13-8

DISK BURSTING SPEED

Type of Disk	Bursting Speed (per cent of nominal)	
	Maximum	Minimum
Disk No. 1	179	163
Disk No. 2	181	165
Disk No. 3	175	153
Disk No. 4	179	163
Disk No. 5	178	162
Disk No. 6	187	171

As the above table shows, the maximum speed at which the unit might run with no disk failure is 175% of nominal. At this speed, disk No. 3 will burst. As one of the first disks ruptures, the steam flow between the blades of the remaining disks is significantly reduced, the turbine-generator is slowed down, and further disk failures are not anticipated. Since the actual value of the bursing speed of each disk will be between the maximum and minimum previously mentioned, the potentiality of bursting each one of the first five disks exists. The probability of disk No. 6 bursting is more remote. The consequences of rupture of any one of these disks at the maximum speed that the unit might approach in case of turbine runaway have been evaluated and the results are summarized in the following pages.

Table 14.1.13-9 lists the values of the rim radius, the weight, the ejection velocity and ejection translational energy of each disk quarter, at 175% of nominal speed. Table 14.1.13-10 lists the same parameters for half disks.

TABLE 14.1.13-9

RUPTURE IN FOUR QUARTERS AT 175% OF NOMINAL SPEED

Type of Disk	Rim Radius (inches)	Weight (lb)	Ejection Velocity (ft/sec)	Ejection Translation Kinetic Energy (ft lb)
Quarter of Disk No. 1	51.875	2050	855	23.3×10^6
Quarter of Disk No. 2	51.875	1912.5	855	21.7×10^6
Quarter of Disk No. 3	51.875	2455	855	25.0×10^6
Quarter of Disk No. 4	51.234	2575	845	28.6×10^6
Quarter of Disk No. 5	49.162	2900	810	29.6×10^6
Quarter of Disk No. 6	43.800	3100	722	25.1×10^6

TABLE 14.1.13-10

RUPTURE IN TWO HALVES AT 175% OF NOMINAL SPEED

Type of Disk	Rim Radius (inches)	Weight (lb)	Ejection Velocity (ft/sec)	Ejection Translational Kinetic Energy (ft lb)
Half of Disk No. 1	51.875	4100	605	23.3×10^6
Half of Disk No. 2	51.875	3825	605	21.7×10^6
Half of Disk No. 3	51.875	4910	605	25.0×10^6
Half of Disk No. 4	51.234	5150	598	28.6×10^6
Half of Disk No. 5	49.162	5800	573	29.6×10^6
Half of Disk No. 6	43.800	6200	510	25.1×10^6

Disk No. 1, No. 2, and No. 3

Rupture of disk No. 3 has been assumed for purpose of analysis because the four quarters of this disk have more translational kinetic energy than disk No. 1 and No. 2. As the four quarters come loose, they strike and deeply deform the inner cylinder No. 1 and cause some deformation of the inner cylinder No. 2 and of the outer cylinder of less extent.

The rupture is expected to be contained within the unit and no outside missile is anticipated to be generated.

The deformation energy per unit volume of the cylinder material has been evaluated under "static" and "dynamic" loading, based on both minimum specifed and actual averaged mechanical properties. Table 14.1.13-11 summarizes the values of the deformation energy per unit volume up to 100%, 75% and 50% of the total elongation, respectively.

TABLE 14.1.13-11

DEFORMATION ENERGY PER UNIT VOLUME

A. BASED ON THE MINIMUM SPECIFIED MECHANICAL PROPERTIES

	up to 50%ε_u	up to 75%ε_u	up to 100%ε_u
Under "static" loading	4,400 $\frac{in\ lb}{in^3}$	7,000 $\frac{in\ lb}{in^3}$	10,200 $\frac{in\ lb}{in^3}$
Under "dynamic" loading	7,900 $\frac{in\ lb}{in^3}$	11,900 $\frac{in\ lb}{in^3}$	15,800 $\frac{in\ lb}{in^3}$

B. BASED ON THE ACTUAL AVERAGED MECHANICAL PROPERTIES

	up to 50%ε_u	up to 75%ε_u	up to 100%ε_u
Under "static" loading	6,000 $\frac{in\ lb}{in^3}$	9,900 $\frac{in\ lb}{in^3}$	14,300 $\frac{in\ lb}{in^3}$
Under "dynamic" loading	9,000 $\frac{in\ lb}{in^3}$	13,500 $\frac{in\ lb}{in^3}$	18,000 $\frac{in\ lb}{in^3}$

It is expected that in order to penetrate throught the inner cylinder No. 1, the ruptured disk quarters shall have the kinetic energy necessary to deform about 1/3 of the inner cylinder No. 1 volume to between 50% and 75% of the actual total elongation. i.e., between 100×10^6 ft lb and 150×10^6 ft lb. The anticipated kinetic energy of 4 quarters of disk no. 3 is at the lower limit of the above range, i.e., 100×10^6 ft lb.

For disk fragments to become missiles, they do not have to violate only the integrity of the inner cylinder No. 1, but also that of the inner cylinder No. 2 and of the outer cylinder. As mentioned earlier, quarters of disk No. 3 are not expected to violate the integrity of inner cylinder No. 1. Should violation occur for some unknown reasons, the kinetic energy of the quarters would be small. Therefore, for these fragments to leave the unit, they shall have enough kinetic energy to deform a significant amount of inner cylinder No. 2 and outer cylinder, rather than just the energy necessary to perforate the back plates of these cylinder.

For these reasons, we do not expect external missiles to be generated because of failure of one of the first three disks.

Disk No. 4 and No. 5

Rupture of disk No. 5 has been conservatively assumed for purpose of analysis because the four quarters of this disk have more translational kinetic energy than disk No. 4. As the four quarters come loose, they strike and deeply deform the inner cylinder No. 2, and cau some deformation of the outer cylinder.

The rupture is expected to be contained within the unit and no outside missile is anticipated to be generated.

It is expected that, in order to penetrate through the inner cylinder No. 2, the ruptured disk quarters shall have the kinetic energy necessary to deform about 25% of the inner cylinder No. 2 volume to between 50% and 75% of the actual total elongation, i.e., between 136×10^6 and 200×10^6 ft lb. The anticipated kinetic energy of 4 quarters of disk No. 5 is less than 120×10^6 ft lb.

For disk fragments to become missiles, they do not have to violate only the integrity of the inner cylinder no. 2, but also that of the outer cylinder. Therefore, ejection of quarters of disk No. 4 and 5 outside the unit is not expected.

Disk No. 6

This disk is the least stressed disk, and the disk that has the highest bursting speed range, i.e., 171%-187% of nominal. The probability of reaching this speed range is quite remote, because one of the other disks is anticipated to fail at lower speed, preventing the unit from reaching the bursting speed range of disk No. 6. For purpose of analysis it has been postulated the occurrence of bursting this disk at the maximum running speed of 175% of nominal.

The damage cause by this failure is expected to be contained within the unit.

Upon bursting, the ejected quarters will strike the coupling flanges of the outer cylinder center and the outer cylinder side. It is expected that, in order to penetrate through the outer cylinder, the ejected quarters shall have the kinetic energy required to deform the directly opposite blade ring, the above mentioned flanges and a two-disk-hub wide portions of the outer cylinder, for a total of 150,000 in^3, to between 50% and 75% of the actual total elongation, i.e., between 112 x 10^6 and 168 x 10^6 ft lb. Since the anticipated kinetic energy of 4 disk No. 6 quarters, i.e., 100 x 10^6 ft lb, is below the lower limit of the required energy range, no external missile is anticipated.

High Pressure Turbine

Due to the very large margin between the high pressure spindle bursting speed and the maximum speed at which the steam can drive the unit with all the admission valves fully open, the probability of spindle failure is practically zero. Therefore, no harmful missile is anticipated in case of turbine runaway.

Based on the admission steam thermodynamic properties and blade geometry, the maximum theoretical speed at which the unit may run is 208% of nominal.

Based on the stress analysis of the low-pressure disks, the maximum actual speed at which the unit may run is 175% of nominal.

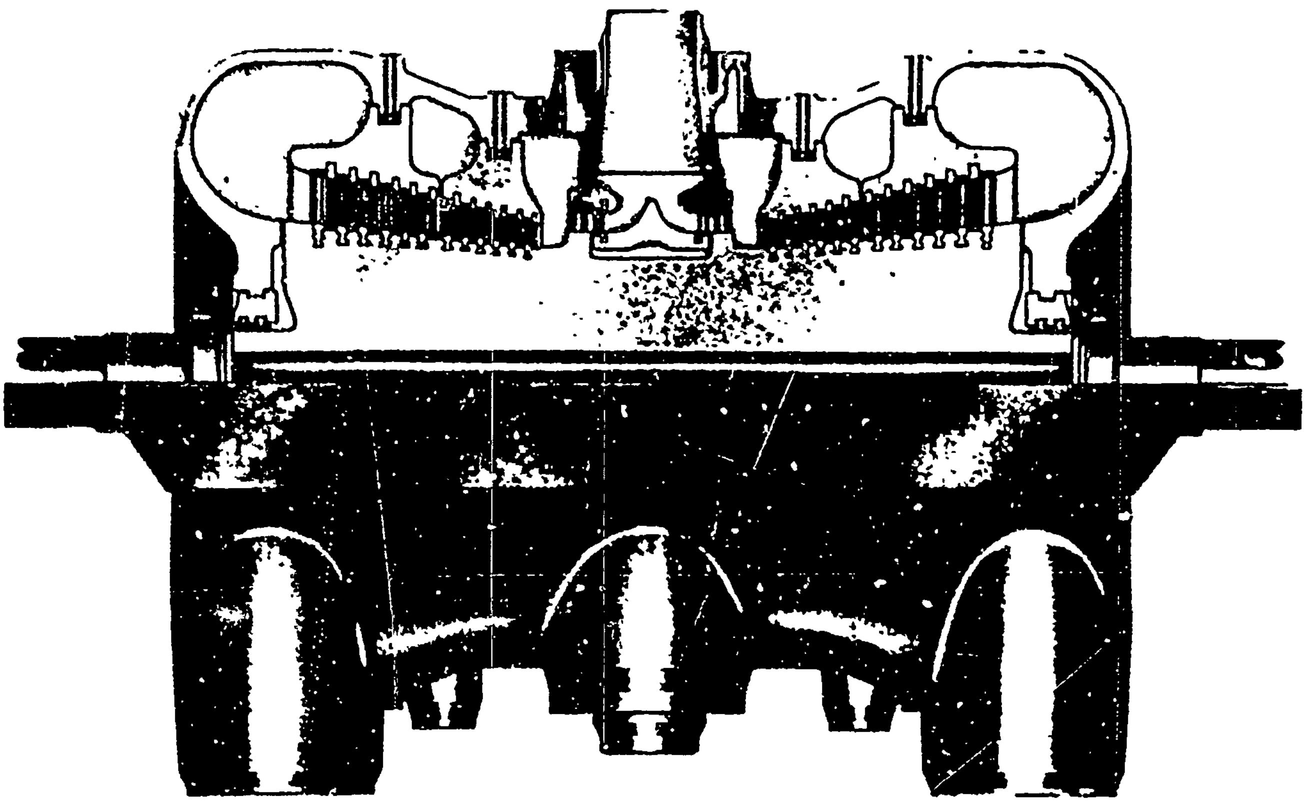
The minimum bursting speed of the spindle, based on the minimum specified mechanical properties of the spindle material, is 270% of nominal. The actual bursting speed is closer to 300% of nominal than 270%.

Hence, the actual margin between the bursting speed and the maximum running speed is of the order of 125% of nominal, i.e., 300%-175%.

No failure of the H. P. turbine is anticipated as a consequence of a unit runaway; and therefore, no missiles are expected to be generated.

HIGH PRESSURE CYLINDER

1800 RPM DOUBLE-FLOW DESIGN

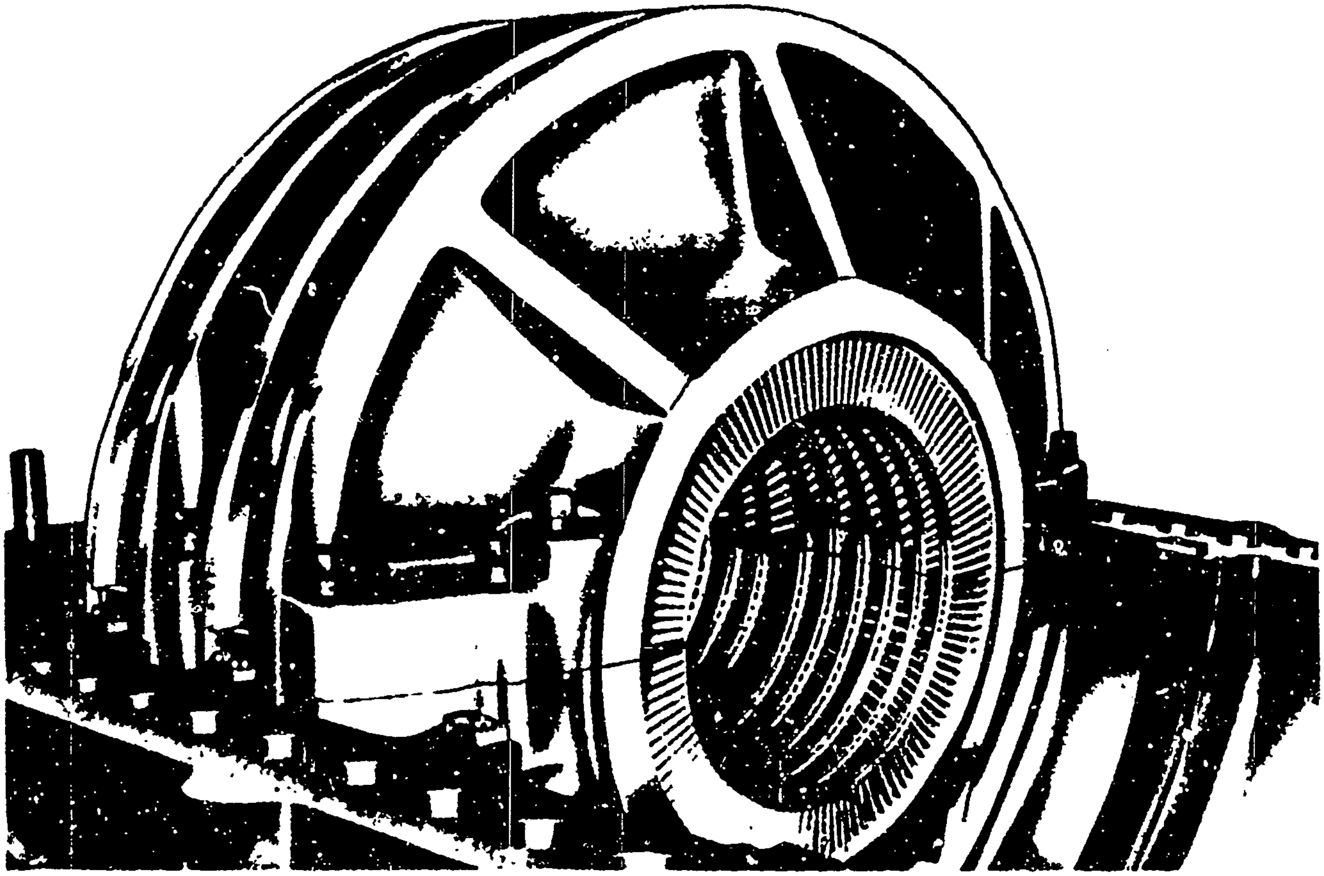


FEATURES

1. Four separate nozzle chambers permit freedom of expansion and contraction during starting and load changes.
2. Double flow design insures thrust balance.
3. Rotor checked in heater box for dynamic balance prior to shipment.
4. Ultrasonic test of rotor performed at steel mill and at the Westinghouse factory.

FIGURE 14.1.13-1

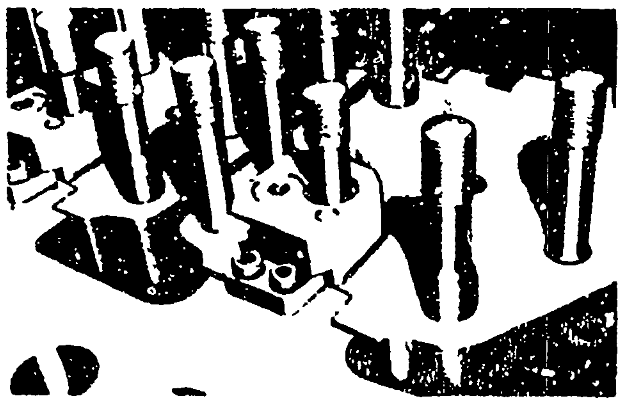
BLADE RINGS



Blade rings of large high-pressure, high temperature turbine, with stationary blades in place.

FEATURES

1. Centerline supporting block insures center alignment while allowing differential expansion between blade ring and cylinder.
2. Blades are inserted in blade ring halves.
3. Tongue and groove holds blade ring in position.
4. Metallic seals between blade rings and cylinder prevent leakage of steam in support grooves.
5. Upper plate, in cylinder cover, prevents any "riding-up" of the blade ring.



View of turbine cylinder and blade ring, showing method of supporting and locking lower blade ring in position.

FIGURE 14.1.13-2

LOW-PRESSURE ELEMENT

1800-RPM DOUBLE-FLOW DESIGN

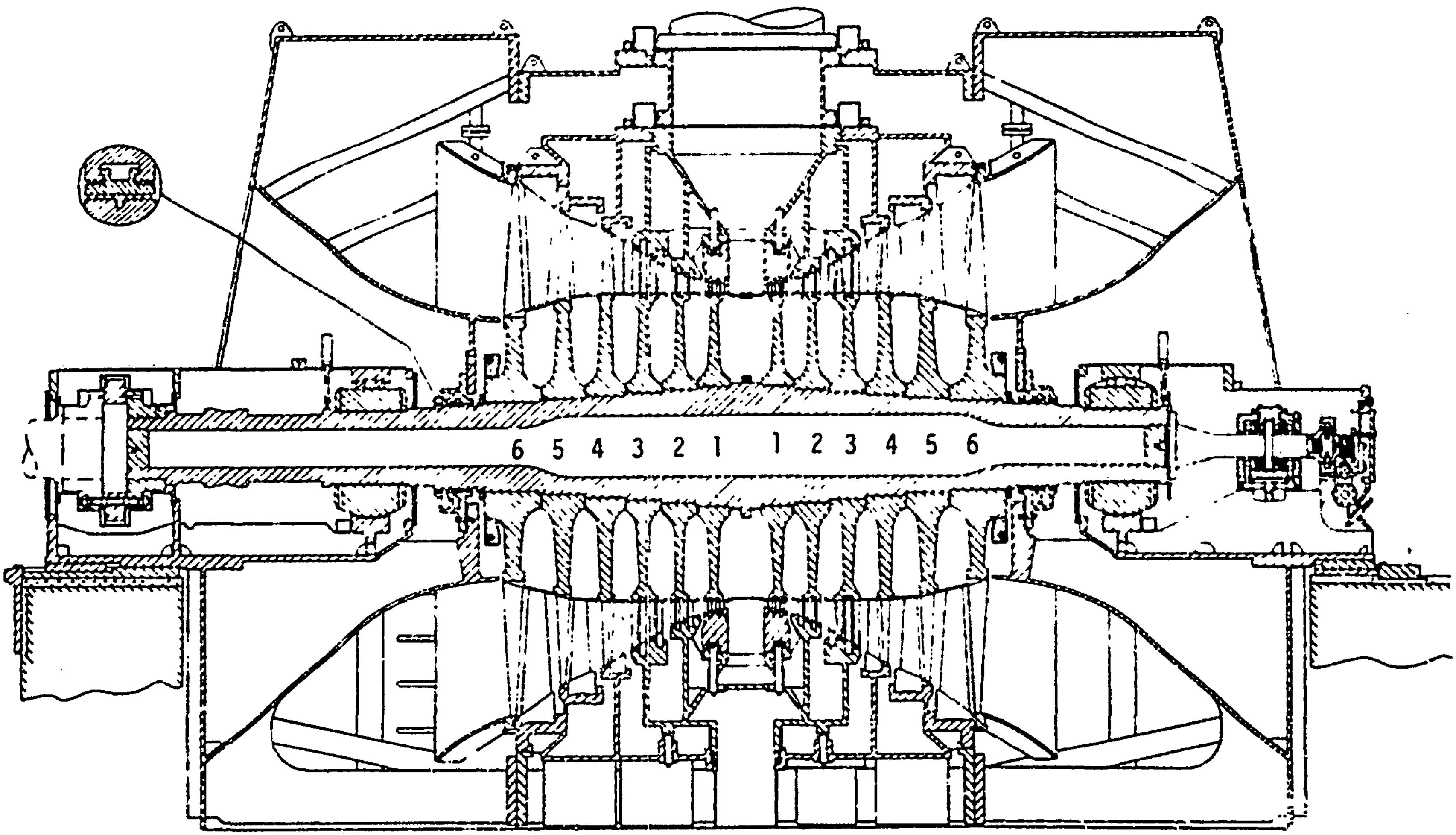
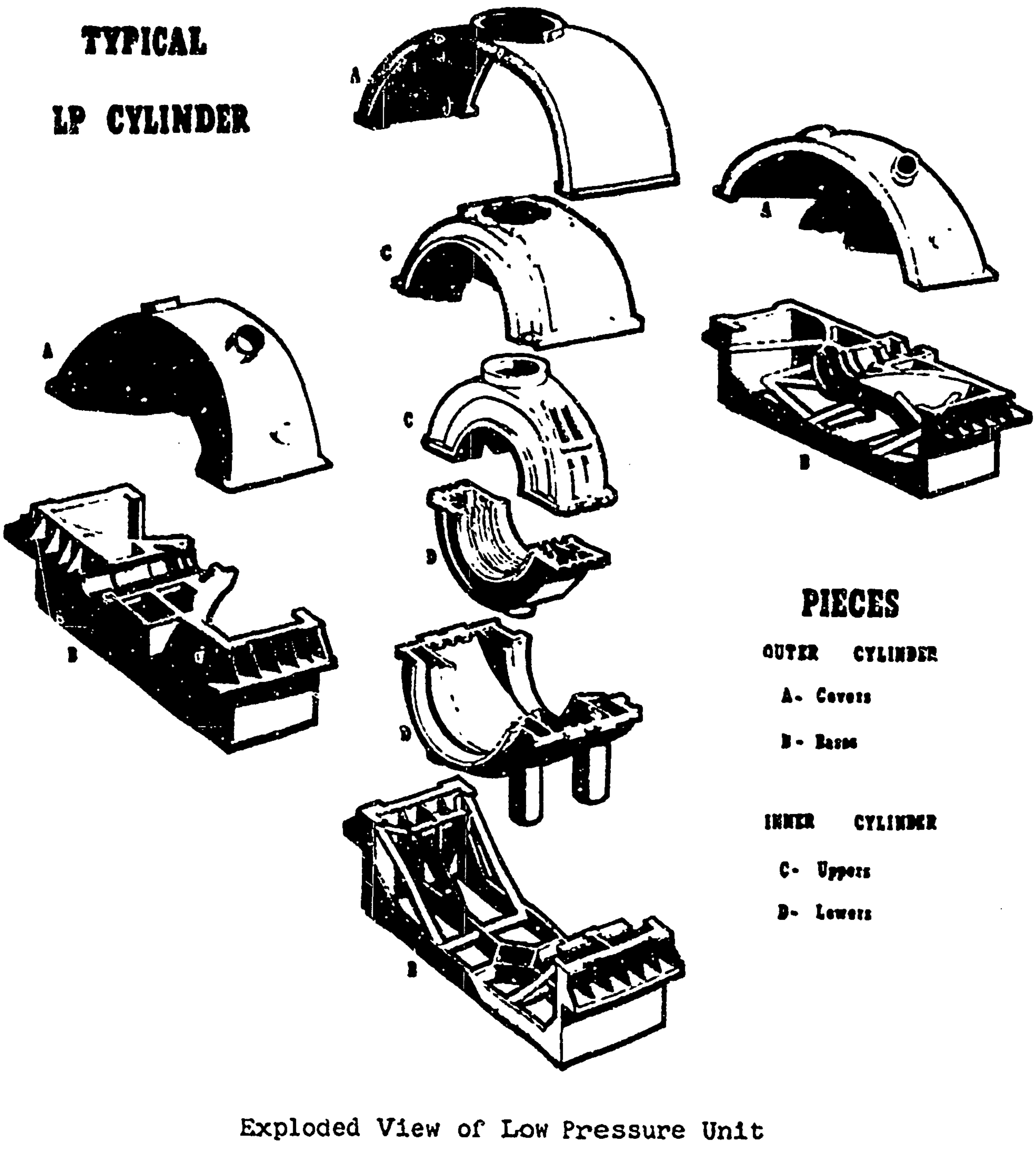


FIGURE 14.1.13-3



Exploded View of Low Pressure Unit

FIGURE 14.1.13-4

APPENDIX 14A

The following report en_itled "Likelihood and Consequences of Turbine Overspeed at the Indian Point Nuclear Generating Unit No. 2" was prepared to amend the previous analyses submitted as section 14.1.13 of the FSAR. The amended study presents the results of recent analytic and experimental work performed following the submitted of the FSAR in 1968.

LIKELIHOOD AND CONSEQUENCES OF
TURBINE OVERSPEED AT THE
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

by

J. N. Fox

July 1970

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1.0 INTRODUCTION AND CONCLUSIONS

The analysis on the consequences of turbine overspeed in the past showed there was reasonable assurance that missiles would not be generated external to the low pressure turbine casing even in the event of a turbine runaway. The basic assumptions used in the analysis that led to this conclusion were deemed reasonable, at the time, by all parties involved. However, because of the potential serious consequences of external missiles, Westinghouse initiated a series of model tests to substantiate these assumptions.

The tests involved bursting of simulated low pressure turbine discs within various stationary steel cylinders modeled to approximate blade rings, inner cylinders and the outer casing. The most significant test findings were: (1) penetration occurs mostly by local punching with little bending or stretching of stationary steel; (2) a disc fragment can wedge a path between two blade rings if a blade ring is not directly opposite the rim of the disc. In either case, the stationary steel has less energy absorbing capability than originally expected, and as such, the energy required to penetrate is minimized.

As a result of these test series, new criteria were evolved for predicting the missile containing ability of the low pressure turbine structures. The previous calculations have been redone using these new criteria and the results show the original position on containment of disc fragments within the turbine casing can no longer be maintained

Because of this, Consolidated Edison of New York and Westinghouse have adopted new objectives:

1. Further reduce the likelihood of a turbine-generator unit overspeeding above the design speed.

2. Give reasonable assurance of public protection following turbine disc rupture at or below design speed.

The first objective is met by providing a completely independent turbine speed detection and value trip initiation system. The addition of this new independent system reduces the likelihood of a runaway to practically zero, and as such, no design consideration will be given to the consequences of a runaway.

For design purposes, rupture of a low pressure turbine disc at or below design speed is postulated, even though this failure is not anticipated because of design conservatism and original quality control. The zero failure record of Westinghouse turbine discs confirm the design adequacy.

Reasonable assurance of public protection following disc rupture at or below design speed is met by demonstrating that:

1. No direct loss of reactor coolant takes place.

2. Reasonable assurance exists that the plant can be maintained in a safe shutdown condition.

3. Off-site exposures are within 10CFR100 guidelines for missile damage resulting in activity release.

The analyses contained herein show that the second objective can and will be met with a limited number of modifications coupled with defined manual actions and timely damage repair.

2.0 ADDITIONAL OVERSPEED PROTECTION

2.1 Description

The additional overspeed protection utilizes the output of magnetic pickups mounted around the turning gear to detect and measure turbine shaft speed. The system generates tripping logic signals on overspeed in three redundant speed channels. These signals are used in a 2/3 logic matrix to de-energize redundant control relays that energize redundant solenoid-operated dump valves attached to the actuator of each turbine stop and control valve (two dump valves per actuator). Opening either dump valve drains the oil under the piston of the actuator (see Figure 2-1), thus closing the corresponding turbine stop and/or control valve. Since the turbine stop and control valve in each line are in series, closure of either one will stop steam flow in that line (see Figure 2-2).

The system is protected against loss of input signal by three redundant failure channels which also operate 2/3 logic to trip the unit. The system will also trip the unit on loss of AC power. In addition, this system does not impair operation of the normal turbine control system.

2.2 Operation

The additional overspeed protection system can be separated for descriptive purposes into three sections. These are detection, signal conditioning and logic, and actuation. Refer to Figures 2-3 and 2-4 to follow the description. Figure 2-5 is a logic diagram describing the functional operation of the system. Figure 2-6 is a block diagram of one of the speed and failure channels.

The detection section consists of three magnetic pickups which generate a high frequency AC voltage, the frequency of which is proportional to the rotational speed of the turbine. This voltage is sent to the signal conditioning and logic equipment.

The input signal is fed to a frequency/analog converter in the signal conditioning section. The frequency/analog converter is a solid-state module which develops a DC voltage proportional to the frequency of the input signal. This DC voltage is applied to solid state comparing amplifiers which compare the signal to an internal setpoint and provide switching output when the signal is below the setpoint (speed 106%). This output is fed to coils of hermetically sealed relays, T_{1A} and T_{1B}, keeping them energized during normal turbine operation. The analog speed signal from the F/A converters is also fed to a solid state integrator with a long time constant to make up a reference signal for the failure channel. This reference signal permits starting the unit and provides for positive switching on loss of input signal. The failure channel is provided to trip the unit on loss of signal from the detector. A signal proportional to the output of the F/A converter is fed to a comparing amplifier with the reference signal. The output voltage of the comparing amplifier (positive for normal operation and negative for turbine speed changes over a set value) switches a transistor to keep relays A_{1A} and A_{1B} energized during normal turbine operation.

The actuation section is a pair of solenoid-operated dump valves (one pair for each throttle stop and control valve actuator) which are de-energized (closed) during normal turbine operation. The contacts of $T_{1A\&B}$ and $A_{1A\&B}$ are in 2/3 matrices and for overspeed or loss of detector output sensed by 2/3 of the boxes will open the circuit to the control relays. The control relays de-energized, closing the circuit to the solenoids, opening the valves, and dumping the fluid from the cylinders of the valve actuators, thus closing the associated turbine valve. Each solenoid valve is actuated by an independent logic train and is powered by an independent DC power source.

2.3 Testing Provisions

The system will be capable of testing at power. The valves will be tested individually at least at monthly intervals during periods of reduced power operation. The detection and logic sections of the system will be provided with test switches and indicating lights to permit testing of a single channel and a 2/3 matrix at power. The method of testing will be to switch to a fixed test set point on the speed channel and observe the channel response to the simulated overspeed. Similarly a test sutich will be used to interrupt the F/A output signal and observe the failure channel response to simulated input failure. Indicating lights and test switches will be mounted in the same cabinets with the analog and logic equipment.

2.4 Reliability Analysis

The additional overspeed protection system is designed to meet nuclear protection system criteria of redundancy, separation, and reliability. A single active component failure cannot cause failure to trip the unit on overspeed. Loss of power will also trip the unit. Fault trees are presented (Fig. 2.7 thru 2.10) to show the possible combinations of component failures which can lead to loss of function. The overall probability of failure to shut off steam to the turbine considering only the additional overspeed detection and tripping system is computed to be 4.8×10^{-8}/demand. This is based on a figure of 7.5×10^{-7}/demand for the probability of failure to dump oil from a single steam valve actuator based on monthly test intervals and generic failure rates for components extracted from various government and industry sources.

The redundant overspeed trip system meets the intent of IEEE-279 operationally 15
and schematically.

FIGURE 2-1

Valve Actuator Dump

(Typical for each turbine stop and control valve.)

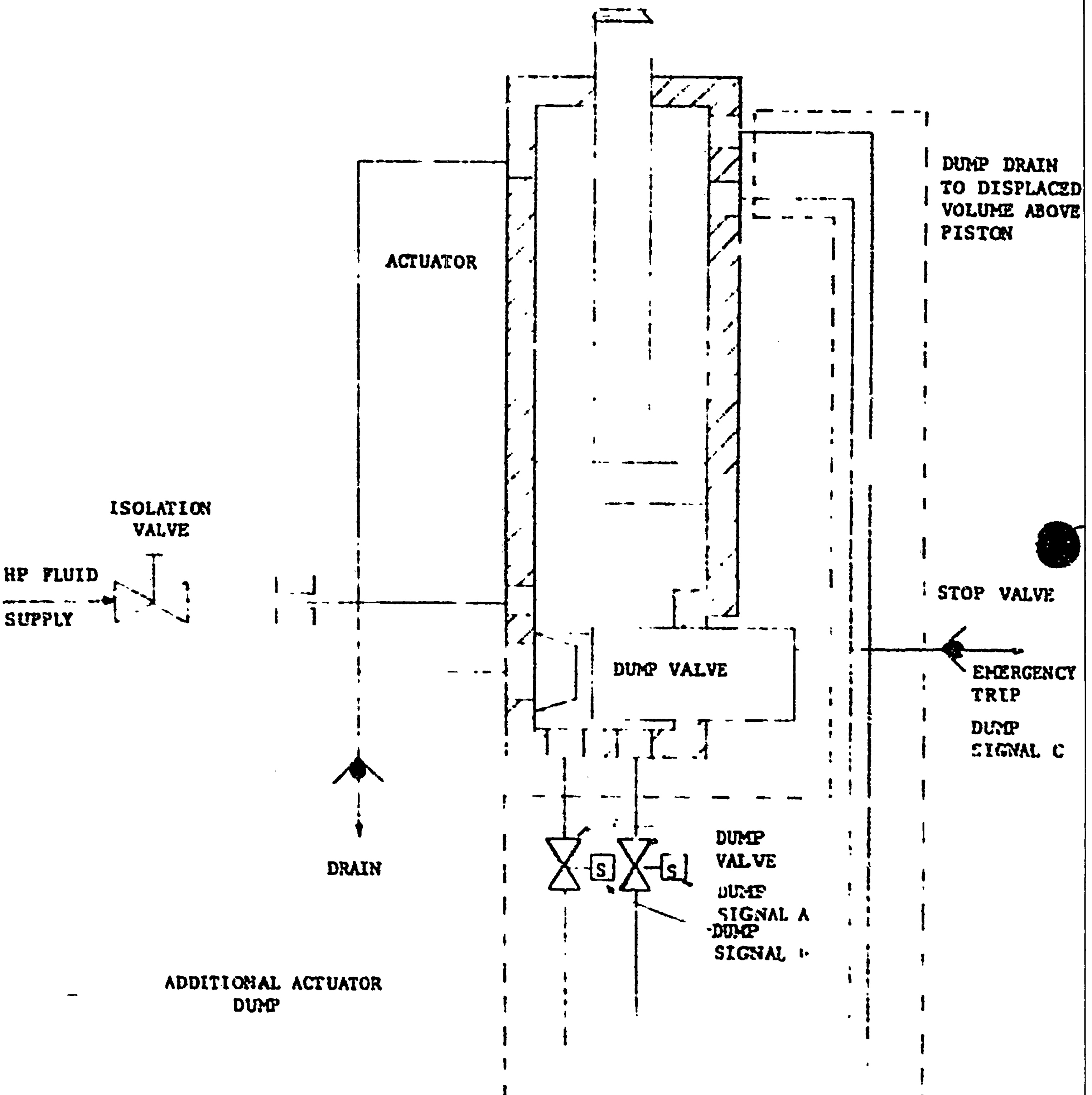
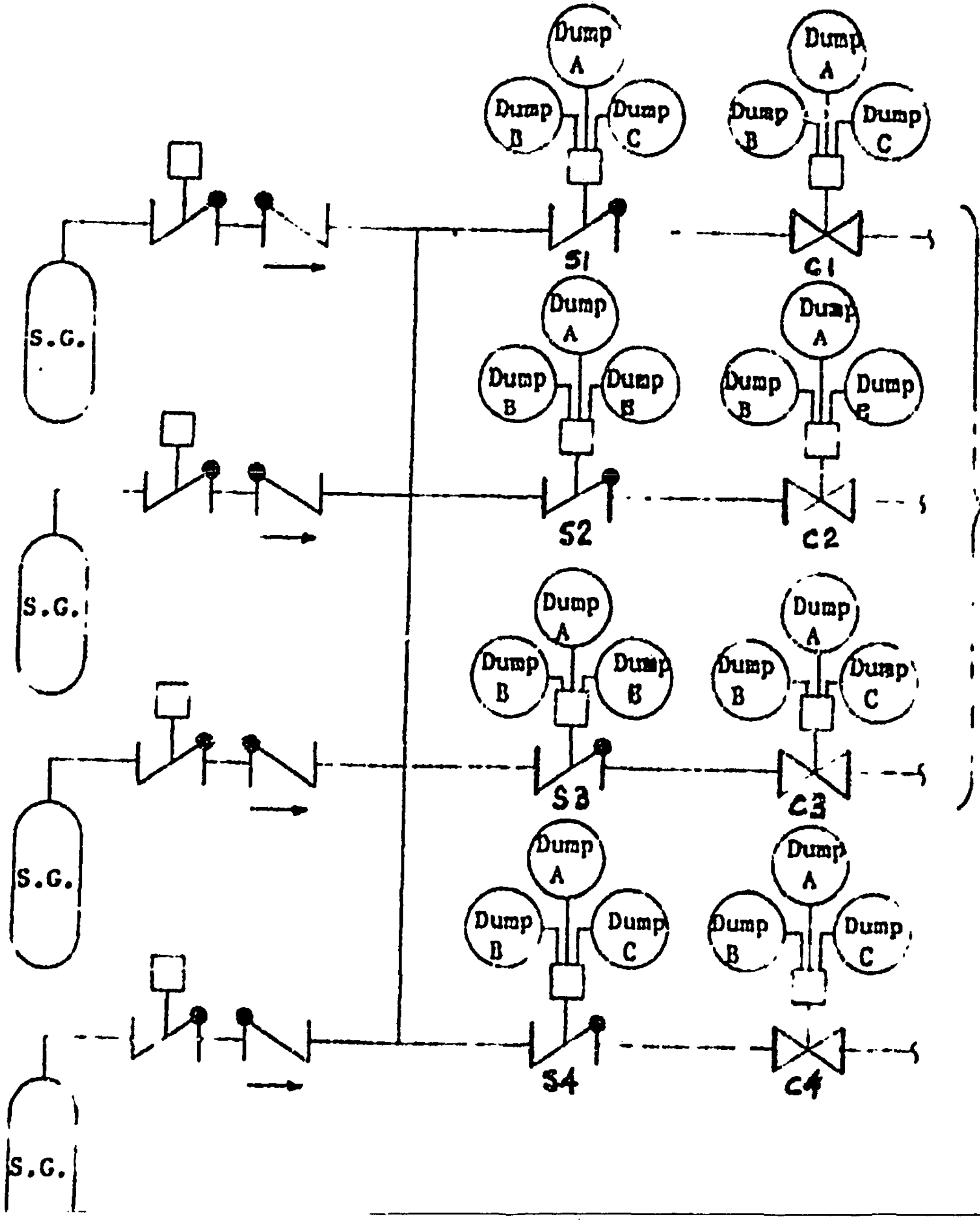


FIGURE 2-2

TURBINE TRIP ACTUATION



LEGEND

S1,S2,S3,S4 – Stop Valves 1,2,3 and 4

C1,C2,C3,C4 – Control Valves 1,2,3 and 4

S.G. – Steam Generator

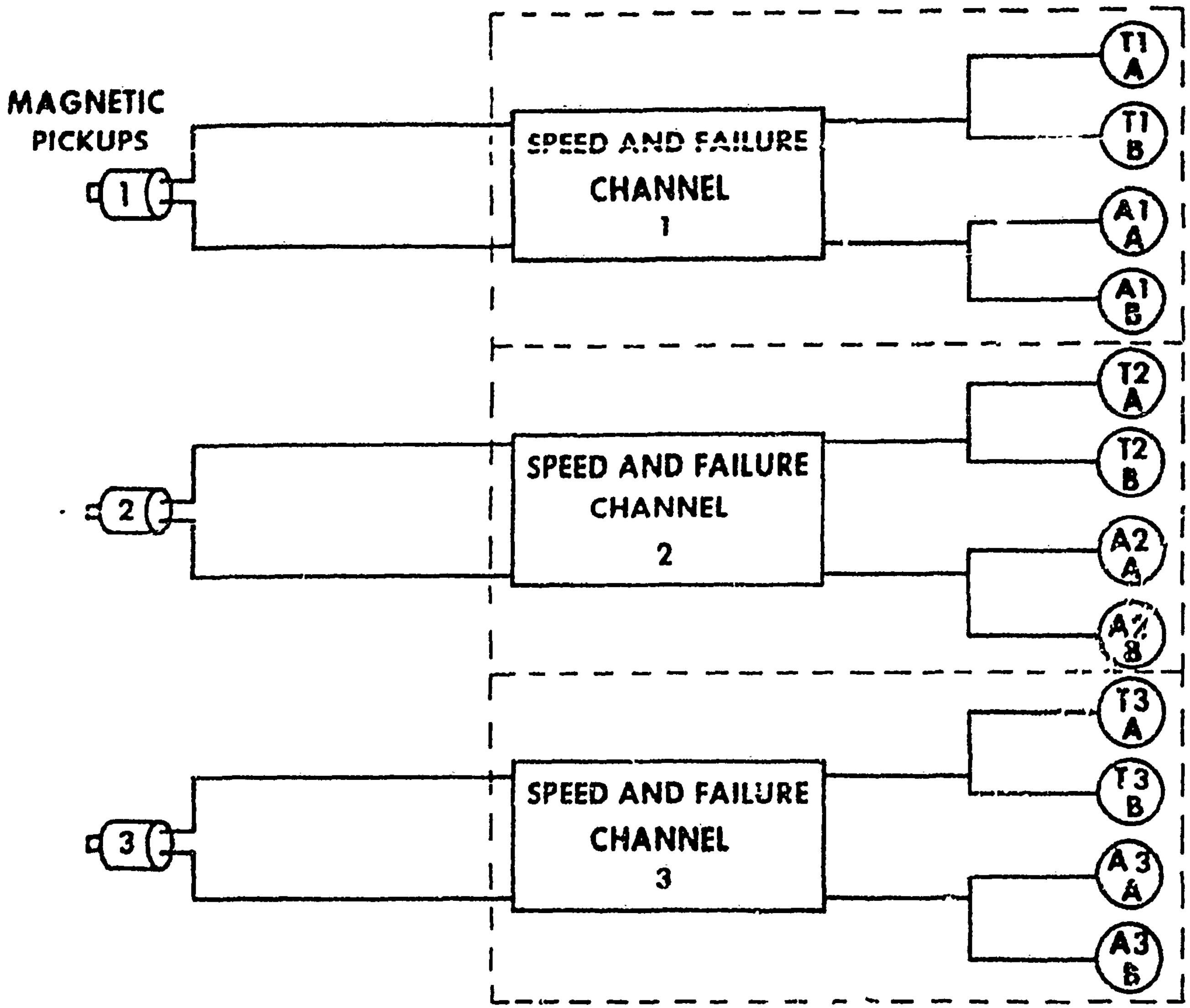


Figure 2-3. Signal Conditioners and Logic

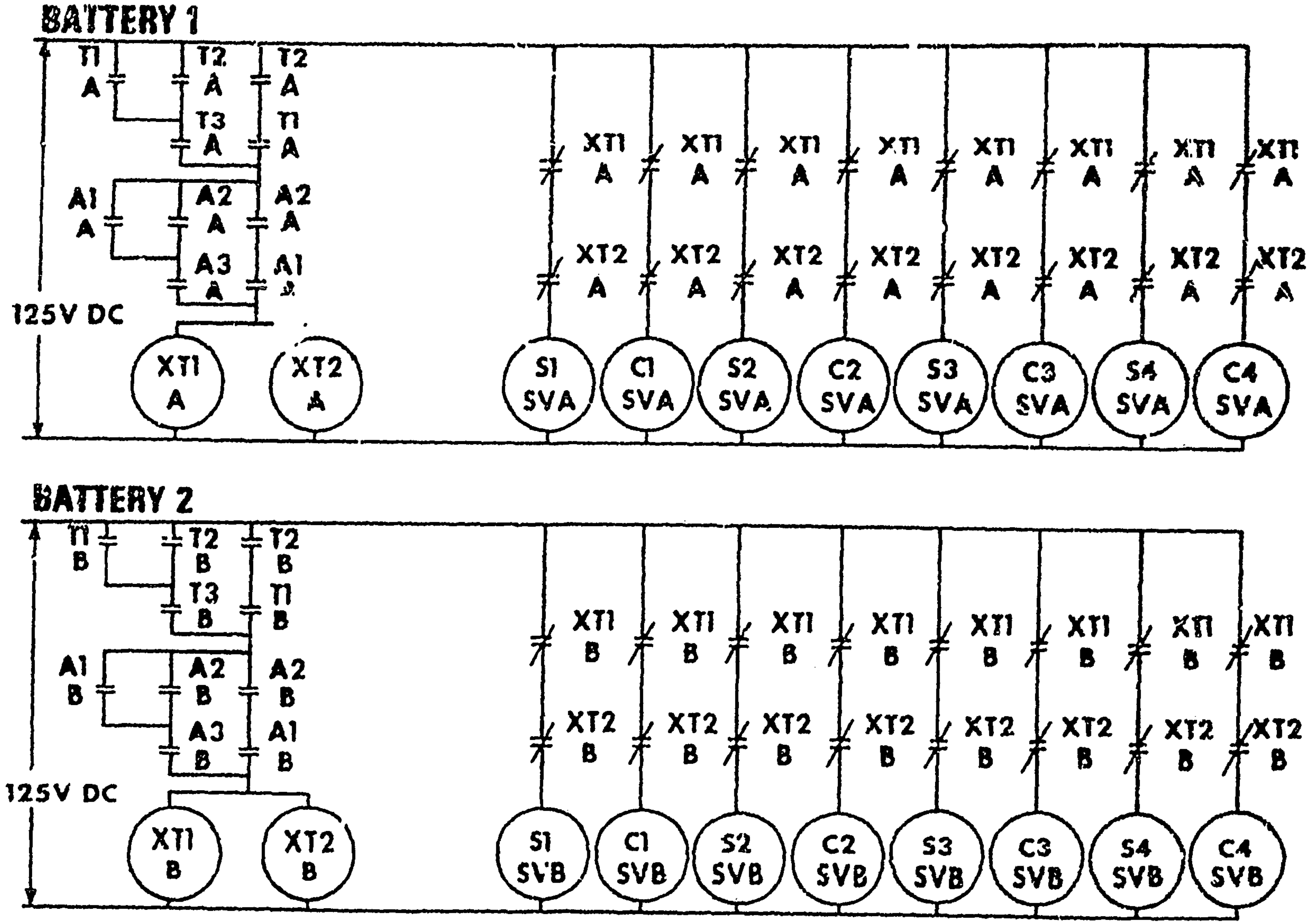


Figure 2-4. Actuation Logic

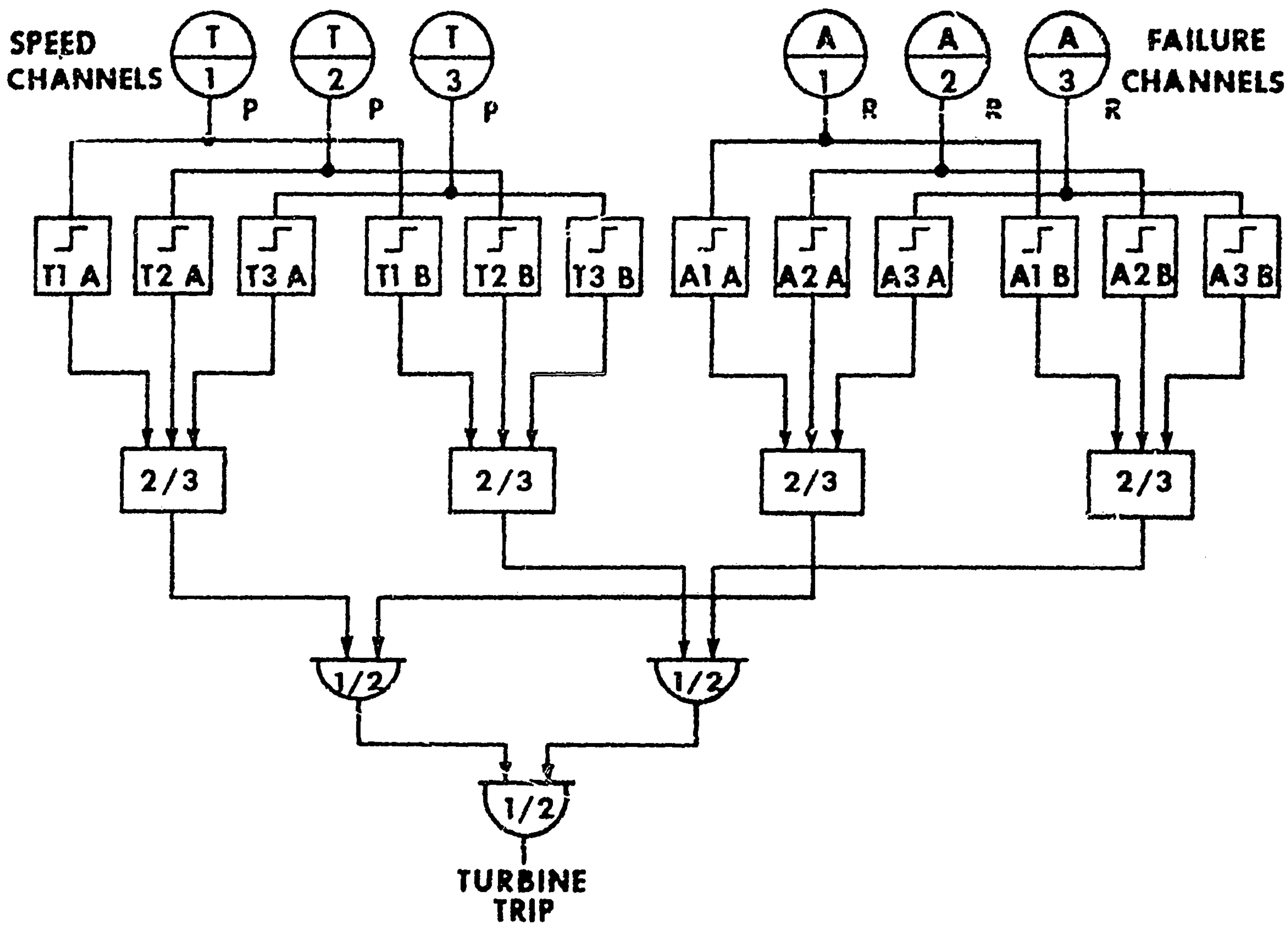


Figure 2-5. System Logic Diagram

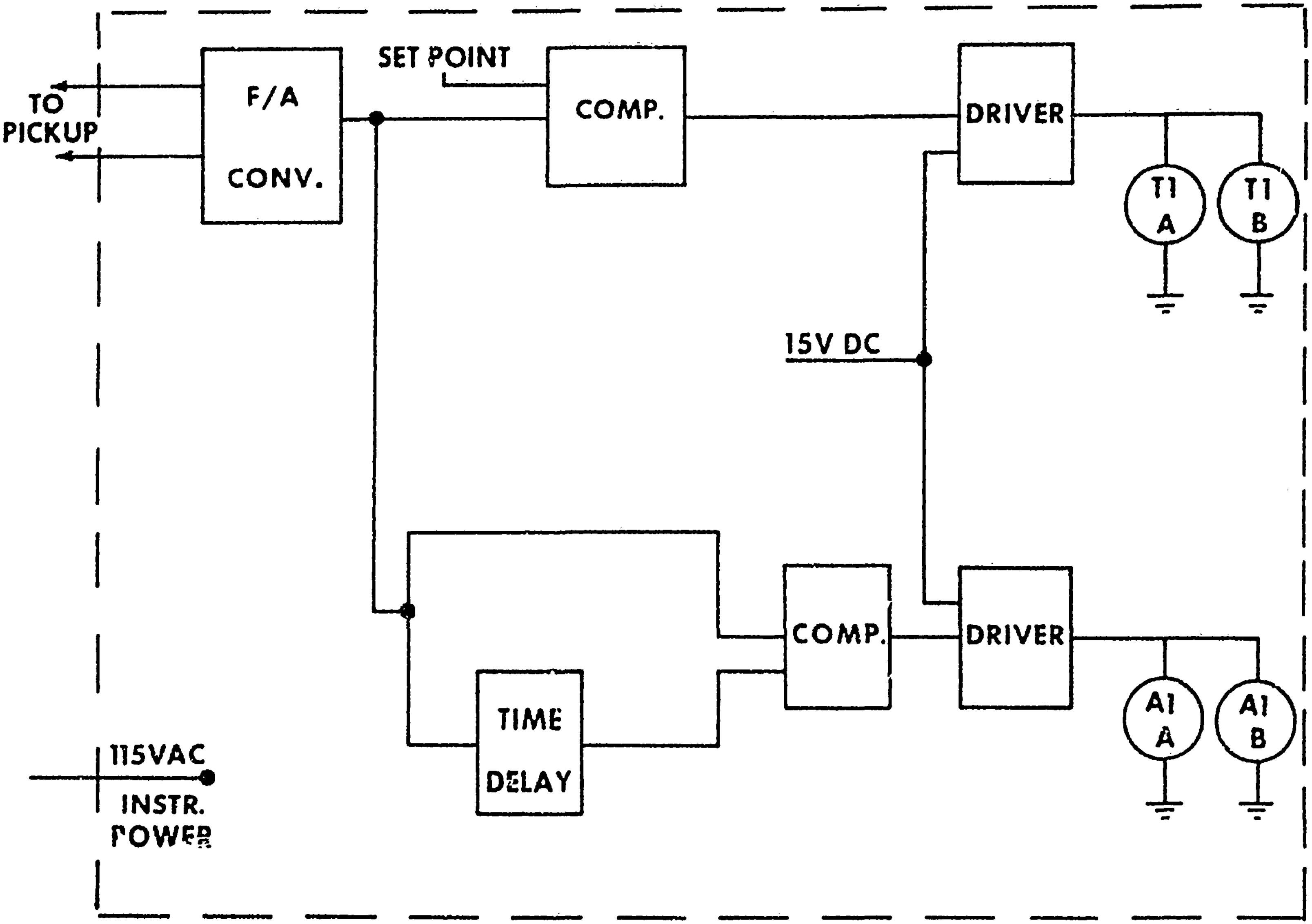


Figure 2-6. Block Diagram for Speed and Failure Channel 1

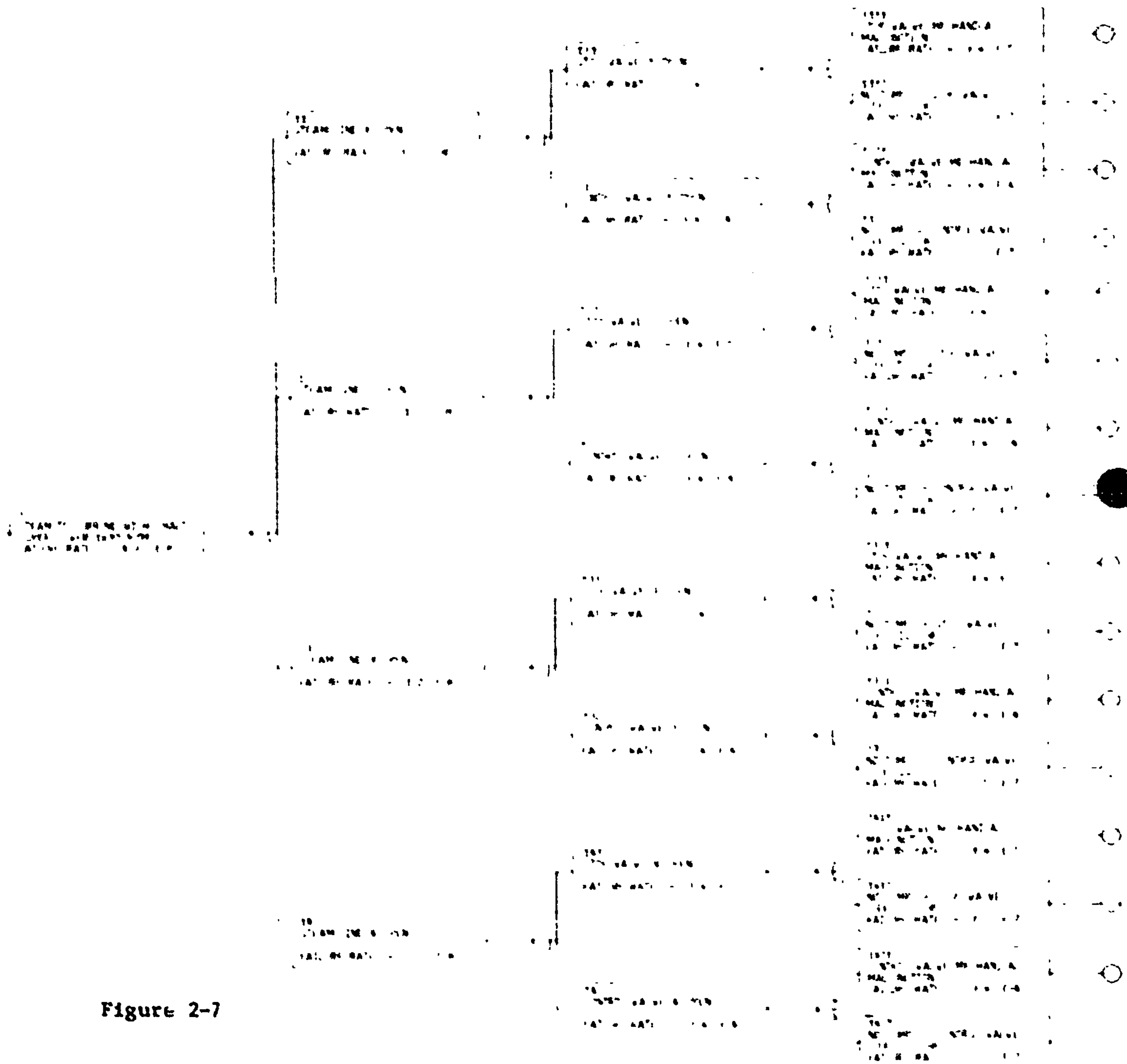


Figure 2-7

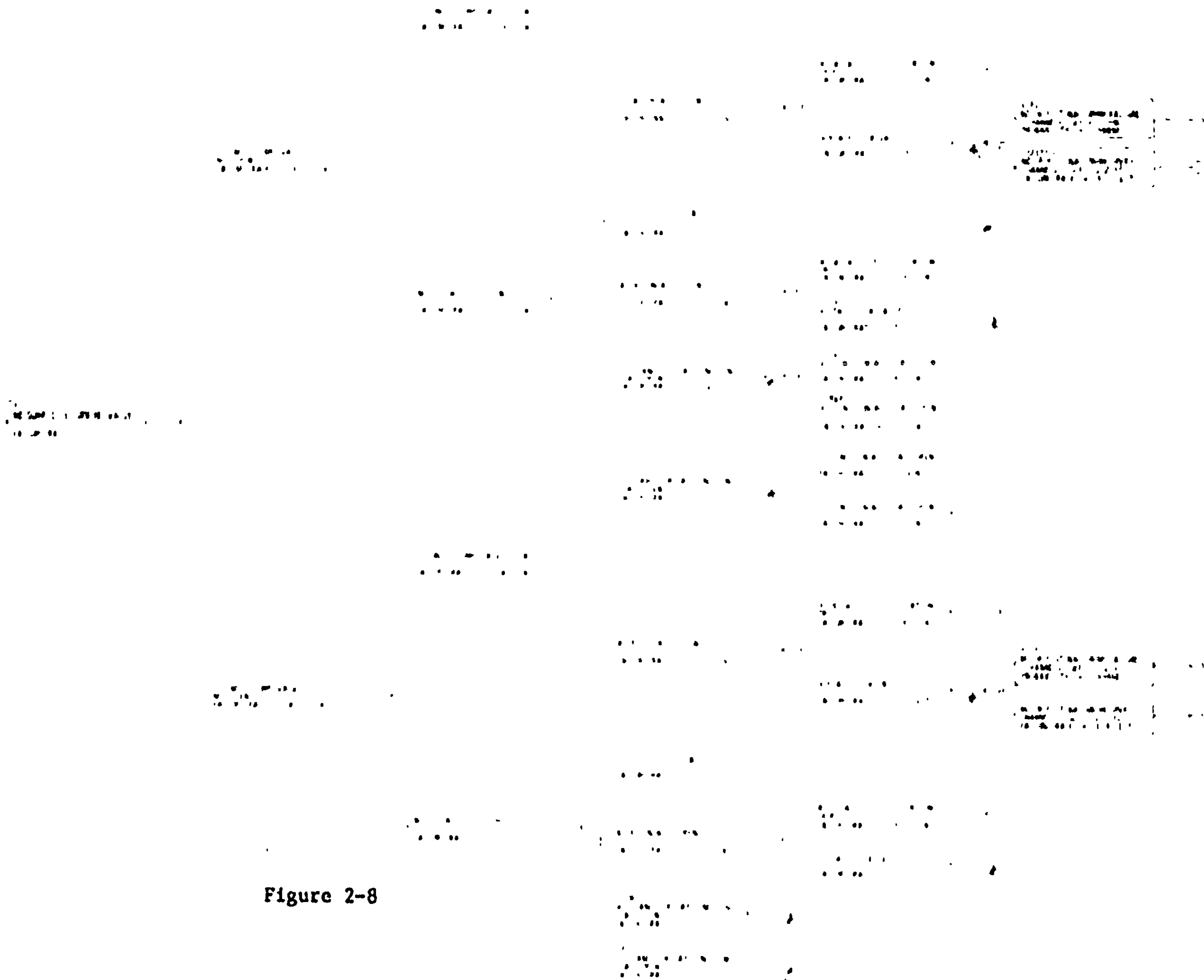


Figure 2-8

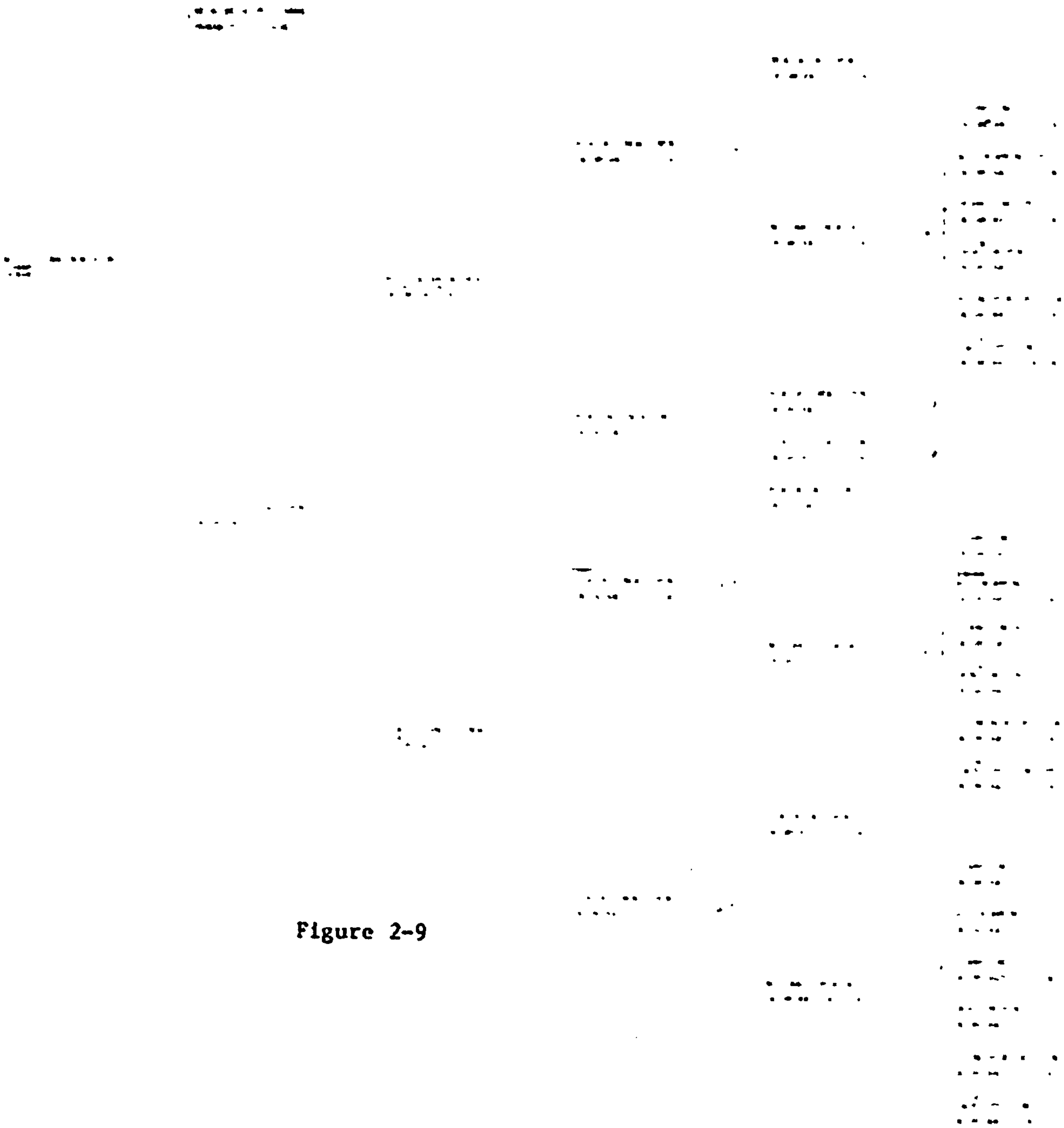


Figure 2-9

Figure 2-10

3.0 BURSTING SPEED MARGIN

3.1 Overspeed Analysis

For determination of the maximum design overspeed, the following conservative sequence is assumed:

a. The unit is operating at full load with all turbine valves wide open.

b. The entire turbine load is dropped instantaneously (no credit taken for plant auxiliary load).

c. The auxiliary governor is assumed to operate improperly (i.e., does not respond to turbine load mismatch).

d. Trip is initiated at the emergency overspeed set point (106% overspeed).

e. From this point on, the turbine valves operate in the prescribed manner.

At the instant the load is dropped, the unit is assumed to accelerate at a constant maximum rate corresponding to the initial steam flow and rotational inertia of the unit until the unit reaches the emergency overspeed trip set point plus a pure time delay of 0.1 second. Flow into the turbine is then calculated during valve closure and is modified for flow versus lift characteristics. It takes approximately 0.15 second to fully close all of the turbine valves following the initial 0.1 second delay. Once the valves close completely, additional overspeeding is calculated using the energy stored in the turbine, the moisture separators and the related piping.

The resulting maximum overspeed calculated in this manner for Units No. 2 and 3 is nominally 131%. Considering the uncertainties in valve characteristics and variations in closing times, it is estimated that the calculated overspeed is within 2%. Thus, for the determination of the maximum energy of a postulated disc failure, an overspeed of 133% is conservatively assumed.

3.2 Margin Between Bursting and Maximum Speed

In Section 2.0, it was shown that between test intervals, the probability of failing to stop steam flow to the turbine upon reaching the emergency overspeed signal is less than 5×10^{-8}/demand. Thus, for all practical purposes, the maximum attainable speed of the turbine-generator unit is the maximum design overspeed of 133%. Hence, the minimum bursting speed margin is the difference between bursting speed of a disc and 133%.

Bursting speeds of the discs are calculated using the experimentally-demonstrated cirteria that bursting will occur when the average tangetial stress reaches the yield strength of the material. The low pressure turbine discs of both Unit No. 2 and 3 are made of Ni Cr Mo V alloy steel with the following yield strength specification:

	Yield Strength, psi (0.2% offset)	
	Minimum	Maximum
Disc No. 1, 2, 4 and 5	120,000	135,000
Disc No. 3	140,000	155,000
Disc No. 6	110,000	125,000

Apply the above bursting speed criteria and specified minimum and maximum yield strengths results in the disc bursting speeds shown in Table 3-1. Since the margin between the minimum bursting speed and the maximum design overspeed is of the order of 51% of nominal (i.e., 184% - 133%), no failures are anticipated in the low pressure turbines for speeds at or below the maximum design speed of either unit.

3.3 Non-Destructive Testing of Low Pressure Turbine Discs

In addition to an extensive destructive testing program to ensure the specified mechanical properties (such as tensile strength, yield strength, ductility, and impact strength), each low pressure disc of Unit No. 2 and 3 undergoes a complete ultrasonic test after completion of major heat treatment and smooth turning. This testing is performed at the supplier's shop and is followed by a magnaflux after rough machining and stress relief.

The above non-destructive testing ensures the integrity of each low pressure turbine disc for adequacy of the bursting speed margins discussed previously.

3.4 Performance of Westinghouse Turbine Discs

The adequancy of the bursting speed margin is further confirmed by the zero failure record of Westinghouse turbine discs. The following applies to Westinghouse turbines in operation from 1938 to 1969:

No. of units with discs: 792
No. of units with shrunk-on discs: 453
Total number of discs: 1429
Total number of disc-years: 15,430
Average number of years per disc: 10.8
Maximum number of years per disc: 10.8
Maximum number of years per disc: 31
Disc failures: none

TABLE 3-1

DISC BURSTING SPEEDS

	Bursting Speed (% of Nominal)	
<u>Disc No.</u>	<u>Minimum</u>	<u>Maximum</u>
1	190	202
2	189	201
3	18[illegible]	194
4	189	201
5	189	201
6	215	229

4.0 TURBINE MISSILE CHARACTERISTICS FOR RUPTURE AT OR BELOW DESIGN SPEED

4.1 Missile Energies External to Turbine Casing

4.1.1 General Approach

The method of determining the residual energy of any disc fragment consists of a step-by-step examination of the various stationary parts that a disc fragment can impact. In the analysis, it is assumed that the disc fragments are of equal size, and in particular for the missiles considered in this report, they are assumed to be 1/4 segments of a low pressure disc. Once the maximum overspeed is determined, the translational kinetic energy of a 1/4 disc segment is calculated using the mass of the segment and the velocity of the center of gravity of the segment (i.e., $KE_{initial} = \frac{1}{2} M_1 V_1^2$). The rotational kinetic energy is assumed to dissipate in the form of heat caused by a circumferential scraping action in the impacting process.

Test results indicate that containment of a missile involves a two phase process. In phase one an inelastic collision occurs between the missile and some effective mass of the containment. Linear and angular momentum are conserved. If the available strain energy of local plastic deformation is less than the energy change as a result of the inelastic collision, local perforation usually occurs. If not, local perforation does not occur but the remaining kinetic energy of the system must be absorbed in phase two, a gross stretching of the containment.

4.1.2 Calculational Procedure

With the initial translational kinetic energy and the mass of the fragment at hand, impact of the first (in an outward radial sense) stationary part is examined. Assuming non-perforation and applying conservation of momentum requires the change in kinetic energy (ΔKE) in the first stage of the impacting process to be:

$$\Delta KE = \frac{1}{2} M_1 V_1^2 \quad (1 - \frac{M_1}{M_1 + M_2})$$

Where M_2 equals the effective mass of the stuck region of the stationary part. Thus, the criterion for non-perforation of the stationary part by the fragment is:

$$\Delta KE < W_c + W_s$$

Where W_c is the work of the local compression and W_s is the shear work required to shear out the plug. The work of compression and of shear are calculated by the following formulae which have been sustantiated experimentally:

$$W_c = 0.085 \, t \, \sigma_{dyn} \, A_F$$

where:

t = thickness of stationary part

σ_{dyn} = dynamic ultimate strength of the stationary part (σ_{dyn} = 1.4 times the static ultimate strength)

A_F = contact area between disc and the stationary part

and

$$W_s = \frac{1}{3} P \, t^2 \, \tau_{dyn}$$

where

P = perimeter of the shear region

t = thickness of the stationary part

τ_{dyn} = dynamic shear strength of the stationary part
(τ_{dyn} = static ultimate)

If in fact $\Delta KE < W_D + W_s$ (i.e., non-perforation), then the stationary part must be examined to see if it is capable of absorbing the residual kinetic energy in stretching. The residual kinetic energy is given by the expression:

$$KE_{residual} = \frac{1}{2} M_1 V_1^2 \left(\frac{M_1}{M_1 + M_2}\right)$$

The material used in the stationary steel parts has been shown to withstand a dynamic elongation of 11% before breaking. Applying this elongation, the maximum energy absorption capability in stretching is given by the expression:

$$W_{stretch} = 0.11 \; L \; A \; \sigma_{dyn}$$

where

L = effective circumferential length of stationary part

A = effective cross-sectional area of stationary part

σ_{dyn} = dynamic ultimate strength of stationary part.

If $KE_{residual} < W_{stretch}$, then the disc fragment will be contained. If on the other hand $KE_{residual} > W_{stretch}$, then the disc fragment will break that particular stationary part in stretching the disc fragment and the related portion of the stationary will continue out radially and will impact the next stationary part. The new residual kinetic energy before the next impact is taken to be the initial kinetic energy minus the stretching energy.

Referring back to the condition of initial impact with the first stationary part, if the non-perforation criterion is not met (i.e.,

$\Delta KE \geq W_D + W_s$) local perforation is assumed and the residual kinetic energy is taken to be the initial kinetic energy of the fragment minus the sum of the work of local compression and shear. Similar calculations are then performed on the remaining stationary parts.

Referring to Figure 4-1, a longitudinal section of the low pressure turbine, it is noted that the hub is greater in width than the rim. This second impact of the disc segment is also considered in the loss of energy. Also note that the blade ring adjacent to some of the discs are not directly opposite the center of the disc. In these cases, the energy required to stretch and/or bend the blade ring out of the path of a radially ejected disc segment is calculated and compared to the sum of the compression and shear work. The smaller of the two is used in subsequent calculations. Figure 4-2 shows the remaining structural parts of the low pressure turbine.

4.2 Summary of Missile Characteristics to be Considered

Although bursting of a turbine disc is extremely remote, for design purposes it is postulated that any one turbine disc ruptures between rated speed and the maximum design speed of 133%. Applying the calculational method described in Section 4.1, it was determined that all of the low pressure disc will be contained within the casing at rated speed except for discs No. 4 and 6; at 133% of rated all discs will be contained except for discs No. 4, 5, and 6. The initial translational kinetic energies for these one-quarter segments at rated speed would be 11.5×10^6 ft-lb, 12.4×10^6 ft-lb and 8.8×10^6 ft-lb for discs No. 4, 5 and 6 respectively. At the 133% speed the corresponding initial translational kinetic energies would be 20.3×10^6 ft-lb, 21.9×10^6 ft-lb and 15.6×10^6 ft-lb. The translational kinetic energies upon leaving the turbine casing are tabulated in Table 4-1. Weights, areas, maximum dimensions and exit translational velocities are also tabulated in Table 4-1. Translational velocities on impact have been estimated for low trajectory missiles (LTM) and high trajectory missiles (HTM) by neglecting air drag in the case of the LTM and taking credit for air drag for the HTM.

Also included in Table 4-1 are the expected range of trajectory angles. The range of values of the angle measured with respect to the original plane of disc rotation are based on the maximum deviations that could be imparted by the adjacent stationary parts.

4.3 Concrete and Soil Penetrating Capabilities of Missiles

4.3.1 General Equations

For this analysis, the work of Amirikian[1] appear most applicable where the missile penetration into an infinite slab is predicted by the empirical relation:

$$D = k\, A_p\, V'$$

where

D = penetration depth, ft.

k = penetration coefficient (experimentally determined), $ft^3\ lb^{-1}$

A_p = sectional pressure of the missile, obtained by dividing missile weight by appropriate frontal area, lb/ft^2

$$V' = \log_{10}\left[1 + \frac{V^2}{215,000}\right]$$

V = missile velocity, ft/sec.

For finite slabs, as would be the case for concrete walls, Amirikian reports Navy experiments which resulted in a correction factor for finite slabs:

$$D' = D\left[1 + e^{-4\left(\frac{T}{D} - 2\right)}\right]$$

where

---

(1) A. Amirikian, "Design of Protective Structures," NAVDOCKS P-51, August 1950 (Presented at annual meeting of the ASCE, Chicago, Illinois, October 11-14, 1950).

D = penetration depth in an infinite slab, ft.

D′ = penetration depth in a finite thickness slab, ft.

T = slab thickness, ft.

Rearrangement of the above equation shows that $D = T/2$ gives complete penetration. Substituting this result in the first equation gives

$$T = 2k\ A_p\ V'$$

where T can be interpreted as the thickest slab that will be perforated by a missile.

4.3.2 Soil Penetration Depth

For estimating the depth of missile penetration into soil, the equation for penetration into an infinite slab applies (i.e. $D = k\ A_p\ V'$). The penetration coefficient is taken as 0.0482 $ft^3\ lb^{-1}$ corresponding to soil with vegetation which would approximate the soil conditions above a buried component or structure. In general the penetration in soil is sufficiently deep so that missile rotation (rotation in the plane of the disc) during penetration will tend to expose the rim, edge and hub regions. Hence, a reasonable estimate of the sectional pressure for penetration into the soil is the weight of the missile divided by the average of the minimum and maximum projected area of the disc segment.

Considering the impact angle of the missile with the generally horizontal ground surface, the HTM yields the deepest penetration measured normal to the ground surface. The LTM has a greater absolute penetration capability but because of the angle of impact results in a much reduced effective penetration. Thus, local missile protection by soil is dictated by the HTM. The 1/4-segment missile of disc no. 4 results in the deepest penetration which is 4.0 ft at rated speed and 10.7 ft at 133% of rated. These penetration depths are included in Table 4-2.

4.3.3 Concrete Thickness to Prevent Perforation

For determination of the concrete thickness required to prevent local perforation, the latter equation applies (i.e., $T = 2k\ A_p\ V'$). The sectional pressure to be used in this case is conservatively taken to be the weight of the missile divided by the projected area of the rim that is engaged. If the depth of penetration is equal to or greater than the maximum distance from the edge of the rim to the rim chord, h (illustrated in Figure 4-3), then the projected rim area is equal to the maximum projected rim area, A_1. For penetration depths less than h, the projected area of the rim is reduced accordingly.

The penetration coefficients, k, for reinforced concrete are shown in Figure 4-4. The specified compressive strength for nearly all of the concrete structures, including containment concrete, is 3000 psi minimum. This corresponds to a penetration coefficient of 0.0035 ft^3-lb^{-1}.

The missile resulting from disc no. 4 has the greatest penetrating capability. Applying the above equation and this penetration coefficient to the characteristics of the disc no. 4 missile at rated speed result in concrete thickness of 1.21 ft and 1.30 for the HTM and LTM, respectively; at 133% of rated, the corresponding values are 3.24 ft and 4.01 ft. These values are also included in Table 4-2. It should be noted that these thicknesses could be substantially reduced for the containment if credit is taken for anti-scabbing afforded by the steel liner.

4.3.4 Concrete Thickness to Stop Missile

Since leaktightness is not required, it is sufficient that a concrete structure be adequate to simply stop the missile, i.e., local perforation can occur but the missile does not completely pass through. In this case, there may be some low energy concrete missiles ejected from the side of the structure opposite the initially impacted surface.

To determine the minimum concrete thickness sufficient to stop the missile requires a "double impact" calculation. For design considerations, the missile is assumed to first impact on the rim followed by a second impact involving the hub. The steps used in this analysis are (refer to Figure 4-3 to follow the discussion):

1) Using the initial impact velocity, and impact area A_1, determine

$$D_1 = k \frac{W}{A_1} V_1'$$

2) If $2D_1 > T$, determine the residual kinetic energy, KE_2, by the expression $KE_2 = KE_1 \left[1 - \frac{T}{2D}\right]$, where KE_1 is the translational kinetic energy of the missile just prior to first impact.

3) Determine the residual velocity

$$V_2 = V_1 \left(1 - \frac{T}{2D_1}\right)^{1/2}$$

4) Using V_2 and the impact area A_2' (equal to the "net" hub area), determine

$$D_2 = k \frac{W}{A_2'} V_2'$$

5) If $2D_2 \leq T$, the thickness is adequate to stop the missile.

Applying the above procedure to the disc no. 4 missile at rated speed for 3000 psi concrete results in a minimum concrete thickness to stop the missile of 0.51 ft and 0.55 ft for the HTM and LTM, respectively; at 133% of rated, the corresponding values are 1.5 ft and 1.83 ft. These values are also listed in Table 4-2.

TABLE 4-1

SUMMARY OF MISSILE CHARACTERISTICS

Characteristics for 1/4 Segments	Disc #4	Disc #5	Disc #6
Weight, Lbs.	2865	3330	3711
Areas, $Ft.^2$			
- Impact on rim (A_2')*	1.66	2.42	3.01
- Impact on hub (A_2)*	3.14	3.24	3.96
- Net hub with initial rim impact (A_2')*	2.36	2.13	2.50
- Average Projected inplane of initial rotation	3.45	3.26	3.93
Largest Dimension, Ft.	6.03	5.80	5.16
Exit Translational Kinetic Energy, 10^6 Ft-Lb.			
Rated Speed	2.7	Contained	1.5
133% of Rated	11.0	2.3	7.7
Exit Translational Velocity, Ft/Sec.			
Rated Speed	246	Contained	161
133% of Rated	497	212	365
Impact Translational Velocity, Ft/Sec.			
- Low Trajectory Missile			
Rated Speed	246	Contained	161
133% of Rated	497	212	365

* Refer to Figure 4-3

TABLE 4-1 (Cont'd)

SUMMARY OF MISSILE CHARACTERISTICS

Characteristics for 1/4 Segments	Disc #4	Disc #5	Disc #6
Impact Translational Velocity, Ft/Sec. (cont'd)			
- High Trajectory Missile			
Rated Speed	236	Contained	158
133% of Rated	429	206	339
Exit Trajectory Angle, Degrees			
- Angle Measured with Respect to Original Plane of Disc Rotation	-5 to +5	-5 to +5	0 to 25
- Angle Measured with Respect to Original Horizontal Axis of Disc	0 to 360	0 to 360	0 to 360
(One missile per quadrant, 90° between adjacent missiles)			

TABLE 4-2

CONCRETE THICKNESSES AND SOIL
PENETRATION DEPTHS FOR MOST
ENERGETIC MISSILE

Type of Missile	Minimum Concrete Thickness Required (3000 Psi Concrete) (feet)		Penetration Depth in Soil (feet)
	For No Perforation	For Stopping Missile	
HTM			
Rated Speed	1.21	0.51	4.0
133% of Rated	3.24	1.50	10.7
LTM			
Rated Speed	1.30	0.55	- -
133% of Rated	4.01	1.83	- -

LOW-PRESSURE ELEMENT

1800-RPM DOUBLE-FLOW DESIGN

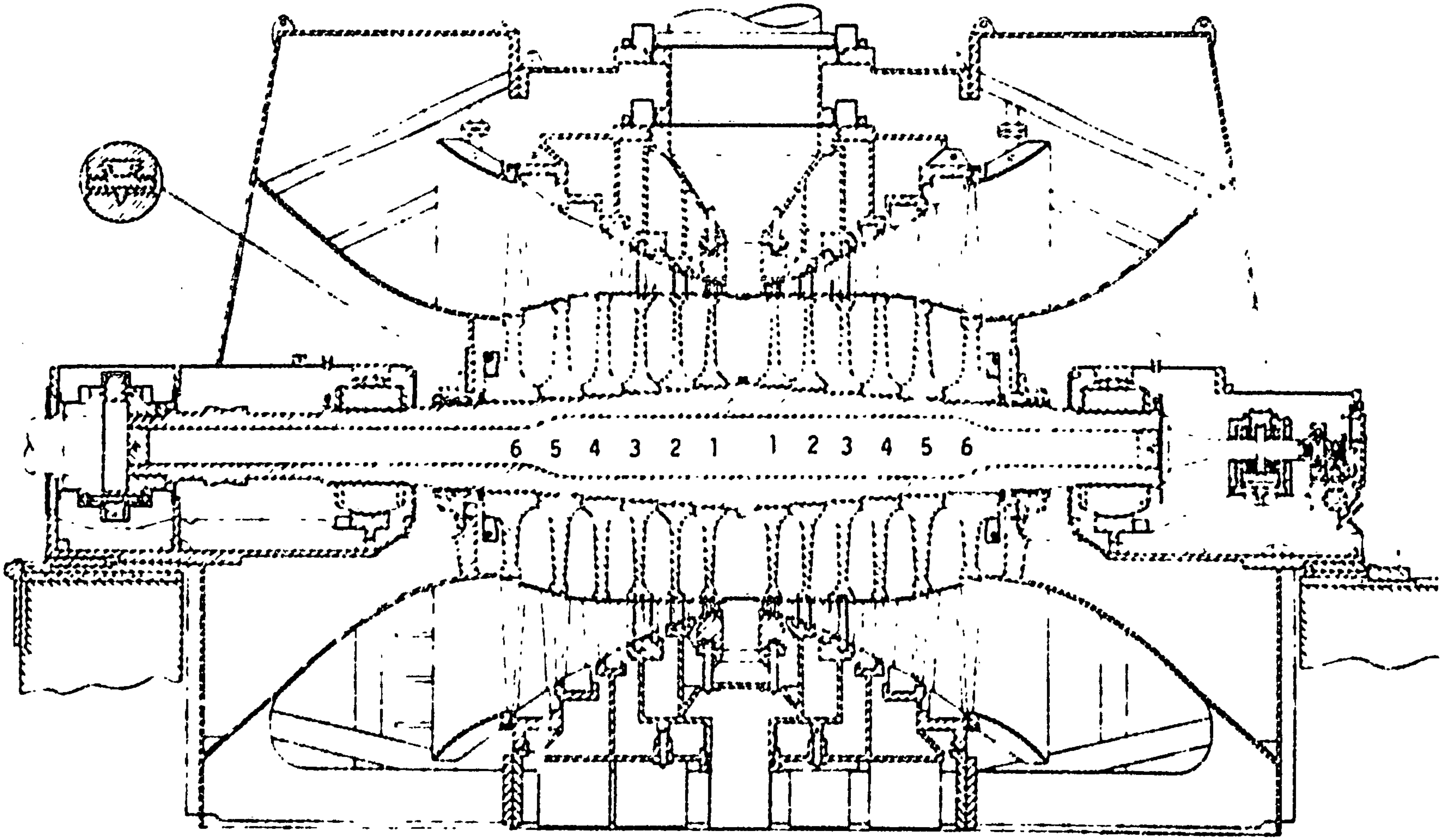
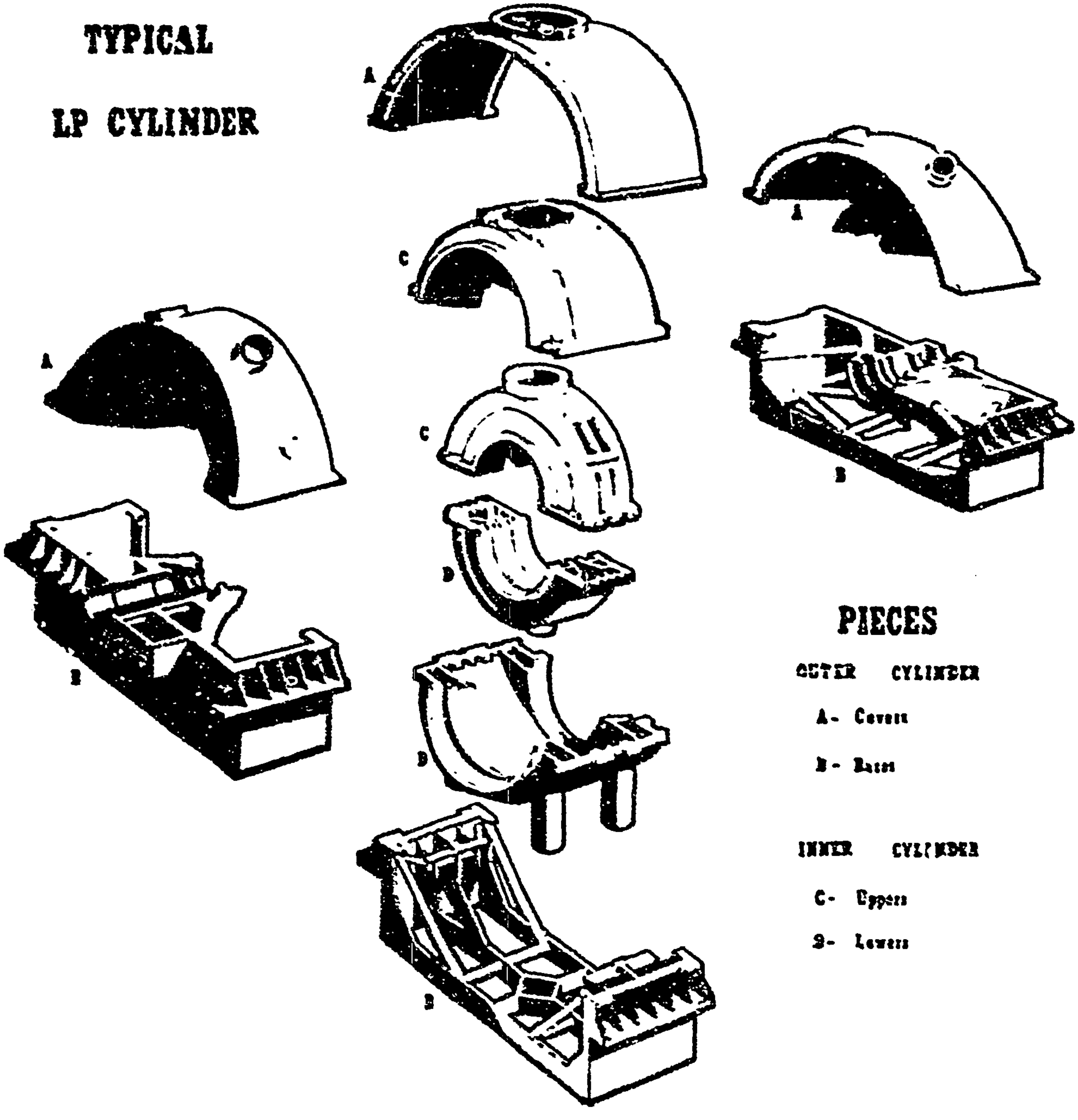


FIGURE 4-1
Supplement 12
7/70



Exploded View of Low Pressure Unit

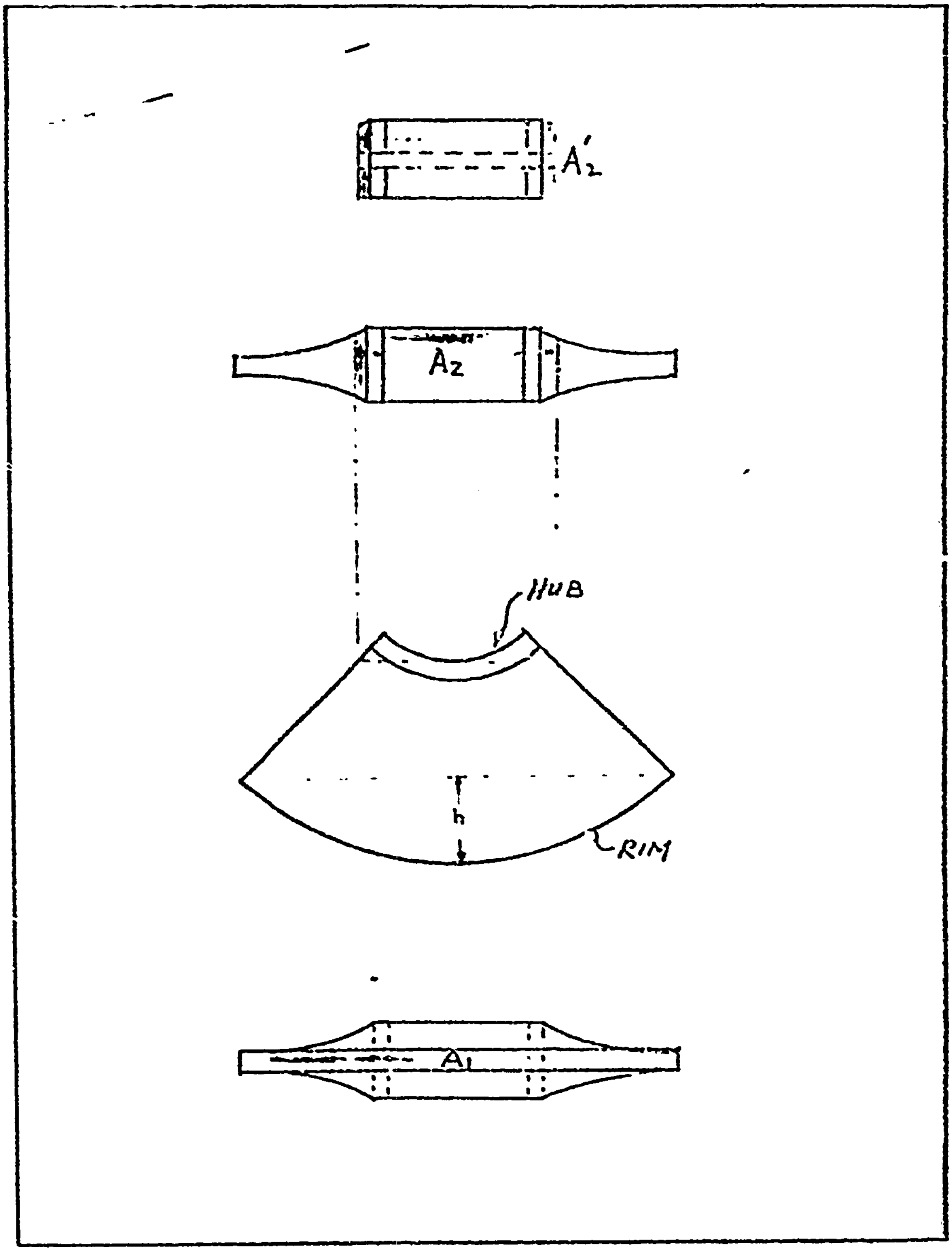


Fig. 4-3

TURBINE MISSILE IMPACT AREAS

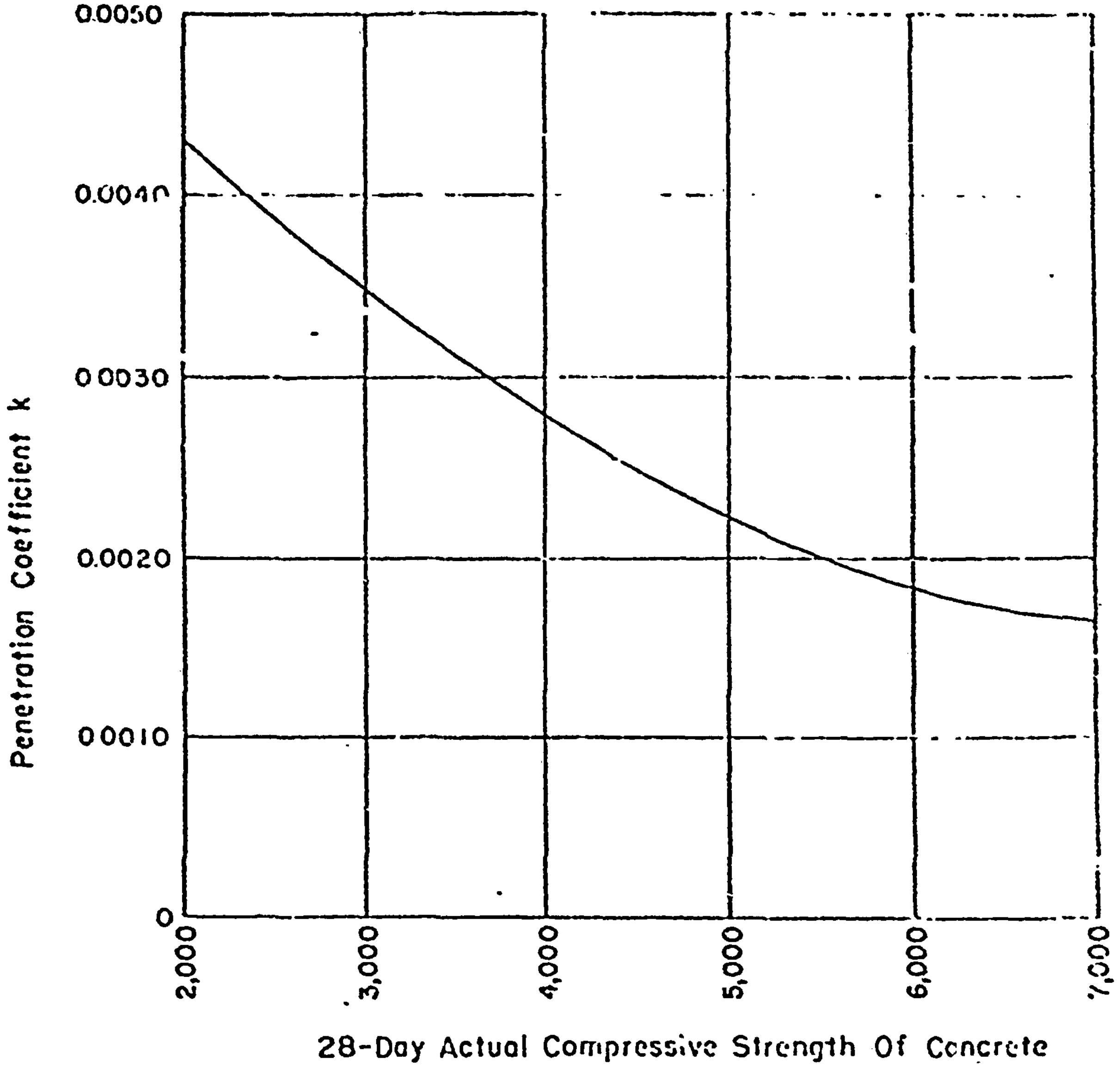


Fig. 4-4

VALUES OF PENETRATION COEFFICIENT (k) FOR REINFORCED CONCRETE

from (1) A. Amirikian, "Design of Protective Structures," NAVDOCKS P-51 August 1950 (Presented at annual meeting of the ASCE, Chicago, Illinois, October 11-14, 1950).

5.0 ASSESSMENT OF PLANT CAPABILITY TO WITHSTAND POSTULATED TURBINE MISSILE

As indicated in Section 1, rupture of a low pressure turbine disc at or below design speed is postulated for design purposes, even though this failure is deemed unlikely because of design conservatism and original quality control.

In the subsequent study, it is postulated that a disc ruptures in four quarters, thus generating four missiles.

For the purpose of evaluating damage to vital equipment or structures, only one missile need be considered because:

a) The four quarters are ejected 90° apart;

b) The two quarters exiting near or below the horizontal axis of the turbine will be stopped by the turbine pedestal or the structures below the level of the turbine operating deck;

c) The remaining two quarters would be ejected above the turbine operating deck;

d) All of the vital equipment but a portion of the service water system is located on the east side of the turbine axis (as shown in Figures 5-1 and 5-2). Therefore, one missile could impact a vital structure on the east side and the other, ejected 90° apart, could hit the portion of the service water system located on the west side. However, as it will be described later, loss of service water can and will be accommodated by making use of the existing city water or hydrant connections, shown in Figure 5-3. This provision will be such that all the vital equipment required for safe shutdown are located on the east side of the turbine axis. Hence, the missile ejected westward will be irrelevent from the safe shutdown standpoint.

5.1 Consideration of Direct Loss of Reactor Coolant

The reactor coolant system is contained inside the reactor containment structure. This is a reinforced concrete vertical right cylinder with a flat base and hemispherical dome. A welded steel liner with a minimum thickness of 1/4 inch is attached to the inside face of the concrete shell to ensure a high degree of leak-tightness. As shown in Figure 5-4, the wall of the vertical cylinder is 4.5 feet thick and the dome is 3.5 feet thick. The 28-day standard compressive strength of the concrete is 3000 psi.

For assessing the adequacy of these concrete thicknesses, it is conservatively assumed that the turbine missile impacts normal to the containment surfaces. Hence the effect of HTM is considered for the dome and the LTM for the side-walls. Referring to Table 4-2, at 133% of rated, the minimum concrete thickness required to prevent local perforation of 3000 psi concrete (with no credit for anti-scabbing provided by a steel liner) for the HTM is 3.24 feet and 4.01 feet for the LTM. Since the corresponding thicknesses of the containment concrete are greater than these values, the containment structure will not be locally perforated even for missiles generated at the maximum design speed. This is more than sufficient to prevent direct loss of reactor coolant since turbine missile stoppage rather than no perforation is adequate.

5.2 Considerations to Maintain Plant in a Safe Shutdown Condition

Rupture of a low pressure turbine disc at speeds below the emergency overspeed set point (106%) will trip the turbine due to loss of condenser vacuum resulting from the damage produced by the ruptured disc. Rupture at or above this set point requires that turbine trip has occurred. Since the reactor trips automatically following a turbine trip, both turbine and reactor trip are assured in the event of the turbine missile incident. Hence, maintaining the plant in a safe shutdown condition requires only minimal performance of the decay heat removal, reactor coolant makeup and boration functions.

Before proceeding with the evaluation of the capability of maintaining the plant in a safe shutdown condition, the components related to the normal performance of these functions following a turbine trip and reactor trip will be identified.

Decay Heat Removal

With sufficient fluid in the reactor coolant system, adequate decay heat removal depends on the performance of the steam generator secondary side since the core decay heat removal is assured by the circulating reactor coolant. In the first few minutes the reactor coolant is circulated by mechanical coastdown of the reactor coolant pumps and subsequently by natural circulation. Decay heat removal from the secondary side depends on the steam relief system and the auxiliary feedwater system.

The steam relief system removes thermal energy by releasing steam to the atmosphere via the steam relief valves or the condenser via the turbine bypass. For the turbine missile incident, credit cannot be taken for the turbine bypass since the bypass valves will not open with loss of condenser vacuum.

The steam dump to the atmosphere consists of five safety valves located on each of the four main steam lines outside the reactor containment and upstream of the non-return valves as illustrated in Figure 5-5. The five safety valves in each main steam line are set to relieve at 1065, 1080, 1095, 1110, and 1120 psig, respectively. These twenty valves have a total capacity in excess of the equivalent nominal rated steam flow. In addition, there are four power operated relief valves which are capable of releasing 10% of the equivalent nominal rated steam flow. These valves are automatically controlled by pressure or may be manually operated from the main control board. Provisions are presently being made to manually control these valves from the auxiliary feed pump room.

The auxiliary feedwater system supplies high pressure feedwater to the steam generators in order to maintain a water inventory for heat removal from the reactor coolant system upon inoperability of the main feedwater system. Upon loss of condenser vacuum the valves in the lines supplying steam to the turbine drive of main feed pumps close automatically. Hence, the auxiliary feedwater system must come into operation following the turbine missile incident.

The auxiliary feedwater system is basically composed of:

1) Two motor driven feedwater pumps
2) One turbine driven feedwater pump
3) Auxiliary steam admission to the drive of the turbine driven feedwater pump.
4) Auxiliary feedwater discharge piping
5) Main feedwater lines
6) Auxiliary feedwater suction piping
7) Auxiliary feedwater source

This system is sized so that any of the auxiliary feedwater pumps can supply the required auxiliary feed. These components, except for the auxiliary steam admission to the drive of the turbine driven feedwater pump, are illustrated in Figures 5- 7 through 5-11. Steam to drive the turbine is supplied from two of the main steam lines upstream of the stop valves just outside of the containment. The turbine is started by opening a pressure reducing valve located in the auxiliary feed pump room. This valve opens automatically upon loss of power.

Reactor Coolant Makeup

Reactor coolant makeup is required to maintain sufficient fluid in the reactor coolant system so as to guarantee that decay heat is removed continuously from the core. At the same time, however, the boration concentration of the reactor coolant system should not be reduced substanttially in order to maintain a sufficient shutdown margin. Hence, for the incident under consideration, the makeup source would normally be from the refueling water storage tank. Makeup from the refueling water storage tank involves the following components:

1) Refueling water storage tank
2) Discharge piping from the refueling water storage tank to the suction of the charging pumps
3) Three pumps (one is sufficient)
4) Discharge piping from the charging pumps
5) Component cooling system to provide cooling to the charging pump fluid drive
6) Service water system to cool the component cooling water

Boration

Boration is required to compensate for the long term xenon decay transient. The normal boration system includes the following components:

1) Two boric acid tanks and boric acid batching tank and heaters
2) Two boric acid transfer pumps (one is sufficient)
3) One boric acid filter
4) Piping and heat tracing from the tanks to the suction of the charging pumps
5) Three charging pumps (one is sufficient)
6) Discharge piping from the charging pumps to the RCS.
7) Component cooling system to provide cooling to the charging pump fluid drive coupling
8) Service water system to cool the component cooling water

The areas of the plant related to the normal performance of decay heat removal, reactor coolant makeup and boration functions were reviewed to possible damage by the turbine missiles. Those areas, illustrated in Figure 5-12, identified as requiring consideration of missile impact one:

1. Service water system
2. Condensate storage tank and piping
3. Refueling water storage tank and piping
4. Shield wall area
5. West end of auxiliary building
6. Electrical penetration area
7. Diesel generators
8. Control room

Areas not included in the above are either afforded enough concrete protection to stop the missile or house equipment not related to the normal performance of these functions.

Impact by the missile is postulated on one of the eight regions at a time and an evaluation made with respect to performing the required functions. In some of these areas the evaluation consists of identifying the available back up and the appropriate plant personnel actions. In other areas plant changes are identified as required to maintain the necessary functions.

In the foregoing evaluations credit is taken for restoration of outside power within one hour.

...1 Service Water System

The service water pumps located on top of the intake structure and the discharge piping up to the distribution header are sufficiently redundant and separated to preclude the loss of service water. The redundant service water piping from the header to the primary auxiliary building (Figure 5-2) could conceivably be damaged by the turbine missile since they are not sufficiently separated.

These pipes rise out of the ground adjacent to the west end of the auxiliary building and could both be damaged by a missile generated at 100% and 133% speed. The remainder of these pipes are buried nearly six feet underground and this is sufficient to prevent damage by the 100% speed missile. The 133% speed missile may be capable of penetrating this soil and damaging both pipes.

The functions affected by loss of service water are reactor coolant makeup and boration. Service water is required for cooling the diesel generators and remove heat from the component cooling system which cools the fluid drive coupling of the charging pumps.

As far as the diesel generators are concerned, they are not considered vital for safe shutdown since there is sufficient amount of time (about 24 hours: reference is made to section 5.2.8) to restore outside power in case it is lost following turbine disc rupture.

Cooling to the charging pumps will be accomodated by making permanent connections to the charging pump cooling water header that will allow direct cooling via the city water supply. On the inlet side of the header, new isolation valves will be installed in parallel with the existing isolation valve. A flanged connection upstream of the new isolation valves will permit the operators to make the connection to the city water supply. New isolation and drain valves will be installed parallel to the existing isolation valve on the outlet side of the header. Initial drainage flow would go to the floor drain and would be eventually piped outside the building. The operator will have sufficient time (reference is made to section 5.2.8) to open and close the manual valves as required and to make the necessary piping connections.

5.2.2 Condensate Storage Tank and Piping

The condensate storage tank is not shielded and could be damaged by a missile generated at either 100% speed on 133% speed. Piping from the tank to the auxiliary feed pump room is buried sufficiently deep so that it would not be damaged by a 100% speed missile but could possibly be damaged by the 133% speed missile.

The function related to maintain the plant in a safe shutdown condition that would be affected by loss of this tank (or piping) is decay heat removal since this is the normal auxiliary feedwater source. However, the back up supply (i.e. the city water supply) and its related piping are on the opposite side of the auxiliary feed pump room as shown in Figure 5-11.

Hence, a missile damaging the condensate storage tank or piping would leave the city water supply intact and available. Opening of an air-operated valve which can be accomplished from the control room is the only action required to take suction from the city water supply.

5.2.3 Refueling Water Storage Tank and Piping

As in the case of the condensate storage tank, the refueling water storage tank is not shielded and could be damaged by the 100% and 133% speed missiles. The shutdown function related to the use of this tank is that of reactor coolant makeup.

In the event the missile makes this makeup source unavailable, initial makeup is available from the boric acid tanks and the volume control tank which have a sufficient capacity to allow the operators to provide an alternate source of makeup if necessary. This alternate source of makeup can be provided from processed water into the batching tank. No action is required for makeup from the normally aligned volume control tank. To take suction from the boric acid tanks, one boric acid transfer pump must be started and one remote operated valve opened.

5.2.4 Shield Wall Area

The shield wall area houses equipment and components associated with the decay heat removal function. As discussed previously, performance of this function involves the steam relief system and auxiliary feedwater system. The effect of the missile on these functions is discussed separately.

Steam Relief System

The portion of the steam system outside the containment is shown in Figures 5-5 and 5-6. As indicated, there is a 4 foot thick shield wall that is adequate to stop the most energetic missile, and as such, protects most of the steam system from a LTM.

The steam lines and their stop and check valves extend beyond this wall and are vulnerable to a LTM. Because of adequate separation between steam lines, a LTM missile could damage only one of these lines and/or their corresponding valve. This has been shown to be a tolerable situation in Section 14.2.5 of the FSAR.

The HTM on the other hand could conceivably damage two steam lines and possibly the corresponding main feedwater lines. However, recent analysis shows that this situation is not more severe than the rupture of the single steam line. The basic reason for this is that although the primary side cools down more rapidly giving rise to more rapid reactivity increase (via the temperature decrease), the pressure is also reduced more rapidly thus allowing more prompt actuation of the safety injection system. The beneficial effect of the latter overrides the deleterious effect of the former.

The following two conclusions can be drawn:

a) Potential steam line ruptures do not expose the core to untolerable transients; and

b) Steam relief capability required to remove decay heat during shutdown is not jeopardized.

Auxiliary Feedwater System

As illustrated in Figures 5-6 through 5-8, the auxiliary feedwater pumps, located in the auxiliary feed pump room, have the 4 foot thick shield wall to stop a LTM and two floors of 2 foot thick concrete to stop the HTM. Since these pumps are protected and the motor driven pumps are powered by the emergency diesels, missile impact in this area does not impair feedwater pump operation (one of the three is sufficient). In addition, since the motor driven pumps will be operable, reliance on the turbine driven auxiliary feedwater pump is not required.

The auxiliary feedwater discharge piping is illustrated in Figure 5-9 and 5-10. It is conceivable even though very unlikely, that the HTM could fail between the steam lines and possibly damage all four auxiliary feedwater pipes. The area of concern is the piping run from the point they come up through the second concrete roof to the second main feedwater connection. Local protection will be provided to prevent rupture of all four auxiliary feedwater lines, should a HTM land in this area. If a missile lands on the remaining auxiliary feedwater pipe runs, the ruptured lines can be isolated by closing valves located in the protected auxiliary feedpump room. Auxiliary feedwater to two steam generators provides adequate cooling.

A portion of main feedwater lines is required for the introduction of feedwater since the auxiliary feedwater lines are connected individually to these lines near the containment wall. Only two of these main feedwater lines need be intact for the reasons discussed before, and this will be insured by their separation.

The redundant and separated auxiliary feedwater sources were discussed previously.

Thus, with the addition of local protection on the critical region of the auxiliary feedwater lines, impact of the postulated missile in shield wall area will permit the required decay heat removal function.

5.2.5 East End of Auxiliary Building

The east end of the auxiliary building has a sheet metal roof and the equipment located there is exposed to damage by both the 100% and 133% speed missiles. It is conceivable that a single missile could damage the boric acid tank piping and component cooling water system located there. Impact in this area could affect both the reactor coolant makeup and boration functions.

The permanent connection from city water to the component water headers discussed previously allows operation of the charging pumps without reliance on the component cooling water system. The charging pumps themselves are located in a protected area of the auxiliary building. Hence, makeup and boration can be accommodated via a charging pump with suction from the refueling water storage tank. Borating by this method may require operation of the air operated pressurizer relief valves. It is estimated that it would require 8 to 10 hours to change the reactor coolant boron concentration by 200 ppm to compensate for the xenon decay. This time is acceptable since the positive reactivity associated with xenon decay need not be compensated for within a period in excess of 24 hours (Reference is made to Section 5.7.8). To take suction from the refueling water storage tank requires opening one remote operated valve.

5.2.6 Electrical Penetration Area

This area is shown in Figures 5-13 and 5-14. The vertical wall here is 2 feet thick which is adequate to stop the LTM at 133% of rated speed. The roof of the electrical penetration area is light weight concrete with no reinforcement and has a maximum thickness of 8 inches. Thus, the electrical penetration area might be damaged by an HTM generated at rated or the maximum design speed. The roof area has a probability of less than 1/1000 of being hit. Damage is anticipated but due to physical separation at the penetration (reference is made to Figure

5-15), loss of all pressurizer pressure and level channels and loss of all steam generator level channels is incredible. Therefore, sufficient information on reactor and steam systems will be available, if the missile hits the electrical penetration area.

5.2.7 Diesel Generators

The diesel generator annex could possibly be damaged by the turbine missile to the extent that emergency power from this facility is unavailable. However, since electrical power is not needed for approximately 24 hours (reference is made to section 5.2.8) reliance on outside power is justified.

5.2.8 Control Room

Due to the relative location between the Unit 2 control room and turbine-generators of Unit 2 and 3 (see Figure 5-1) the control room can only be hit by a HTM. The probability of having a HTM hit somewhere on top of the control room is less than 4×10^{-3}.

To stop the most energetic HTM generated at rated speed requires slightly more than 6 inches of concrete. Since the floors in the control building are 6 inches thick, such a missile may pass through the upper floor but would easily be stopped by the second floor. In this case, damage could be caused to equipment located on the upper two floors: control room pannels and board on the upper floor and distribution and control panels on the second floor. The 480V switchgear located on the bottom floor would not be damaged. At the maximum design overspeed, a HTM could possibly damage equipment located at all three floors. Hence, in this case, some of the 480V switchgear may also be damaged.

The results of missile damage to the control room are of course difficult to predict in precise terms. Depending on where the missile lands, direct control and power to some vital equipment could be lost. On the other hand, some or all control signals may persist and keep the controlled components in the state at the time of the accident. Still another condition could be loss of an electrical and/or air signal causing the component to change to the loss of signal mode. In order to provide a basis for evaluation, the following conservative assumptions have been made regarding the status of the control room just following impact by the missile:

1. Electrical cabling supplying power to vital equipment are severed.

2. Components (e.g., valves) remain in their state at the time of the accident.

3. All normal instrumentation is lost.

In applying the second assumption, the most severe state of all the components (with respect to safe shutdown) is employed. However, those possible states that do not normally persist for extended periods are not included as a possible state at the time of the accident. For example, the steam generator blowdown valves are assumed to be open since they can be open during periods where the plant is changing power. Alternately, the power operated pressurizer relief valves are normally not exercised during normal operation and as such they are assumed to be closed. The valve status shown in Table 5-1 is assumed for this evaluation.

5.2.8.1 Time Requirements

Within the framework of these three assumptions and the corresponding valve status, analysis was performed to establish the times required for manual actions and damage repair. This is summarized in the next few paragraphs.

a) Decay Heat Removal

To ensure the decay heat removal function, feedwater must be provided to the steam generators within 30 minutes of reactor scram if steam generator boil-off is to be avoided.

As indicated in Table 5-1, the steam generator blowdown valves and sample isolation valves are assumed to be open initially. This mass outflow rate is small compared to the outflow from the safety valves and becomes even less important when the turbine driven auxiliary feed pump is started. Thus, the open status of these valves is not critical but should be closed in about one hour to limit unwanted secondary side losses.

While adequate steam generator level is assured by operation of the turbine driven auxiliary feedwater pump within 1/2 hour, there could be carryover from the steam generator turbine drive of this pump. It is estimated that full feedwater flow from this pump would not produce water carryover within the first two hours of operation. Hence, within about two hours it would be desirable to have indication of steam generator level.

b) Reactor Coolant Makeup

In estimating the minimum time required to provide reactor coolant makeup, two cases were considered without charging or safety injection: the first corresponding to reactor coolant discharge from the normal letdown line, the excess letdown line and through the seals of the four reactor coolant pumps; and the second case with leakage only through the seals of the reactor coolant pumps. The total initial leakage rate for the first case is approximately 260 gpm and 40 gpm for the second.

The SLAP code (described in Section 14.3.2 of the FSAR) employed in this analysis resulted in core uncoverage in approximately 7 hours in the first case and 40 hours in the latter.

The rate of decrease of reactor coolant volume in both instances were found to be nearly linear in time. Hence, assuming it took the operators as long as two hours to close the required valves, it would require additional 28 hours to uncover the core with the seal pump leakage. This means that reactor coolant makeup would not be required for over 30 hours.

c) Boration

To estimate the minimum time required to compensate for the xenon transient it was assumed that at the time of the turbine incident, the xenon decay is at its maximum rate of approximately 0.13%/hour. This would correspond to operating the plant for a long period of time (xenon at its equilibrium value), going to a hot shutdown condition for 10 to 14 hour period and then returning to full power.

For the end of life conditions, there would be a design minimum of 2% shutdown margin (assuming a stuck rod). In addition to this, there is an additional 0.6% corresponding to the nuclear design criteria of 10% control rod control margin above the design value of 6%. Nuclear calculations on the depressurization transient result in a negative reactivity of 0.6%. Thus, for the end of life core with a stuck rod, a total of 3.2% negative reactivity would exist during the period of time without makeup or boration capability. With the maximum positive reactivity addition rate of 0.13%/hour associated xenon decay, it would take 24 hours to return to critical. Before this time it would require an increase of 200 ppm boron to compensate for the positive xenon reactivity. Borating from the boric acid tanks would take approximately 30 minutes to change the boration concentration by 200 ppm.

Without the stuck rod assumption, there would be an additional 1% negative reactivity available or a total of 4.2%. Taking into account the decreasing rate of positive reactivity addition at the time of the xenon decay, it would require more than 40 hours for the reactor to return to critical. Prior to this point in time it would require an increase in boron concentration of about 100 ppm. This change in concentration could be made in about 15 minutes by borating from the boric acid tanks.

While boration capability itself is not required for more than 24 hours, the ability to borate from the boric acid tanks depends on the solubility of the borated water in the boric acid tanks and the piping from these tanks to the suction of the charging pumps. The tanks themselves would not freeze up before four or five days without power to the heaters. The borated water in the piping, on the other hand, could freeze within about an hour without power to the heat tracing. Hence, to assure borating capability at the required time, these lines must be flushed with clean water within this hour period. To accomplish this flushing action, use will be made of the existing primary water flushing provisions. These provisions will be futher augmented by installing a cross connection (at the discharge of the boric acid tanks) to the city water supply.

Summarizing, the minimum time requirements are:

Function	Requirement	Time
Decay Heat Removal	Feedwater to steam generators	1/2 hour
	Closure of steam generator blowdown valves and sample isolation valves	1 hour
	Steam generator level indication	2 hours
Makeup	Closure of normal and excess letdown isolation valves	2 hours
	Charging Capability	30 hours
Boration	Flushing of piping between boric acid tanks and charging pumps	1 hour
	Boration from boric acid tanks	>24 hours

5.2.8.2 Short Term Operator Actions

These actions pertain to those required in the first few hours following the postulated impact of the missile on the control room.

1. The operators have approximately 30 minutes to make sure there is sufficient auxiliary feedwater available (~800 gpm). To be assured of this, the following steps are necessary:

   a) Check that the pressure reducing valve on the steam admissions line to the turbine driven auxiliary feedwater pump is open (located in the auxiliary feed pump room). This valve should automatically open upon loss of power. If this valve is closed it can be opened manually.

   b) Check that the auxiliary feedwater control valves are open. These are air operated valves and fail open upon loss of air. These valves, located in the auxiliary feed pump room, can be opened manually.

2. To make sure that the inventory of water in the steam generators remains sufficient, within about one hour the operators must:

   a) Check that the steam generator blowdown valves are closed (these valves may or may not be open depending on the mode of plant operation). These valves located in the piping tunnel are air operated and fail closed. They can be closed manually.

   b) Close the normally open steam generator sample isolation valves (these valves are also air operated and fail closed). These valves located in the piping tunnel can be closed manually.

3. To assure that boration from the boric acid tanks can be accomplished when required (>24 hours), the operators have approximately one hour to flush the piping between the boric acid tanks and the suction of the charging pumps. This can be accomplished by connecting the city water supply to the new flushing valve at the discharge of the boric acid tanks and opening the existing drain valve at the suction of the charging pump. Once the flushing operation is complete, these valves can be closed and the lines will be available for the boration operation.

4. Within about two hours, the reactor coolant boundary must be assured. The operators must check that the normal and excess letdown line isolation valves are closed. These valves are air operated and close upon loss of air pressure or electrical power. The most direct means of shutting these valves is to interrupt the electrical power.

5. To prevent water carryover from the steam generators to the drive of the turbine driven auxiliary feedwater pump, within about two hours, the operators should check that the steam generators are functioning. If not, emergency steam generator level instrumentation consisting of a portable emergency instrumentation pack (battery powered) and charts identifying penetration wire numbers and channel conversion factors will be employed. The instrumentation pack will be connected to tagged wires at the containment penetration or the control end of the tunnel. Level indication of one steam generator would be adequate. Flow control can be accomplished by manually adjusting the auxiliary feedwater control valves located in the auxiliary feedpump room.

5.2.8.3 Long Term Operator Action and Damage Repairs

With the completion of the short term operator actions, the plant can be maintained in a safe shutdown condition without electrical power or any other hydraulic system for about 24 hours. Thus, these actions and damage repair deal with the 24 hour period following the accident.

1. The condensate storage tanks contain a 24 hour supply (minimum) of auxiliary feedwater. Within this time period the city water supply should be aligned to the suction of the turbine driver auxiliary feedwater pump. The valve in the city water line can be opened from the control room or in the auxiliary feed pump room with the nitrogen bottles located there.

2. The electrical equipment required for makeup is one charging pump and for boration one charging pump and one boric acid transfer pump. Charging capability is required in about 30 hours and boration from 24 to 40 hours depending on the initial shutdown margin. The required electrical cable splicing and switchgear repair depends on the extent of control room damage and the location of missile impact.

   For missile impact in the control room (Figure 5-16) areas of concern would include the 480V switchgear and the entrance to the cable tunnel. Computer studies show that the probability of hitting the tunnel entrance is less than 2×10^{-5} and the switchgear is less than 3×10^{-5}. Thus either event is very unlikely, even assuming a missile is generated.

   If the missile damages 480V switchgear, it is not possible to damage all the redundant switchgear associated with the required equipment, and power can be supplied to the necessary equipment once the damaged area has been isolated. In the extremely unlikely event that missile impact is in the small area where power and control wiring enters the electrical tunnel, it is possible to sever all redundant paths of supplying power to charging pumps, etc. However, in this case, switchgear will be left intact, and repair crews can make emergency power connections to the few pieces of vital equipment by bridging damaged cables. The necessary lines will be tagged in locations which will permit identification following such an accident, and in view of the time available, the repairs above described can be performed to provide emergency power for makeup and boration.

5.2.8.4 Summary

As discussed previously, the operator actions depend on the damage caused by the missile, the objective is, however, clearly to satisfy the three essential functions: decay heat removal, reactor coolant makeup and boration. With these functions assured by the actions and repairs presented in the preceeding sections, the operators may now consider whether or not sufficient equipment is available for plant cooldown. The hot shutdown condition may, however, be held almost indefinitely.

5.3 Consequences of Radioactivity Releases

The possibility of a turbine missile causing release of fission product activity has been considered. The reactor coolant system is protected by the containment. The gas decay tanks, the volume control tank, and demineralizers are protected by the auxiliary building structure. The liquid holdup tanks are protected by the holdup tank vault structure.

Should an HTM land in the spent fuel pit pool from above (LTM will not perforate the 6'-3" vertical walls above ground level), damage to some spent fuel assemblies would occur. The impact area of a quarter disc would affect four storage cells. Allowance for secondary effects by consideration of damage to one row around the four cells gives a maximum of 18 cells damaged. Damage to fuel in these cells would not result in criticality. For purpose of determining the limiting site boundary dose, it is assumed that all 18 cells contain fuel assemblies and that these assemblies are all freshly removed from the core having decayed only 100 hours since plant shutdown.

Two cases are considered: an expected case, in which the expected characteristics of the 18 highest rated assemblies normally to be discharged at end-of-life are assumed, along with best-estimate behavior of fission products determined by tests; and a design case, in which factors are introduced to allow for uncertainities.

The expected case is summarized in Table 5-2. The resultant maximum site boundary doses are calculated to be 53.5 rem thyroid and 6.5 rem whole body.

The design case is summarized in Table 5-3. The resultant maximum site boundary doses are calculated to be 198 rem thyroid and 15.5 rem whold body.

Assuming a turbine missile is ejected, the probability of its hitting the fuel pit pool is estimated to be less than 1/1000. Secondly, assuming the turbine missile hit the pool, it is probable the dose would be much less than the maximum doses given above. For example, with one refueling per year the probability of a turbine rupture within the first 16 days (two I-131 half lives) after transfer of fuel to the storage pool is 16/365 the probability of its rupture. Further, the three turbines of Unit #2 would not be in operation for these first 16 days. Since Units 1 and 3 have a total of 5 turbines, the probability of missile generation is reduced by 5/8 during these 16 days. Since 16 days is equivalent to two I-131 half lives and three Xe-133 half lives, assuming a turbine missile-fuel pool accident, there is 97% $[1 - (\frac{16}{365} \times \frac{5}{8})]$ probability that the site boundary dose would be less than 1/4 the maximum doses given above.

Should the turbine missile hit the vicinity of the steam lines, no more than two steam lines could be damaged. Analysis for blowdown of the two steam generators shows that no core damage would result. Activity release would be dependent on RCS activity from operation with fuel defects and steam generator tube leakage, if any, during the period to cool and depressurize the RCS after the accident. With RCS activity concentration corresponding to operation with 1% clad defects and a 25 gpm tube leak (the tech. spec. limit) for 8 hours, the released activity from the RCS leakage for the duration of the accident is 13,800 Ci equivalent to Xe-133 and 120 Ci equivalent to I-131. In addition, the iodine activity in the two steam generators which blowdown is 88.4 Ci equivalent I-131 (based on the 0.48 μc/cc tech. spec. limit). The site boundary dose would be 80.5 rem thyroid and 0.6 rem whole body.

The refueling wate   rage tank and the monitor tanks are vulnerable to a turbine missi   The maximum tritium concentration in the refueling water storage tank should not exceed 2.5 μc/cc corresponding to a total of 3000 Ci tritium in the tank. The maximum concentration of tritium in the river at Chelsea from a burst release of this tritium would be 7.5×10^{-7} μc/cc which is 2.5×10^{-4} m.p.c. (See answer to question 11.9 supplied in Supplement 3 to the FSAR for discussion of the river water dilution factors.) The release of the activity contained in the monitor tanks to the river has been examined in the answer to question 11.8 of Supplement 4 to FSAR where it is shown that the resultant river concentrations at Chelsea are less than 10^{-7} m.p.c.

It is concluded that the probability of a turbine missile causing a large release of fission product activity is very low. Further, with worst case assumptions the turbine missile would not cause offsite exposure in excess of the 10CFR100 guideline.

TABLE 5-1

VALVE STATUS JUST AFTER
IMPACT OF CONTROL ROOM

Valve	Status
Pressurizer Power Operated Relief Valves	Closed
Letdown Line Valves (Normal and Excess)	Open
Steam Generator Blowdown and Sample Isolation Valves	Open
Power Operated Steam Relief Valves (Atmospheric Dump)	Closed
Turbine Bypass Valves	Closed
Reactor Isol...ion Valves	Open
Pressure Reducing Valve (in admission line to drive of turbine driven auxiliary feed pump)	Closed
Auxiliary Feedwater Control Valves	Closed

TABLE 5-2

TURBINE MISSILE ACCIDENT - EXPECTED CASE

Fuel Parameters

Reactor Power (102%)	3280 MWt.
No. of Assemblies	193
Fuel Rods per Assembly	204
Normalized Power, 18 Highest Rated Discharged Assemblies	1.16
Normalized Power, Highest Rated Discharged Assembly	1.29
Axial Peak/Avg., Highest Rated Discharged Assembly	1.37

Activity Release Data

Isotope	Release Fraction	Bubble Decontamination Factor	Total Curie Release To Environment
I-131	0.0165	740	138
Xe-133	0.0135	1	156,500
Kr-85	0.242	1	31,100

TABLE 5-3

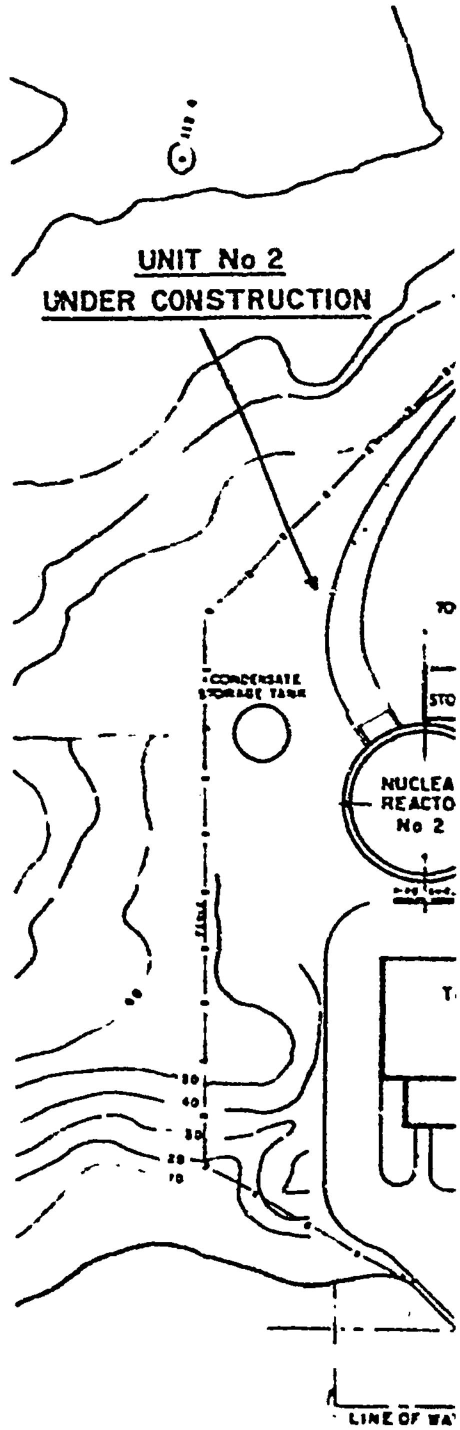
TURBINE MISSILE ACCIDENT - DESIGN CASE

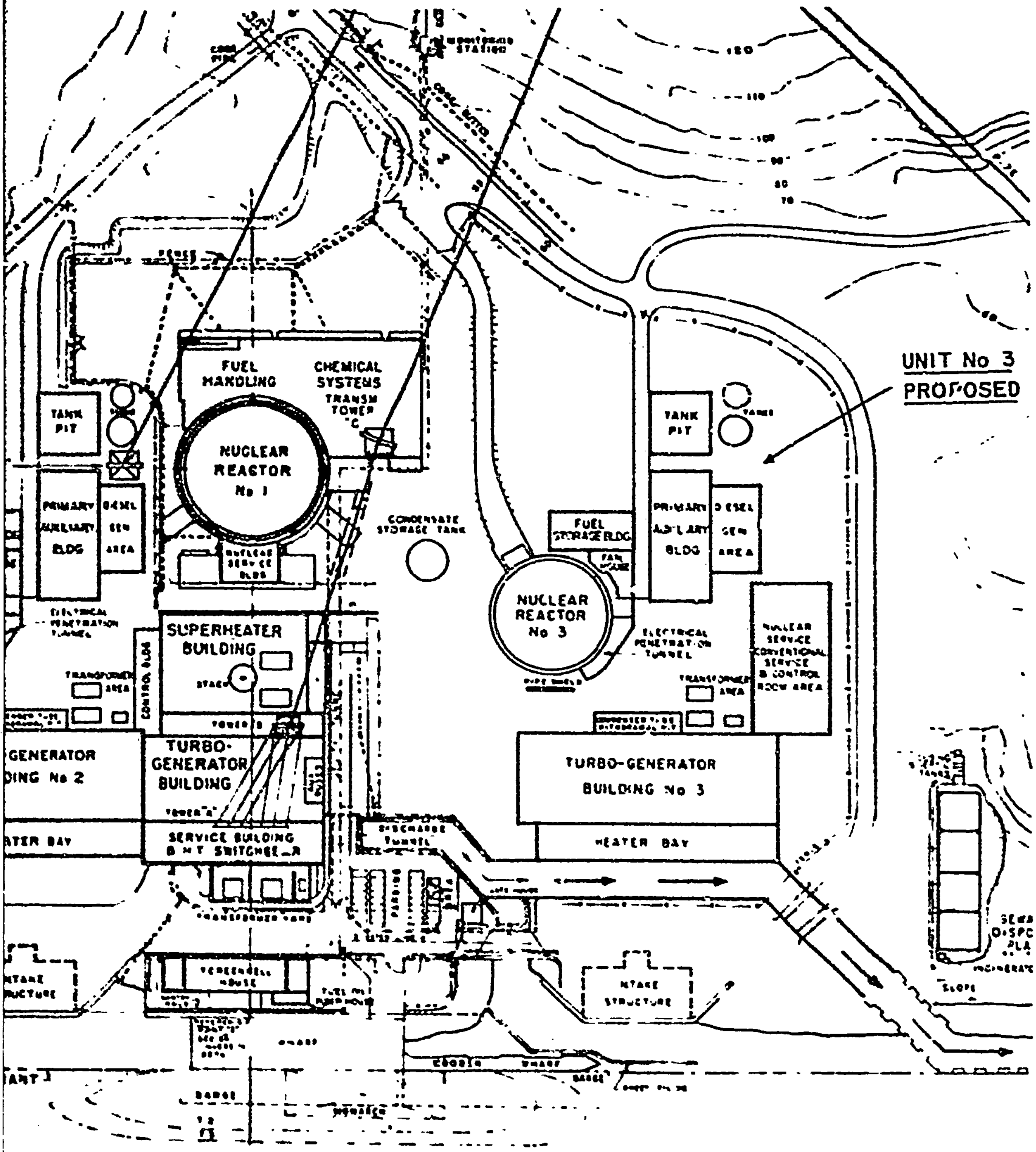
Fuel Parameters

Reactor Power (102%)	3280 MWt
No. of Assemblies	193
Fuel Rods per Assembly	204
Normalized Power, 18 Highest Rated Discharged Assemblies	1.27
Normalized Power, Highest Rated Discharged Assembly	1.63
Axial Peak/Avg., Highest Rated Discharged Assembly	1.72

Activity Release Data

Isotope	Release Fraction	Bubble Decontamination Factor	Total Curie Release To Environment
I-131	0.0368	500	508
Xe-133	0.0296	1	362,000
Kr-85	0.345	1	48,700


UNIT No 2
UNDER CONSTRUCTION



SITE PLOT PLAN
FIGURE 5-1

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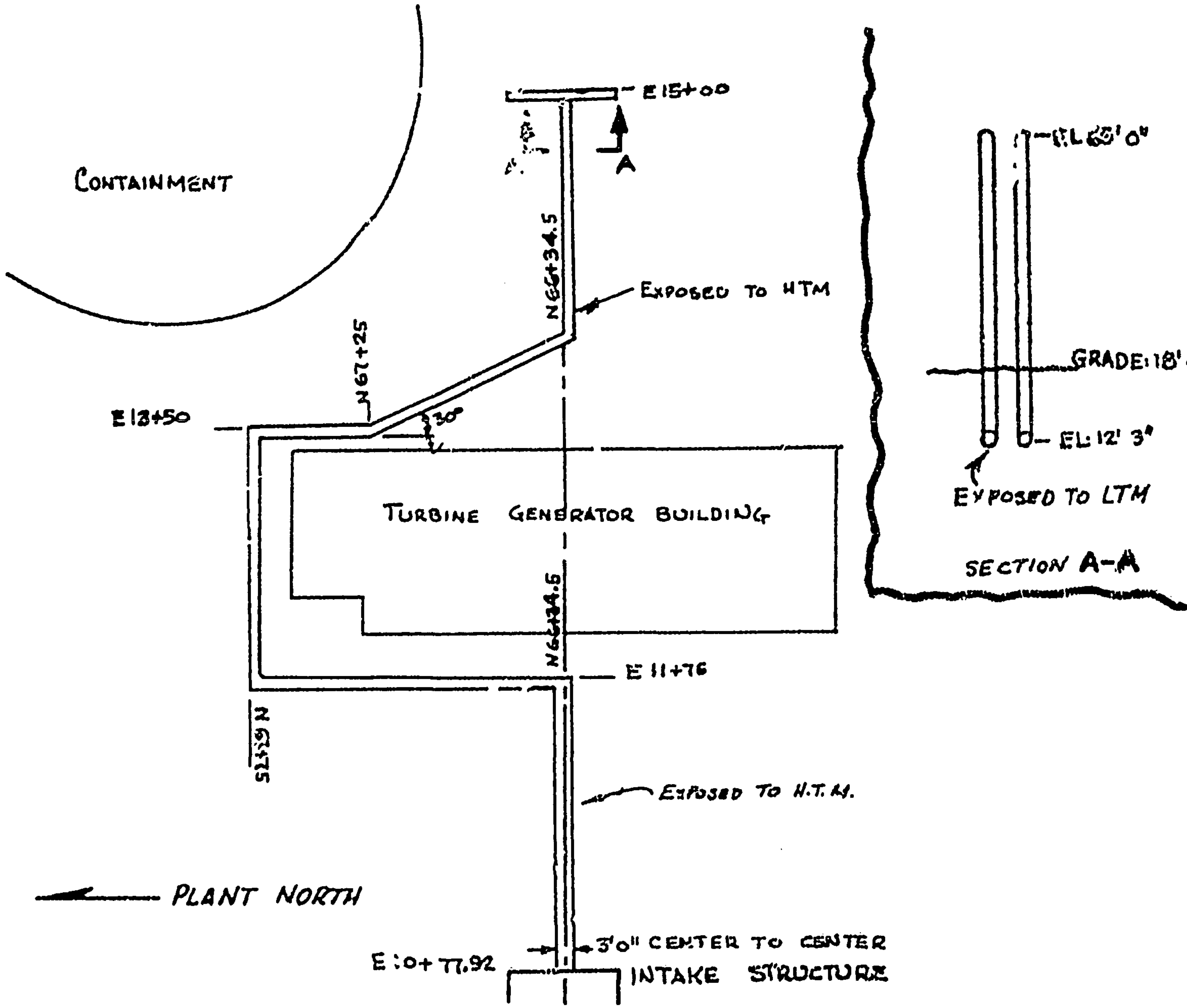

SERVICE WATER PIPING
FIGURE 5-2
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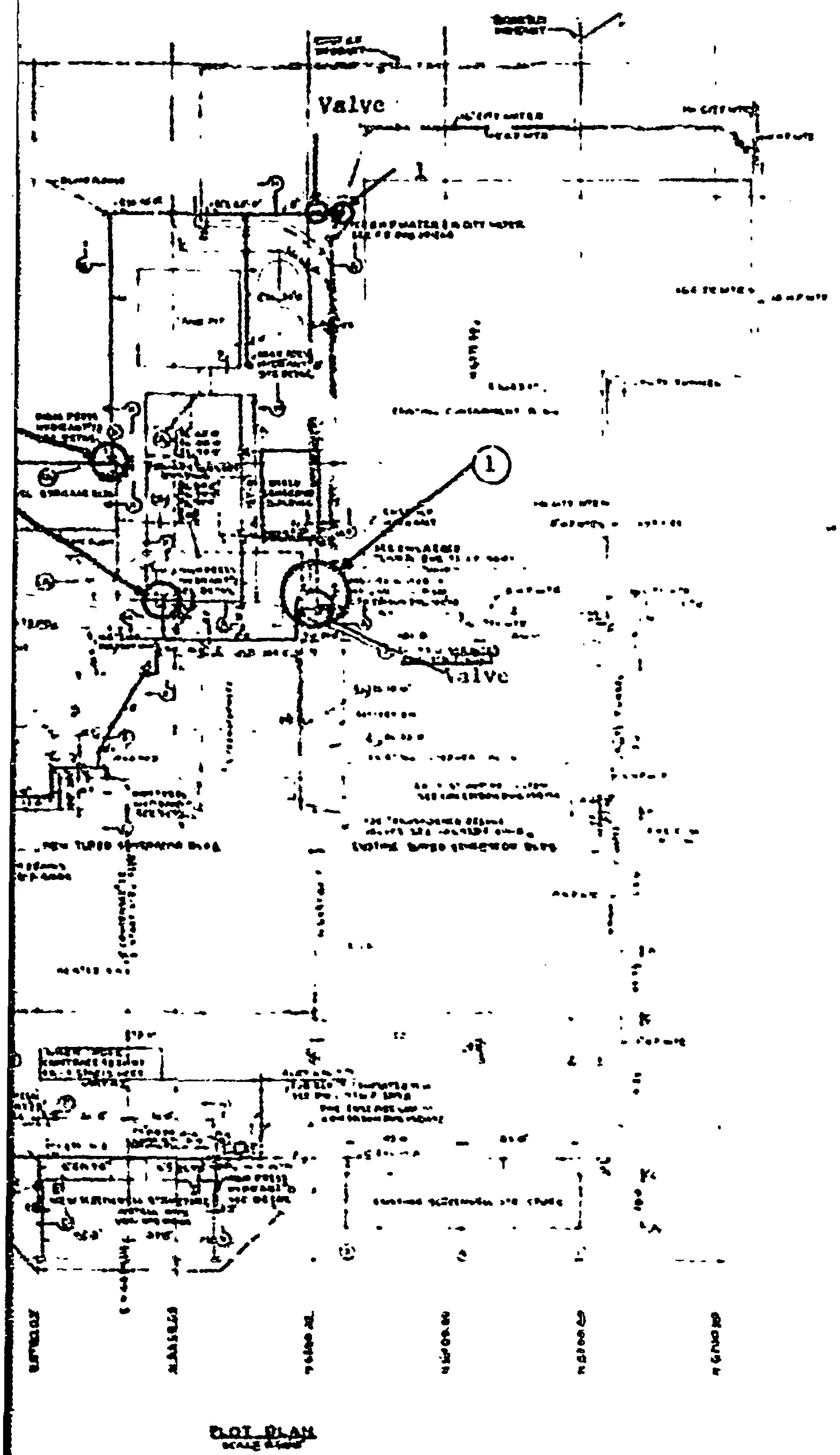
1) Connecti
2) Hig : Pr



onto City Water Header
ro Hydrants

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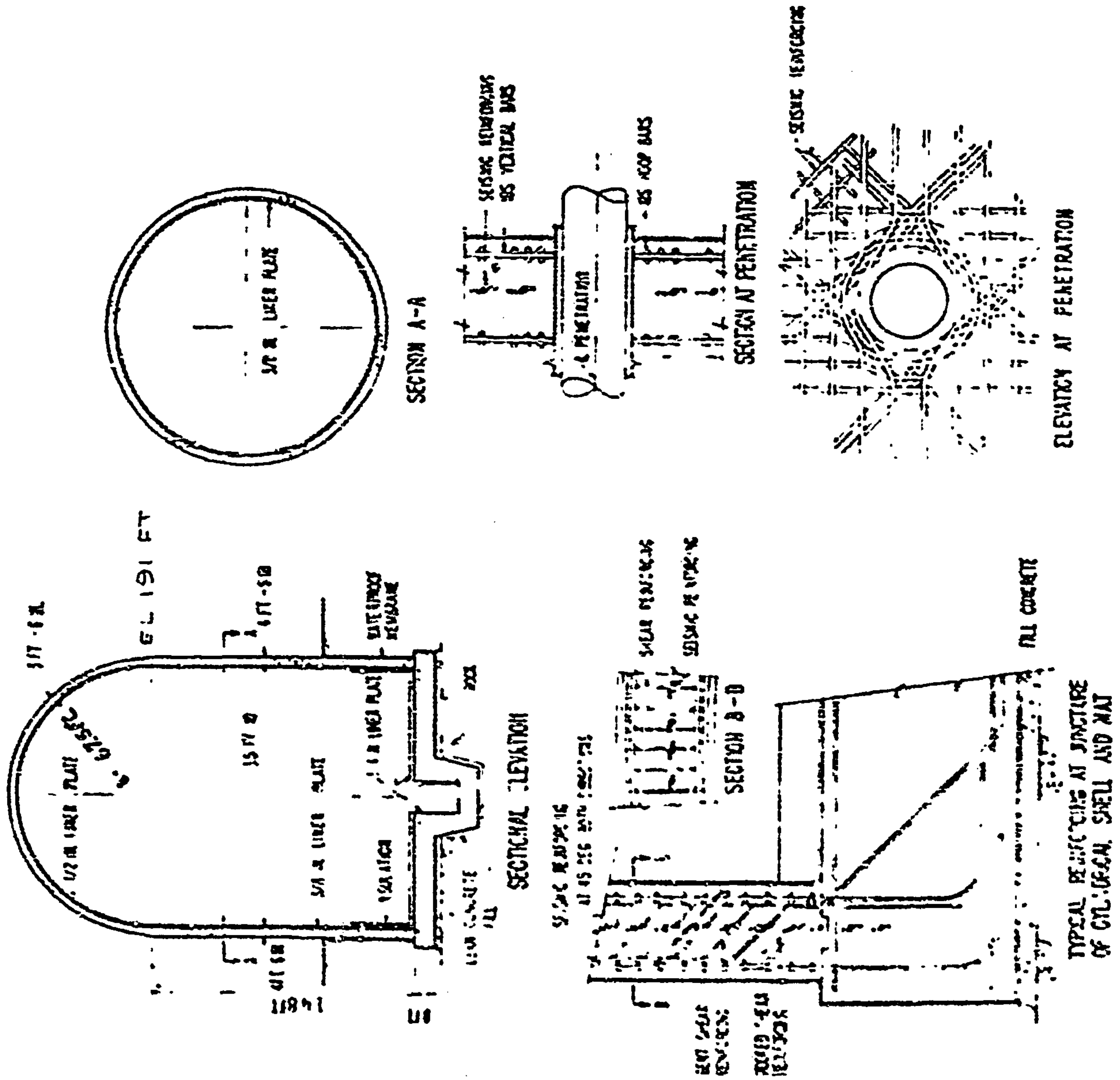
14A-66

City Water and hydrant Locations
FIGURE 5-3

CONTAINMENT STRUCTURE



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Indian Point Unit No. 2
CONTAINMENT STRUCTURE
FIGURE 5-4
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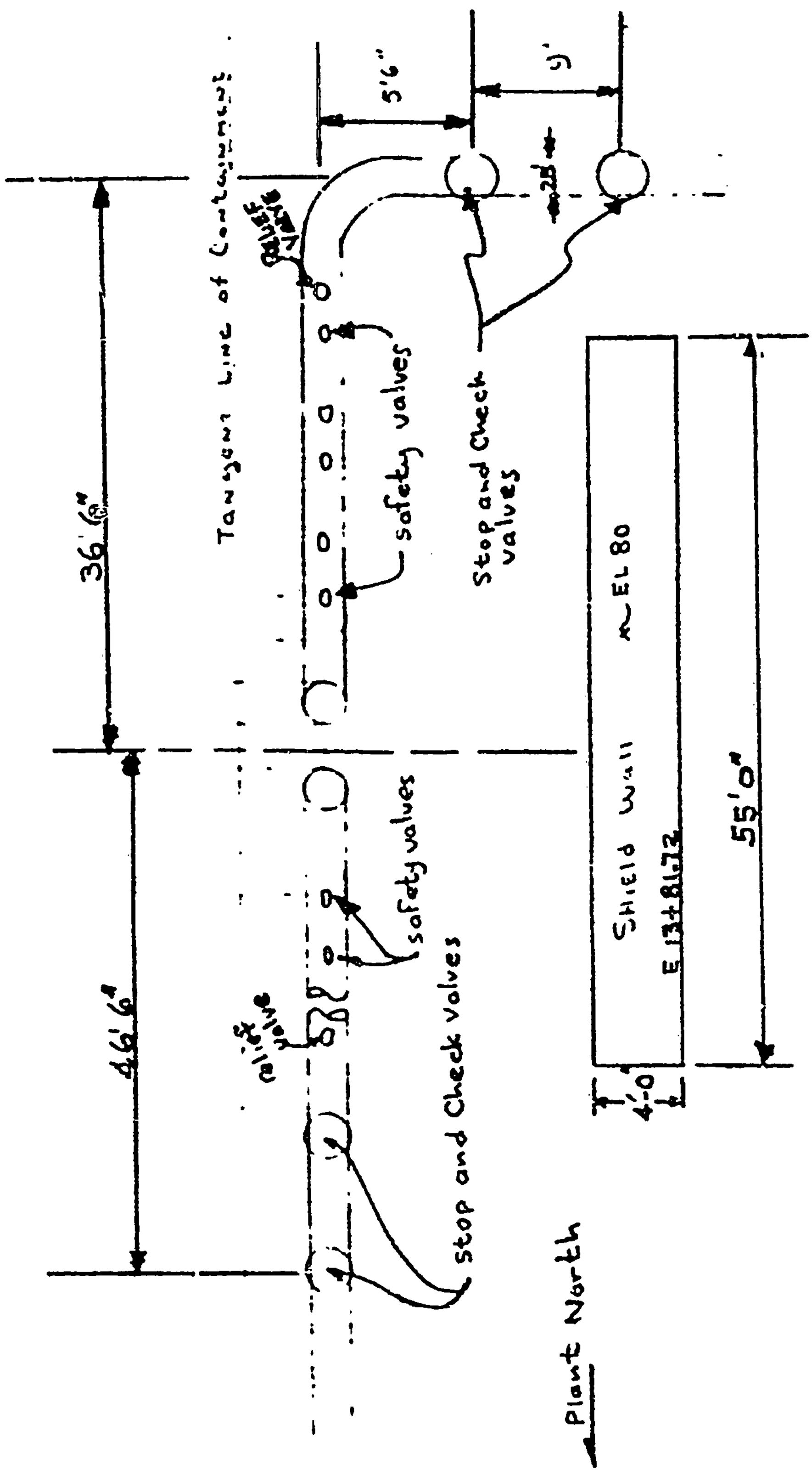


Plan View of Shield Wall Area
FIGURE 5-5

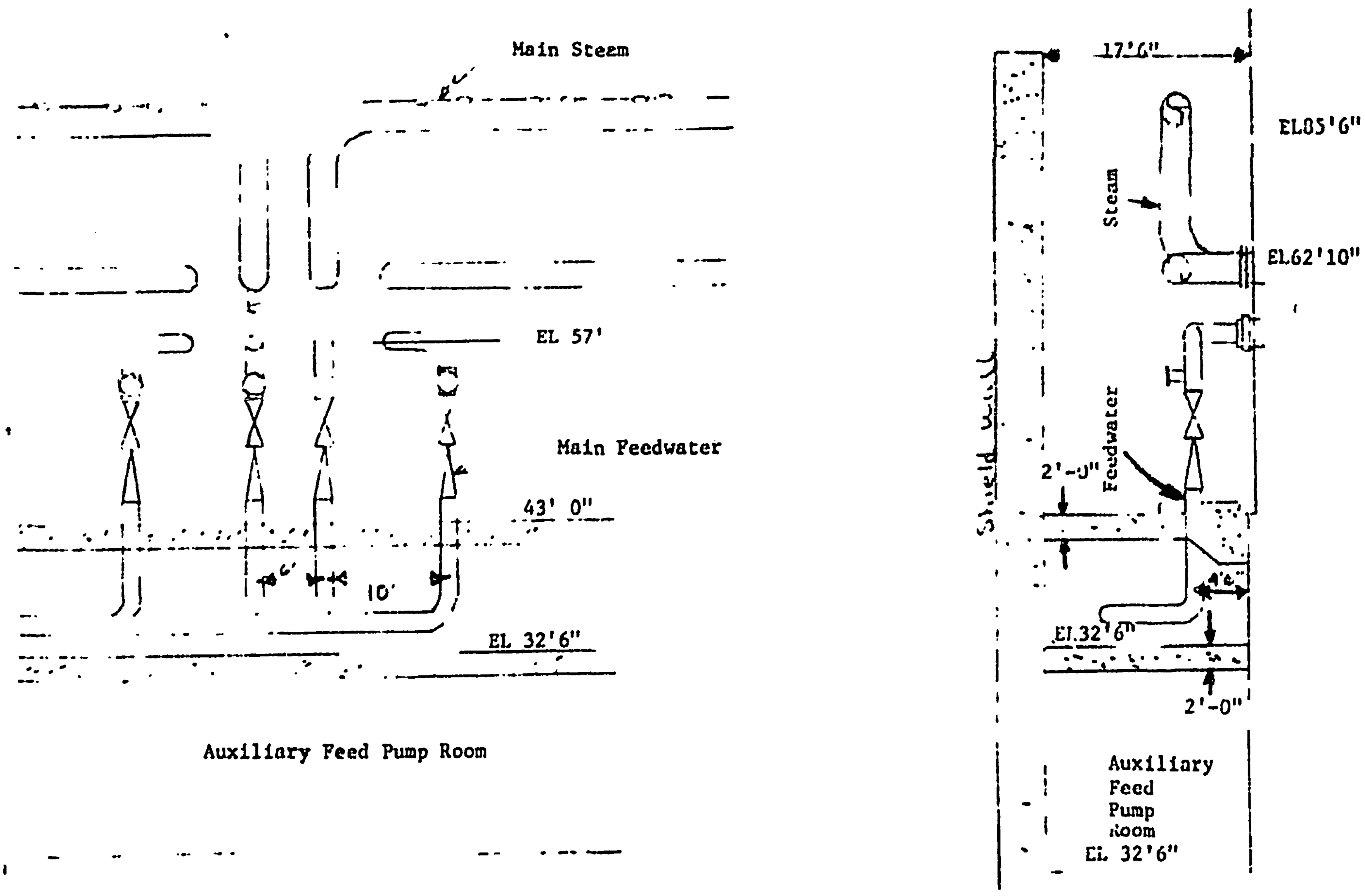
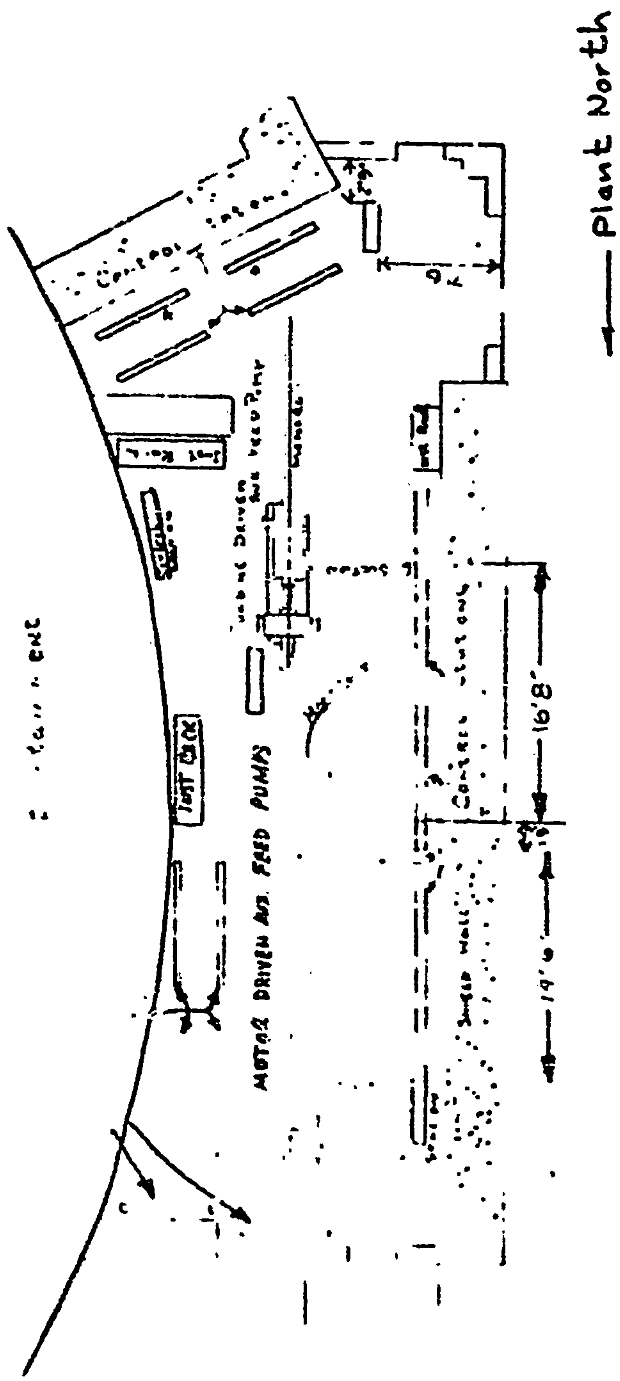


FIGURE 5-6
Elevation View of Shield Wall Region



Auxiliary Feedpump Room -
Plan View

FIGURE 5-7

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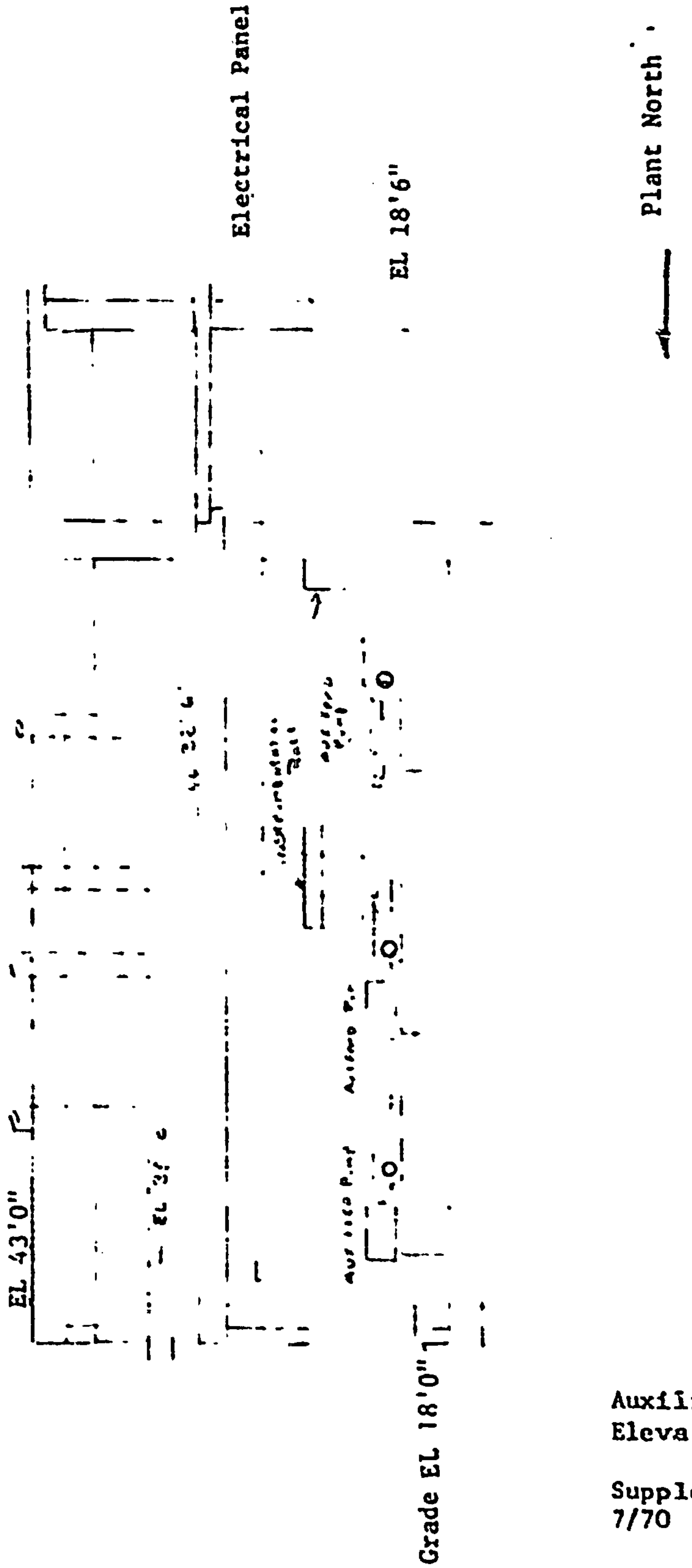

Auxiliary Pump Room -
Elevation View
FIGURE 5-8
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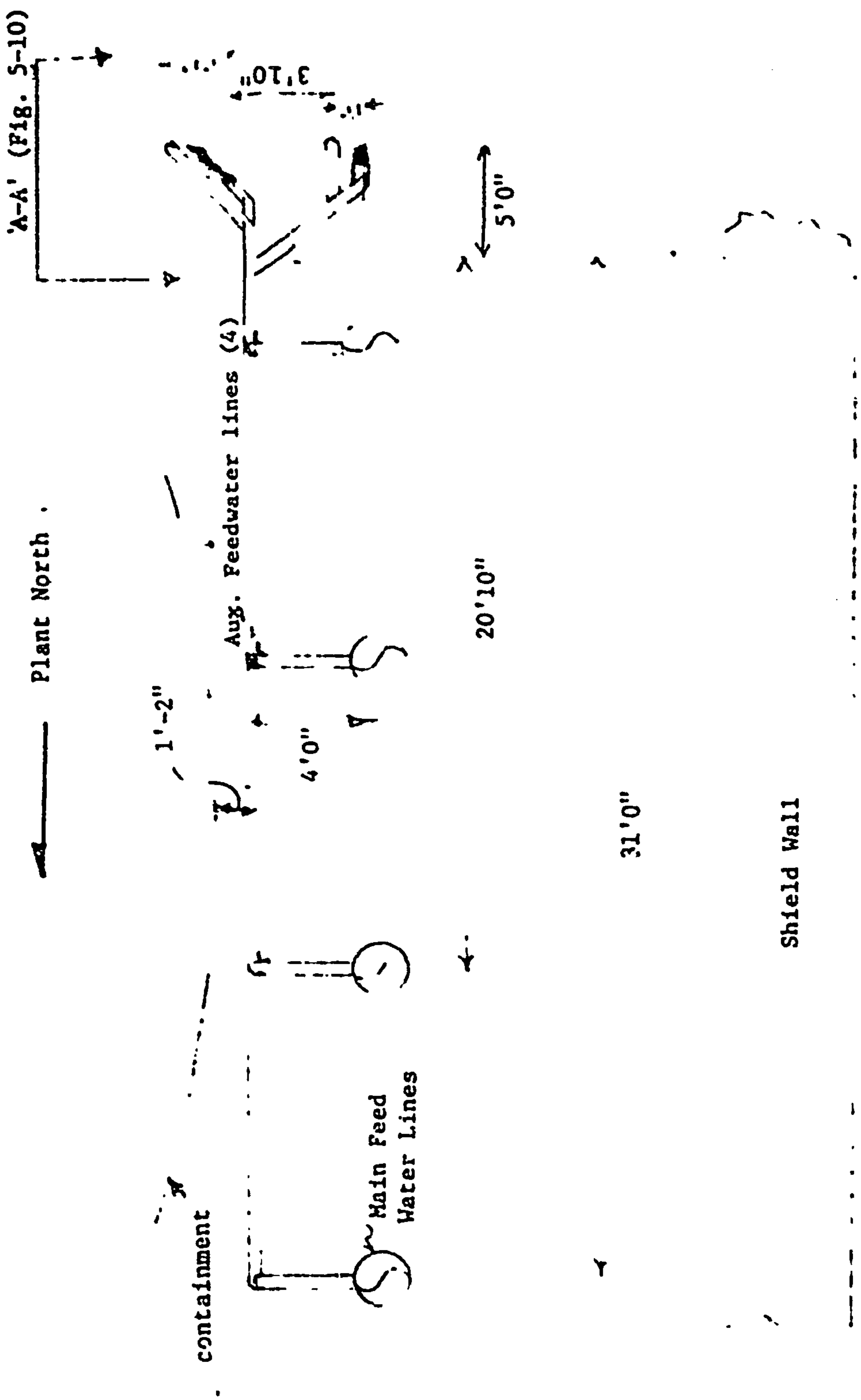
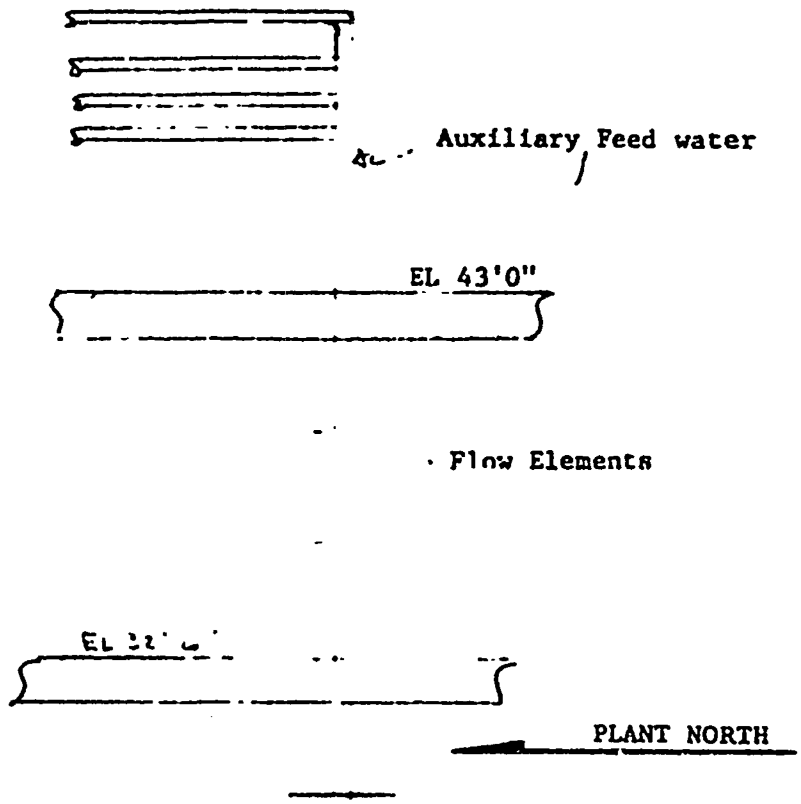


Auxialiary Feedwater Piping - Plan View
FIGURE 5-9



Auxiliary Feedwater Piping - Elevation View

FIGURE 5-10

FIGURE 5-11
Auxiliary Feedwater Supplies and Piping

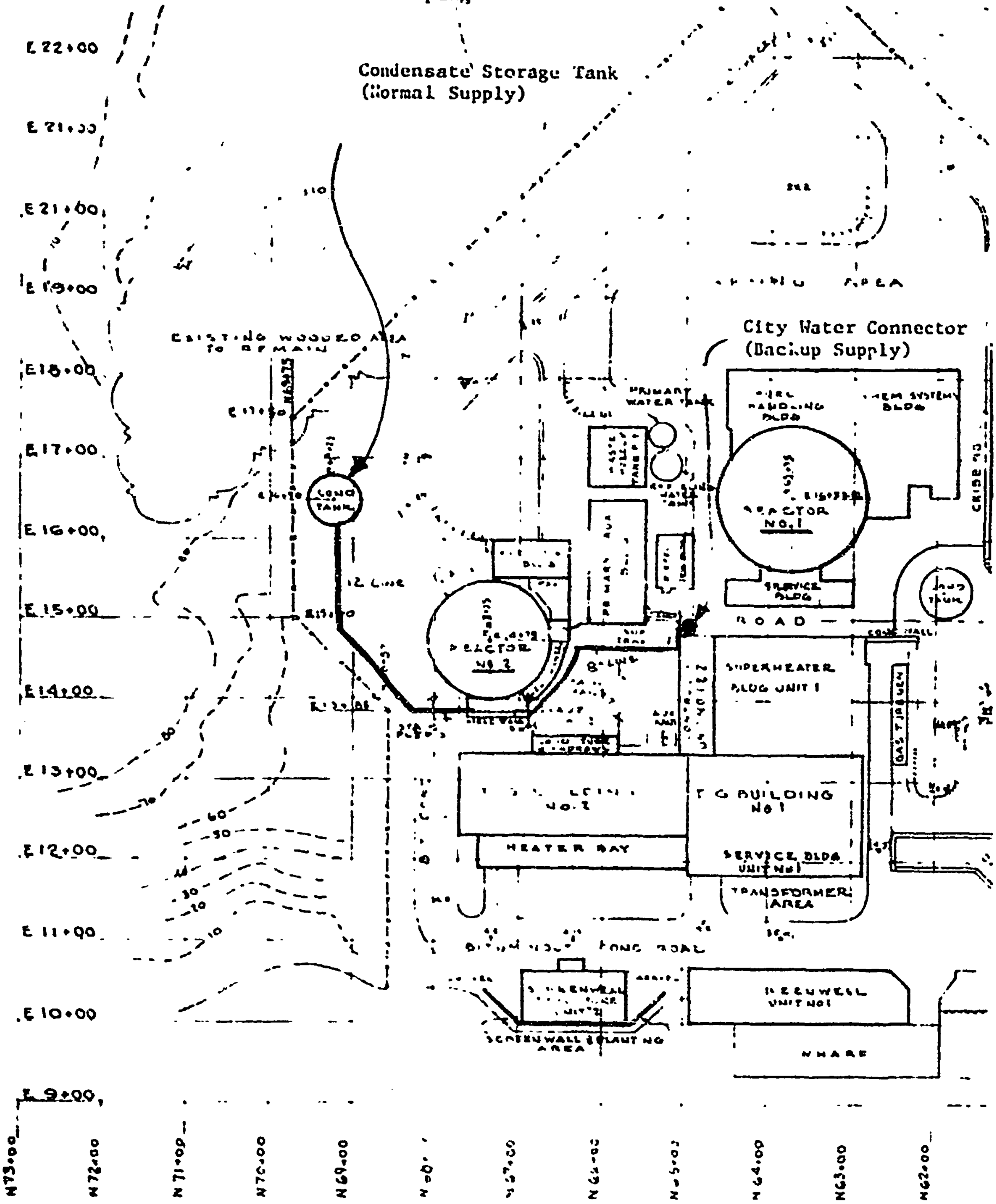
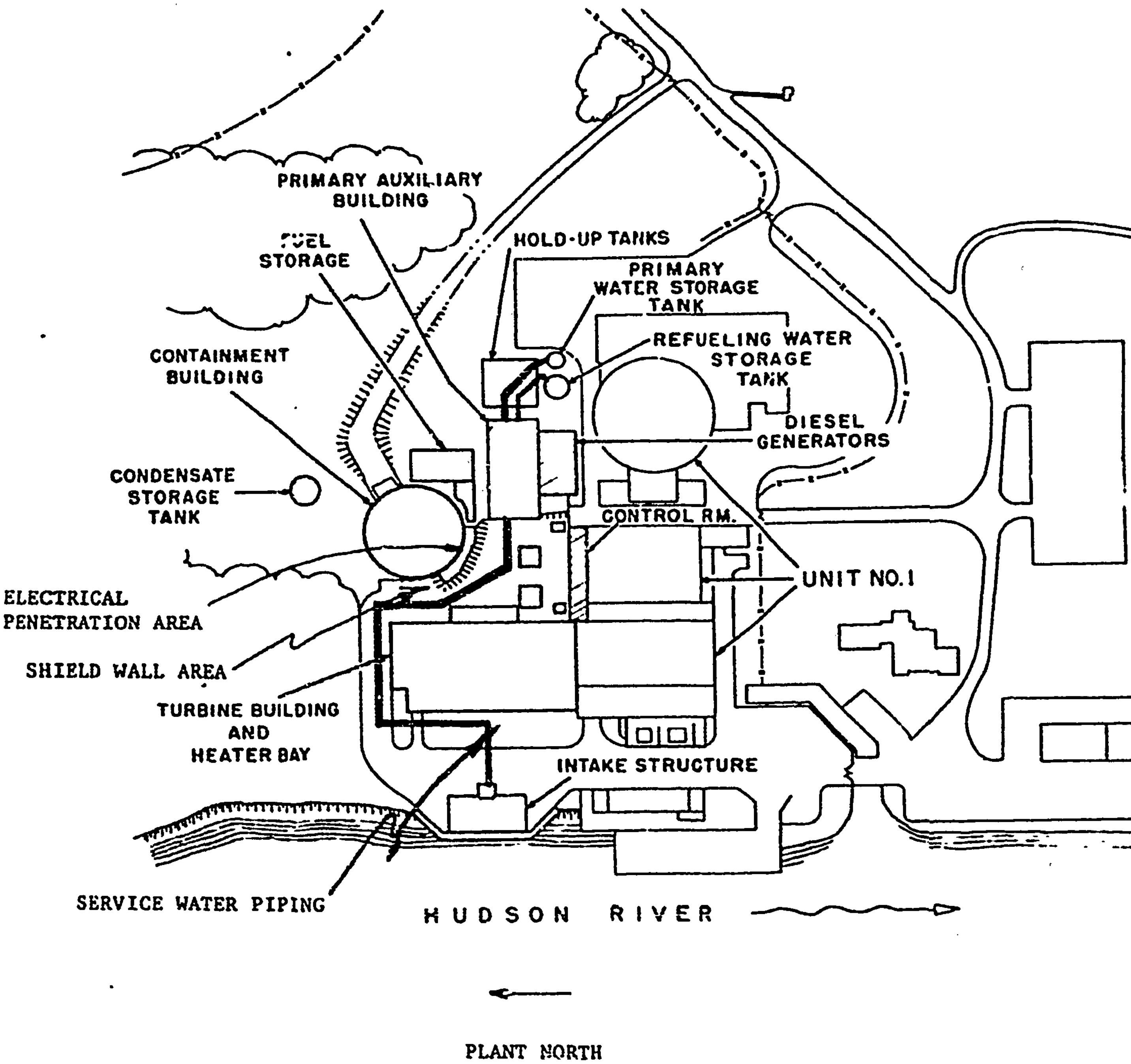


FIGURE 5-12



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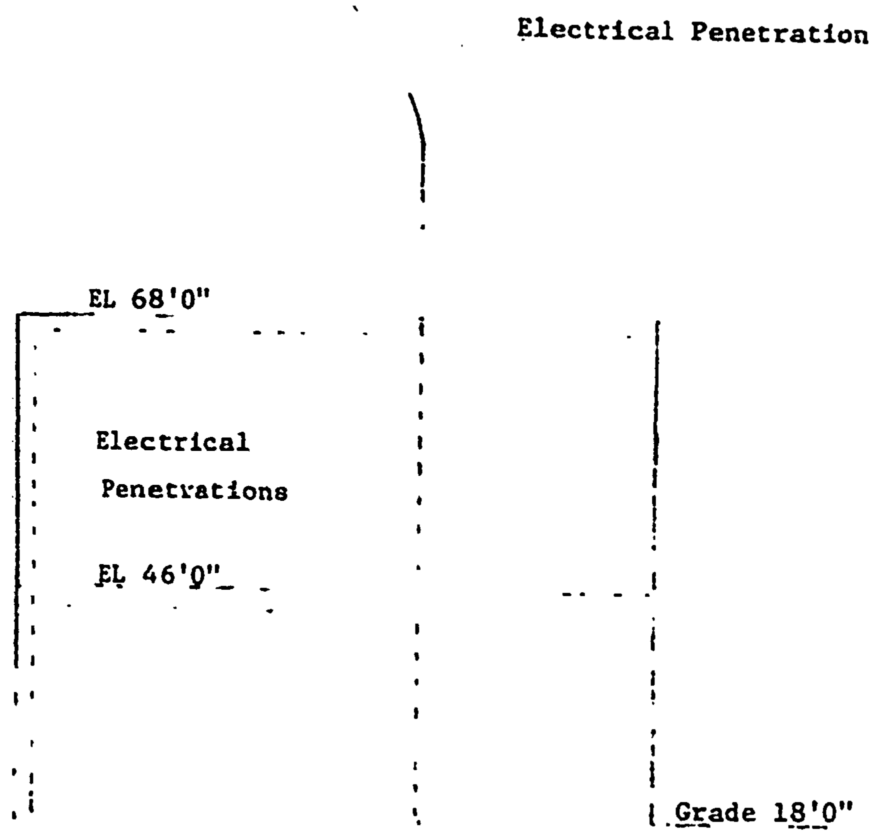

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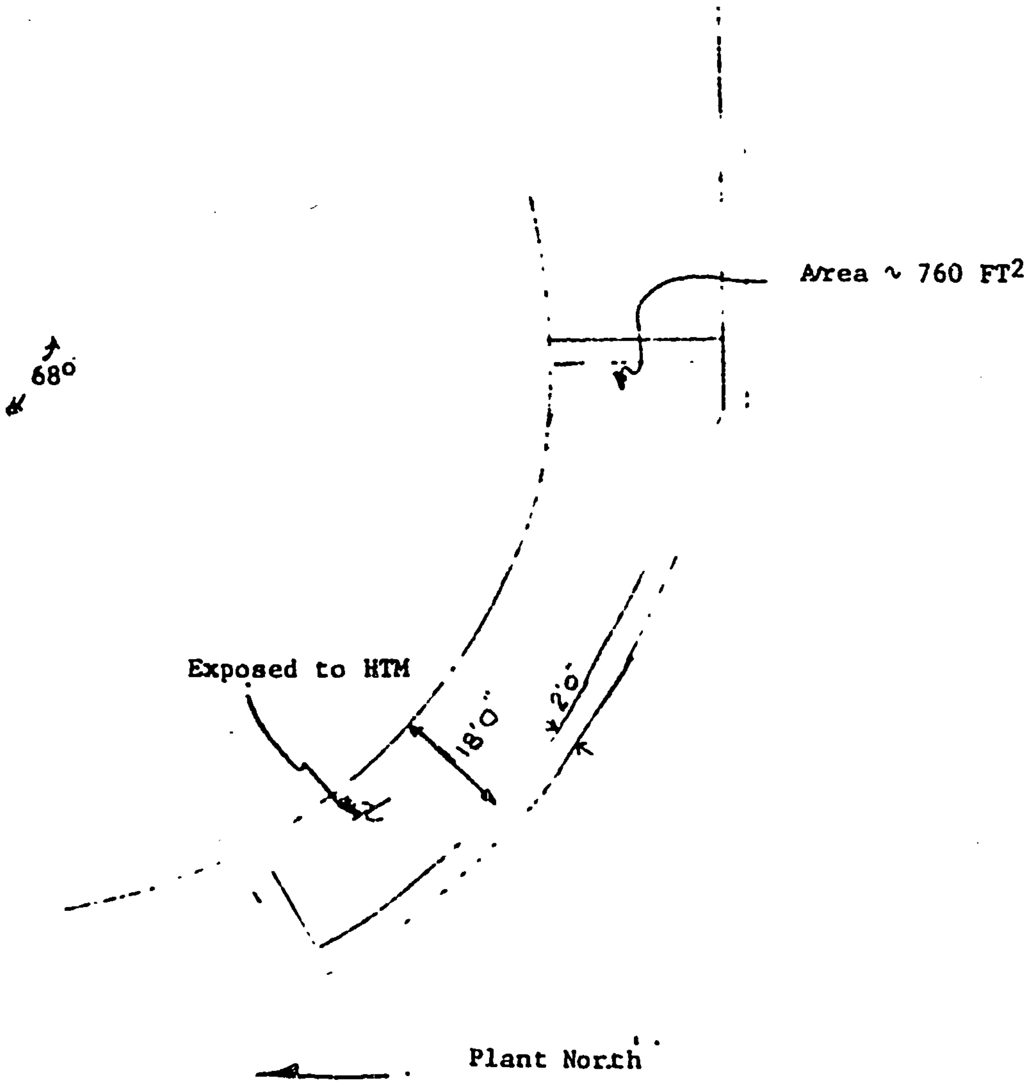
Electrical Penetrat.on Area - Elevation View

FIGURE 5-13



Electrical Penetration Area - Plan View

FIGURE 5-14

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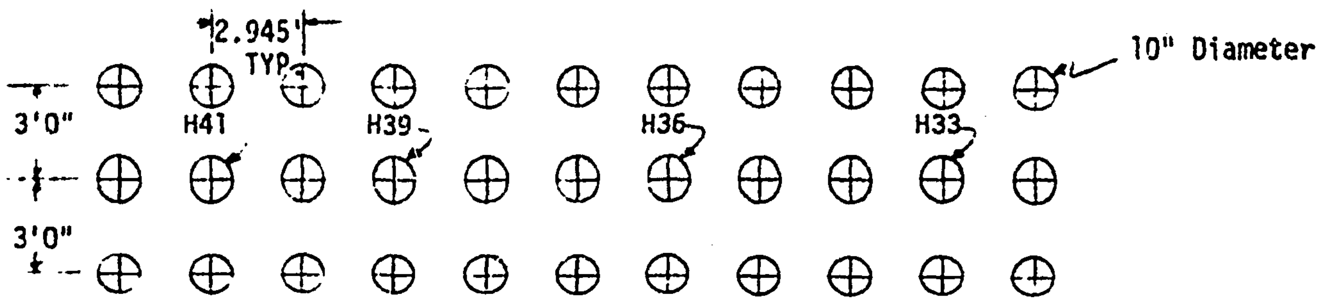
INDIAN POINT UNIT 2 ELECTRICAL PENETRATION ARRANGEMENT

Portion of Penetration Matrix



CHANNEL, PENETRATION NUMBER CORRESPONDENCE

PENETRATION NUMBER	PRESSURIZER PRESSURE	PRESSURIZER LEVEL	STEAM GENERATOR LEVEL
H33	P 457	L 461	L 417B L 427B L 437B L 447B
H36	P 474	-	L 417A L 427A L 437B L 447B
H39	P 455	L 459	L 427C L 437C
H41	P 456	L 460	L 417C L 447C

FIGURE 5-15

Indian Point
Final Facility Description and Safety Analysis Report
Dated: October 15, 1968

Ex 4

Figure 5-16 (Appendix 14a, Supplement 12, 7/70), Titled "Control Building General Plan"

The above was **redacted** by NRC staff as **sensitive information** to allow release of this document to the public.

APPENDIX 14B

INDIAN POINT UNIT NO. 2 CORE COOLING ANALYSIS SUPPLEMENT

Introduction

Additional analyses have been performed for the Indian Point Unit No. 2 postulated loss-of-coolant accident. The analyses utilize improved calculational techniques and are reported in WCAP-7422-L (Westinghouse proprietary report). Subsequent to the preparation of the report, sensitivity studies were performed which resulted in some modified input parameters and revisions to the model. The results of these additional studies are reported below and supplement the basic report, a summary of which follows:

> This report is intended to provide a complete description of the behavior of a Westinghouse PWR core during a loss of coolant accident. Detailed discussions of the codes and assumptions used in the analysis are discussed as well as the experimental verification of the codes or correlations used in the code.
>
> The report introduces the improved analytical techniques that have been developed to analyze the loss of coolant accident. These techniques represent the culmination of extensive research and development programs carried out by W and augmented by AEC funded research and development. These are the BLODWN-2 code which replaces the BLODWN-1 code for the core internals analysis, and the SATAN V code which replaces the FLASH-R code for evaluating the pertinent parameters for the core cooling analysis such as core flow and pressure. A transition boiling heat transfer correlation for evaluating the heat removal from the fuel rod after DNB has also been incorporated.

A parameter study is presented that shows the sensitivity to the various important core cooling phenomenon.

Previously issued WCAP topical reports describing some codes and experimental work are only summarized in this report.

The results of this report are generally applicable to all Westinghouse PWR plants. The specific calculations are based on the Indian Point No. 2 plant parameters. For completeness this report presents the principal results of the design evaluation of the ECCS, core, and internals published in the Indian Point No. 2 FSAR. These conservative evaluations demonstrated that the above criteria were met with a substantial margin. The improved analytical techniques and experimentally verified correlations discussed in this report demonstrate that additional margin exists in meeting the core cooling design criteria beyond that described in the Indian Point No. 2 FSAR.

Supplementary Analysis

A multi-control volume analysis (SATAN-V) was used to analyze a loss of coolant in the Indian Point No. 2 Plant. The analysis was divided into two stages:

A. A parametric survey to determine the most conservative combination of assumptions as input to the SATAN code. These surveys were run at 102% of the maximum calculated thermal power (3216 MWt, 18.9 Kw/ft).

B. A design basis analysis for a full range of breaks sizes. These cases were run for the most conservative design case established in (A) above at 102% of the application rating (2759 MWt, 18.8 Kw/ft).

A. Parametric Survey

In the SATAN-V analysis the Reactor Coolant System was divided into 70 control volumes. The control volumes in the loop were divided into broken and unbroken loops and two control volumes were used to describe the coolant in the core region. This analysis supersedes the analysis presented in the FSAR where FLASH code with a total of three control volumes was used. The current multi control volume analysis, particularly with two control volumes in the core, provides the opportunity to perform a detailed parametric survey of the important phenomena affecting the blowdown process.

The parametric analyses were:

1. Heat transfer from core to coolant during blowdown
2. Reactor Coolant Pump Characteristics
3. Steam Generator Heat Transfer Characteristics
4. Loop Resistances and Break Location
5. Accumulator Performance

In the parametric survey several break locations were analyzed to establish that the double ended cold leg break was the worst break location. All the remaining cases were for the double ended cold leg break.

1. Heat Transfer from Core to Coolant During Blowdown

a. SATAN/LOCTA Heat Transfer

Detailed fuel and clad thermal transients were analyzed with the LOCTA-R2 code using the blowdown results from the SATAN code. It is important, however, that the feedback effect on blowdown from core heat transfer be properly determined. This study indicated the need for core control volumes to properly treat the problem. This capability was not available previously when the FLASH code was used. A series of SATAN/LOCTA iterations were run for the double ended cold leg break until the total heat transferred from the core to the coolant converged as a function of time were essentially identical in both cases. The peak clad temperature calculated was 2120°F. This case is presented as Case 1 in Table 1. It served as the starting point for additional parametric analyses.

In this transient, a first peak clad temperature of 2030°F was reached at 6 seconds. This resulted from the combination of low flow and high quality in the core region. The limited number of core control volumes (two) results in a conservative treatment of this region of the transient. It is expected that a more detailed representation of the core region will result in lower first peak clad temperatures. Additional volumes in the core provides a more realistic coolant temperature distribution which will allow each control volume to saturate at a different time and result in an improved flow distribution and heat transfer early in the transient.

b. Flow Stagnation Analysis Assuming Zero Pressure Drop Across Core

During the blowdown phase of the LOCA, the core pressure drop may be reduced to values of approximately 1 psi. It has been postulated that because of the near zero values for the core pressure drop, a period of flow stagnation or negligible core cooling would result. To resolve this question, a core heat transfer analysis was performed employing the most pessimistic assumption of zero pressure differential across the core. The results of the analysis indicate that for small time periods of zero core pressure drop, sufficient heat transfer is available to cool the core.

The study was performed with the computer code CHARM which calculates the time varying coolant temperatures, enthalpies, velocities, pressures, and densities in the core by solving the two dimensional continuity, momentum, and energy equations with the method of characteristics.

In this analysis, the geometry of the reactor core, lower plenum, and upper plenum were simulated as shown in Figure 1; the boundary conditions consisted of the pressure transient at the bottom and top of the system (Figure 2). The first portion of the transient (0-0.065 sec) represents a simplified pressure transient typical of a double ended cold leg break accident and was imposed to obtain the core flow conditions representative of the flow stagnation period. In the second portion of the transient, the pressure at the upper boundary was arbitrarily set equal to the pressure at the lower boundary less the elevation head based on the average core density. This condition corresponds to the worst case since all external driving forces are set equal to zero and only the local internal effects are left to cause the movement of the fluid.

The resulting flow patterns inside the core were illustrated by the plots of the velocity vectors at different times (Figure 3 shows a typical result). The analysis of these diagrams indicates that the stagnation point does not remain fixed at one location, but moves in the core and that, owing to the heating effect, the fluid velocities can be quite large even though no core pressure difference is considered. The local stable film boiling heat transfer coefficient calculated with the Rohsenow & Dougal correlation is generally above 100 btu/hr-ft^2-F and the time during which the heat transfer at the hot spot is below 100 is less than 100 msec. It should be noted that, due to the high system pressure, a coolant velocity of only 2 ft/sec is required to generate a value of heat transfer of 100 btu/hr-ft^2-F.

Therefore, we can conclude that, if water is present in the core, the condition of no pressure difference across the core does not yield to core flow stagnation. On the contrary, due to non-uniform heat flux and enthalpy distribution, considerable coolant motion and heat transfer exist in the core. This analysis, carried on for long periods of time (more than 2.5 sec), will result in a completely voided core if no water can be supplied. In this case the coolant motion and the heat transfer will be considerably reduced. However, it should be pointed out that under the conditions of steam cooling and zero core pressure difference (after completion of blowdown), the fuel rod temperature transient in LOCA analyses is calculated by assuming adiabatic conditions at the clad surface.

2. Reactor Coolant Pump Characteristics

In the heat transfer survey above it was apparent (for example see Figure 4) that following the first peak, a significant clad temperature reduction occurred due to reverse flow through the core. The water still remaining in the unbroken loops provides the source of coolant flow through the core. Parametric studies were then run to determine if any combination of Reactor Coolant pump assumptions could reduce this effect and degrade core cooling further. Since the normal pump action provides flow in the normal forward direction, these studies were to determine if the pumps could impede the reverse flow cooling. The results are summarized in Table 1. The modeling available allowed for separate treatment of the RC pumps in the broken and unbroken loops. The principal assumptions examined were RC pump trip, impeller rotor condition at cavitation, and pump head developed after cavitation. The most conservative case, Case 4 (clad temperature 2336°F), assumed the pumps tripped at the time of reactor trip, coasted down until cavitation conditions were reached, and then the rotor did not lock but continued to develop a head according to the local density at the pump suction.

TABLE 1

INDIAN POINT UNIT NO. 2 SENSITIVITY STUDY

3216 MWt

Cold Leg Break

Reactor Coolant Pump Characteristics

Case	Pump Trip		Locked Rotor		Pump Head After Cavitation		T_{clad} °F
	Broken Loop	Unbroken Loop	Broken Loop	Unbroken Loop	Broken Loop	Unbroken Loop	
1	Yes	Yes	Yes	Yes	None	None	2120
2	No	No	No	No	Yes	Yes	2233
3	Yes	Yes	Yes	No	None	Yes	2244
4	Yes	Yes	No	No	Yes	Yes	2336

3. <u>Steam Generator Heat Transfer Assumptions</u>

Using Case 4 as a base the heat transfer during blowdown to and from the steam generator was exam. The results are presented in Table 2. The results indicate a small variation in clad temperature with large variations in heat transfer assumptions. Case 5 was established as the most conservative case.

TABLE 2

INDIAN POINT NO. 2 SENSITIVITY STUDY

3216 MWt

Cold Leg Break

Steam Generator Heat Transfer Characteristics

Case	Heat Transfer Assumption	T_{clad} °F
4	One Control Volume for Secondary Side Nominal Correlations	2356
5	Two Control Volumes for Secondary Side Nominal Correlations	2427
6	Two Control Volumes for Secondary Side 1.2 x Nominal Correlations	2366
7	Two Control Volume for Secondary Side Nominal for Cooling - 0.5 x Nominal for Heating	2402

4. Loop Resistance and Break Location

Using Case 5 as a basis, changes were made to the loop resistances during blowdown by modifying the steam generator pressure drop and the two phase multiplier. In one case, the steam generator pressure drop was modified to increase during two phase flow periods. In addition, large variations were made to the Armand Coefficient[1] which is used to determine the two phase pressure drop. During the saturated blowdown regime flow through the core is dependent upon the distribution of flow through the broken and unbroken loop. Three cases were run to determine the sensitivity to pressure drop assumptions. Table 3 presents the results. Small variations in clad temperature resulted. The reason for this is the parameters were varied consistently

1 A. A. Armand, "The Resistance During the Movement of a Two-Phase System in Horizontal Pipes", Report No AERE, Trans 828, (1959) United Kingdom, Atomic Energy Authority Research Group Translation.

in the broken and unbroken loops and a balancing effect results, together with the fact that during this period heat transfer from the core was already quite low. Because Case 9 resulted in a very slight increase in peak temperature (increase+30°F), case 5 was used as the base case for further studies.

Two other break locations were studied, hot leg and pump suction, and the clad temperatures calculated for these cases were less than the corresponding case for the cold leg.

TABLE 3

INDIAN POINT NO. 2 SENSITIVITY STUDY
3216 MWt
Cold Leg Break

Loop Resistances

Case	Pressure Drop Assumptions	
5	Nominal Correlations (High Armand Coefficient) Steady State Pressure Drop Distribution	2427
8	Nominal Correlations Steam Generator ΔP adjusted to increase during two phase	2373
9	Nominal Correlation (Low Armand Coefficient) Steady State Pressure Drop Distribution	2457

5. Accumulator Performance

Two additional cases were run to determine the sensitivity to accumulator injection performance, and results are presented in Table 4. The standard assumption is to assume that three accumulators deliver to the intact loop while the accumulator delivering to the broken loop is lost to the containment floor. In case 10 the effect of the fourth accumulator delivering upstream of the break was examined. Although the subcooling introduced by the accumulator at the break location did increase break flow near the end of blowdown, there was an insignificant increase in core flow and reflood time so a LOCTA calculation was not made to determine the reduction in peak clad temperature from Case 5.

For the final parametric analysis, use was made of the accumulator line resistances verified by in plant performance tests of the Rochester Gas and Electric - Ginna accumulator system. These tests demonstrated that the accumulator piping resistance is about two thirds of the value used in the FSAR analysis providing a flow margin about 15% over the flow used in the above analysis. This resistance was used in Case 11 and although the more rapid accumulator delivery causes a little more water to be lost during blowdown the net effect is faster core reflooding and a peak clad temperature of 2278°F.

TABLE 4

INDIAN POINT NO. 2 SENSITIVITY STUDY

3216 MWt

Cold Leg Break

Accumulator Performance

Case	Accumulator Assumption	T_{clad} °F
5	3 Delivering 1 Spilling to Containment Conservative Line Resistances	2427
10	3 Delivering 1 Injecting Upstream of Break Conservative Line Resistance	<2427
11	3 Delivering 1 Spilling to Containment R. Ginna Verified Resistances	2278

It was concluded from the above parametric studies that with a conservative combination of assumptions, and particularly conservative treatment of the heat transfer during the early stages of blowdown the peak clad temperatures are limited to 2457°F. If credit is taken for the known accumulator performance the peak clad temperature is below 2300°F.

To assure effective cooling of the core, limits on peak clad temperature and local metal water reaction, have been established. It has been demonstrated in the single rod burst test phase of Westinghouse Rod Burst Program (2) that for conditions within the area of safe operation as shown in Figure 5, fuel rod integrity is maintained. Additional experimental data could further increase this area of safe operation. Results of the multi-rod burst test phase of the Westinghouse Rod Burst Program (3) show that peak clad temperatures calculated during a LOCA increase less that 100°F due to geometry distortion. Thus peak clad temperatures determined on the bases of no geometry distortion should be limited to 100°F below the limits presented in Figure 5.

B. Design Basis Analysis

A design basis analysis was performed for the application rating (2758 MWt) using the conservative assumptions established in Case 5 above. A full range of breaks sizes was analyzed. The results are contained in Table 5. The clad temperature as a function of time is presented in Figure 4 for the double ended cold leg break.

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(2) "Topical Report - Performance of Zircaloy Clad Fuel Rods During a Simulated Loss-of-Coolant Accident - Single Rod Tests," WCAP-7379-L, Volume I (Proprietary) and Volume II (Non-proprietary), September 15, 1969.

(3) "Topical Report - Performance of Zircaloy Clad Fuel Rods During a Simulated Loss-of-Coolant Accident - Multi-Rod Tests," WCAP-7495-L, Proprietary, Volume I - Test setup and Results; Volume II - Analyses of Results.

TABLE 5

Break Size (Cold Leg)	Peak Clad Temperature, °F
Double Ended	2015
4.5 ft^5	1840
3.0 ft^2	1490
0.5 ft^2	1350
0.196 ft^2 (6 in)	870

Conclusion:

The maximum clad temperature calculated (2015°F) at 102% of the application rating of 2758 MWt (18.8 Kw/ft) is well below the Westinghouse 2700°F maximum temperature criterion and the local hot spot metal water reaction (0.6%) is well below the 16.0% metal-water reaction criterion indicating considerable margin for flow blockage effects and calculational uncertainties.

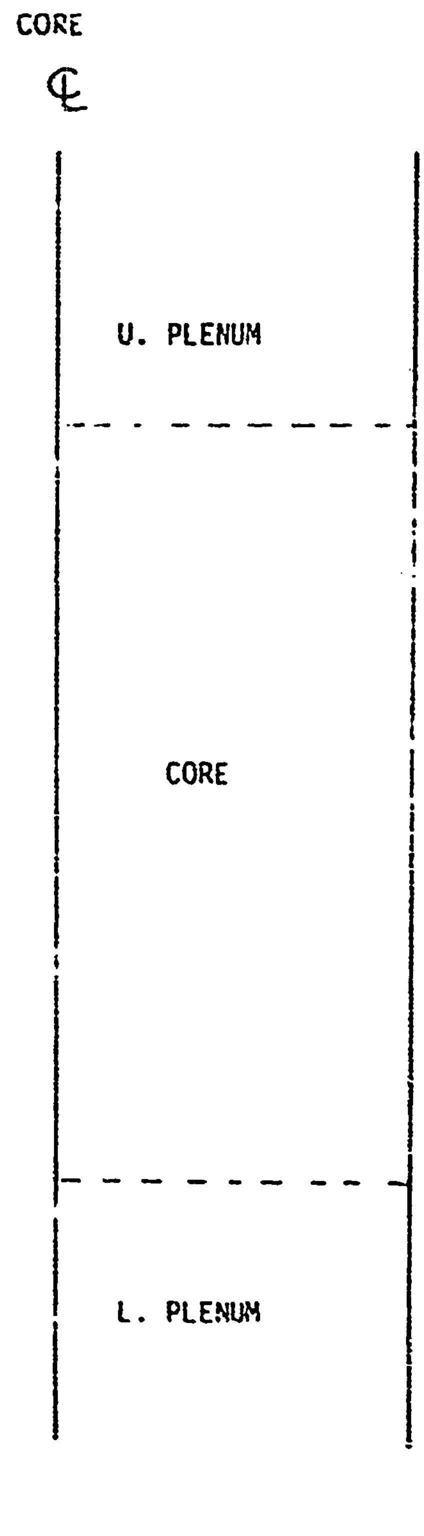
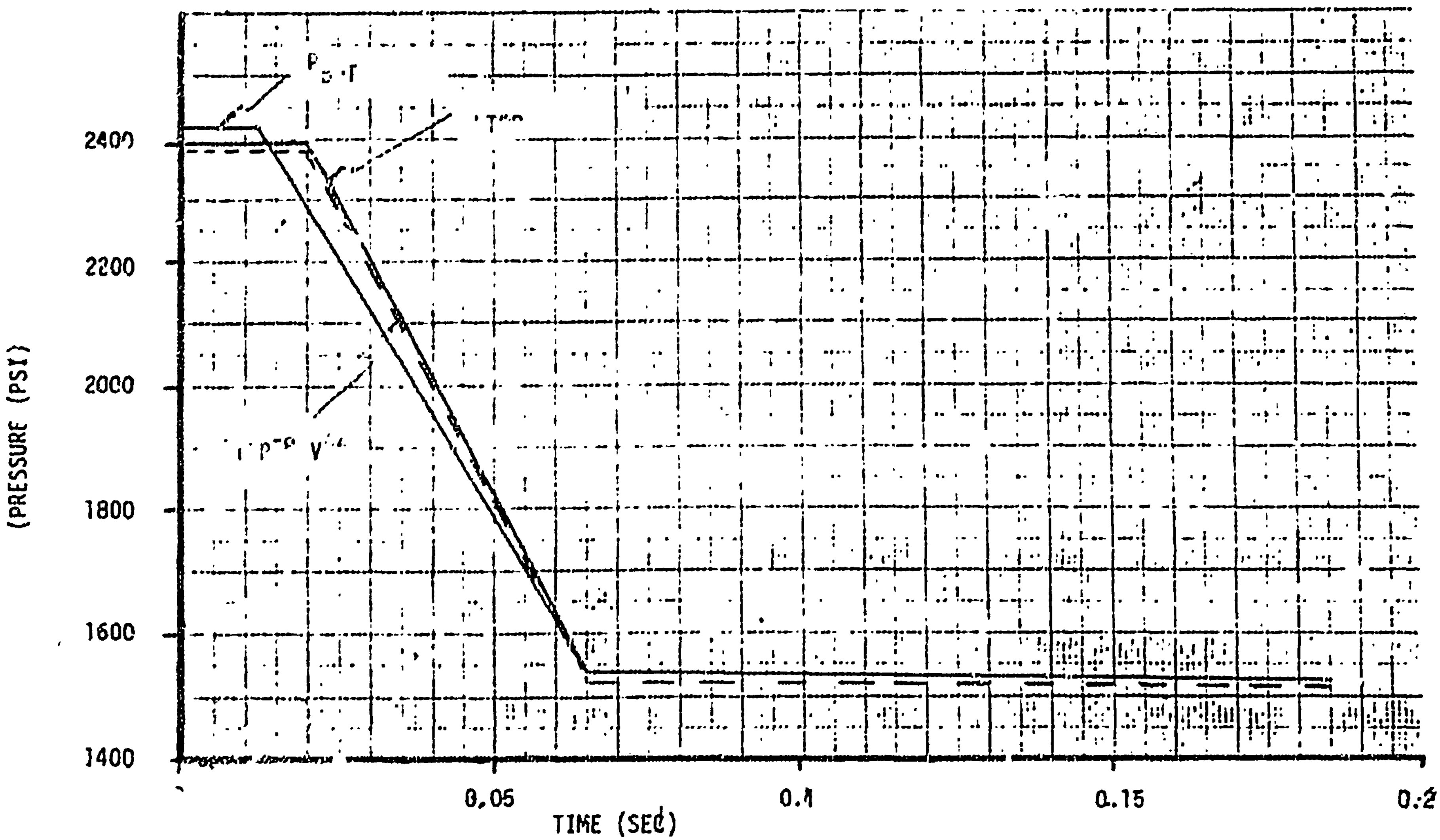


FIGURE 1
REACTOR SYSTEM GEOMETRY

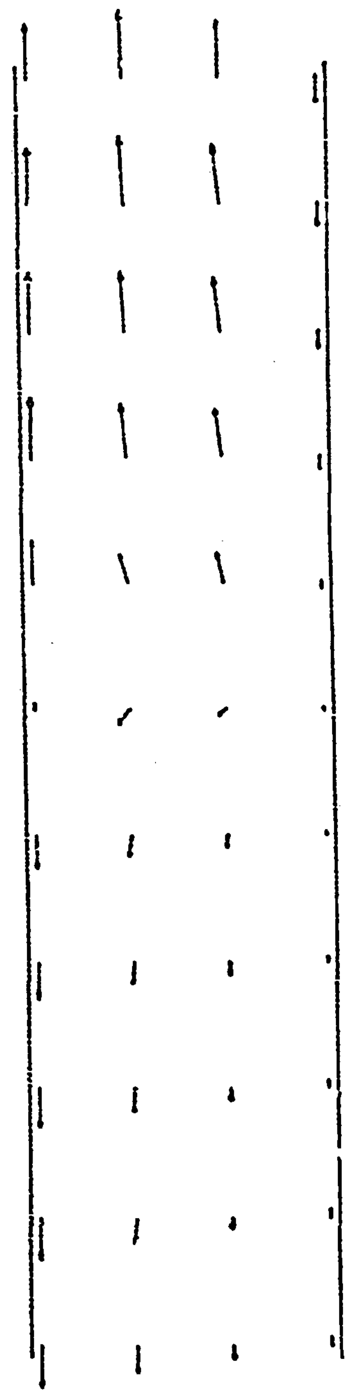


PRESSURE TRANSIENT AT THE BOTTOM
AND TOP OF SYSTEM

FIGURE 2

FIGURE 3

TYPICAL FLOW PATTERN INSIDE THE CORE



TIME 0.106 SECONDS

CLAD TEMPERATURE VS TIME
DOUBLE ENDED COLD LEG BREAK
INDIAN POINT UNIT NO. 2

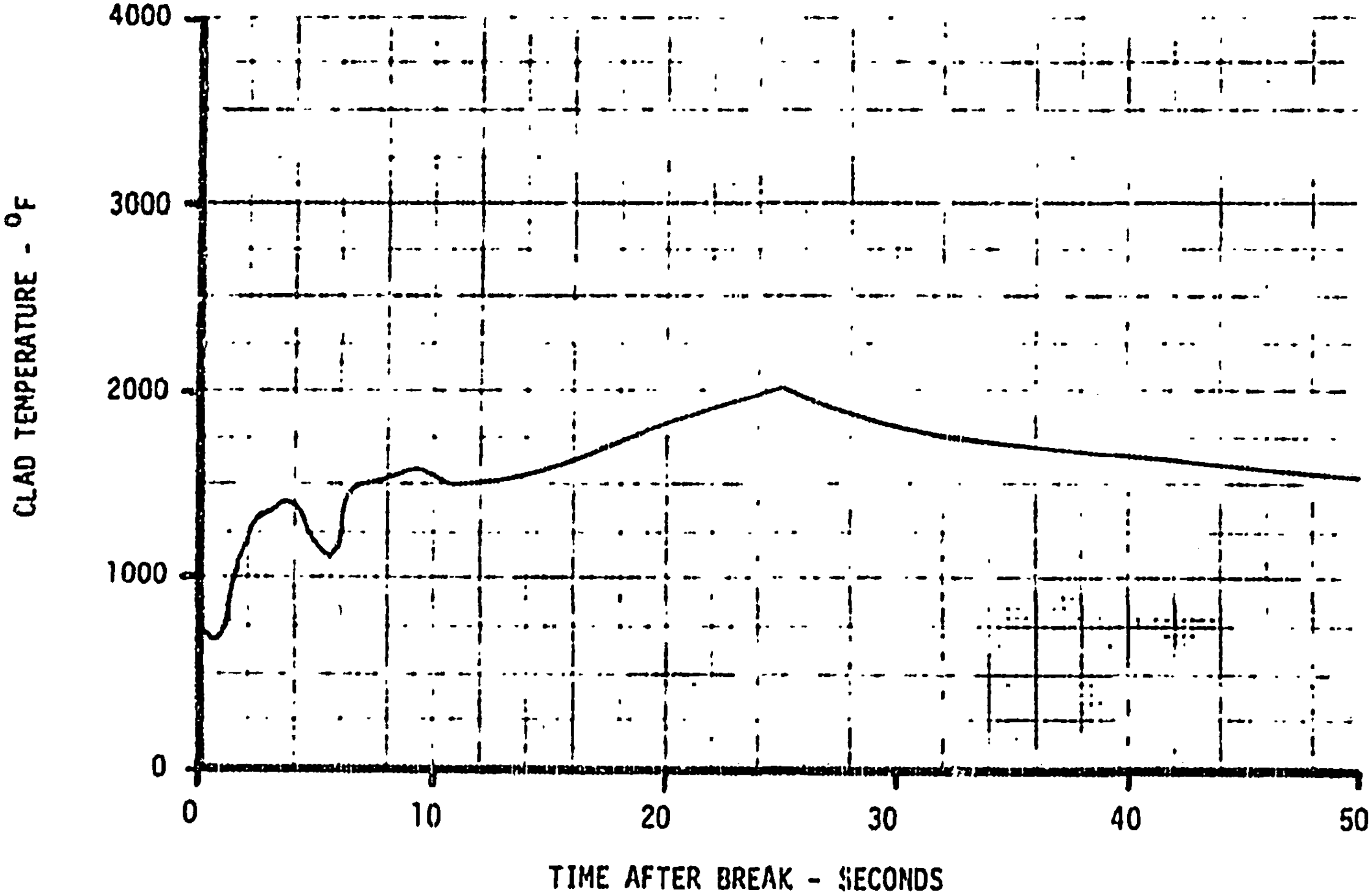


FIGURE 4

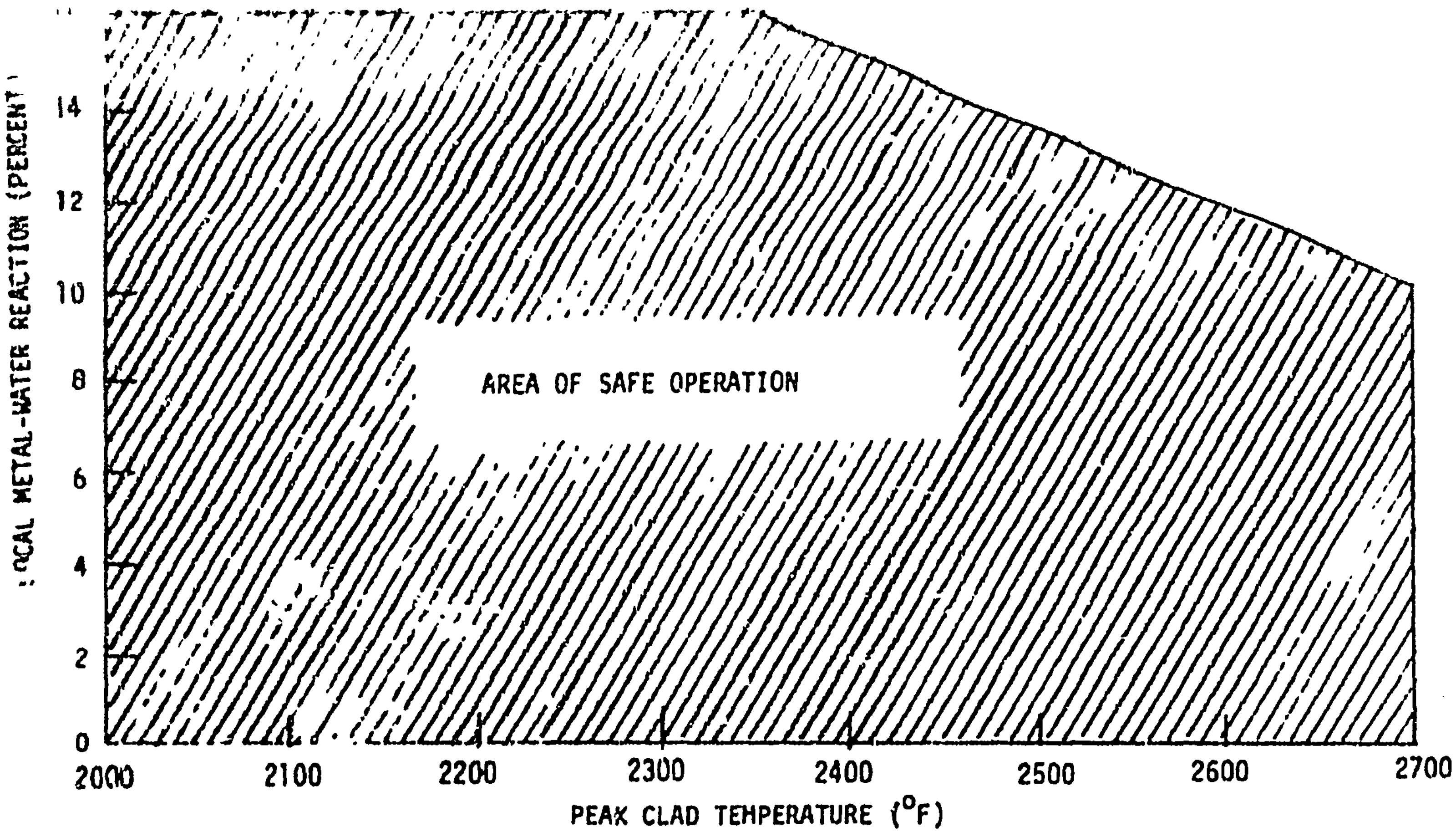


FIGURE 5 - PEAK CLAD TEMPERATURE AND METAL WATER REACTION LIMIT FOR A LOSS-OF-COOLANT ACCIDENT

14.2 STANDBY SAFETY FEATURES ANALYSIS

Adequate provisions have been included in the design of the plant and its standby engineered safety features to limit potential exposure of the public to well below the limits of 10 CFR 100 for situations which could conceivably involve uncontrolled releases of radioactive materials to the environment. The situations which have been considered are:

a) Fuel Handling Accidents
b) Accidental Release of Waste Liquid
c) Accidental Release of Waste Gases
d) Rupture of a Steam Generator Tube
e) Rupture of a Steam Pipe
f) Rupture of a Control Rod Drive Mechanism Housing - Rod Cluster Control Assembly (RCCA) Ejection

14.2.1 FUEL HANDLING ACCIDENTS

The following fuel handling accidents are evaluated to ensure that no hazards are created:

a) A fuel assembly becomes stuck inside reactor vessel.

b) A fuel assembly or control rod cluster is dropped onto the floor of the reactor cavity or spent fuel pit.

c) A fuel assembly becomes stuck in the penetration valve.

d) A fuel assembly becomes stuck in the transfer carriage or the carriage becomes stuck.

Causes and Assumptions

The possibility of a fuel handling incident is very remote because of the many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety. Also, before any refueling operations begin, verification of complete rod cluster control assembly insertion is obtained by tripping each rod individually to obtain indication of rod drop and disengagement from the control rod drive mechanisms. Boron concentration in the coolant is raised to the refueling concentration and verified by sampling. Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical with all rod cluster assemblies withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications.

After the vessel head is removed, the rod cluster control drive shafts are disconnected from their respective assemblies using the manipulator crane and the shaft unlatching tool. A spring scale is used to indicate that the drive shaft is free of the control cluster as the lifting force is applied.

The fuel handling manipulators and hoists are designed so that fuel cannot be raised above a position which provides adequate shield water depth for the safety of operating personnel. This safety feature applies to handling facilities in both the containment and in the spent fuel pit area. In the spent fuel pit, the design of storage racks and manipulation facilities is such that:

> Fuel at rest is positioned by positive restraints in an eversafe, always subcritical, geometrical array, with no credit for boric acid in the water.
>
> Fuel can be manipulated only one assembly at a time.

Violation of procedures by placing one fuel assembly in juxtaposition with any group of assemblies in racks will not result in criticality.

Crane facilities do not permit the handling of heavy objects, such as a spent fuel shipping container, above the fuel racks.

Adequate cooling of fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel assembly is immersed continuously while in the refueling cavity or spent fuel pit.

Even if a spent fuel assembly becomes stuck in the transfer tube, the fuel assembly is completely immersed and natural convection will maintain adequate cooling to remove the decay heat. The fuel handling equipment is described in detail in Section 9.5.

Two Nuclear Instrumentation System source range channels are continuously in operation and provide warning of any approach to critical during refueling operations. This instrumentation provides a continuous audible signal in the containment, and would annunciate a local horn and a horn and light in the plant control room if the count rate increased above a preset low level.

Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical by at least 10 per cent with all rod cluster control assemblies inserted. At this boron concentration the core would also be more than 2 per cent subcritical with all 53 control rods withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications.

All these safety features make the probability of a fuel handling incident very low. Nevertheless, it is possible that a fuel assembly could be dropped during the handling operations. Therefore, this incident is analyzed both from the standpoint of radiation exposure and accidental criticality.

Special precautions are taken in all fuel handling operations to minimize the possibility of damage to fuel assemblies during transport to and from the spent fuel pit and during installation in the reactor. All handling operations on irradated fuel are conducted under water. The handling tools used in the fuel handling operations are conservatively designed and the associated devices are of a fail-safe design.

In the fuel storage area, the fuel assemblies are spaced in a pattern which prevents any possibility of a criticality accident. The design of the facility is such that it is not possible to carry heavy objects, such as a spent fuel transfer case, over the fuel assemblies in the storage racks. The design is such that only one fuel assembly can be handled at a given time.

The motions of the cranes which move the fuel assemblies are limited to a low maximum speed. Caution is exercised during fuel handling to prevent the fuel assembly from striking another fuel assembly or structures in the containment or fuel storage building.

The fuel handling equipment suspends the fuel assembly in the vertical position during fuel movements, except when the fuel is moved through the transport tube.

The design of the fuel assembly is such that the fuel rods are restrained by grid clips which provide a total restraining force of approximately 60 pounds on each fuel rod. If the fuel rods are in contact with the bottom plate of the fuel assembly, any force transmitted to the fuel rods is limited due to the restraining force of the grid clips. The force transmitted to the fuel rods during fuel handling is not sufficient to breech the fuel rod cladding. If the fuel rods are not in contact with the bottom plate of the assembly, the rods would have to slide against the 60 pound friction force. This would absorb the shock and thus limit the force on the individual fuel rods.

After the reactor is shut down, the fuel rods contract during the subsequent cooldown and would not be in contact with the bottom plate of the assembly.

Considerable deformation would have to occur before the rod would make contact with the top plate and apply any appreciable load on the fuel rod. Based on the above, it is felt that it is unlikely that any damage would occur to the individual fuel rods during handling. If one assembly is lowered on top of another, no damage to the fuel rods would occur that would breech the integrity of the cladding.

If during handling the fuel assembly strikes against a flat surface, the loads would be distributed across the fuel assemblies and grid clips and essentially no damage would be expected in any fuel rods.

If the fuel assembly were to strike a sharp object, it is possible that the sharp object might damage the fuel rods with which it comes in contact but breeching of the cladding is not expected. It is on this basis that the assumption of the failure of an entire row of fuel rods (15) is a very conservative upper limit.

Analyses have been made assuming the extremely remote situation where a fuel assembly is dropped and strikes a flat surface, where one assembly is dropped on another, and where one assembly strikes a sharp object. The analysis of a fuel assembly assumed to be dropped and strikes a flat surface considered the stresses the fuel cladding was subjected to and any possible buckling of the fuel rods between the grip clip supports. The results showed that the buckling load at the bottom section of the fuel rod, which would receive the highest loading, was below the critical buckling load and the stresses were relatively low and below the yield stress. For the case where one assembly to be dropped on top of another fuel assembly, the loads will be transmitted through the end plates and the RCC guide tubes of the stuck assembly before any of the loads reach the fuel rods.

The end plates and guide thimbles absorb a large portion of the kinetic energy as a result of bending in the lower plate of the falling assembly. Also, energy is absorbed in the struck assembly top end plate before any load can be transmitted to the fuel rods. The results of this analysis indicated that the buckling load on the fuel rods was below the critical buckling loads and the stresses in the cladding were relatively low and below yield.

The refueling operation experience that has been obtained with Westinghouse reactors has verified the fact that no fuel cladding integrity failures are expected to occur during any fuel handling operations.

Rupture of one compl te outer row of fuel elements in a withdrawn assembly is assumed as a conservative limit for evaluating the environmental consequences of a fuel handling incident. The remaining fuel assemblies are so protected by the storage rack structure that no lateral bending loads would be protected. No damage resulted from an axially applied load of 2200 lb. to a fuel assembly. The maximum column load expected to be experienced in service is approximately 1000 lb. This information was used in the fuel handling equipment design to establish the limits for inadvertent axial loads.

Activity Release Characteristics

For the assumed accident there would be a sudden release of the gaseous fission products held in the voids between the pellets and cladding of fifteen fuel rods. The low temperature of the fuel during handling operations precludes further significant release of gases from the pellets themselves after the cladding is breached. Halogen release is also greatly minimized due to their low volatility at these temperatures. The strong tendency for iodine in vapor and particulate form to be scrubbed out of gas bubbles during their ascent to the water surface further reduces the quantity released from the water surface.

Decontamination factors of 10^{-3}(1) have been measured with much shallower water depth and much higher gas-to-water ratios. In a Westinghouse laboratory apparatus, elemental iodine (I_2) was passed in an air stream through a solution of 2000 ppm boron as boric acid. This solution is chemically similar to that in the spent fuel storage pit. The contact time in this apparatus corresponded to a bubble rise of 1.6 cm. Initially, the iodine decontamination factor (D.F.) in this apparatus was about 90%. The value decreased with time as the concentration of iodine in solution approached saturation, as expected. The D.F. at zero aqueous iodine concentration agreed with that obtained with an iodine fixing regent (sodium thiosulfate) in solution, indicating that gas phase diffusion to the bubble wall was controlling when the iodine laden bubbles contacted fresh solution. This condition can be assumed to represent the scrubbing of gas bubbles released from an accidental cladding failure as they rise through a vast reservoir of iodine free solution in the spent fuel pit. The calculated contact time in the accident can be related to the experiment by the ratio of the submergence, which is 24 feet, in the case of the plant, compared with 1.6 cm in the experiment. Assuming the same mass transfer rate in the bubble the D.F. of 10^{-3} would be obtained in a rise of only 9.9 cm. While this extrapolation is undoubtedly optimistic, it indicates that a large margin is available in the height of bubble rise in the pool to compensate for differences in bubble size and the decay of eddy motion inside the bubble with time. Conservatively assuming all of the iodine in the gap of the 15 rods (500 curies I-131) is present in the gaseous phase and taking credit for the 10^{-3} decontamination factor, the estimated I-131 release from the water surface is 0.5 curie.

An evaluation of the diffusion of the various iodine isotopes to the pellet-clad gap over a full core cycle indicates that when fuel is to be removed from the core to the spent fuel pit only about 0.6% of the gap iodine inventory is I-131 the remainder being principally I-127, I-129 and I-31. Further, from iodine vapor pressure considerations at the low temperature associated with refueling and the volume present for the iodine vapor to occupy, only about 2.6% of the total iodine present in the gap can be present in the form of I_2 vapor the remainder plating out during the cooldown. Inclusion of this effect would reduce the above assumed release by a factor of about 40.

The noble gas activity calculated present in the gas gap of the 15 rods is given in Table 14.2.1-1. These activities are based on a 90 hour decay period following operation at 3216 MWt for a full core cycle. This is the minimum delay before the reactor vessel head can be removed and a fuel assembly can be moved following shutdown.

The model for determining gap activity during operation is discussed in Section 14.3.5.

Method of Analysis

The activity could be released either in the containment or in the auxiliary (fuel storage) building. Both areas ventilation systems are in operation under administrative control during refueling hence in calculating dose inside the structures uniform dilution is assumed within the structure. Radioactivity monitors would immediately indicate and alarm the increased activity level, directing evacuation of the respective area. Activity in the containment would automatically close the purge ducts. In evaluating dose to refueling personnel inside the containment 15 minutes is assumed a reasonable time for evacuation. In the spent fuel storage area the integrated dose is evaluated based on the 33,000 cfm ventilation rate and the 180,000 ft^3 free volume.

In calculating offsite exposure it is assumed that the incident occurs in the spent fuel pit and that the activity is discharged to the atmosphere at the ground level through doors in the fuel storage building. This results in maximum ground level doses. This assumption is very conservative for two reasons: the ventilation system normally draws air across the fuel pit and exhausts to the vent stack, and during fuel transfers, the containment and fuel storage building doors are closed.

Dispersion of this activity is computed using the Gaussian plume dispersion formula and taking credit for building wake dilution as included in the two hour dispersion factor developed in Section 14.3.5.

Summary of Calculated Doses

The calculated doses are summarized in Table 14.2.1-2 and are seen well less than 10 CFR 100 limits. The permissible containment re-entry time after reduction to Xe-133 occupational MPC is nine hours.

It is concluded that a dropped fuel assembly would present no criticality hazard and would not result in significant radiation exposure at the site boundary (530 meters).

[1]Diffey, H. R., et. al., "Iodine Cleanup in a Steam Suppression System," AERE-R4882, 1965.

TABLE 14.2.1-1

NOBLE GAS ACTIVITY RELEASE FROM FUEL HANDLING INCIDENT

Isotope	Activity (Curies)
Kr-85	77
Xe-133 (m)	73
Xe-133	1220

TABLE 14.2.1-2

SUMMARY OF FUEL HANDLING INCIDENT DOSES

	Whole Body Dose (rem)	Thyroid Dose (rem)
Dose for 15 min. exposure inside Containment	1	4
Dose in Spent Fuel Building for Duration	4	16
Dose at Site Boundary	<0.1	0.2

14.2.2 ACCIDENTAL RELEASE-RECYCLE OR WASTE LIQUID

Accidents which would result in the release of radioactive liquids are those which may involve the rupture or leaking of system pipe lines or storage tanks. The largest vessels are the three liquid holdup tanks (CVCS), each sized to hold two-thirds (2/3) of the reactor coolant liquid volume, which are used to process the normal recycle or waste fluids produced. The contents of one tank will be passed through the liquid processing train while another tank is being filled.

All liquid waste components (WDS) except the reactor coolant drain tank and the waste holdup tank are located in the auxiliary building and any leakage from the tank or piping will be collected in the building sump to be pumped back into the liquid waste system. The waste holdup and the liquid holdup tanks are located in a thick concrete underground vault. The vault volume is sufficient to hold the full volume of any tank without overflowing to areas outside the vault. The reactor coolant drain tank is located in the containment building. holdup tanks are equipped with safety pressure relief and designed to accept the established seismic forces at the site. Liquids in the Chemical and Volume Control System flowing into and out of these tanks are controlled by manual valve operation and governed by prescribed administrative procedures.

The volume control tank design philosophy is similar in many respects to that applied for the holdup tanks. Level alarms, pressure relief valves and automatic tank isolation and valve control assure that a safe condition is maintained during system operation. Excess letdown flow is directed to the holdup tanks via the reactor coolant drain tank.

Piping external to the containment running between the containment and the auxiliary building and between the auxiliary building and liquid holdup tank vault is run below grade in concrete trenches. Any liquid spillage from pipe rupture or leaks in these trenches would drain to sump and be pumped to the sump tank and to the waste holdup tank.

The incipient hazard from these process or waste liquid releases is derived only from the volatilized components. The releases are described and their effects summarized in Section 14.2.3.

No credible mechanism exists for accidental release of waste liquids to the river. A river diffusion analysis was performed, however, to determine the concentrations which would result at the Chelsea reservoir if a release was assumed. The results of the analysis show that even the instantaneous release of the entire primary coolant system maximum activity corresponding to operation with 1% defects would not result in peak concentrations at Chelsea in excess of 10 CFR 20 MPC limits. Drought conditions were assumed to exist at the time of and for a period following the spill limiting the total run off flow to 4000 cfs. The mean longitudinal diffusion coefficient corresponding to this flow was 8.74 square miles per day. These data represent a drought similar to conditions existing in late summer of 1964, which can be verified by data in Section 2.5.

The unlikely event of a loss of water from a spent resin storage tank actuates a low level alarm to warn the operator. Resin contained in the tank can then be cooled by periodically flushing water from the primary storage water tank through the resin. Two pathways are available for the water; (a) through the primary storage water injection pipeline used when resin is removed from the tank or (b) through the primary storage pipeline used when resin is sluiced from the demineralizers into the tank.

Conservative assumptions made to determine the frequency of flushing to cool the resin are as follows:

1) The tank contains only the mixed bed resin from one mixed bed demineralizer discharged to the spent resin storage tank following operation of the plant for one cycle with 1% fuel defects. This assumption yields the maximum heat generation per unit volume of resin in the tank and the maximum level of radioactivity in the tank.

2) There are no heat losses through the tank walls.

3) Water is lost immediately following discharge of a mixed bed resin into the spent resin storage tank. This yields a maximum heat generation rate due to fission product decay.

These assumptions result in the following relationships:

1) The heat generation rate, q (BTU/hr), due to fission product decay is approximated closely as a function of time, t (hours), by

$$q = 178e^{-0.0116t} + 50e^{-0.00127t} + 37.5$$

where the first term is a short lived, the second an intermediate lived, and the last term is a long lived isotope contribution.

2) The mean heat capacity of resin is 0.31 BTU/lb-°F.

3) Resin volume is 25 ft^3 consistent with assumption number (1) and with the coefficients of the heat generation formula given in item (1).

4) Resin specific gravity is 1.14 with a void fraction of 0.4 giving a resin density of 43 lb/ft^3.

5) The amount of radioactivity in the tank is 21,250 curies.

On this basis, the resin bed temperature, T (°F), as a function of time, t (hours), is

$$T = 47\,(1 - e^{-0.0116t}) + 141\,(1 - e^{-0.00127t}) + 0.11t + T_o,$$

where T_o is the initial resin temperature. If T_o is assumed to be 90°F, it will take four days for the bed temperature to rise to 140°F, the normal resin operating limit. At or below a temperature of 140°F, the radioactivity will not be rel ed from the resin. The actual time to heat to 140°F will be greater than four days because of the conservative assumptions made in the calculation. The heat accumulated in the resin through the initial four days will be 18750 BTU. The bed can be maintained at 140°F or less by back flushing the resin with primary water at four day intervals. Flush water will be collected by the floor drain system and be pumped to the waste holdup tank. If a 10°F rise is taken in the flush water, the total quantity of water required will be about 250 gallons per back flush operation to remove the 18750 BTU accumulated in the resin.

Hence the loss of water from the spent resin storage tank presents no hazard off-site or on-site because means are available both to detect the situation occurring and to keep the resin temperature under control until the resin can be removed to burial facilities.

14.2.3 ACCIDENTAL RELEASE - WASTE GAS

The leakage of fission products through cladding defects can result in a buildup of radioactive gases in the reactor coolant. Based on experience with other operational, closed cycle, pressurized water reactors, the number of defective fuel elements and the gaseous coolant activity is expected to be low. The shielding and sizing of components such as demineralizers and the waste handling system are based on activity corresponding to 1% defective fuel which is at least an order of magnitude greater than expected. Tanks accumulating significant quantities of radioactive gases during operation are the gas decay tanks, the volume control tank, and the liquid holdup tanks.

The volume control tank accumulates gases over a core cycle by stripping action of the entering spray. Equilibrium gaseous activity for the tank based on operation with 1% defective fuel is tabulated in Table 14.2.3-1. During a refueling shutdown this activity is vented to the waste gas system and stored for decay. Rupture of this tank is assumed to release all of the contained noble gases plus that small amount contained in the 75 gpm flow from the demineralizers which would continue for up to five minutes before isolation would occur. The released activity would be 15,000 curies equivalent Xe-133.

The liquid holdup tanks receive reactor coolant, after passing through demineralizers, during the process of coolant purification. The liquid is stored and then processed through the boric acid evaporators for recycle. The contents of one tank are passed through the liquid processing train while another tank is being filled. In analyzing the consequence of rupture of a holdup tank it is assumed that all of the contained noble gas activity is released. A major tank failure would be required to cause release of all the contained noble gas. Since the tanks operate at low pressure, approximately 2 psig, a gas phase leak would result in expulsion of approximately 12% of the contained gases and then the pressure would be in equilibrium with atmosphere. The tank pits are vented to the ventilation system so that any gaseous leakage would be discharged to atmosphere by this route. Any liquid leaks from the tanks or piping will be collected in the tank sump pit to be pumped back into the liquid waste system.

The waste gas decay tanks receive the radioactive gases from the radioactive liquids from the various laboratories and drains processed by the waste disposal system. The maximum storage of waste gases occurs after a refueling shutdown at which time the gas decay tanks store the radioactive gases stripped from the reactor coolant. As discussed in Section 11.1, six shutdown gas decay tanks are provided in addition to the four gas decay tanks used during power operation to reduce the gaseous activity release as a result of an assumed rupture of one of the tanks during the decay period following a refueling shutdown.

Dose Evaluation

The provisions and operating procedures for this plant will preclude an offsite whole body dose in excess of 0.5 rem as a result of accidental release of the gases stored in any of the above tanks. This is accomplished by limiting the noble gas activity stored in any one tank to less than the amount computed to result in a 0.5 rem whole body dose at the site boundary as calculated below:

1. The dispersion factor described in Section 14.3 including the effect of dilution in the wake of the containment building and a one m/sec wind velocity is used in calculating maximum activity concentration at the site boundary, i.e. $X/Q = 7.5 \times 10^{-3}$ sec/m^3.

2. The cloud shape is approximated by an infinitely long semi-cylinder of semi-area, A_s, defined by the assumption that the cloud is of uniform concentration $X_{max.\ site\ boundary}$, throughout and the continuity relation, $Q = (X_{max.\ site\ boundary})\ (A_s)\ (\bar{u})$, which results in a cylinder radius,

$$R_S = \frac{2}{\pi \bar{u}} \left(\frac{Q}{X}\right)$$

3. The dose for a given γ emitter is computed from

$$D_\gamma = K \int_{\substack{\text{semi-cyl.}\\ \text{volume}}} \frac{S_{v-\gamma}\, e^{-\mu_t \bar{r}}}{4\pi\ \bar{r}^2}\ B(\bar{r})\ dV$$

where the buildup factor, $B(\bar{r})$, is approximated by $1.5e^{(\mu_t - \mu_a)\bar{r}}$,

$$S_{v-\gamma} = 3.7 \times 10^{10} \frac{\text{dis/sec}}{\text{curie}} \; C \text{ (curies) } \frac{X}{Q} \left(\frac{\text{sec}}{m^3}\right) f \left(\frac{\gamma}{\text{dis}}\right) E \left(\frac{\text{Mev}}{\gamma}\right)$$

and $K^{-1} = 6.8 \times 10^{10} \left(\frac{\text{Mev}/m^3}{r}\right)$

4. The beta dose is evaluated assuming 2r exposure at the skin surface, i e.

$$D_B = \frac{1}{2} K S_{v-B}$$

Where S_{v-B} is the volumetric beta source at the site boundary, $\left(\frac{\text{Mev}}{\text{sec}-m^3}\right)$

The maximum reactor coolant noble gas activity corresponding to operation with 1% fuel defects as listed on an isotopic basis in Table 9.2-5 is 110,000 curies equivalent Xe-133. The total reactor coolant system noble gas gamma and beta sources corresponding to operation with 1% fuel defects are 1.91×10^{14} Mev/sec and 5.17×10^{14} Mev/sec respectively. Based on the above, complete release of the reactor coolant system noble gas inventory would result in an offsite gamma dose of 0.32 rem and an offsite beta dose of 2.95 rem or a total dose of 3.27 rem. Restricting the maximum inventory of noble gases in any gas or liquid tank to 15% of the total maximum reactor coolant system inventory or to 16,500 curies equivalent Xe-133, will result in a total offsite exposure of less than 0.5 rem for complete release of the noble gas activity stored in the tank.

The 16,500 curie equivalent Xe-133 limit exceeds the quantity calculated present in the volume control tank during operation with 1% defective fuel and also exceeds the quantity that would be placed in one of the six shutdown gas decay tanks during a refueling shutdown when decay of the shorter half-lived xenon and krypton isotopes is considered during the processing time. A radiation monitor counts activity in a gas decay tank sample to the gas analyzer, and sounds an alarm if activity approaches the above limit.

In the process of filling a holdup tank during recycle operation approximately 50% of the coolant activity could be put in a single holdup tank if decay is neglected during the filling time and it if were assumed none of the holdup tank liquid was processed during the filling time. Hence if coolant activity should approach that corresponding to 1% defects, sampling and inventory balances will be utilized to ensure that the 16,500 curies equivalent Xe-133 limit is not exceeded.

The iodine present in the vapors of the above tanks would be minimal based on past operating experience. With an iodine removal factor of 10 in the mixed bed demineralizers the maximum iodine concentration in the liquid of the volume control tank or the liquid holdup tank would be less than 0.2 μc/cc I-131 when operating with 1% defective fuel. An iodine partition factor on the order of 10^{-4} is expected between the liquid and vapor. The corresponding gaseous iodine release from a holdup tank, which is the largest of the above tanks, would be less than 10 millicuries I-131. In comparison, the evaluation of the fuel handling incident resulted in an offsite thyroid dose of 0.2 rem for release of 500 millicuries to the atmosphere.

TABLE 14.2.3-1

VOLUME CONTROL TANK NOBLE GAS ACTIVITY

Isotope	Specific activity cur/m^3 vapor) Kr-85
Kr-85	6.4
Kr-85m	17.5
Kr-87	5.8
Kr-88	27.7
Xe-133	3880
Xe-135m	41
Xe-135	128

The above activities are computed for the vapor space (3.7 m^3) when operating with 1% fuel defects. The activity present in the liquid is negligible by comparison.

14.2.4 STEAM GENERATOR TUBE RUPTURE

The event examined is a complete tube break adjacent to the tube sheet, since a minor leak may not necessitate immediate action depending on the particular circumstances. If a tube breaks, reactor coolant would discharge into the secondary system. Since the reactor coolant is radioactive, methods of operation to limit uncontrolled condensate release have to be considered.

Once the reactor coolant system pressure is below the steam generator design pressure the faulty steam generator will be isolated by redundant valves and the possibility of uncontrolled leakage removed.

The following sequence of events is initiated by a tube rupture:

1. Rapidly falling pressure and level in the pressurizer will initiate a safety injection signal, tripping the unit. The safety injection signal automatically terminates normal feedwater and initiates auxiliary feedwater.

2. The steam generator liquid monitor and the air ejector radiation monitor will alarm, indicating the passage of primary fluid into the secondary system. The air ejector discharge is automatically diverted back to the containment and the setam generator blowdown and sampling systems are isolated automatically.

3. The unit tirp will automatically shut off steam flow through the turbine and will open steam bypass valves and bypass steam to the condenser.

4. In the unlikely event of concurrent blackout, the loss of circulating water through the condenser would eventually result in loss of condenser vacuum and valves in the condenser bypass lines would automatically close to protect the condenser, thereby causing steam relief to be to atmosphere.

5. Cooldown procedures are followed which entail:

   a) boration

   b) regulating pressurizer level with spray or relief valves

   c) reducing safety injection flow in order to reduce reactor coolant pressure to several hundred psi subcooling.

   d) condenser relief (if available) or atmospheric relief in order to reduce the reactor coolant temperature.

6. Isolation of the faulty steam generator is achieved by:

   a) further reducing safety injection flow so as to drop the RCS pressure below 1100 psi (steam generator design pressure)

   b) close the steamline isolation valve connected to the affected steam generator (determined by steam generator liquid sample activity monitor) and block the atmospheric relief

   c) turn off the auxiliary feedwater flow to that steam generator

7. Ordinarily this would end the leakage during the interval while cooldown is continuing by steam bypass from the intact steam generators. Should the faulty steam generator's outlet valve not close, then the main steamline bypass valve would be closed and atmospheric relief from the intact steam generators would be used for plant cooldown.

8. After the residual heat removal system is in operation, the condensate accumulated in the secondary system can be examined at the operators leisure. If the radioactivity level is in excess of that allowed, the condensate can be processed through the waste disposal system.

The faulty unit will be isolated by a steam line isolation valve once the reactor coolant pressure is reduced below 1100 psia. This can be accomplished in approximately 30 minutes and will terminate the mass flow into the secondary system and steam relief from the faulty steam generator.

With power available to the circulating water pumps the steam is bypassed to the condenser. The air ejector discharge is diverted to the containment within a few seconds after the air ejector monitor high activity signal and the activity release to atmosphere is not significant.

With concurrent blackout a portion of the reactor coolant system activity is released to atmosphere in steam relief during the 30 minutes to isolate the faulty steam generator.

All of the noble gas activity contained in the portion of reactor coolant discharged into the steam generator during the 30 minutes to isolate are assumed released to atmosphere.

The iodine transferred into the steam generator is assumed to partition between the liquid and vapor phases of the steam generator and the portion contained in the steam relief is assumed released to atmosphere. A distribution factor of 4×10^{-3} has been selected from the data of Reference (2) as being representative of the pH and pressure conditions within the steam generator.

During the 30 minute period to isolate the faulty steam generator 70,000 lbs of reactor coolant are discharged into the steam generator and 60,000 lbs of steam are relieved to atmosphere. Based on a reactor coolant system activity concentration corresponding to 1% defective fuel the noble gas activity release to atmosphere is 14,000 equivalent curies Xe-133. The corresponding iodine activity discharge into the steam generator is 114 curies equivalent I-131 of which 5.7 curies are released to atmosphere.

The resultant site boundary dose is 0.5 rem whole body and 2.2 rem to the thyroid using the two hour meteorological dispersion factor discussed in Section 14.3.5.

With the circulating water pumps available, the air ejector discharge is diverted to the containment and the only activity released to atmosphere is that contained in the steam flow to the turbine gland seal (5000 $\frac{lb}{hr}$ For this case the activity release to atmosphere during the 30 minute period would be 1.1% of the values given above. It is concluded that a tube rupture accident would not result in significant radiation exposure.

REFERENCES

1. L. C. Watson, A. R. Bancroft and C. W. Howlke, "Iodine Containment by Dousing in NPD-11" AECL-1130 Atomic Energy of Canada Limited, Chalk River, Ontario, October 27, 1960.

2. M. A. Styrikovich, O. I. Martynova, K. Ya. Katkovskaya, I. Ya. Dubrovskii, and I. N. Smirnova, "Transfer of Iodine from Aqueous Solutions to Saturated Vapor," Atomnaya Energiya, Vol. 17, No. 1, pp. 45-49, July 1964.

14.2.5 RUPTURE OF A STEAM PIPE

A rupture of a steam pipe is assumed to include any accident which results in an uncontrolled steam release from a steam generator. The release can occur due to a break in a pipe line or due to a valve malfunction. The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant temperature and pressure. With a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive control rod is assumed stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power even with the remaining control rods inserted. A return to power following a steam pipe rupture is a potential problem only because of the high hot channel factors which may exist when the most reactive rod is assumed stuck in its fully withdrawn position. Assuming the most pessimistic combination of circumstances which could lead to power generation following a steam line break, the core is ultimately shut down by the boric acid in the Safety Injection System.

The analysis of a steam pipe rupture is performed to demonstrate that:

1) With a stuck rod and minimum engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact.

2) With no stuck rod and all equipment operating at design capacity, insignificant (or no) cladding rupture occurs.

Although DNB and possible clad perforation (no clad melting or zirconium - water reaction) following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive rod stuck in its fully withdrawn position.

The following systems provide the necessary protection against a steam pipe rupture:

1) Safety Injection System Actuation from any one of the following:*

   a. One out of three pressurizer coincident low pressure and low level signals

   b. Six sets of two out of three high differential pressure signals between steam lines

   c. High steam flow in two out of four lines (one out of two per line) in coincidence with either low reactor coolant system average temperature (three out of four) or low steam line pressure (three out of four)

   d. Two out of three high containment pressure signals

2) The overpower reactor trips (nuclear flux and ΔT) and the reactor trip occurring upon actuation of the Safety Injection System.

3) Redundant isolation of the main feedwater lines. Sustained high feedwater flow would cause additional cooldown, thus, in addition to the normal control action which will close the main feedwater valves, any safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

4) Closing of the fast acting steam line stop valves (designed to close in less than 5 seconds with no flow) on:

   a. High steam flow in two out of four lines (one out of two per line) in coincidence with either low reactor coolant system average temperature (three out of four) or low steam line pressure (three out of four).

   b. Two out of three high containment pressure signals.

*The details of the logic used to actuate Safety Injection are discussed in Section 7.2.

Each steam line has a fast closing stop valve and a check valve. These eight valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For example, for a break upstream of the stop valve in one line, closure of either the check valve in that line or the stop valves in the other lines will prevent blowdown of the other steam generators.

Steam flow is measured by monitoring dynamic head in nozzles inside the steam pipes. The nozzles (16" I.D. vs a pipe diameter of 28" I.D.) are located inside the containment near the steam generators and also serve to limit the maximum steam flow for any break further downstream. In particular, the nozzles limit the flow for all breaks outside the containment and those inside the containment which are downstream of the flow measuring nozzles. A schematic showing the location of the stop valves, check valves, and nozzles is shown in Figure 14.2.5-10.

Method of Analysis

The analysis of the steam pipe rupture has been performed to determine:

1) The core heat flux and reactor coolant system temperature and pressure resulting from the cooldown following the steam line break. A full plant digital computer simulation has been used.

2) The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital computer calculation has been used to determine if DNB occurs for the core conditions computed in (1) above. This calculation solves the continuity, momentum, and energy equations of fluid flow in the core and with the Macbeth critical heat flux correlation (see reference in paragraph 7 below) determines the margin to DNB.

The following assumptions were made:

1) The rods give 0.0195 shutdown reactivity at no load. This is the end of life design value including design margins with the most reactive rod stuck in its fully withdrawn position. The actual shutdown capability is expected to be significantly greater.

2) The negative moderator coefficient corresponding to the end of life core with all but the most reactive rod inserted. The variation of the coefficient with temperature and pressure has been included. The k versus temperature at 800 psia corresponding to the negative moderator coefficient used is shown in Figure 14.2.5-1. In computing the power generation following a steam line break, the local reactivity feedback from the high neutron flux in the region of the core near the stuck control rod has been included in the overall reactivity balance. The local reactivity feedback is composed of doppler reactivity from the high fuel temperatures near the stuck control rod and moderator feedback from the high water enthalpy near the stuck rod. For the cases analyzed where steam generation occurs in the high flux regions of the core, the effect of void formation on the reactivity has been included. The effect of power generation in the core on overall reactivity is shown in Figure 14.2.5-2. The curve assumes end of life core conditions with all rods in except the most reactive rod which is assumed stuck in its fully withdrawn position (completely removed from the core).

3) Minimum safety injection capability corresponding to two out of three safety injection pumps in operation and three out of four safety injection lines available for flow to the reactor coolant system. 20,000 ppm boron is assumed in the boric acid tank at the suction of the Safety Injection pumps. The time delays required to sweep the low concentration boric acid from the safety injection piping prior to the delivery of the 20,000 ppm boron have been included in the analysis.

4) A steam generator heat transfer parameter of 13800 BTU/sec F. This is considered conservative since no allowance for reduction of the heat transfer UA as the water level falls into the tube region has been made. Furthermore, higher steam generator UA values will result in lower reactor coolant temperatures at full power which in turn will result in an increase in the available shutdown margin at zero load. This heat transfer coefficient with the reactor coolant system temperatures is used in determining the shutdown margin of 0.0195.

5) Hot channel factors corresponding to one stuck rod — the rod giving the highest factor at end of life. The hot channel factors account for the fact that at the reactor coolant pressures existing during the return to power following a steam pipe rupture void exists in the region of the core near the stuck rod. This void in conjunction with the large negative moderator coefficient offsets the effect of the stuck rod. The hot channel factors depend upon the core temperature, pressure, and flow and, thus, are different for each case studied. The values used for each case are given in Table 14.2.5-1. The calculations used to obtain the hot channel factors again assume end of life core conditions with all rods in except the most reactive rod.

6) Five steam pipe ruptures at end of core life, starting from the hot standby condition of the plant, have been considered in determining the core power and Reactor Coolant System transient.

   a) Complete severence of a pipe outside the containment, downstream of the steam flow measuring nozzle starting at no load conditions with outside power available.

   b) Complete severence of a pipe inside the containment at the outlet of the steam generator starting at no load conditions with outside power available.

c) Case (a) above with loss of outside power simultaneous with the steam break.

d) Case (b) above with loss of outside power simultaneous with the steam break.

e) A break equivalent to steam release through one steam generator safety valve with outside power available.

The cases above all assume the accident starts from hot shutdown conditions with the rods inserted (except for one stuck rod) at time zero. Should the reactor be just critical or operating at power at the time of a steam line break the reactor will be tripped by the normal overpower protection system when the power level reaches a trip point. These cases have been analyzed and are discussed along with the results of the five cases above. Following a trip at power the reactor coolant system contains more stored energy than at no load, the average coolant temperature is higher than at no load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no load conditions of reactor coolant system temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analyses which assume no load conditions at time zero.

7) In determining the critical heat flux the Macbeth correlation based on local coolant conditions has been used. This correlation is discussed in the report AEEW-R-267 dated August, 1963, "Burnout Analyses Part 4: Application of a Local Conditions Hypothesis to World Data for Uniformily Heated Round Tubes and Rectangular Channels", by R. V. Macbeth. This correlation was used since plant parameters are outside the range of validity of the W-3 correlation.

8) The Moody Curve for $fL/D = 0$ contained in Figure 3 of the article by F. J. Moody in Transactions of the ASME, Journal of Heat Transfer, February 1965, page 134 was used to calculate the steam flow through a steam line break.

Results

The results presented are a conservative indication of the events which would occur assuming a steam line rupture. The worst case assumes that all of the following occur simultaneously.

1) Minimum shutdown reactivity margin equal to 1.95%.

2) The most negative moderator temperature coefficient for the rodded core at end of life.

3) The rod having the most reactivity stuck in its fully withdrawn position.

4) One safety injection pump and one safety injection valve fails to function as designed.

Core Power and Reactor Coolant System Transient

Figure 14.1.5-3 shows the Reactor Coolant System transient and core heat flux following a steam pipe rupture (complete severence of a pipe) outside the containment, downstream of the flow measuring nozzle at initial no load conditions. The break assumed is the largest break which can occur anywhere outside the containment either upstream or downstream of the stop valves. Outside power is assumed available such that full reactor coolant flow exists. The transient shown assumes the rods inserted at time 0 (with one rod stuck in its fully withdrawn position and steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by high differential pressure between steam lines or by high steam flow signals in coincidence with either low reactor coolant system temperature or low steam line pressure will trip the reactor. Steam release from at least three steam generators will be prevented by either the check valve or by automatic closing of the fast acting stop valves in the steam lines by the high steam flow signals in coincidence with either low reactor coolant system temperature or low steam line pressure. Even with the failure of one valve, release is limited to no more than 5 seconds for three steam generators while the fourth generator blows down. (The steam line stop valves are designed to be fully closed in less than 5 seconds with no flow through them. With the high flow existing during a steam line rupture, the valves will close considerably faster).

As shown in Figure 14.2.5-3, the core becomes critical with the rods inserted (with the design shutdown assuming one stuck rod) at 45 seconds. Boron solution at 20,000 ppm enters the Reactor Coolant System from the Safety Injection System at 131 seconds with a delay of 80 seconds required to clear the Safety Injection System lines of low concentration boric acid. The 80 seconds delay is taken from the time at which the system pressure has fallen to 1250 psia and in the delay for sweeping the Safety Injection System of low concentration boric acid if the system pressure were constant at 1000 psia. Since the pressure is less than 1000 psia during most of the 80 seconds and since the shutoff head of the safety injection pumps is 1500 psia, the 80 seconds is somewhat conservative.

The computer calculation used assumes the boric acid is mixed with and diluted by the water flowing in the reactor coolant system prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the Reactor Coolant System and in the Safety Injection System. The variation of mass flow rate in the Reactor Coolant System due to water density changes is included in the calculation as is the variation of flow rate in the Safety Injection System due to changes in the Reactor Coolant System pressure. The Safety Injection System flow calculation includes the line losses in the system as well as the pump head curve. No credit has been taken for the 2,000 ppm boron which enters the Reactor Coolant System prior to the 20,000 ppm boric acid. It should be noted that the core power transient would be terminated even without the addition of boric acid when the affected steam generator boils dry. For the transient shown in Figure 14.2.5-3 this occurs at 145 seconds. The peak core average heat flux for this case is 14% of the value at 2758 MWt.

Figure 14.2.5-4 shows the case of a steam line rupture at the exit of a steam generator (inside the containment) at no load. The sequence of events is similar to that described above for the rupture outside the containment. The peak core average heat flux is 27% of 2758 MWt.

Figures 14.2.5-5 and 14.2.5-6 show the responses for the cases assuming a loss of outside power at time 0 which then results in a Reactor Coolant System flow coastdown. The Safety Injection System delay time includes the time required to start safety injection pumps on the diesels. It should be noted that the safety injection pumps cannot supply flow until the Reactor Coolant System pressure has fallen to 1500 psia. Thus, at least 35 seconds (see Figure 14.2.5-6) is available for starting the safety injection pumps on the diesels before the Reactor Coolant System pressure has fallen to 1500 psia. Credit is taken for only the safety injection flow entering one cold leg line since the second cold leg safety injection line can be the line assumed blocked by a valve failure. The flow to the hot legs will be delayed in reaching the core as a result of the low reactor coolant flow. The peak powers are 8% and 13%.

Figure 14.2.5-7 shows the transient following a break equivalent to steam release through one steam generator safety valve (credible break) with steam release from one steam generator. In this case, safety injection is initiated automatically by low pressurizer pressure and level at 212 seconds (the low pressure setpoint is assumed to be 1700 psia and is reached after the pressurizer is empty). Boron solution at 20,000 ppm enters the Reactor Coolant System at 343 seconds. For the transient, there is no return to criticality. The 20,000 ppm boron provides sufficient negative reactivity to keep the reactor shut down and well below criticality while the steam generator empties and causes further cooldown. The cooldown for the case shown in Figure 14.2.5-7 is more rapid than the case of steam release from all steam generators through one safety valve. The transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal or the energy stored in the other steam generators. Since the transient occurs over a period of six minutes, the neglected stored energy is likely to have a significant effect in slowing the cooldown. Figure 14.2.5-7 was computed assuming the design shutdown with all rods except one stuck rod inserted at time zero. The transient is assumed to start from a hot shutdown condition.

Figure 14.2.5-8 shows the initial portion of the transient following the maximum outside containment break assuming an initial power level of 100%. The transient was computed assuming the reactor is tripped by the high flux trip. In actuality, the reactor will be tripped before an appreciable power increase occurs by the high steam flow signal in coincidence with low steam line pressure. In the figure the high flux trip is assumed to occur conservatively at 118%. The steam flow is assumed limited by only the four steam line flow measuring nozzles and the steam line stop valves are assumed fully open. For a break inside the containment with the assumption that the check valve in the affected line fails to open, calculations have shown that the peak heat flux preceding and immediately following reactor trip does not differ significantly from that shown in Figure 14.2.5-8. The peak heat flux in Figure 14.2.5-8 is such that the DNBR does not approach 1.3.

Using the reactivity insertion rate which occurs with the steam flow limited by only the four steam line nozzles, assuming an initial power level of 10^{-13}, and a high flux trip at 25%, the peak average heat flux was computed to be 35%. (This transient is similar to that occurring during a startup accident -- see Section 14.1.1.) The same peak heat flux was computed using the steam flow existing following an inside containment break with the assumption that the check valve in the affected line fails open. The heat flux will not result in a DNBR of 1.3.

Figure 14.2.5-9 shows the reactivity insertion for a steam line break downstream of the steam line stop valves assuming the steam flow is initially that occurring through the four steam line nozzles. The steam flow is assumed to fall to zero in a step at 3, 5, 7, 10 and 15 seconds. The steam line stop valves are designed to be fully closed in less than five seconds with no flow through them. With the high flows assumed in Figure 14.2.5-9 the valves will close considerably faster. Criticality does not occur during any of the transients shown in 14.2.5-9.

It should be noted that following a steam line break only one steam generator blows down completely. Thus, three steam generators are available for dissipation of decay heat at times after the steam break transient is over.

Margin to Critical Heat Flux

Using the transients of Figures 14.2.5-4 and 14.2.5-6 with breaks assumed inside the containment for reference plant parameters, the Macbeth critical heat flux correlation was used to determine the margin to burn-out. The power and flow conditions were used with various core pressures and core inlet temperatures bracketing the values shown in Figures 14.2.5-4 and 14.2.5-6. The hot channel factors for each case are listed in Table 14.2.5-1. The analyses showed that the heat flux remains below the critical value in all cases. The minimum DNBR was greater than 2 in all the analyses. The lowest value occurred for the break inside the containment with outside power available assuming an inlet temperature of 425°F and a reactor coolant pressure of 700 psia. The breaks outside the containment result in lower core heat fluxes than those inside the containment and, thus, also do not cause DNB anywhere in the core. Therefore, even for the worst double-ended pipe rupture (that occurring inside the containment at the exit of a steam generator) the analysis has shown that no DNB occurs.

For the break inside the containment at the exit of a steam generator, the total mass and energy release to the containment have been conservatively computed as 2000000 lbs, and 182×10^6 Btu, respectively. Assuming an instantaneous release to the containment and no credit for containment safeguards, the containment pressure has been calculated to be 27 psig compared to the containment design value of 47 psig.

Dose Considerations

Assuming a steam line break occurred when operating with a leak in the steam generator, the portion of reactor coolant activity discharged through the leak will be released to the steam generator. For the case of the break outside the containment and the leak occurring in the steam generator with the ruptured steam line, this activity is released to atmosphere. In addition the activity initially present in the steam generator will be released. Following the accident the reactor system would be cooled down and depressurized. Assuming a cooldown rate of 50°F per hour, the system would be depressurized to 350 psig in about 4 hours. At this time the residual heat removal loop would be put into operation and in an additional four hours, the system would be cooled to saturation pressure less than atmospheric. The system would be depressurized and the leakage would terminate.

The dose evaluation is based on the following:

1. The product of the operating tube leak rate, L_o, and reactor coolant I-131 concentration, C_{I-131}, is assumed to be the maximum consistent with steady 10 CFR 20 I-131 MPC in the circulating water canal. i.e.,

$$L_o \times C_{I-131} = F_c \times MPC_{I-131}$$

   Where F_c is the circulating water flow rate and MPC_{I-131} is the unrestricted area water maximum permissible I-131 concentration.

2. Due to flow choking with the hot fluid the leak rate does not increase significantly when the secondary side of the steam generator is depressurized. Although the pressure is decreasing during the cooldown period, the leak rate is conservatively assumed constant at L_o during the 8 hour leakage period.

The previous analysis has shown that fuel damage is not expected as a result of a steam line break outside the containment, hence the concentration of the I-131 in the leak remains at C_{I-131}.

The initial quantity of I-131 in the steam generator liquid is assumed in equilibrium with the tube leak and steam generator blowdown rate, i.e.,

$$A_{I-131,\ SG} = \frac{L_o \times C_{I-131}}{B} \times V_{SG\ liq}.$$

Where B is the blowdown rate and $V_{SG\ liq}$ is the steam generator liquid volume

Based on the above the total I-131 release is:

$$A_{Total} = L_o \times C_{I-131}\left[T + \frac{V_{SG\ liq}}{B}\right] = F_c \times MPC_{I-131}\left[T + \frac{V_{SG\ liq}}{B}\right]$$

Where T is the 8 hour leakage period in the appropriate units.

With a circulating water rate of 840,000 gpm and a 10 gpm blowdown rate, the total iodine activity release is 2 curies equivalent I-131. Based on a short term site boundary dispersion factor of 7.5×10^{-4} sec/m^3, the thyroid dose is 0.8 rem.

As a measure of margin to dose guide lines suggested in 10 CRF 100 an additional analysis was made using the calculated fuel-clad gap activities, site dispersion factors, and breathing rates presented in Section 14.3.5. and assuming operation with a 1 gpm leak prior to the accident. The results indicate that in excess of 20% of the calculated total core fuel-clad gap activity would have to be released to the reactor coolant to cause thyroid dose levels of the magnitude of 10 CRF 100 when operating with a 1 gpm tube leak prior to an outside containment steam break.

TABLE 14.2.5-1

Nuclear Hot Channel Factors Used in Steam Break Analyses. Values assume the End of Life Rodded Core with One Stuck Rod

Case	Nuclear $F_{\Delta h}$	Nuclear F_z
Figure 14.2.5-3, Outside Containment Break, Outside Power Available	10.8	2.0
Figure 14.2.5-4, Inside Containment Break, Outside Power Available	7.7	1.8
Figure 14.2.5-5, Outside Containment Break, Loss of Outside Power	Not computed since average power is considerably less than the case of Figure 14.2.5-6.	
Figure 14.2.5-6, Inside Containment Break, Loss of Outside Power	7.0	1.8

VARIATION OF REACTIVITY WITH CORE TEMPERATURE
AT 800 PSIA FOR THE END OF LIFE RODDED CORE
WITH ONE ROD STUCK
(ASSUMES 0 POWER)

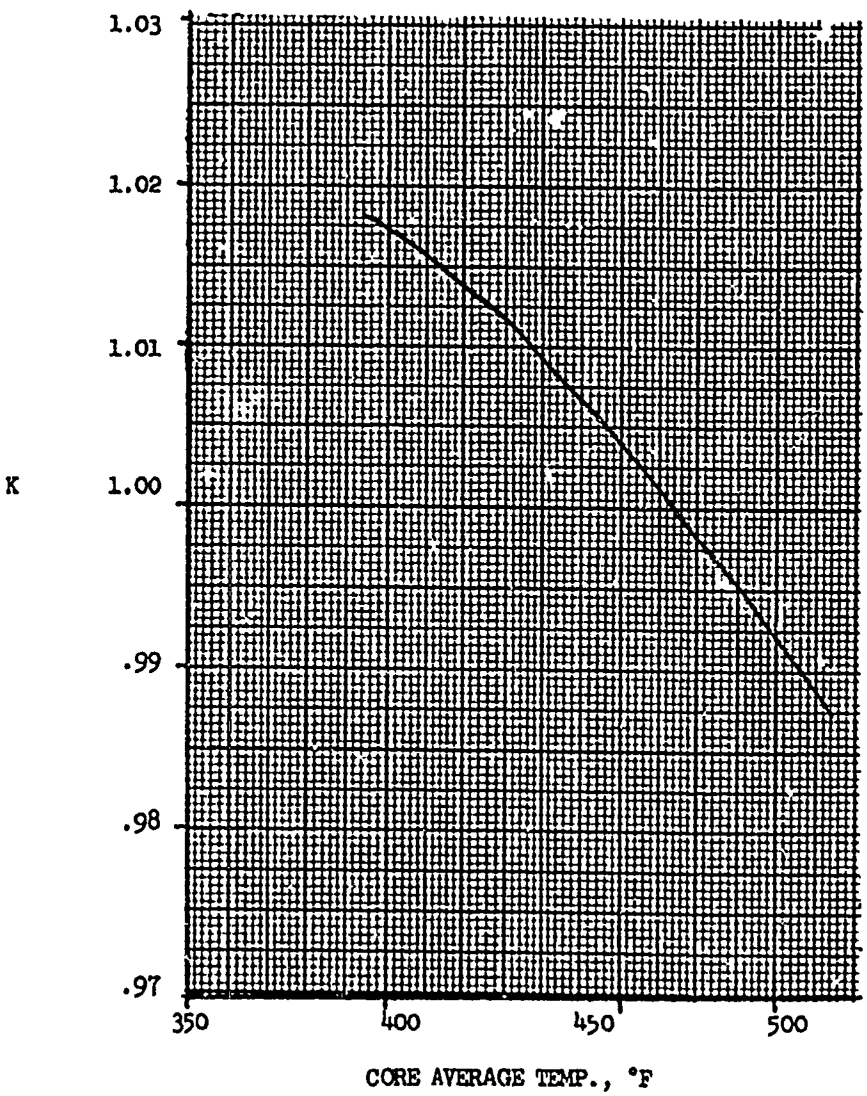


FIGURE 14.2.5-1

VARIATION OF REACTIVITY WITH POWER AT CONSTANT CORE AVERAGE TEMPERATURE. VALUES INDICATED WERE USED IN STEAM PIPE RUPTURE ANALYSES FOR THE END OF LIFE RODDED CORE WITH ONE ROD STUCK.

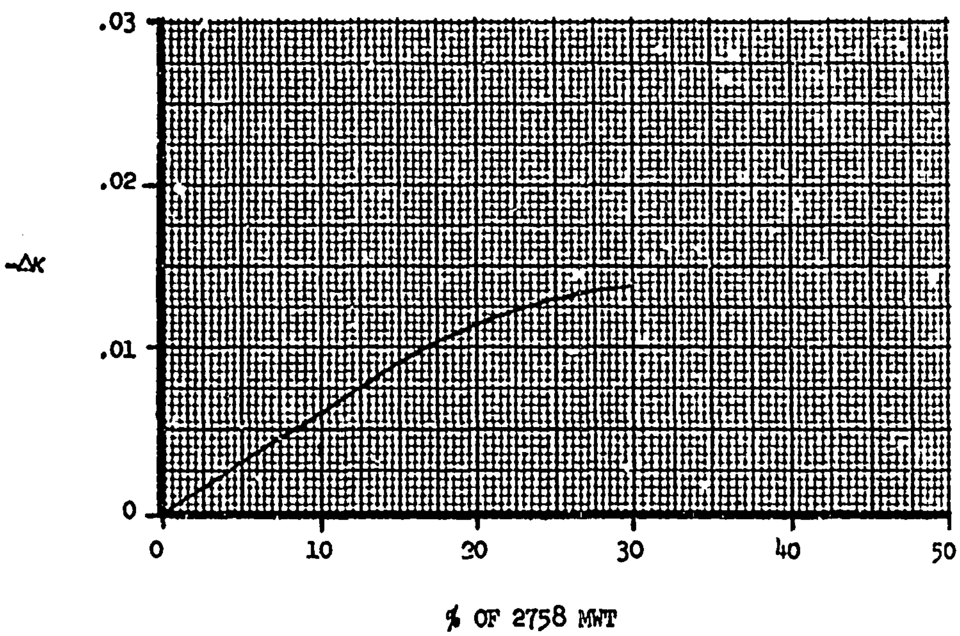


FIGURE 14.2.5-2

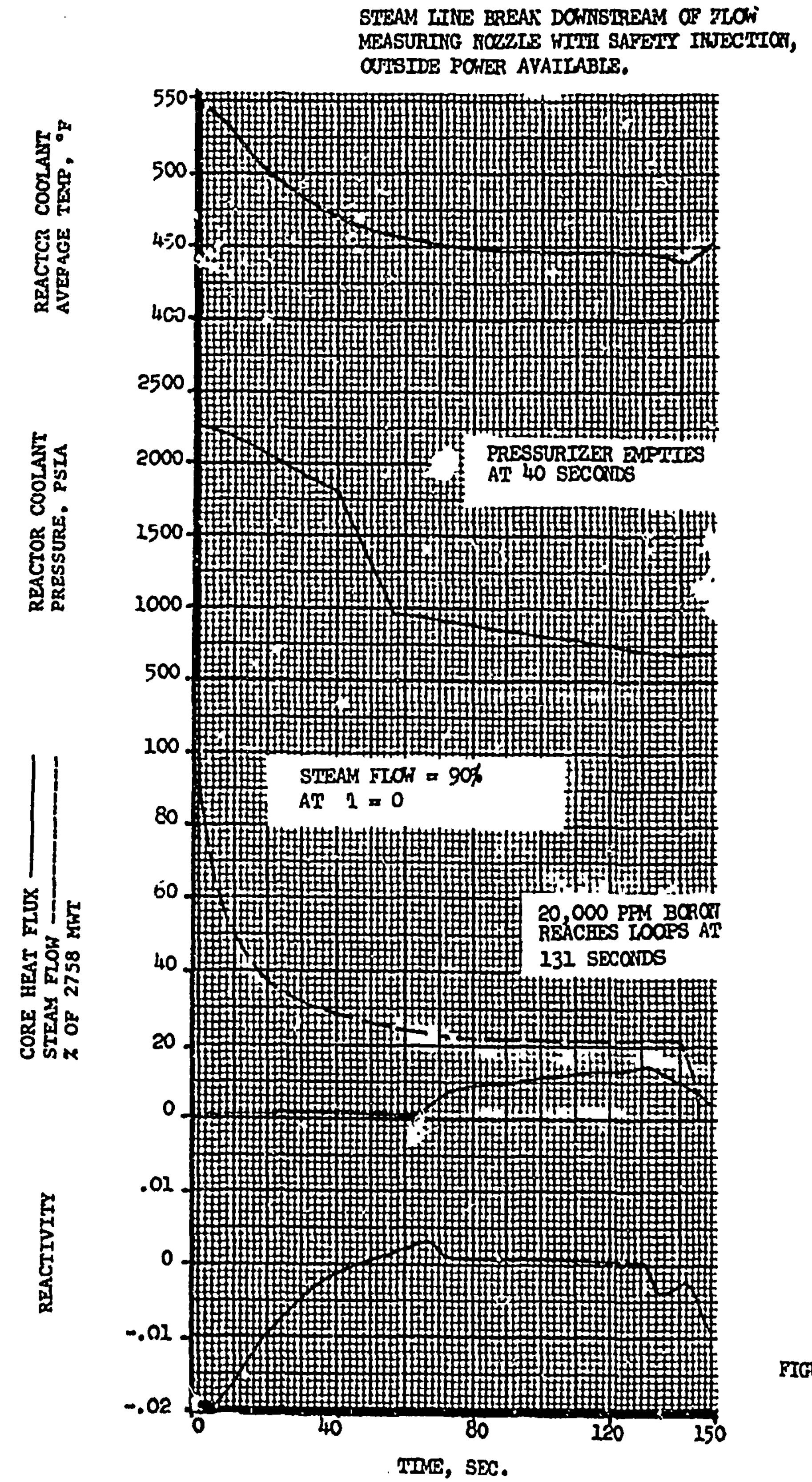


FIGURE 14.2.5-3

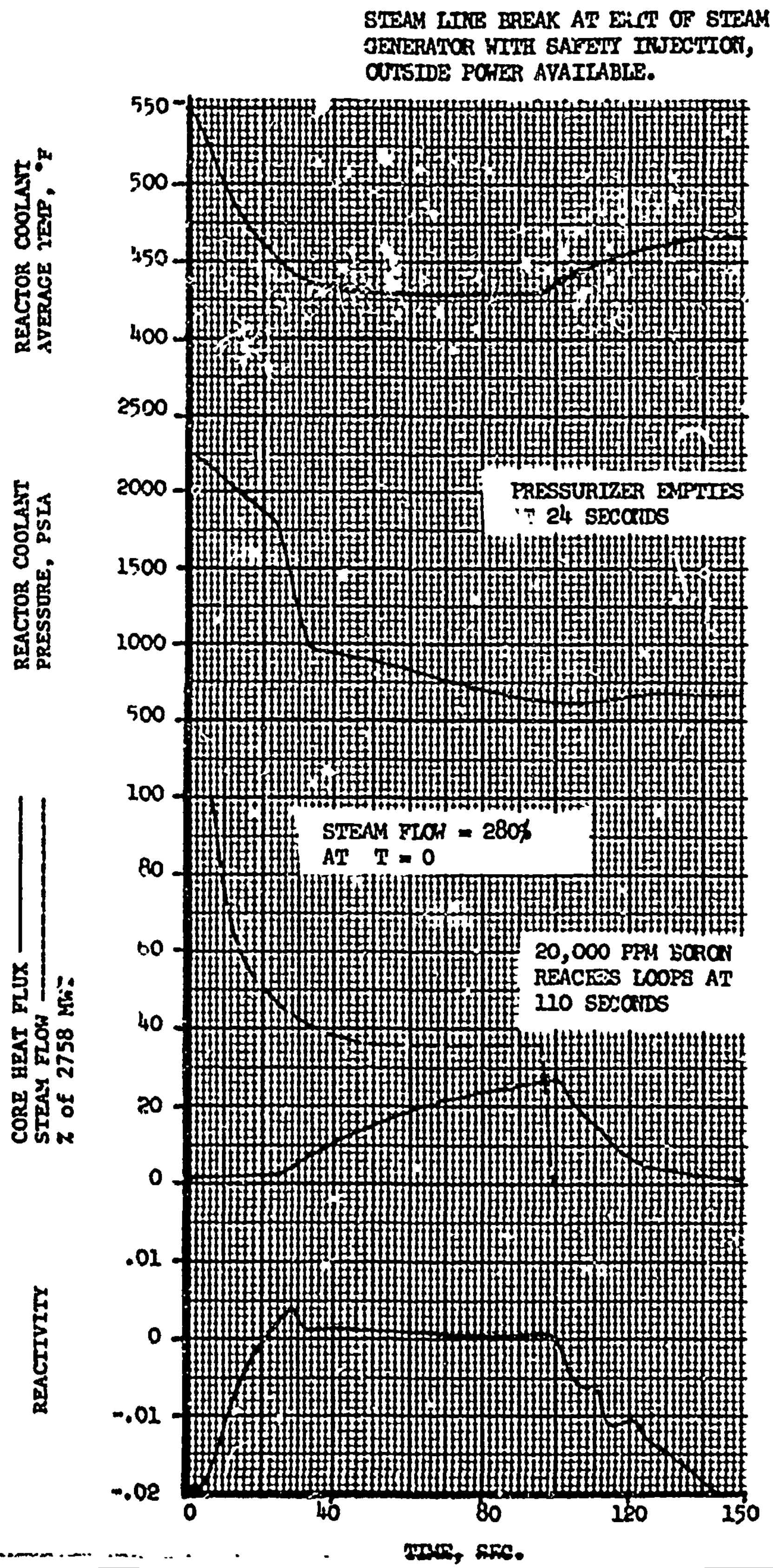


FIGURE 14.2.5-4

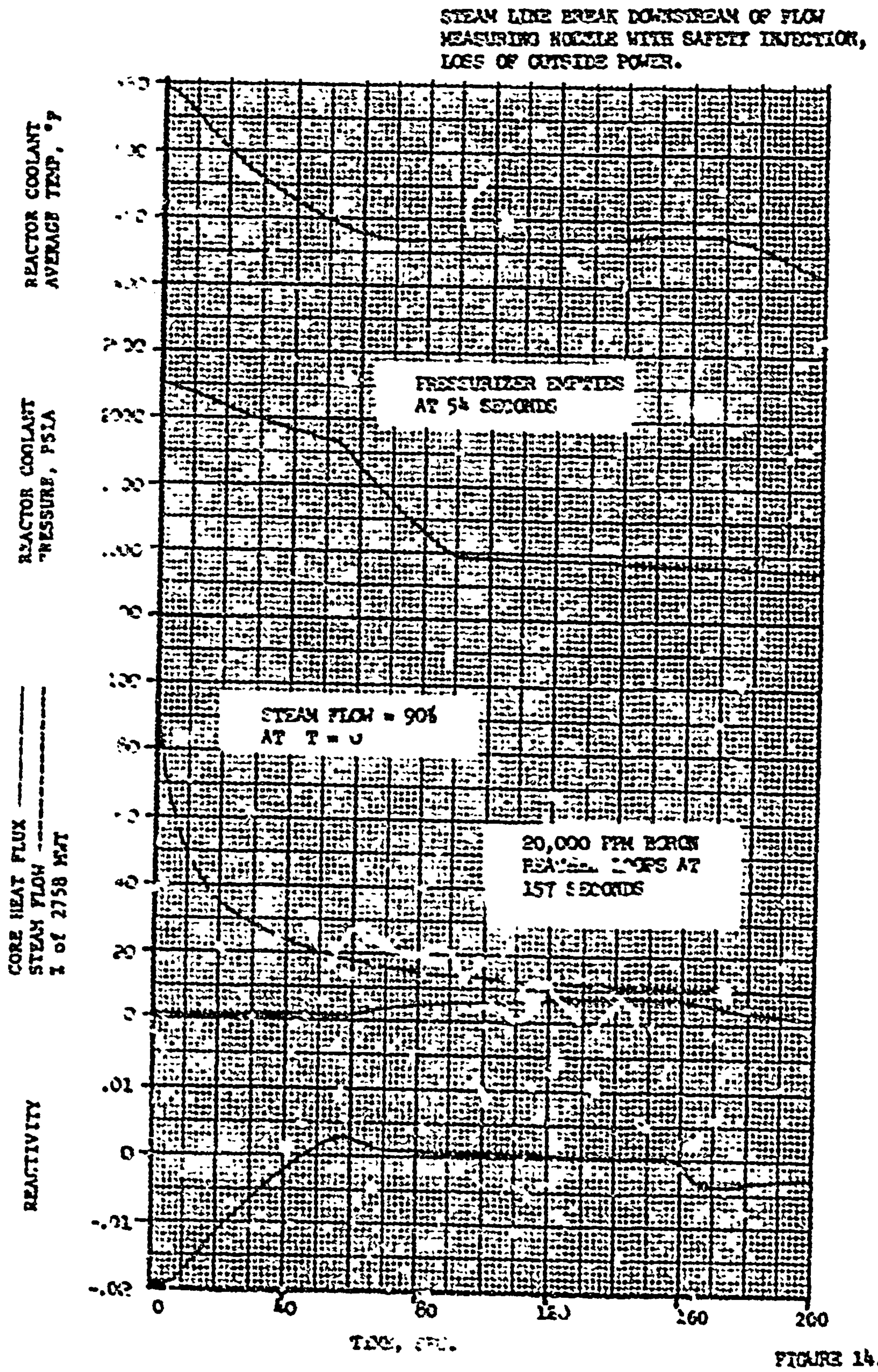


FIGURE 14.2.5-5

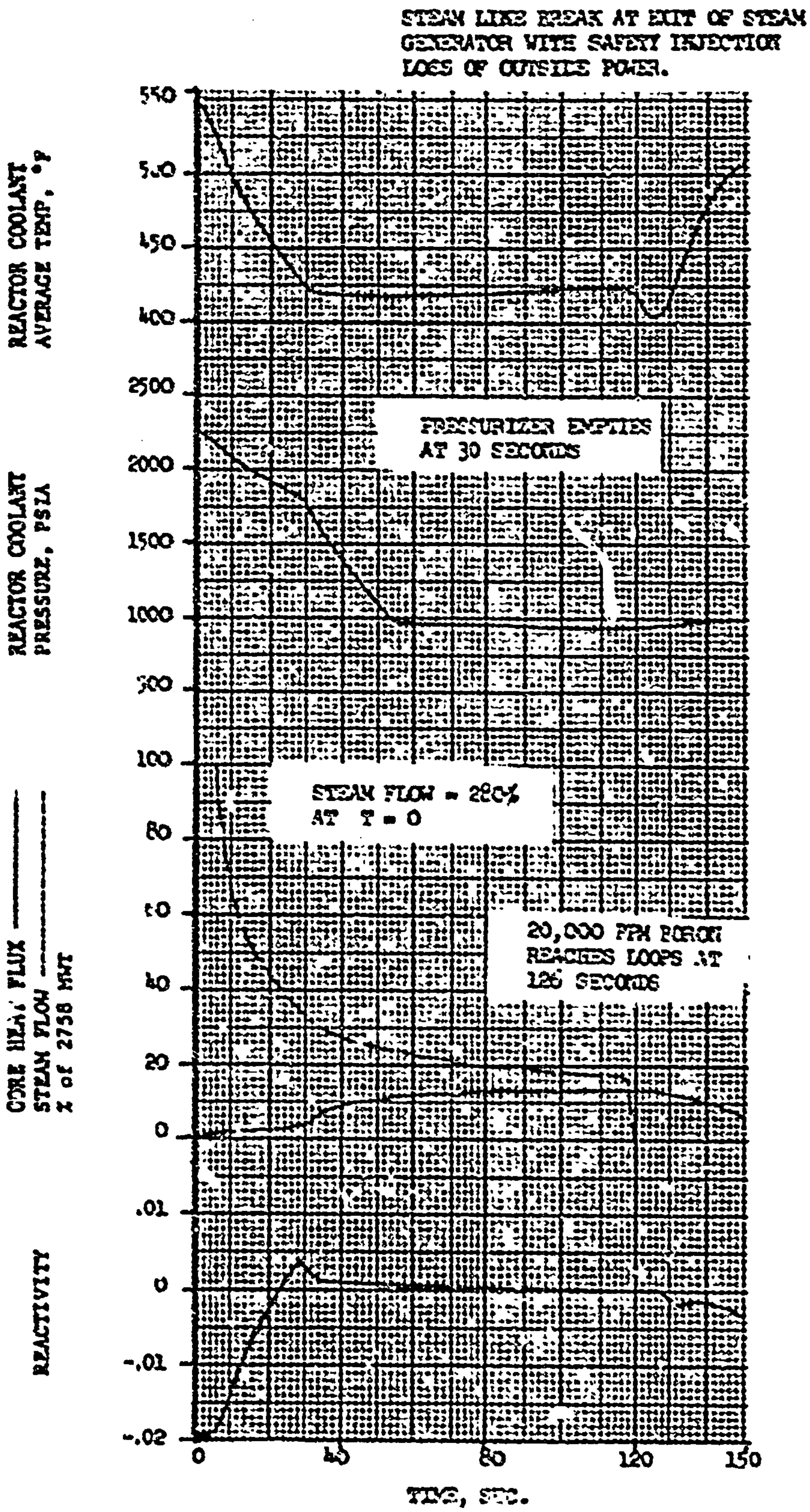


FIGURE 14.2.5-6

STEAM LINE BREAK EQUIVALENT TO
OPENING ONE STEAM GENERATOR SAFETY
VALVE, OUTSIDE POWER AVAILABLE.

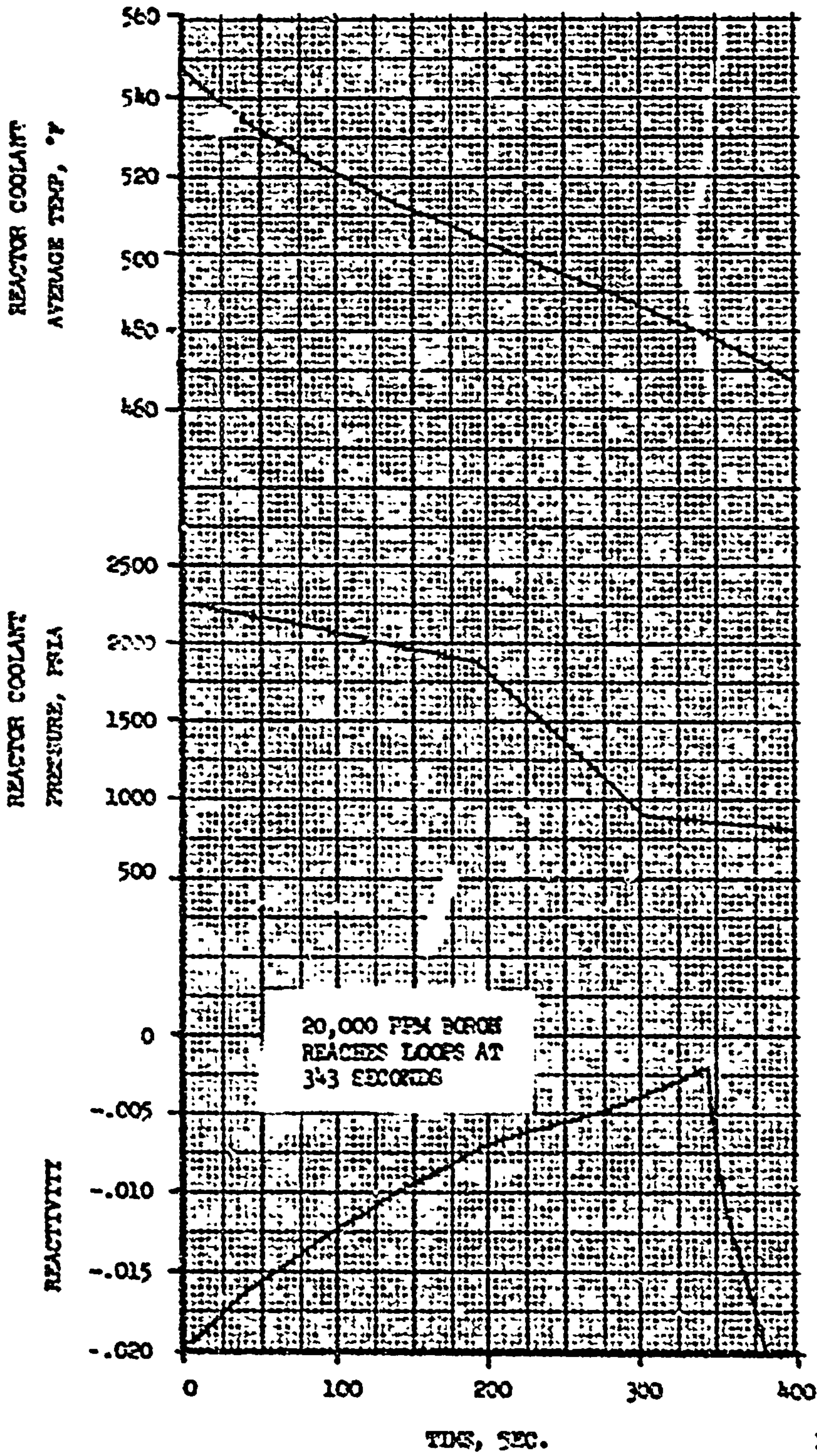


FIGURE 14.2.5-7

STEAM LINE BREAK DOWNSTREAM OF FLOW MEASURING NOZZLES WITH FLOW FROM 4 STEAM GENERATORS, INITIALLY AT 100% POWER.

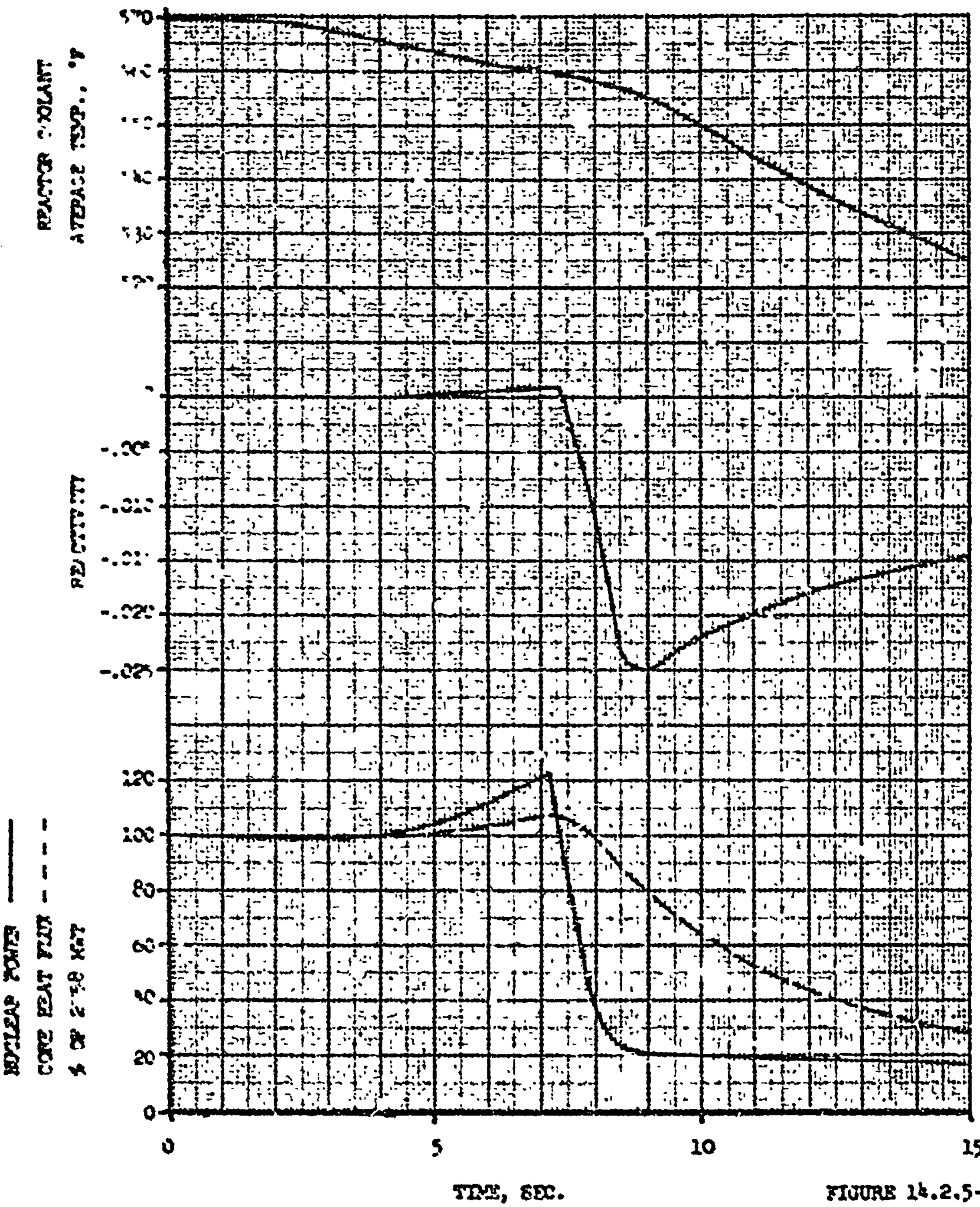
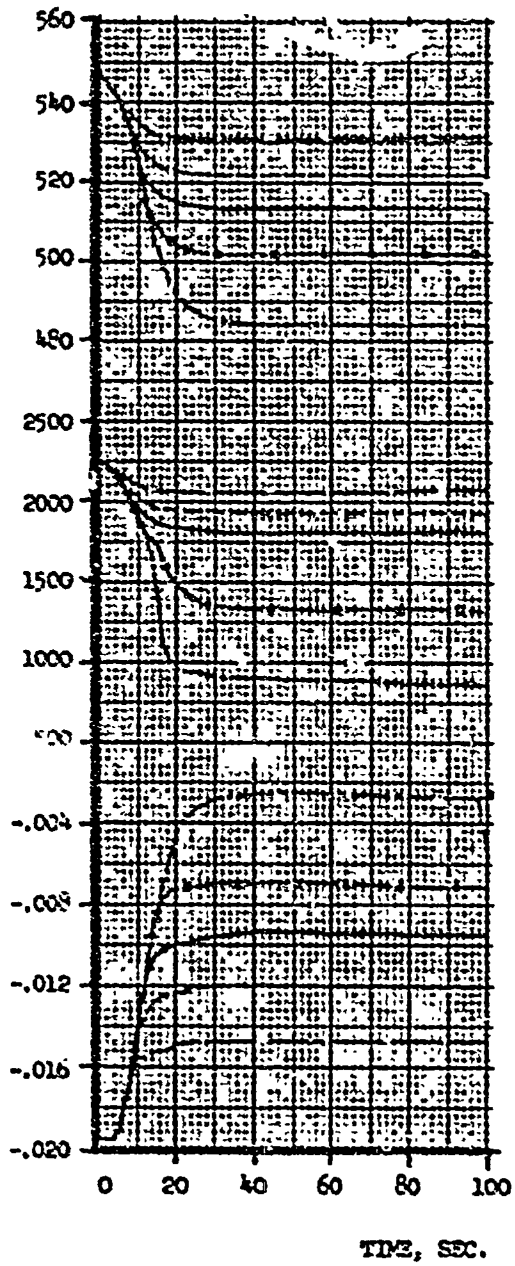


FIGURE 14.2.5-8

STEAM LINE BREAK DOWNSTREAM

OF FAST ACTING ISOLATION VALVES



A. 3 SEC. FLOW FROM 4 STEAM GENERATORS, NO FLOW AFTER 3 SEC.

B. 5 SEC. FLOW FROM 4 STEAM GENERATORS, NO FLOW AFTER 5 SEC.

C. 7 SEC. FLOW FROM 4 STEAM GENERATORS, NO FLOW AFTER 7 SEC.

D. 10 SEC. FLOW FROM 4 STEAM GENERATORS, NO FLOW AFTER 10 SEC.

E. 15 SEC. FLOW FROM 4 STEAM GENERATORS, NO FLOW AFTER 15 SEC.

FIGURE 14.2.5-9

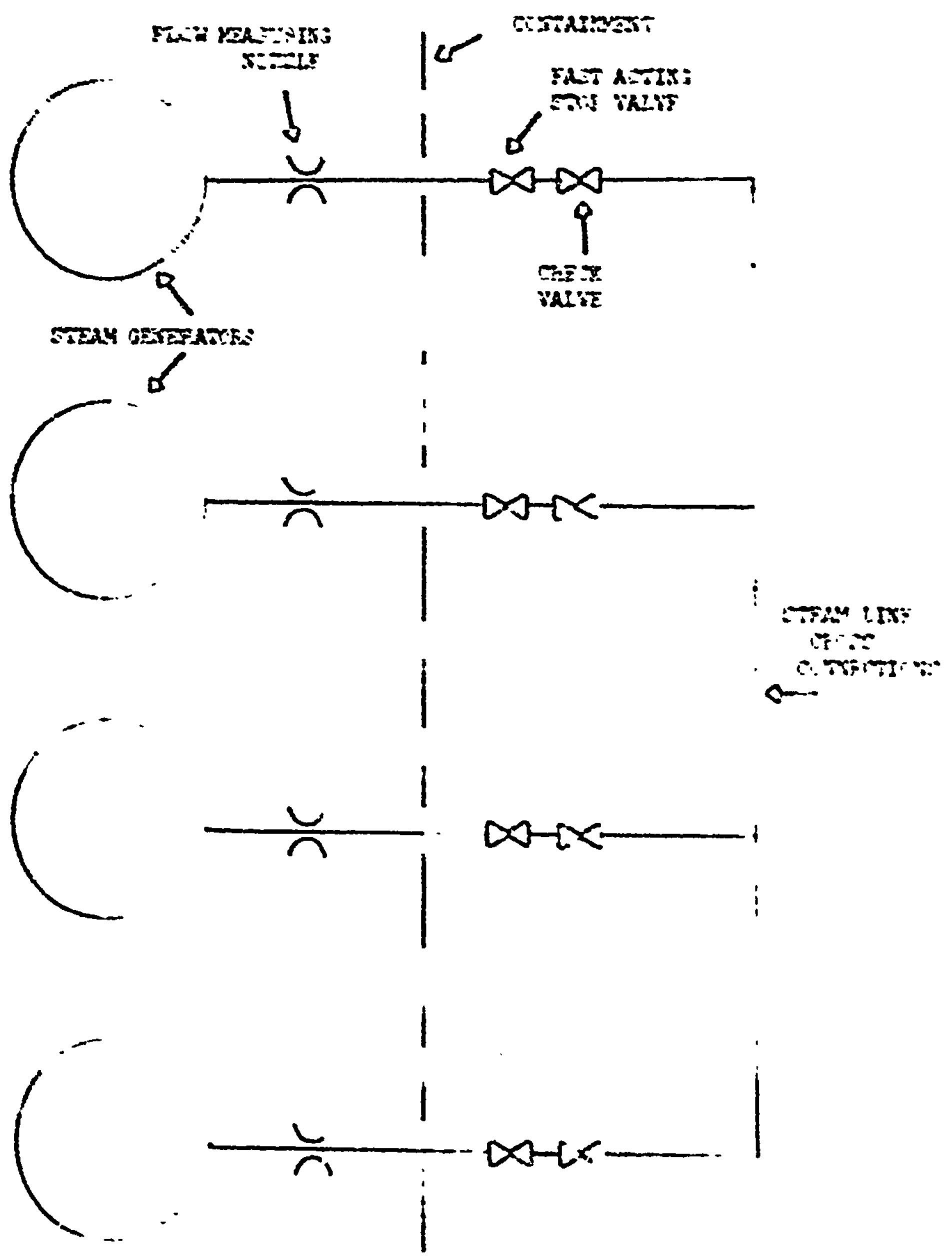


Figure 14.2.5-10 Schematic Showing the Location of the Steam Line Stop Valves, Check Valves, and Flow Measuring Nozzles

14.2.6 RUPTURE OF A CONTROL ROD MECHANISM HOUSING-RCCA EJECTION

In order for this accident to occur, a rupture of the control rod mechanism housing must be postulated creating a full system pressure differential acting on the drive shaft. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals.

A failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

a) Each control rod drive mechanism housing is completely assembled and shop-tested at 4100 psi.

b) The mechanism housing will be individually hydrotested to 3105 psig as they are installed on the reactor vessel head to the head adapters, and checked during the hydrotest of the completed Reactor Coolant System.

c) Stress levels in the mechanism are not affected by system transients at power, or by thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the ASME code, Section III, for Class A components.

d) The latch mechanism housing and rod travel housing are each a single length of forged type-304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered. The joint between latch mechanism and head adapter is a threaded joint, reinforced using a canopy type seal weld. The joint between the latch mechanism and rod travel housings is a Conoseal mechanical joint.

The operation of a chemical shim plant is such that the severity of an ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution, there are only a few rods in the core at full power. Proper positioning of these rods is monitored by a control room alarm system. There are low and low low insertion monitors, Section 7.3, with visual and audio signals. Operating instructions require boration at the low level alarm and emergency boration at the low low alarm. The control rod position monitoring alarm systems are described in detail in Section 7.3 and in reference 6. By utilizing the flexibility in the selection of control rod cluster groupings, radial locations and position as a function of load, the design minimized the peak fuel and clad temperatures for the highest worth ejected rod. It is shown that no fuel or clad melting occurs. Overinsertion of the control rods constitutes a violation of the operating instructions. However, for completeness sensitivity to overinsertion was considered. For the worst case, which is at full power at the end of life, it is shown that considerable margin exists. In fact, the C4 control bank may be fully inserted without causing clad melting on ejection.

This section describes the models used and the results obtained. The worst cases are presented in detail.

Only the initial few seconds of the power transient are discussed, since the long term considerations are the same as for a loss of coolant accident.

Method of Analysis

The calculation of the transient is performed in two stages, first an average core calculation and then a hot region calculation. The average core is analyzed to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator density reactivity. The largest temperature rises, (and hence the largest joint Doppler feedbacks) occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions also have high weights. This means that the reactor Doppler feedback is larger than that predicted by a simple single channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single channel Doppler feedbacks, correct them to effective whole core feedbacks for the appropriate flux shape. The values used in the analysis are listed in Table 14.3.6-2. Enthalpy and temperature transients in the hot spot are determined by adding a multiple of the average core energy generation to the hotter rods and performing a transient heat-transfer calculation. The asymptotic power distribution calculated without feedback is pessimistically assumed to persist throughout the transient.

Average Core

The nuclear power transients are calculated using the CHIC-KIN code developed by the Bettis Atomic Power Laboratory for similar analyses.(1) This code solves the point kinetics equations, with feedback from an axially and radially segmented fuel element. CHIC-KIN results have been compared with SPERT results for two dissimilar cores over a wide range of periods with good agreement.(1)

For this study, six delayed neutron groups were used and the fuel rod was divided into eight radial increments, with a ninth increment for the clad. Five axial segments were employed. The calculation is essentially a single channel analysis representing the core average conditions.

Prompt heat generation directly in the coolant has been calculated to be 2.6 percent of the nuclear power generation using the LEOPARD code. Heat generation in the fuel pellet is assumed to occur non-uniformly radially with a slight reduction in the center due to self-shielding effects.

Hot Region

The average core energy addition calculated as described above is multiplied by the appropriate hot channel factors and the worst cases analyzed using a detailed heat transfer code. The code contains a representation of gap conductance as a function of fuel temperature, clad temperature and differential pressure. The zirconium water reaction is explicitly represented and all material properties are represented as functions of temperature. The Tong, Sandberg and Bishop correlation (as described in section 3.2.2) is used to determine the film boiling heat transfer coefficient. The results indicate that no clad melting occurs. The energy levels assumed for fuel melting are given in Table 14.2.6-1.

Selection of Input Parameters

Pessimistic values for all the input parameters were selected on the basis of calculated values and a parameter study. The parameter study indicates that the parameters to which the transient is most sensitive are the ejected rod worth, the Doppler weighting, the delayed neutron fraction and transient hot channel factors. A margin of 10% has been added to the ejected rod worth. The Doppler weighting has been reduced by 10% to allow for errors in the weighting calculation and a further 10% for errors in the basic Doppler coefficient. The valve of β, the delayed neutron fraction, was not reduced since the method of calculation introduced conservatism, and in any case, a 10% conservatism had already been added to the ejected rod worth. Transient hot channel factors were conservatively calculated without taking any credit for the flattening effects of feedback. The values used in the final analyses are listed in Table 14.2.6-2.

The values of the ejected rod worth and transient hot channel factors, are dependent on the positioning of the part length rods. The overall ejected rod worth is arrived at by carrying out two separate calculations, one for the reactor slab containing the part length rods and one for the non-part length slab, and then determining a weighted total. The ejected rod worth and radial hot channel facts for the part length region are always higher than those for the non-part length region. Since the regional weighting increases with increasing regional flux, the worst overall ejected rod worth is obtained with the part length rods in the axial power peak. However, in general this condition does not result in the maximum value of Fq. If the part length rods are located in the axial flux peak, as they will be under normal conditions, the radial and axial peaks will be coincident. If the part length rods are moved out of the axial peak, the axial hot channel factor will increase. However, the axial peak is now in the non-part length region where the radial peaking factor is lower. This effect tends to reduce the dependance of Fq on the part length position, but the net result is often to increase Fq when the part length rods are moved away from the axial peak.

It can be seen that the worst part length rod position from the point of view of ejected rod worth probably does not correspond to the worst condition for Fq. Analyses indicate that the worst hot spot transient occures when the part length rods are located in the axial peak. The ejected rod worth is the dominant effect. All analyses have been conducted with the worst part length rod position with respect to the hot spot transient.

Pessimisms Included in the Analyses

Apart from the conservatism resulting from the pessimistic parameter selection, considerable conservatism exists in the model for the reasons noted below.

It is assumed that the transient and steady-state hot spots are coincident. This means that the largest temperature rise is combined with the highest steady-state temperature. In practice, the transient peak occurs in the immediate area of the ejected control rod, where under steady-state conditions the power is well below normal.

The Doppler feedback calculation is based on the asymptotic power distribution with feedback (i.e., flattened power distribution), but the hot spot transient power is computed on the basis of the asymptotic power distribution without feedback

Detailed physics calculations indicate that the total core moderator feedback will be considerably larger than that predicted on the basis of the average channel. However, the moderator feedback is pessimistically assumed equal to the average core value. Physics calculations were carried out to determine whether it is possible for local pockets of positive temperature coefficients to occur. An infinite medium moderator temperature coefficient was obtained by performing two 2-dimensional unit cell calculations with no transverse buckling. The calculations were carried out at two different temperature coefficients.

For the conditions which give the least negative temperature coefficient for the core as a whole, the above method yielded a local infinite moderator coefficient of zero. For positions in the core which have a higher than average local power density, there will be a net leakage out of the region, and so the local temperature coefficient will be more negative than this calculation indicates.

Reactivity insertion as a result of lattice deformation was considered. In the region of the hot spot there will be a large power gradient. Since the fuel rods are free to move in a vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradient across individual rods may produce a force tending to bow the midpoint of the rods toward the hot spot. Physics calculations indicate that the net result of this would be a negative reactivity insertion. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced.

Boiling in the hot spot region will produce a net fluid flow away from that region. However, the fuel heat . s released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It is concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is ignored in the following analyses.

Cases Considered

In general the end of life cases are worse than the beginning of life cases. This is because the delayed neutron fraction is smaller at the end of life, and also because ejected rod worths and hot channel factors tend to be worse at the end of life. However, for completeness, the full (27[illegible] MWt) and zero power cases for beginning of life are presented. The reactivity contributions and fuel rod temperatures for the cases which follow are presented in Figure 14.2.2-1 through 14.2.6-10. The power transients for the power cases and for the zero power cases are shown in Figures 14.2.6-11 and 14.2.6-12 respectively.

At the end of life, ejection of the worst control bank rods at full and zero power were studied as end point cases. At the end of life, both the worst ejected rod worth and worst hot channel factor occur when two banks are fully inserted with the third bank at the bottom limit of its solo movement (i.e., immediately before the fourth control bank begins to move in). The rod program is such that the maximum power at which this situation may occur is 4% of full power. An analysis has been carried out for this condition. The ejection of a part length rod was also considered.

Beginning of Life Full Power

The rod program limits the control bank holding to 0.5% ΔK for this condition. The reactor is sub-prompt critical with the worst ejected rod worth of 0.156%. The peak power reached is 1.27 times normal full power, and the peak hot spot clad and center fuel temperatures are respectively 1700°F and 4410°F. The results are shown in Figure 14.2.6-1 and 14.2.6-6.

Beginning of Life Zero Power

For this condition there will be one control bank fully inserted, and a second bank partially inserted. For conservatism the worst ejected rod from two fully inserted banks was used. The value of 0.75% ΔK results in the core becoming weakly prompt critical. At the peak hot spot heat flux, film boiling would not occur. (Even through only two of the four main coolant pumps are assumed to be running). The peak hot spot center fuel temperature is 1880°F. The results are shown in Figures 14.2.6-2 and 14.2.6-7.

End of Life Full Power, Ejection of the Worst Control Bank Rod

Results for this case are shown in Figures 14.2.6-3 and 14.2.6-8. Again the rod program limits the control bank reactivity holding to 0.5% ΔK. The worst ejected rod worth is then 0.2% ΔK. This results in a peak power of 1.55 times the normal full power and the peak hot spot heat flux of 684,000 Btu/ft^{2}hr (1.2 times design maximum). This heat flux will not result in film boiling. However, two hot spot cases were considered, one with DNB and one without DNB. For the case with DNB the peak clad and center fuel temperatures were respectively 1690°F and 4300°F.

Based on a steady-state hot channel factor of 3.23 (design), and a coincident transient hot channel factor of 3.8, the model indicates that an ejected rod of worth 0.3% ΔK would be required to initiate center melting. For this condition the peak cladding temperature would be less than 2100°F. In practice, the transient and steady-state hot spots cannot be coincident. It is therefore concluded that a 0.3% ΔK ejected rod can be tolerated with some considerable margin. The ejected rod worth for full power operation with the C4 control bank fully inserted is 0.28% ΔK. This condition constitutes a large deviation from the operating instructions. A further insertion of the control rod , beyond this condition, would of course increase the reactivity of the accident, and would result in a limited amount of fuel melting on the pellet center line at the leak spot. This would not produce fuel dispersal in the coolant. A considerable margin to clad melting would still exist.

Part Power End of Life

The worst part power ejected rod worth and hot channel factors occur when the first two control banks are fully inserted, and the third bank almost fully inserted. Physics calculations indicate that the maximum possible power at which this condition can occur is 6% of full power. For an ejected rod worth of 0.8% and assuming that only two main coolant pumps are running, this results in a peak cladding temperature of 2150°F and a peak center fuel temperature of 3320°F. These results are included in Figures 14.2.6-4 and 14.2.6-9.

Zero Power End of Life

The worst ejected rod worth at zero power is less than the value of 0.8% used in the part power case above. For conservatism, the value of 0.8% has been used. The peak cladding temperature is 1830°F, and the peak center fuel temperature is 2780°F. The result of this case are included in Figures 14.2.6-5 and 14.2.6-10.

Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered only a small fraction of the core (less than 2%) enters DNB. The position with regard to fission product release is therefore much better than for the double ended coolant pipe break (the maximum hypothetical accident) for which the majority of the core enters DNB.

Pressure Surge

Because there is no fuel or clad melting, even in the worst case with the most pessimistic assumptions, there is no danger of a sudden pressure rise due to heat transfer from dispersed molten fuel or from massive sudden metal-water reaction. Thus, the pressure conditions can be judged on the basis of relatively conventional heat transfer rates. The most severe excess addition of energy to the coolant occurs for the full power end of life case, and so this case results in the worst pressure transient. In order to estimate the magnitude of this pressure transient, average channel and hot spot heat transfer calculations were performed using a high gap conductance and without assuming DNB. The power curves used for these calculations represented a limiting case which almost initiated center melting at the hot spot. Using the heat flux data obtained above, a THINC 3 run was conducted to determine the volume surge (without the benefit of pressure feedback). This volume surge was subsequently used as the basis for a pressure calculation. The results indicated that starting at 2250 psi a peak pressure of about 2340 psi occurs some 1.5 seconds after rod ejection.

Conclusions

Even on the most pessimistic bases, the analyses indicate no fuel or clad melting. The amount of fission products released as a result of clad rupture during DNB is considerable less than in the case of the double ended main coolant pipe break (the maximum hypothetical accident). Furthermore, the resulting pressure surge is insufficient to produce consequential damage to the primary circuit.

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2. Conway and Hein, Journal of Nuclear Materials, (15.1), 1965

3. Ogard & Leary "High Temperature Heat Content and Heat Capacity of Uranium Dioxide - Plutonium Dioxide Solid Solutions." LA-DC-8620.

4. Barry, R.F., "Fab-7, One-Dimensional Diffusion Code," WCAP—2995 (1966).

5. Barry, R.F., "The Revised LEOPARD Code - A Spectrum Dependent Non-Spatial Depletion Program," WCAP-2759 (1965).

6. "Power Distribution Control of Westinghouse PWR's," WCAP 7208.

TABLE 14.2.6-1

UO_2 PROPERTIES

	Beginning of Life Enthalpy Btu/lb	End of Life Enthalpy Btu/lb
Solid at Room Temperature	0	0
Solid at Melting Temperature	414	391
Liquid at Melting Temperature	533	510

TABLE 14.2.6-2

Time of life	Beginning	Beginning	End	End	End
Power Level	0	Full	0	Full	4%
Parameter & Units					
Ejected worth %ΔK	0.75	0.156	0.80	0.20	0.80
Delayed neutron fraction	0.0069	0.0069	0.0052	0.0052	0.0052
Inlet temperature°F	543	539	543	539	543
Pressure psia	2275	2275	2275	2275	2275
Doppler temperature weighing	4.45	1.45	4.6	1.54	3.18
Scram time sec.	0.5	0.5	0.5	0.5	0.5
Scram shutdown %ΔK	1.0	3.0	0.9	3.0	0.9
Gap heat transfer coefficient for average core calculation Btu/hr ft^2 °F	150	150	150	150	150
Fuel thermal conductivity for average core calculation Btu/hr ft °F	1.5	1.5	1.5	1.5	1.5
Neutron lifetime ℓ* micro sec.	15	15	15	15	15
Initial moderator density coefficient ΔK/gr cm^{-3}	0.0219	0.0152	0.200	0.201	0.200
Transient F_q	20	4.22	21.6	3.8	17.6
Number of pumps running	2	4	2	4	2

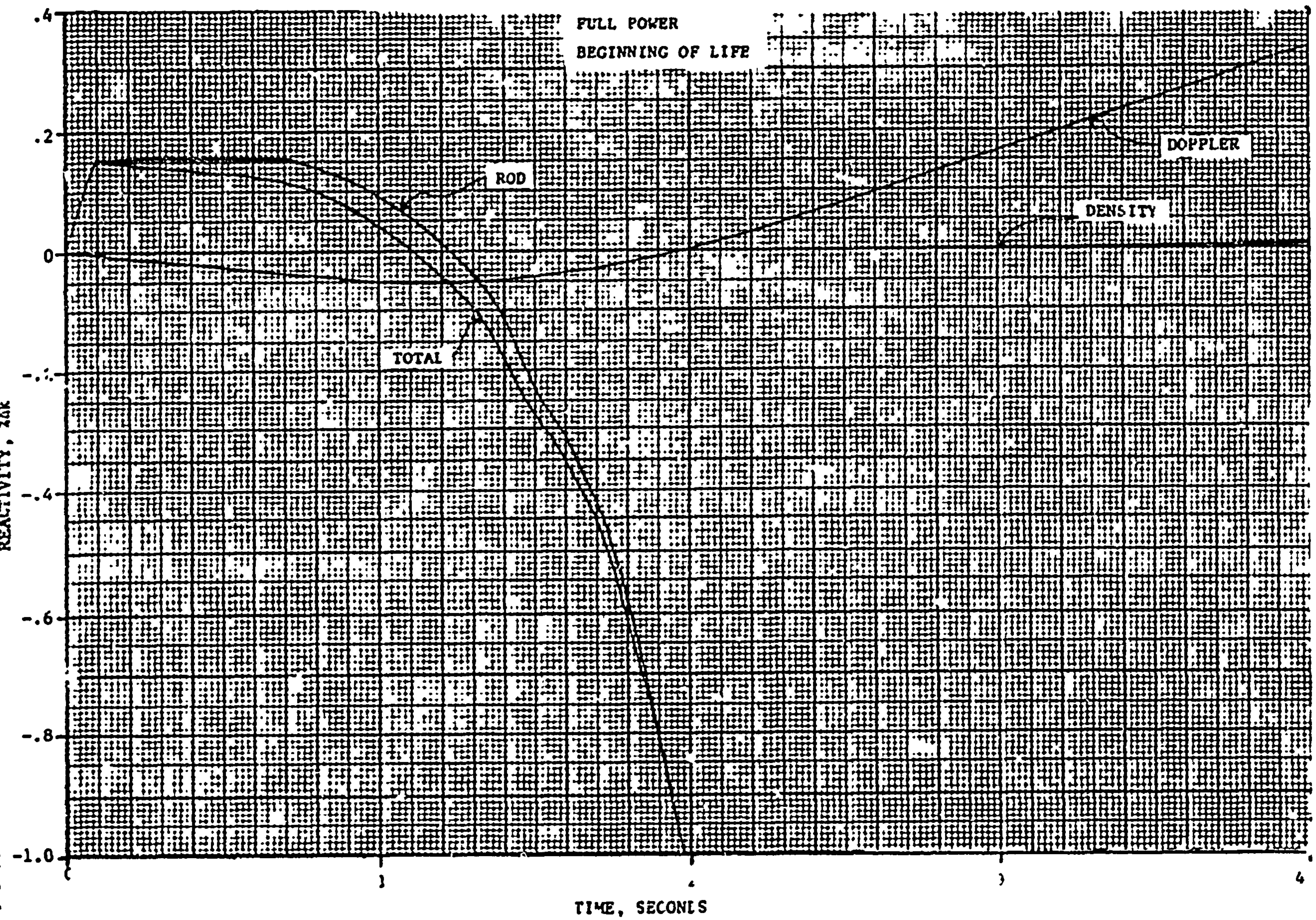

FIGURE 14.2.6-1

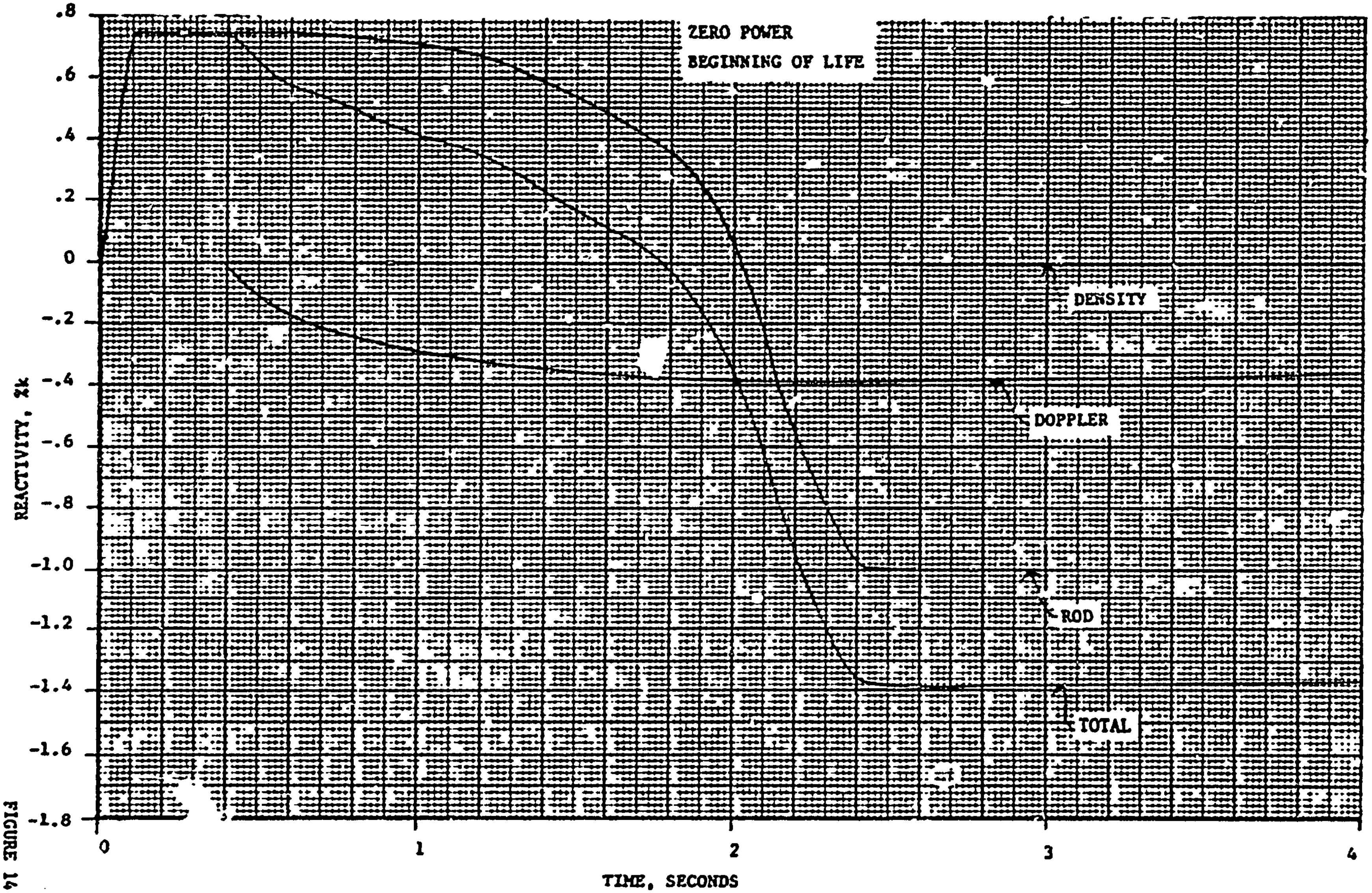

FIGURE 14.2.6-2

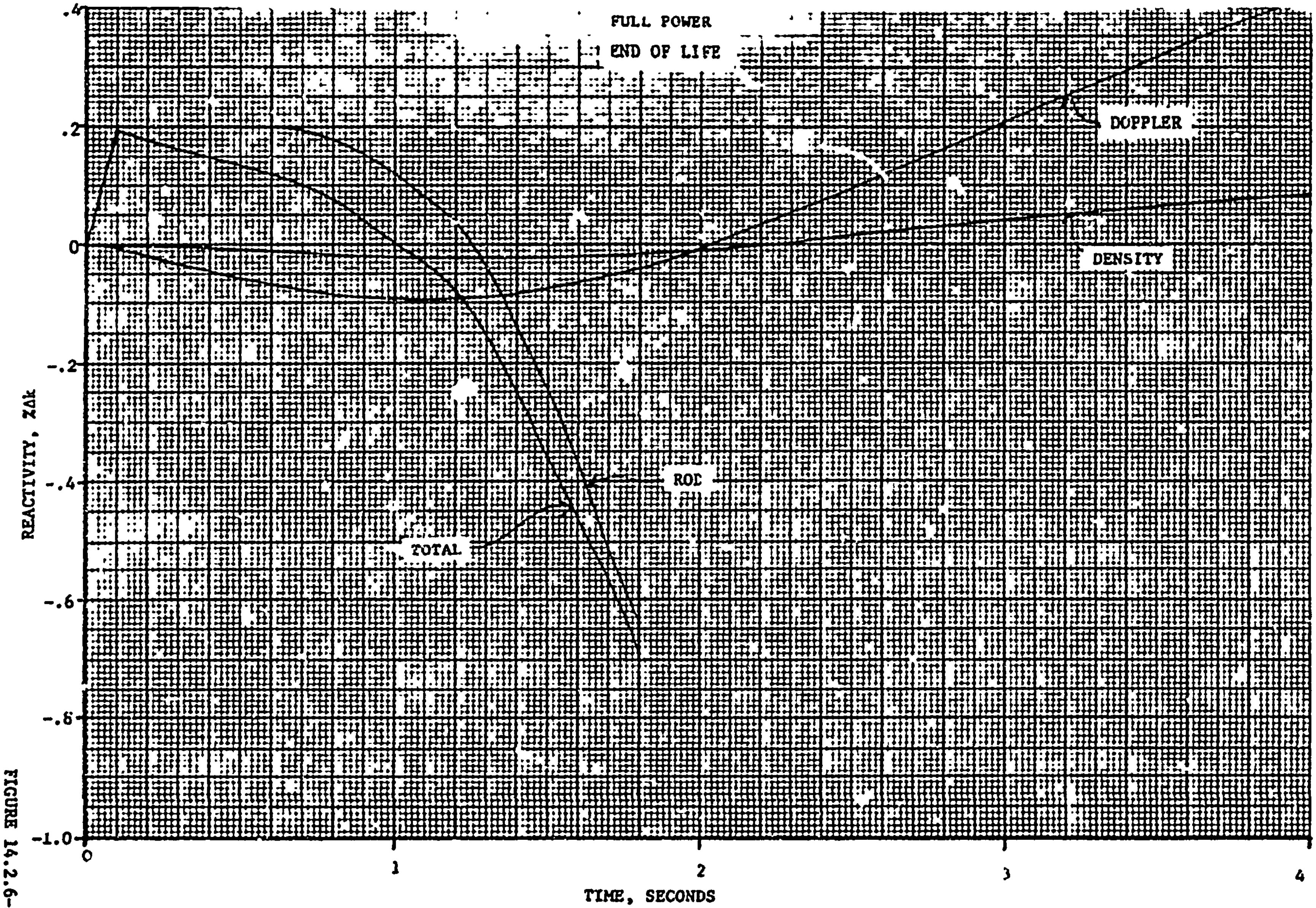

FIGURE 14.2.6-3

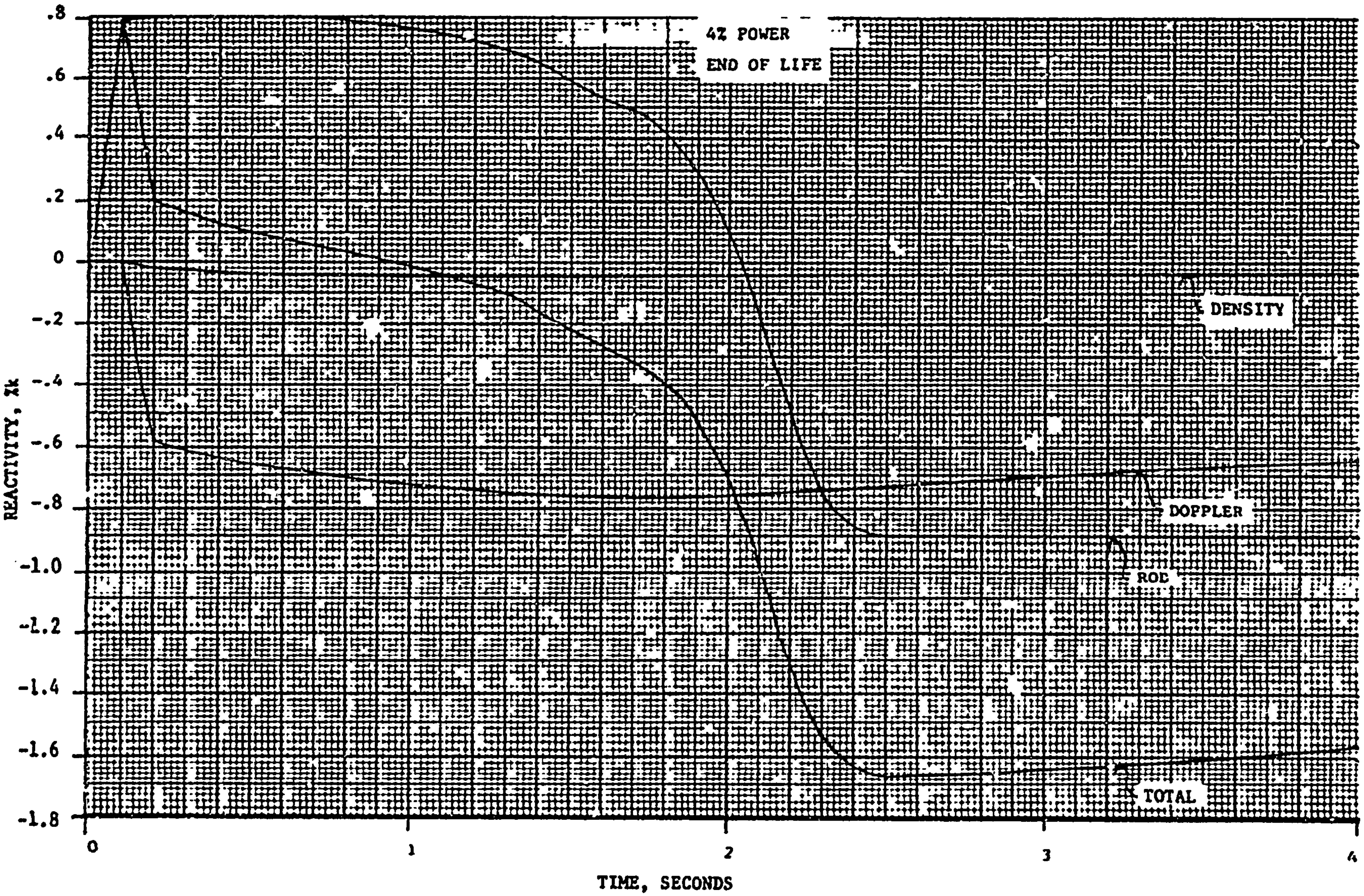

FIGURE 14.2.6-4

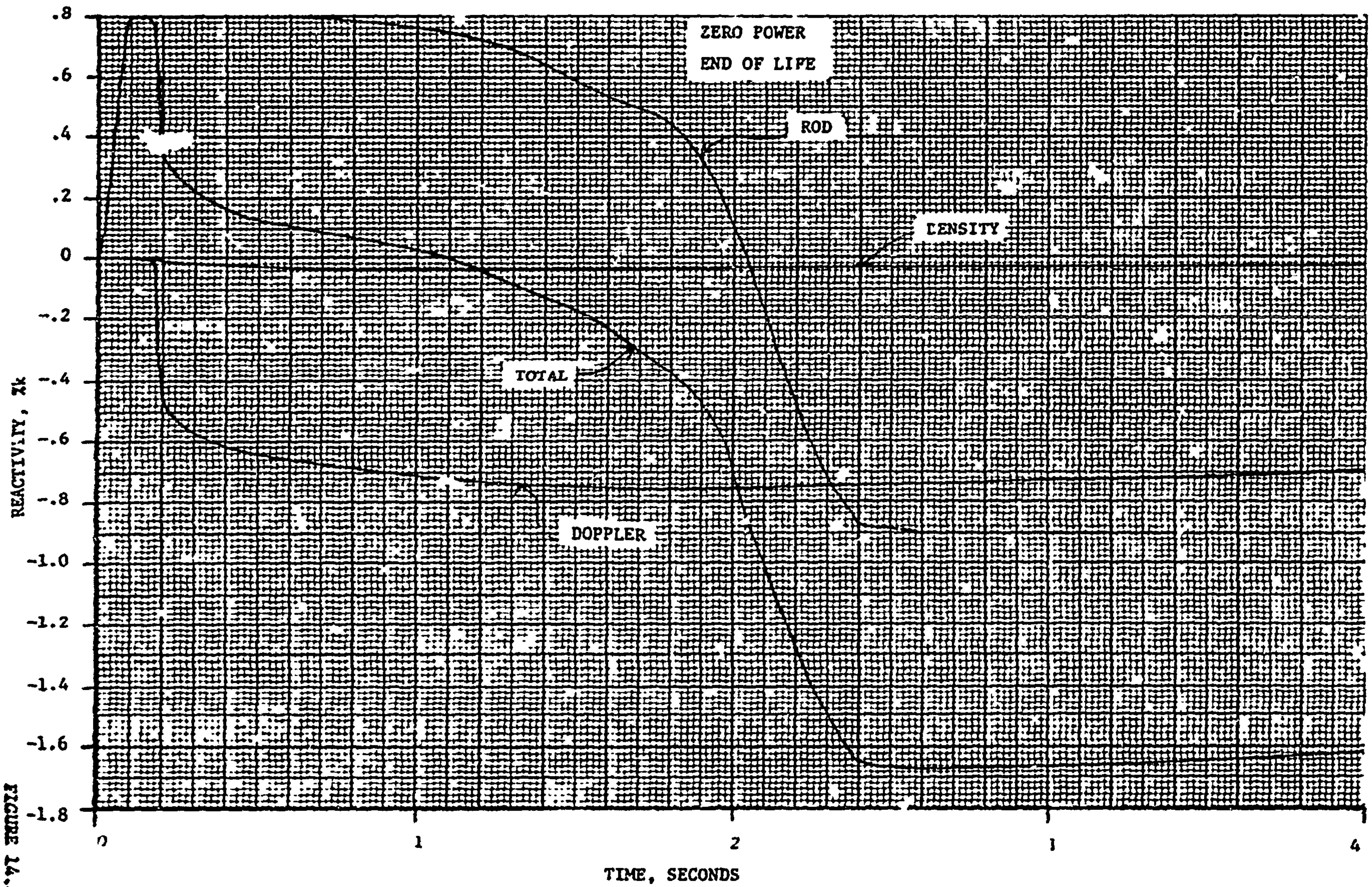

FIGURE 14.2.6-5

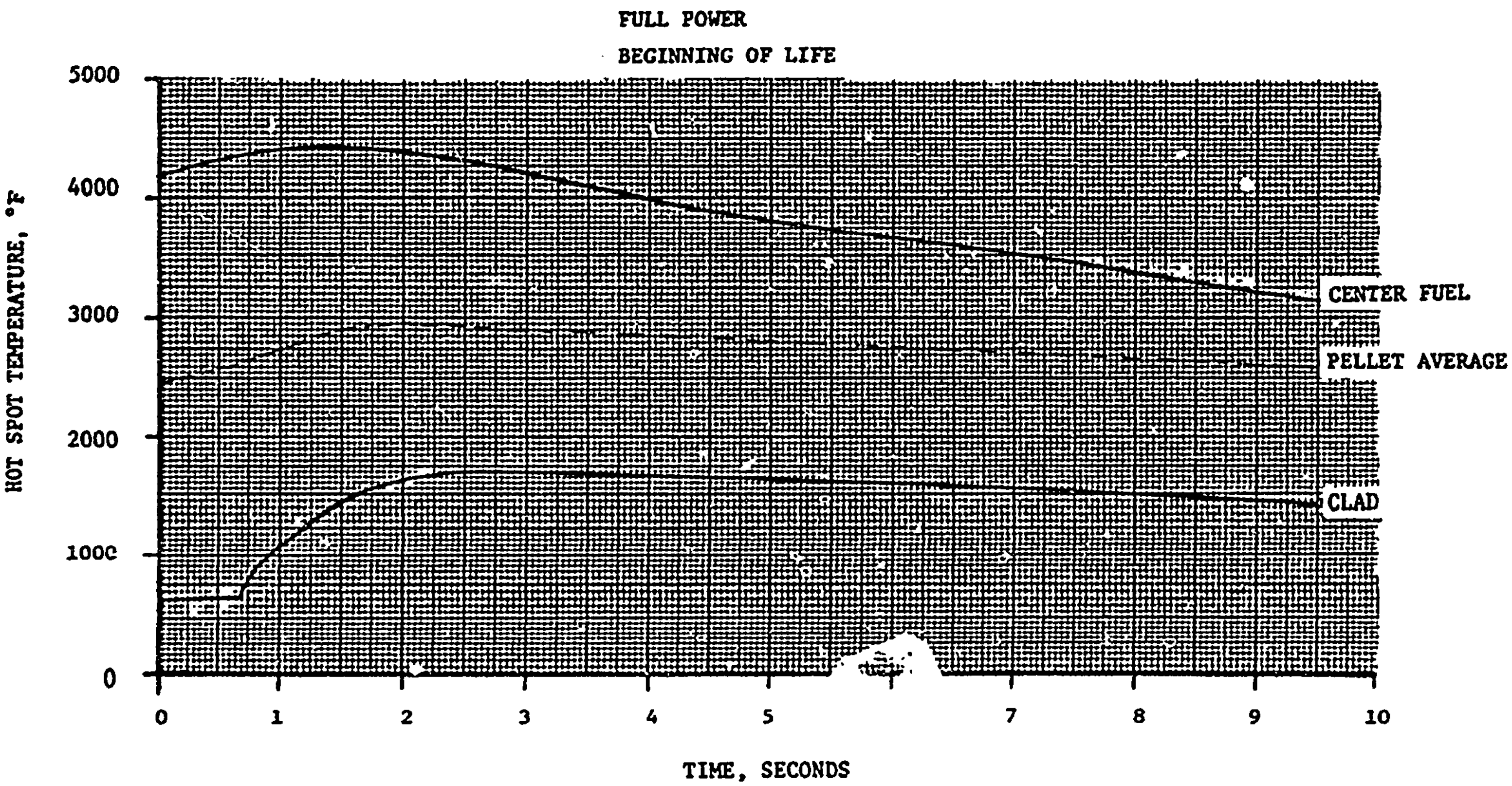

FIGURE 14.2.6-6

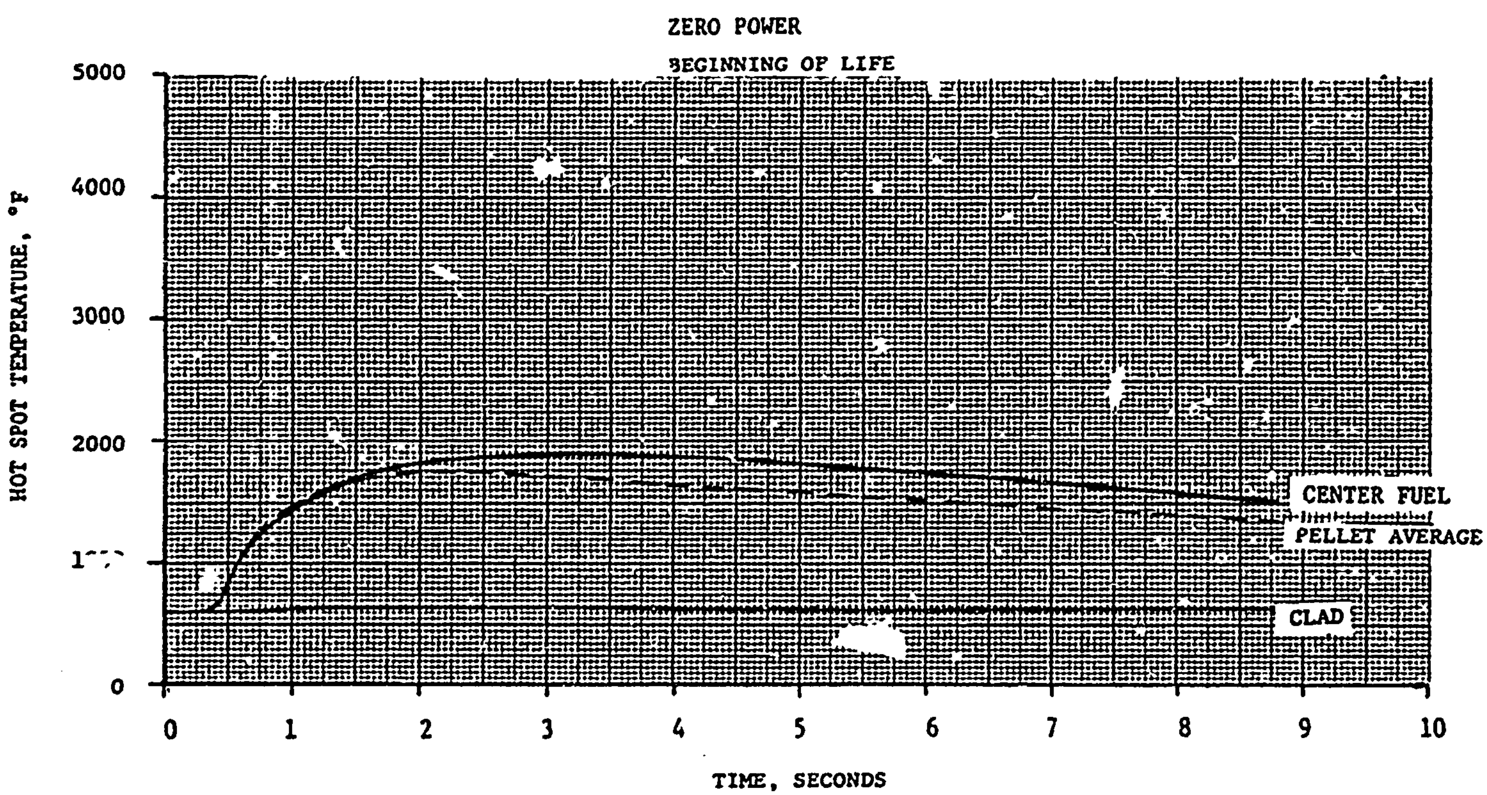

FIGURE 14.2.6-7

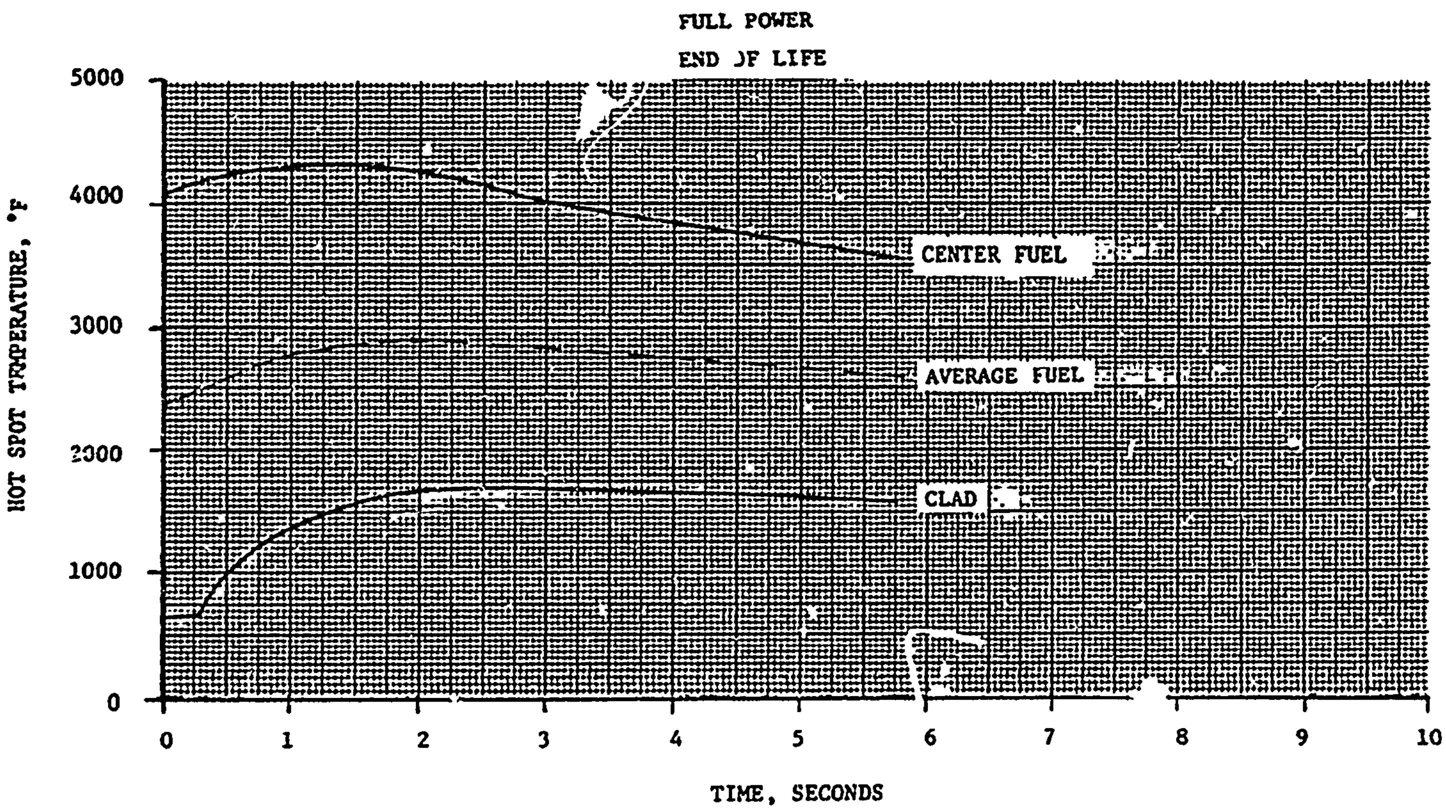

FIGURE 14.2.6-8

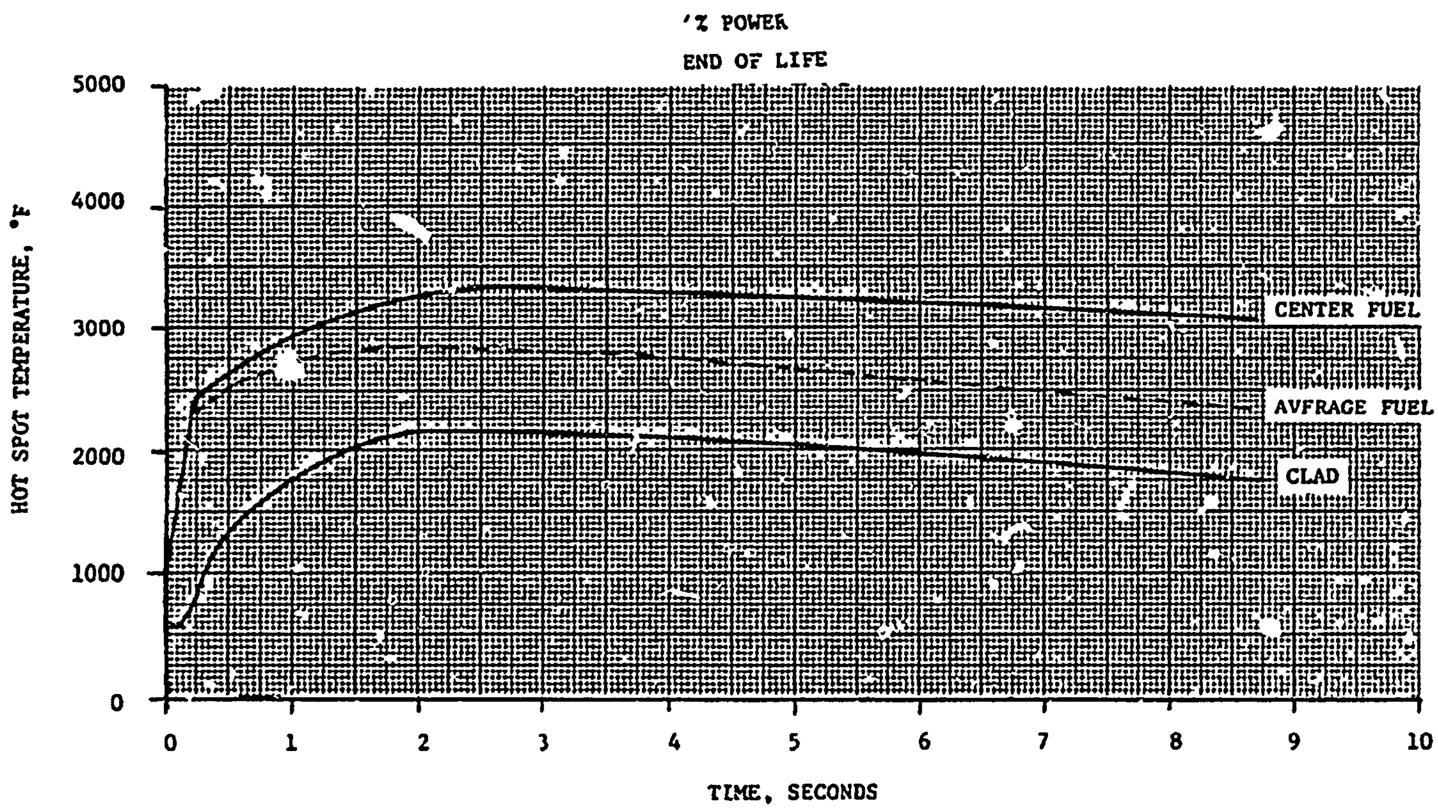

FIGURE 14.2.6-9

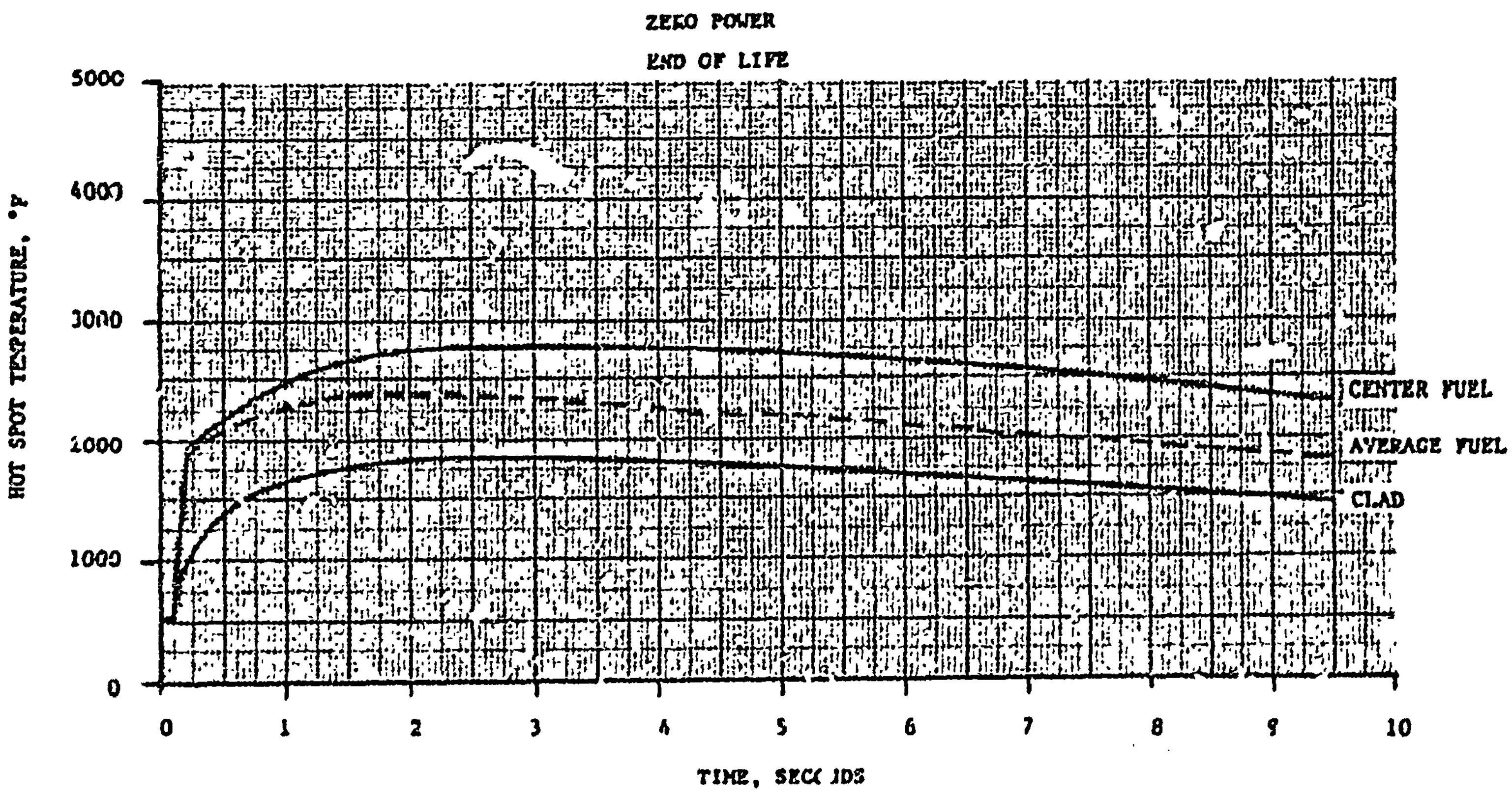

FIGURE 14.2.6-10

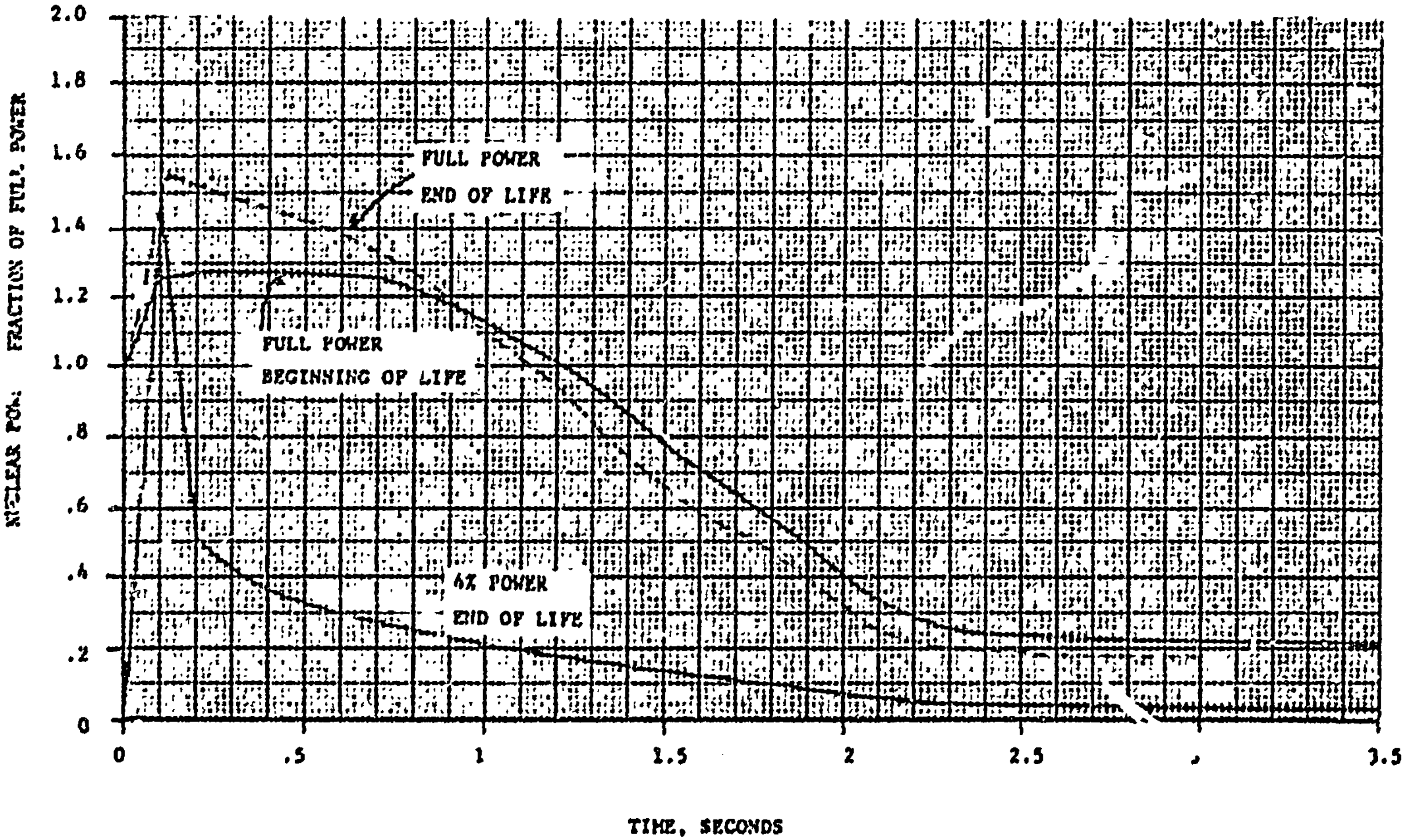

FIGURE 14.2.6-11

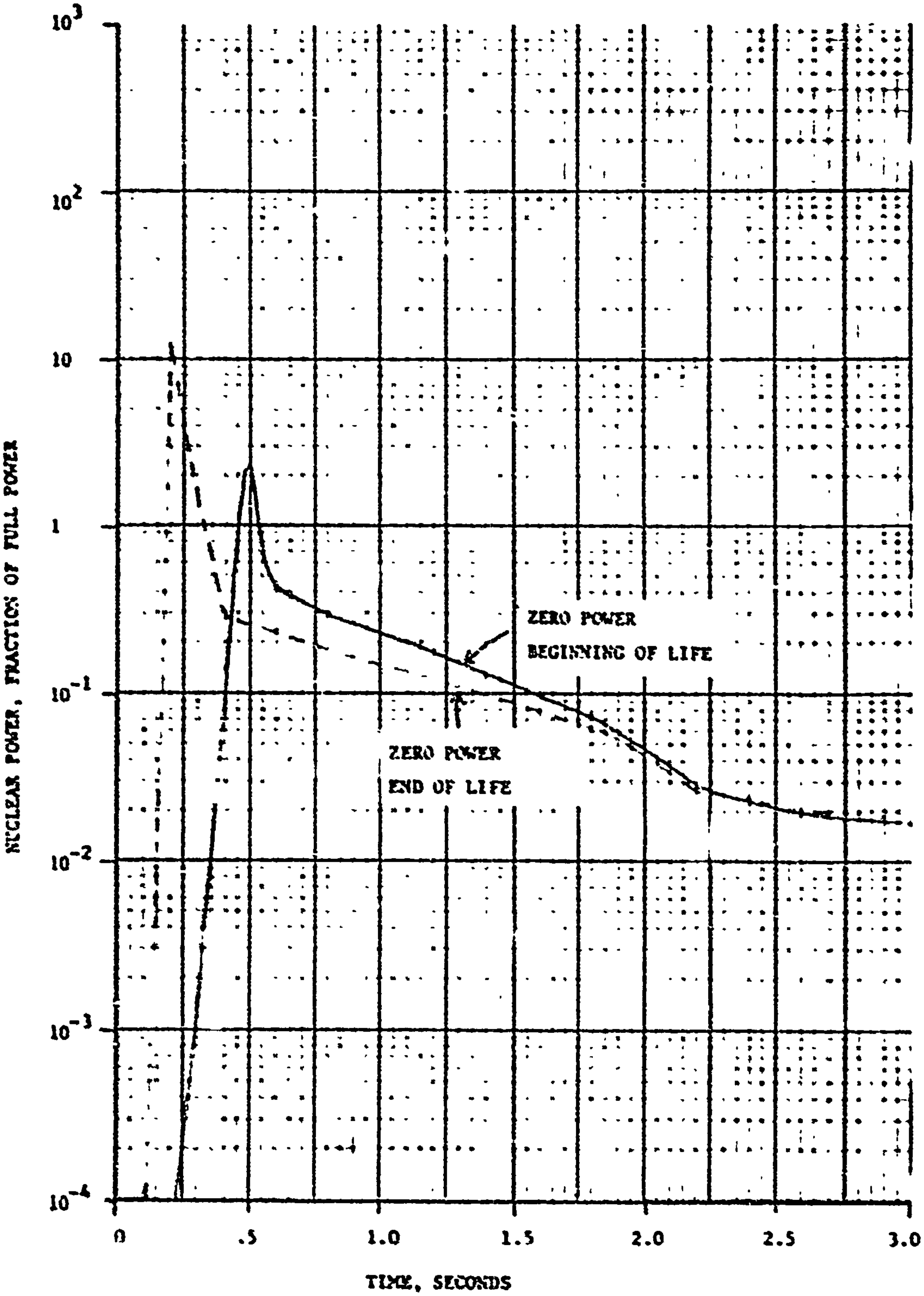

FIGURE 14.2.6-12

14.3 PRIMARY SYSTEM PIPE RUPTURE

14.3.1 GENERAL

A loss-of-coolant accident may result from a rupture of the Reactor Coolant System or of any line connected to that system up to the first closed valve. Ruptures of very small cross section will cause expulsion of coolant at a rate which can be accommodated by the charging pumps. Should such a small rupture occur, these pumps would maintain an operational level of water in the pressurizer, permitting the operator to execute an orderly shutdown. A moderate quantity of coolant containing such radioactive impurities as would normally be present in the coolant, would be released to the containment.

Should a larger break occur, resultant loss of pressure and pressurizer liquid level will cause reactor trip and initiation of safety injection. These countermeasures will limit the consequences of the accident in two ways:

a) Reactor trip and borated water injection will supplement void formation in causing rapid reduction of the nuclear power to a residual level corresponding to delayed fissions and fission product decay.

b) Injection of borated water ensures sufficient flooding of the core to prevent excessive temperatures.

The safety injection system, even when operating on emergency power, limits the cladding temperature to below the melting temperature of Zircaloy-4 and below the temperature at which gross core geometry distortion, including clad fragmentation, may be expected. In addition the total core metal-water reaction is limited to less than 1%. This is valid for reactor coolant piping ruptures up to and including the double ended rupture of a reactor coolant loop. Consequences of these ruptures are well within those described later in this section for the hypothetical accident and are therefore well within the limit of 10 CFR 100.

The following paragraphs will describe the method of analysis used and results of the calculations to demonstrate that the Safety Injection System meets the core cooling requirements for the full range of break sizes.

14.3.2 CORE THERMAL TRANSIENT

Method of Analysis

The analysis of the loss-of-coolant accident is divided into three major phases:

1) Blowdown. This calculation provides a description of the thermal and hydraulic response of the Reactor Coolant System to a rupture, through depressurization and the operation of the emergency cooling systems. The basic information concerning the dynamic environment of the reactor core is thus provided for use in reactor kinetics and core cooling analysis.

2) Reactor Kinetics. The nuclear transient is forced by the blowdown dynamics and in turn affects the blowdown. The kinetics calculation determines the energy added to the core, an essential input to the core cooling analysis.

3) Core cooling. Based on the above information, a detailed analysis of reactor core cooling is performed to determine the core clad temperature.

The division of the study into these three phases permits a careful evaluation of the importance of various assumptions on each significant aspect of the overall problem. These three phases are described in detail in the following paragraphs.

Blowdown Analysis - FLASH Code

The blowdown analysis is performed using FLASH-R[1], a digital computer code developed at Bettis and slightly modified at WAPD to better conform with commercial PWR systems. This code uses three regions, each at a different pressure, to simulate the Reactor Coolant System. Two regions are the upper and lower volumes, corresponding to the hot and cold volumes in the coolant loop, and these are connected by flow paths through the reactor core and through the intact loop piping and steam generators. The third region is the pressurizer, connected to the upper region by the surge line. Inertia and pressure losses are calculated in each connecting line, including the main coolant pump coastdown and cavitation.

The above simulation of the Reactor Coolant System permits a lumping into control volumes of relatively uniform pressure and temperature regions. In normal system operation, all of the pressure rise occurs across the reactor coolant pumps and most of the pressure drop occurs across the core. Similarly the temperature rise occurs across the core, while the temperature drop occurs across the steam generators. During blowdown, the core and the reactor coolant pumps offer most of the resistance to flow.

Heat is removed from the main coolant system by the steam generators until the coolant temperature drops below the steam-side temperature. Heat addition in the core for the current study was input in the form of heat flux as function of time, based on more detailed core studies, described in the next section.

Flow through a leak is calculated, in subcooled conditions, using Fauske's[2] model for metastable flow for short pipes, or Moody's[3] model for homogeneous equilibrium flow for long pipes. Once the leaking region reaches saturation, Moody's correlation is used for both cases. A double-ended break is represented as a "short-pipe" leak from the near region and a "long-pipe" leak from the farther region, while smaller breaks are treated as "short-pipe" leaks from the near region only.

The code contains a bubble-separation model assuming an upward velocity of two feet/second of bubbles through water. This determines the quality of flow through breaks and through the core and piping during blowdown. The actual times of uncovering of the core, however, are calculated using FLASH code results, with the assumption of complete separation of steam from water. This is a conservative measure to avoid taking credit for froth height in determining core water level. The use of the above flow correlations and bubble separation model conservatively overpredicts the mass loss through the break.

Modifications were made to the original FLASH program to account for the specific system configurations of the system: these included the singlepass rod-type core, and the location of the reactor coolant pump, the accumulator and injection pumps characteristics. A calculation was added to the FLASH program which determines the flow rate into the lower volume of the Reactor Coolant System for the accumulator. The flow rate calculation is based on the pressure difference between the Reactor Coolant System and accumulator gas pressure and the resistance of the accumulator lines. The accumulator tank gas pressure is assumed to expand isentropically to replace the injected accumulator water. The accumulator pressure, and liquid and gas inventories are continually calculated. Accumulator injection continues until the tanks are emptied. In addition, applicability of the FLASH Code was extended by the incorporation of a detailed core flooding calculation. This calculation considers the steam bubble in the core formed by steam generation when the core is reflooded. The water in the downcomer rises at a faster rate than the core water level. Thus, a static driving head is developed to drive the generated steam through the resistance of the loop piping and Reactor Coolant System components.

In summary, this phase of the analysis requires as input the reactor coolant system description and initial conditions, break size and location, energy addition in the core, and emergency core cooling system characteristics. The analysis produces as output the blowdown pressure, enthalpy, uncovery and recovery times, core flow, core pressure drop, and the conditions required to determine reactor trip. FLASH also provides the transient steam flow and steam quality in the hot channel, for use in determining the extent of DNB.

The FLASH-R Code has been used to predict the blowdown and refill transient presented for this plant. Several investigators [14] have compared this analysis to experiments and found the code conservative in two principal areas: rate of depressurization and mass of water left after blowdown.

This conservatism is generally attributed to the critical flow boundary condition. The correlation, as used in this analysis, overpredicts the mass flow through the break and thus the analysis overstates the severity of the transient. A better comparison between analysis and experiment is obtained when the break area used in the analysis is reduced to about 80% of the experimental value.

Thus the transient which is ascribed to the double ended rupture would, in fact, occur for a larger break and conversely the analysis done for a smaller rupture represents the true double-ended break.

Since the whole spectrum of break sizes less than or equal to the double-ended rupture has been analyzed, adequate protection has therefore been demonstrated even when the break flow takes on the smaller, more realistic, values.

Core Power Transient CHIC-KIN

The basic tool used for the reactor kinetics calculation is the CHIC-KIN [4] code, which has a point kinetics model and a single channel fuel and coolant description. In this study the channel was divided axially into five sections, with density in each section a function of pressure and enthalpy, plus nucleate boiling void. A nucleate boiling model for highly subcooled conditions was used, even though a large part of the coolant is saturated throughout the transient. That is, when the clad surface is hotter than saturation temperature, 10% of the heat flux goes into local boiling void, which has a mean lifetime of 0.1 second. This was done to minimize apparent void formation in order to retard reactor shutdown and yield a conservatively high energy input. Since hot channels of the core have greater than average void fraction, use of an average channel model and neglecting hot channel effects also reduces the apparent void, yielding a conservatively high energy input. In addition, coolant bypass around the core was neglected, reducing the calculated void.

Each axial fuel rod section was divided into nine radial regions for the heat transfer calculation. A high conductivity of 2.65 Btu/hr- ft -°F for the UO_2, and a high fuel-to-gap heat transfer coefficient of 2000 Btu/hr-ft^2-°F, were chosen for the kinetics calculation. These numbers give a minimum reasonable initial average fuel temperature, thus reducing heat transfer rate during shutdown. This minimizes void formation and decreases the rate of power decay.

For moderator density reactivity feedback the calculated density coefficient as a function of density for beginning of life with no control rods, which corresponds to a temperature coefficient of -0.3×10^{-4} δk/°F at nominal temperature, was uniformly reduced to correspond to zero coefficient at the initial average density. The curve yields a 1% negative reactivity with a density reduction of 25%; whereas the calculated curve would show twice this feedback for the same density change.

Doppler reactivity feedback was simulated as a function of the average fuel temperature, with a weighting factor of 1.6 used as an upper limit for the initially unrodded core to reduce the rate of power decrease during shutdown.

Six groups of delayed neutrons were used. For the total effective fraction, a conservative maximum of 0.0072 was used to slow down power decay. Average core pressure was input as a function of time from the FLASH output. For the 1/2 ft^2 breaks and the 3 ft^2 hot leg breaks, the core inlet flow as shown by the FLASH code calculations was used as input to CHIC-KIN. For all other breaks, with violent flow reversal and then near-stagnation, the core pressure drop as indicated by FLASH was assumed to be a reasonable representation of the forcing action between the two large liquid regions of the system. This pressure drop was used as input to CHIC-KIN, which calculated flow response taking into account inertia and losses at inlet and outlet and due to grids and friction in the fuel. The resulting flow transients were very close to those obtained by FLASH.

Trip would be activated in all cases by low pressurizer pressure. For the 1/2 ft^2 break case and the 3 ft^2 hot leg break, this would occur in approximately 1.25 seconds. Significant rod reactivity insertion was calculated to start in 1.95 seconds after the break, with -0.026 δk being inserted over the following one second.

For the larger breaks, trip would be similarly actuated, but because void formation is adequate for shutdown, trip was not simulated in these studies.

Core Cooling Analysis - LOCTA Code

The LOCTA-R2 transient digital computer program was developed for evaluating fuel pellet and cladding temperatures during a loss-of-coolant accident. It determines the extent of the Zircaloy-steam reaction and the magnitude of the resulting energy release in Zircaloy clad cores.

The transient heat condition equation is solved by means of finite differences, considering only heat flow in the radial direction. A lumped parameter method is used; the fuel containing three radial nodes and the cladding one radial node.

Internal heat generation can be specified as a function of time. The decay heat from any initial power level can be calculated by the code. The decay heat is based on the heat generated from

a) fission products,
b) capture products, and
c) delayed neutrons

It is assumed that the core has been irradiated for an infinite period of time.

In addition to decay heat, the code calculates the heat generated due to the Zircaloy-steam reaction. The Zr-H_2O reaction is governed by the parabolic rate equation unless there is an insufficient supply of steam available, then a "steam limited" evaluation is made. However, for the cases considered, the parabolic rate equation was used. The buildup of the Zircaloy-oxide film is calculated as a function of time, and its effect on heat transfer is considered. An isothermal clad melt is considered based on the heat of fusion of Zircaloy. Once the Zircaloy metal melts, it is retained by Zirc-oxide, and slumps against the fuel. The Zircaloy-steam reaction may continue until the oxide melts. If the oxide melts the remaining Zircaloy is assumed to fall, and 10% of this metal is assumed to react with additional water which is available in the vessel.

The code has been developed to stack axial sections and thereby describe the behavior of a full length region as a function of time. A mass and energy balance is used in evaluating the temperature rise in the steam as it flows through the core.

The initial conditions of the fuel rod are specified as a function of power. The following core conditions are also introduced as a function of time, as determined by the FLASH-R Code:

1. Mass flow rate through the core
2. Coolant quality
3. Pressure

Heat transfer coefficients during the various phases of the accident are evaluated in the following manner:

1. Nucleate boiling film coefficients on the order of 20,000 Btu/hr-ft^2-°F are used until DNB. The correlation applied during this period is:[5]

$$\Delta T_{Sat} = 1.9e^{-P/900} (q'')^{1/4}$$

2. When DNB occurs, it is assumed that the fuel rods can immediately develop a condition of stable film boiling. No credit is taken for higher transition boiling coefficients that exist prior to the establishing of a stable film on the fuel rods. The correlation used during this period is:[6]

$$h = 0.023 \frac{k_v}{D_e} \left[\frac{\rho_v D_e}{\mu_v} \left(\frac{Q_l + Q_v}{A_c} \right) \right]^{0.8} \left[\frac{C_p \mu}{k} \right]_v^{0.4}$$

3. During the time the core is uncovered (period of steam flow through the core), laminar or turbulent forced convective coefficients and radiative coefficients are evaluated.

For laminar forced convection to steam:[7,8]

$$\left(\frac{hD}{k}\right)_{iso} = 3.66$$

$$h/h_{iso} = \left[\frac{T_b}{T_w} \right]^{0.25}$$

For turbulent forced convection to steam:[9,10]

$$\frac{hD}{k} = 0.020 \left[Re_b \right]^{0.8} \left[Pr_b \right]^{0.4} \left[\frac{T_b}{T_w} \right]^{-0.5}$$

Where:

h	-	Heat transfer coefficient on outer surface of fuel rod (Btu/hr-ft^2-°F)
D_e	-	Equivalent diameter of flow channel - (ft)
ρ	-	Density (lbs/ft^3)
μ	-	Viscosity (lbs/ft-hr)
Q	-	Volumetric flow rate (ft^3/hr)
A_c	-	Area of flow channel (ft^2)

C_p - Specific heat (Btu/lb-°F)

k - Thermal conductivity (Btu/hr-ft-°F)

T - Temperature °F

P - System pressure, psia

q" - Heat flux (Btu/hr-ft^2)

Subscripts:

v - Evaluation of the property at the saturated vapor condition

l - Evaluation of the property at the saturated liquid condition

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

w - Evaluation of the property at the clad surface temperature

4. The rising clad temperature transient is turned around after the lower portion of the core has been reflooded. Conservative heat transfer coefficients of the order of 25 Btu/hr-ft^2 are calculated for this initial reflooding period using the dispersed flow theory heat transfer correlation[7] and result from the two-phase flow which is present due to entrainment. This entrainment process is initiated when a steam velocity of approximately 7 ft/sec based on work of R.F. Davis [11] is evaluated leaving the flooded region of the core.

   The analytical model used during the core reflooding phase of the accident has been compared to experimental data obtained from the FHUST[12] experimental tests in conjunction with the LOFT program. With the same geometr configuration, flow conditions, etc. as that used in the experimental studies, the Westinghouse design model predicts clad temperature turn around times to be greater than those obtained from the FHUST data, indicating the Westinghouse model is conservative.

Information generated by LOCTA R-2 as a function of time includes:

1. Fuel temperature,
2. Clad temperature,
3. Steam temperature,
4. Amount of metal-water reaction,
5. Volume of core metal, and
6. Total heat released to coolant

Small Break Analysis - SLAP Code

For small breaks up to about a 6 inch diameter hole the digital computer code, SLAP, is employed to calculate the transient depressurization of the reactor coolant system as well as to describe the mass and enthalpy flow through the break. The code considers three volumes.

1. The reactor coolant system
2. The pressurizer, and
3. The steam generators (shell side).

Fluid can flow between the pressurizer and the reactor coolant system, while heat can be transferred between the reactor coolant system and the secondary. The code uses the equations of state, continuity and energy conservation to define the condition in each volume as a function of time. Fluid flow between the pressurizer and the reactor coolant system is defined by the momentum equation. Heat is transferred between the steam generators and the reactor coolant system unless the liquid level falls below the level of the tubes. The heat transfer rate is assumed to be zero for that portion of the tubes not covered by shell-side water. Heat transfer to the steam generator decreases as the temperature difference between the primary and secondary is reduced. Heat transfer in the reverse direction is also accounted for in the code, when the reactor coolant temperature is below the steam generator temperature.

Thermodynamic conditions are initialized by designating the size of each volume as well as

1. the reactor coolant system pressure and temperature
2. the pressurizer level, and
3. the secondary pressure and level.

The initial fluid flow between the pressurizer and the reactor coolant system is zero since the pressure in these volumes is essentially equal. The initial heat transfer rate to the steam generator is equal to the operating power.

When a break occurs in the reactor coolant system, subcooled water is assumed to initially flow out the opening. The flow is defined by the correlation of Fauske[2]. He concluded that for sharp-edged orifices test data can be accurately evaluated using the incompressible flow equations for a nozzle.

Once the reactor coolant system fluid becomes a two phase mixture, a different break flow correlation is used. The new flow scheme is defined by the correlation of Moody,[3] which specifies the two phase critical discharge out the break.

The pressure decrease in the reactor coolant system causes fluid to flow from the pressurizer, resulting in a pressure decrease in the pressurizer. Reactor trip occurs when the pressurizer low pressure set point is reached. Safety injection is actuated when the pressurizer low pressure and low level set point are reached.

Injection water flow into the reactor coolant system is defined by an input table of injection flow rate as a function of system pressure. A start up delay time is also included. Injection water is allowed to flow once the safety injection signal is generated and the delay time is exceeded. The accumulators automatically discharge their fluid when the reactor coolant system pressure drops below the accumulator set point.

Before the reactor trip signal occurs, it is assumed that the heat being generated in the core is removed via the secondary. The mass and energy entering and leaving the secondary are assumed to be equal. When reactor trip occurs, isolation valves are assumed to close, preventing secondary flow to or from the steam generator. Heat from decay, hot internals, and the vessel enters the reactor coolant system fluid. The pressure in the secondary increases and heat enters from the hotter reactor coolant system. Secondary steam discharges from the steam generators when safety valve set pressure is reached. Emergency feedwater flow as a function of time is specified by input tables. Steam flow as a function of dump valve flow area is specified to simulate the operation of the power operated dump valves.

The code follows the pressure and mass in each volume as a function of time.

Results

The capability of the Emergency Core Cooling System to meet the design criterion was analyzed for the following range of break sizes:

1. Large breaks, both hot and cold leg (FLASH)
   a) Double ended severance of the Reactor Coolant Pipe
   b) 6 ft^2
   c) 3 ft^2, and
   d) .5 ft^2

2. Small breaks, cold leg (SLAP)
   a) 6 inch
   b) 4 inch
   c) 3 inch
   d) 2 inch
   e) 1 inch

For all of the above cold leg breaks the clad temperature transient is presented for the case where the contents of one accumulator tank was assumed spilled through the break. For the hot leg breaks all of the accumulators empty into the reactor vessel. In addition the temperature transient for the double ended cold leg break with only 2 or 4 accumulators operating is also presented. Full flow from the safety injection pumps was assumed at 25 seconds after the SIS actuation signal.

Results - Large Area Ruptures

Blowdown and Refill

Figures 14.3.2-1 to 14.3.2-9 are plots of the water volume in the reactor vessel for the large area ruptures. During blowdown the volumes plotted represent an equivalent liquid volume which would result if the liquid and gas phases were completely separated. No credit is taken for an increased froth height due to voids created by boiling in the core. The volume of liquid remaining in the vessel after blowdown is used as a starting point to predict the liquid level during the refilling phase. It should be noted here that the FLASH code conservatively underpredicts this quantity of water remaining in the vessel at the end of blowdown, when compared to experimental data (LOFT semiscale tests, etc.), so that this conservatism is carried throughout the refill phase of the predicted water levels.

Several factors have been considered in the analysis that could adversely affect the flow of emergency cooling water to the core. These are:

a. Accumulator water carried out of the break, or to other parts of the system during blowdown.

b. Steam bubble formation when accumulator water refloods the core.

c. The affect of the nitrogen gas entering the vessel.

The method of determining the affect of these factors is discussed in the following paragraphs.

<u>Accumulator By Pass</u>

The flow from each accumulator enters the cold leg pipe between the outlet of the reactor coolant pumps and the cold leg nozzles. The maximum accumulator flow rate is 6600 lbs/sec. This occurs for the double ended break shortly after the beginning of injection. This flow rate is approximately 17.5% of the steady state flow rate of 37,800 lbs/sec for normal plant operation, and thereby there is no possibility of choking the downcomer and backing the flow to other parts of the system.

Flow into the inlet of the vessel is also enhanced by the reactor coolant pump, which would be coasting down during the transient and would tend to force coolant in the direction of the reactor. Further, a characteristic of the reactor coolant pumps prevent back-flow through the pumps under the injection condition. Each pump has a diffuser which effectively serves as a weir to impede back-flow through the pump. The weir effect of the suction/diffuser section of the pump can be seen in the schematic Figure 14.3.2-10a. The discharge pipe has to be full of water before the weir can be over-topped. The water required to fill the discharge piping between the reactor coolant pipe and the vessel is accounted for in the calculations.

The possibility of accumulator water being carried out with the blowdown was also considered.

The FLASH computer code is used to predict the rate of mass discharge through the break in the loss-of-coolant accident analysis evaluation. A very conservative bubble rise correlation is used to predict the bubble entrainment and corresponding froth level in the annulus of the reactor vessel. This results in an extension in the duration of two phase blowdown. Comparison with LOFT tests has indicated that FLASH overpredicts the amount of water lost through the break. No attempt has been made to modify the bubble rise correlation or to perform a momentum analysis to predict a more realistic blowdown. Such a modification would predict more water remaining in the vessel after blowdown, and a faster reflooding of the core.

In the FLASH analysis of the transient, the accumulators are injecting during blowdown and loss of accumulator water because of the upward velocity and resulting entrainment in the annulus is considered. When the accumulators begin to inject (600 psi) the quiet water level has dropped below the bottom of the core. In the FLASH calculation a conservative froth level remains at the level of the nozzles, more than 17 feet above the quiet level, and a two phase entrained blowdown is continuing. Note that no credit is taken for the froth level in the core cooling analysis.

Only a portion of the accumulator injection takes place during blowdown. For the double-ended break, accumulator injection begins at 7.0 seconds and blowdown is completed at 12 seconds. The entire contents of the accumulator are not injected until 35 seconds. During blowdown only 26,400 lbs. of the total accumulator mass of 126,000 lbs. is injected (3 of 4 accumulators considered). Therefore, only about 20% of the accumulator injection is subject to loss through the break during blowdown, and this is accounted for in the analysis of the transient.

In summary the correlations used in the FLASH code to predict the mass loss through the break, from both the mass initially in the reactor coolant system and the mass added to the system by the accumulators, conservatively account for the water being carried to the break.

Steam Bubble

When the core is reflooded by the accumulators special consideration is given to steam generation in the core which could retard the reflooding process. Steam will be generated around the hot fuel rods, causing a pressure build-up in the core region. This steam must be vented from the system through the break. The flow paths are illustrated in Figures 14.3.2-10a, 10b. The worst break location for this is a cold leg break, where the steam must flow through the reactor coolant pipes, steam generator, and reactor coolant pump to escape. The sketches in Figures 14.3.2-10b and 10c show the path the steam must follow for the cold leg break. (Note that 14.3.2-10c shows one intact and one broken loop for the purpose of illustration, while actually there are three intact loops.)

There are two paths available for the steam to flow to the break. The first path is directly to the break through the broken loop. The other path is through the three intact loops, back into the inlet annulus, and finally to the break through the inlet nozzle in the broken leg. Because of the pressure drop due to steam flow between the core and inlet annulus, the liquid level in the downcomer annulus will rise at a faster rate than the core level. This will continue until the water head between the downcomer and core level equals the pressure drop of steam flowing through the loops to the break.

The relationship between this downcomer head and corresponding steam flow rate through the loops is shown in Figure 14.3.2-10. The resistance to steam flow used for this curve is based on the resistance of the loop piping, steam generators and pumps (assuming the expected condition of empty loop seals) and a saturation pressure of 62 psia at the completion of blowdown, which corresponds to the containment pressure.

The model used for the refill calculation allows for this resistance to the steam flow, and the core liquid level predicted by the revised FLASH code is consistent with the necessary pressure differential between downcomer and core.

The height of downcomer above the calculated liquid level is available as an additional head of water to cause steam flow through the core. This available head would permit steam flow in excess of the calculated flow, or an additional back-pressure build-up due to water filled loop seals (as discussed below) without any loss of safety injection water.

The available downcomer head during the refilling of the core for the double-ended cold leg break is shown in Fig. 14.3.2-11. Comparison of this available head to the head needed for a steam flow sufficient to cool the core shows the considerable margin in potential steam driving head. Only 0.7 feet of head is required to drive the steam to the cold leg break location, while 16.4 feet is available. This allows considerable margins for variations in the pressure drop calculation such as the effect of entrainment and decreases in the system saturation pressure during containment cooldown.

In the unlikely event that all recirculation loop seals (pipe between steam generator and reactor coolant pump) were filled at the end of blowdown, the escape paths for the steam from the core would be temporarily blocked causing a rapid pressure buildup in the core. However, the available downcomer head far exceeds the head needed to blow the liquid out of the loop seals (8.5 ft) as shown in Figure 14.3.2-11. Filled loop seals would, therefore, result in the rapid filling of the downcomer until the head in the downcomer has reached 8.5 feet. This would be followed by a back flow of the water from the downcomer into the core until downcomer head and steam pressure are equalized. No accumulator water would be lost and the delay in covering the first 2 feet of the core would be insignificant.

It is concluded, therefore, that the downcomer head accounted for in the calculation of liquid level in the core is sufficient to drive the calculated steam flow to the cold leg break. In the event of a steam flow lower than that calculated, the liquid level in the core would rise at a faster rate, thereby recovering the core with liquid sooner than predicted, while a steam flow higher than the calculated flow is possible with the available head in the downcomer. In this event the liquid level in the core would rise at a slower rate than that predicted, however, the higher steam flow would increase the margin in the core cooling capacity.

For the hot leg breaks analyzed, when the core is reflooded by accumulator water, steam generation in the core does not retard the refill transient because the steam does not have to travel through the loops to reach the break.

Nitrogen Interference

Nitrogen gas enters the system after the accumulator injection is complete and the core maximum temperatures have already been greatly reduced. For a cold leg break the accumulator gas actually helps prevent the steam binding situation, i.e., the gas pressure at the reactor vessel inlet nozzle would tend to retard the rise of injection water in the downcomer annulus. No credit is taken for this in the steam bubble calculations.

The gas entering the system will, in part, escape through the break by venting around the downcomer annulus to the break location and, in part, occupy the high dead spaces in the reactor coolant system. The latter gas volumes, however, cannot impede the flow of core cooling water.

For the hot leg break some of the gas could bubble into the coolant around the bottom of the core barrel, but the majority of the gas would vent in the reverse flow direction to the break. The nitrogen thus dissolved in the coolant has little effect on the cooling in the core.

Core Power Transient During Blowdown

The core power transients calculated are shown in Figures 14.3.2-12 and 14.3.2-13. For the 0.5 ft^2 breaks, and the 3 ft^2 hot leg break the initial subcooled decompression does not form enough void to shutdown the core. As pressure continues to drop, however, the power drops until it is under 80% when trip becomes effective for the 0.5 ft^2 cold leg break and the 3 ft^2 hot leg break. For the 0.5 ft^2 hot leg break, no significant power decrease occurs before trip becomes effective.

For the larger breaks, the faster subcooled blowdown and subsequent rapid continued depressurization introduce voids much more rapidly and extensively than in the case of a small break. Backflow through the core also forces a saturated steam-water mixture from the reactor outlet plenum down into the core, adding to the voiding. The result is that for the 3 ft^2 cold leg break, 6 ft^2 and double-ended breaks studied, the reactor shuts down immediately.

For these cases, a "standardized" decay heat plus delayed neutron curve was used as a minimum power level in the thermal analysis, even though this power is significantly higher than the power actually calculated with the conservative assumptions listed.

Core Thermal Analysis Results

The core thermal analysis was determined using the blowdown and recovery data and the core power transients which were described in the previous sections.

Figures 14.3.2-14 through 14.3.2-22 present a plot of core pressure and core flow and the calculated heat transfer coefficient used for all breaks. Figures 14.3.2-23 through 14.3.2-26 present the maximum clad temperature transient for the design and adiabatic after blowdown cases associated with the large cold leg breaks. Figures 14.3.2-27 and 14.3.2-28 present the design peak clad temperature transient for the large hot leg breaks. The zirconium-metal water reaction was computed to be less than 1% in all cases and is an insignificant factor in the containment transient. The following table summarizes the important results of the transient.

Break Size, Cold Leg	Maximum Clad Temperature, °F	Total % Clad Burst
Double-ended	2120	54
6 ft^2	1910	42
3 ft^2	1660	28
0.5 ft^2	1940	56

Only the double-ended cold leg break was analyzed for the case where one accumulator is isolated from the system, and the resulting injection of two of the four accumulators to the reactor vessel. No credit was taken for the accumulator line isolation valve opening on the safety injection signal. The maximum clad temperature calculated was 2550°F. (Figure 14.3.2-23).

The double-ended cold leg break is the worst break size for this injection case. This is apparent from the above table, and the adiabatic heat-up curves presented in Figures 14.3.2-23 to 14.3.2-26. For the three accumulator cases a peak clad temperature was calculated for the double-ended break of about 180°F greater than the smaller breaks analyzed. In addition the adiabatic heat-up rate following blowdown is faster for the double-ended break. This is because blowdown ends sooner for the double-ended break, and therefore the decay heat generation during the period the core is uncovered is greater.

Therefore, clad temperatures considerably below 2550°F are expected for smaller break sizes.

It is concluded that the safety injection system with one accumulator isolated thus limits the peak clad temperature to well below the melting point of Zircaloy-4.

Fragmentation of Clad as a Result of Quenching

Westinghouse has partially completed experimental quenching tests of production type Zircaloy-4 tubing. This experimental work was initiated to investigate the possibility of clad fragmentation during the core reflooding phase of the loss-of-coolant accident.

Tests to date include the heating of production type Zircaloy-4 tubing to initial temperatures as high as 2800°F with approximately 2.5% Zr-oxide formation. No rod shattering or any observable change in the geometric appearance of the rod occurred. It should be noted that the temperatures (up to 2800°F) and Zr-H_2O reaction (2.5%) obtained prior to quenching provide ample margin to the design conditions (3 accumulators injecting for cold leg breaks and 4 accumulators injecting for hot leg breaks) during a loss-of-coolant accident which exist for this plant.

Eutectic Formations

Westinghouse has also performed experimental studies to investigate the metallurgical reaction between the Zircaloy fuel rod and the Inconel grid under loss of coolant conditions. These tests were run under more severe conditions than those expected in this reactor during a loss of coolant accident. Tests results indicate that operation at peak fuel cladding temperatures for a period of several minutes is required before you need to consider this phenomena in the evaluation of this accident. For all the design loss of coolant accidents (3 of 4 accumulators for cold leg breaks and 4 accumulators for hot leg breaks) this reactor is operating in the temperature range of concern for eutectic formations for less than one minute.

Clad Perforation and Deformation

During a loss-of-coolant accident, the clad temperature may get sufficiently high (1200-2200°F) so that bursting or swelling of the clad would occur by virtue of the internal gas pressure and the significant reduction of clad strength. Clad bursting or swelling is of concern due to the possibility of releasing large quantities of fuel and/or blocking the flow channel sufficiently so that core cooling would be insufficient to prevent fuel rod melting.

Clad burst failures are predicted to occur at clad temperatures of approximately 1200 to 1300°F. The fuel rods which do not burst due to insufficient internal pressure exhibit maximum clad temperatures of approximately 1350°F. At these temperatures, annealing of the Zircaloy-4 is not expected to occur. Thus, only a minor amount of clad swelling is expected for these rods which do not burst; the flow area blocked by swelling for these rods which do not burst; the flow area blocked by swelling being less than that caused by fuel rod perforation. Thus, it is concluded that the limiting mode of failure which is to be expected under loss of coolant accident conditions is that of clad perforations rather than that due to clad swelling or ballooning.

Clad Perforation Model

Calculations are performed to determine the number of fuel rods that might fail during the thermal transient following a rupture in the primary cooling systems. In this analysis, fuel rods are considered to fail when the differences between the internal and external pressure exceeds the rod burst pressure.

The calculations are performed in the following manner:

A. The maximum clad temperature vs time transients on the rods in the core are calculated assuming no change in the core geometry.

B. For each radial region of the core, a burst pressure vs time curve is obtained by combining the temperature transient curve and the burst pressure vs temperature curve.

C. The hot fuel volumes and the hot clad volumes obtained in the fuel rod transient study are used to determine the hot void volume in the fuel rod as a function of time. The internal gas pressure distribution as a function of time is calculated considering the actual fuel rod power histories at the end of the equilibrium cycle when the maximum internal pressures are expected to exist.

D. All rods are assumed to fail if at any time during the transient the difference between internal gas pressure and external system pressure exceeds the burst pressure of the clad.

E. An evaluation is then performed to determine the rod with the lowest power rating (kw/ft) which fails. All rods above this power level then are considered as exhibiting rod bursting.

Results of the rod burst evaluation is presented in the preceding table. Preliminary results from experimental rod burst tests indicate that the burst pressure versus clad temperature data is as much as 100 to 200% higher than the design curve used in the analytical analysis.

Flow Blockage Analysis

Westinghouse has already completed experimental burst tests on unirradiated, unhydrided, Zircaloy-4 cladding.

Results of this program reveal that the geometry of the rupture (which has been found to be quite consistent) exhibits a small longitudinal split in the cladding with the length of approximately 1/2 inch maximum, and a width approximately 1/32 to 3/16 inch. This would result in a flow area blockage of 10-15% for a single rod. Even if all four rods would burst so as to block flow in the hot channel, 50% of the channel would remain open.

Present Westinghouse reactor safety injection systems supply a cold flooding rate of approximately 10 in/sec to the bottom of the core. Using a conservative flooding rate of only 6 in/sec an evaluation was performed for typical accident conditions which considered a local flow blockage at the core hot spot of 50%. This evaluation utilized a three dimensional analysis for predicting the flow redistribution in the open lattice core. The local mass flow at the location of the blockage was evaluated to be 40% of the nominal value. The effect of this mass flow reduction was to decrease the heat transfer coefficient obtained for entrained liquid immediately downstream of the blockage, the location where the minimum coefficient occurs, and thus, just extend the time at which the clad temperature transient begins to decrease at a continuous rate. The heat transfer coefficients for no flow blockage and for 50% flow blockage from the time of initiation of entrainment is given in the following table.

Time After Initiation of Entrainment (sec)	h, Btu/hr-ft^2-°F (No Blockage)	h, Btu/hr-ft^2-°F (50% Blockage)
1	25	15
2	28	16
3	30	17
4	32	18
5	36	20
6	40	22
7	43	23
8	47	25

The clad temperature transient for the above two cases of no blockage and 50% blockage is presented in Figure 14.3.2-28a. From this figure it can be seen that the peak clad temperature is only increased approximately 50°F and the time to reach this peak temperature is delayed approximately 5 seconds.

Thus, it is concluded that fuel rod failure due to clad bursting or swelling during a loss of coolant accident has a negligible effect on the ability of the emergency core cooling system to effectively cool the core.

Results - Small Breaks

The analysis carried out and presented in the previous section demonstrated the adequacy of the accumulators to terminate core exposure and limit the temperature rise of the core for large area ruptures. For smaller breaks the discharge of fluid through the hole is less severe and for small enough breaks the high head safety injection pump is capable of maintaining flooding of the core hot spot for the entire blowdown. Where the hot spot remains covered no clad damage is expected.

Ruptures of very small cross sections (up to about the equivalent of a 3/4" connecting pipe) will cause expulsion of coolant at a rate which can be accomodated by two of the three charging pumps well before the core is uncovered. Since instrument taps and sample connections are less than 3/4" diameter, protection from rupture of this line is afforded by the charging pumps.

For smaller leaks, (up to about 1/2 inch) these pumps would maintain an operational level of water in the pressurizer, permitting the operator to execute an orderly shutdown.

Should a larger break occur, resultant loss of pressure and pressurizer liquid level will cause reactor trip and initiation of safety injection supplementing the charging flow.

Using the SLAP code, break sizes of 1, 2, 3, 4, and 6 inch equivalent diameters were analyzed. Three combinations of safety injection pump availability were considered. These were:

1. Full system; three pumps delivering through the four injection lines.

2. Single Failure; two pumps delivering through three lines, one injection line isolation valve failed to open (One injection line is assumed to spill to simulate a break near the injection location.)

3. Single failure and broken safety injection line; one pump delivering to intact header and second pump delivering to second header with one line broken.

(This is a special case where the loss of coolant is caused by a break in the safety injection line between the reactor coolant pipe and the check valve in the injection line. In this case not only is the flow lost through the one line, but the effective cut-in pressure for delivery is reduced until the pressure loss due to flow in the spilling line equals the Reactor Coolant back pressure. Since the injection lines are only 2 inches in diameter, this case applies for break sizes 2 inches and smaller.)

The delivery curves for these cases are presented in Figure 14.3.2-29.

For cases 2 and 3 the pumps are operated on diesel power. For the third case it should be noted that effective delivery is provided by only the pump aligned to the header without the broken line. The other pump delivers its contents through the broken line until the reactor coolant pressure has reduced to less than 450 psi. Thus, for the case, protection is afforded by one pump.

The Reactor Coolant System pressure and volume for these cases are presented in Figure 14.3.2-30 through 14.3.2-34 and 14.3.2-35 through 14.3.2-39, respectively.

As indicated on the curves the hot spot remains flooded for all breaks up to and including a 4" diameter hole. It should be noted that the volumes presented are the quiet levels. No credit is taken for the actual froth level that would occur due to void formations in the core.

The existence of a water filled loop seal was considered in the transient. That is, the plot of the water level in the core takes into account the depression of the core water level necessary to maintain a full downcomer and loop seal. This depicts a break for the worst break location, i.e., a cold leg break between the pump outlet and the reactor vessel inlet.

Therefore, from the results of analyses it is concluded that a break size of about 4 inches defines the upper limit of protection afforded by two high head safety injection pumps.

For a 6 inch break the hot spot is uncovered for a short period of time for the minimum injection case, but remains covered for the full injection case.

A core thermal analysis was performed for the 6 inch break with peak clad temperatures being evaluated for the following cases:

1. DNB occurring at 0.5 seconds after the break.

2. No DNB occurs.

The analysis utilized the benefit of the froth level core volume transient shown in Figure 14.3.2-39A for the minimum safety injection case. Assuming DNB occurs (Case 1) the peak clad temperature of 1550°F occurs at the hot spot (core mid-plane). For Case 2, since no DNB occurs and the hot spot of the core never becomes uncovered, the hot spot clad temperature gradually decreases from its initial steady state value of approximately 715°F. However, the upper portion of the hot rod obtains a slightly higher maximum clad temperature of approximately 1010°F due to the portion of the rod being uncovered for approximately 90 seconds. During this uncovery period, the upper part of the hot rod is cooled by the steam generated in the covered portion of that core.

In the previous cases no credit was taken for operator action. Since time is available in a small break accident, it is expected that the operator will take control of the accident. By dumping steam through the steam generator relief valves the Reactor Coolant System can be depressurized.

This depressurization of the Reactor Coolant System would result in less discharge through the break and greater addition from the Safety Injection System. The net result is a great capability to maintain core flooding.

The action the operator would perform for this accident would be very similar to a normal cooldown. In a blackout situation the atmospheric dump valves are used, and when power is available the condenser dump would be used.

Figure 14.3.2-40 presents the volume transient for the several breaks considering only atmospheric steam dump and the minimum safety injection pump case. The pressure transients for these cases is presented in Figure 14.3.2-41. Thus hot spot flooding is maintained up to a six inch break.

Safety Injection Signal Actuation Capability

The time to initiate the safety injection signal from either high containment pressure or a coincidence of low pressurizer level and pressure is presented in Figure 14.3.2-42 as a function of break size. For reference, the time the core mid plane would become uncovered with no safety injection is also included in the figure. This curve illustrates the substantial margin which exists between the time at which initiation occurs and the time at which a signal is required. Note that the passive accumulator system provides the initial core protection with no dependence on an actuating signal for all breaks of greater than 6 inch equivalent diameter.

Conclusion

For breaks up to and including the double-ended severance of a reactor coolant pipe, the Safety Injection System with partial effectiveness will prevent clad melting and assure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved. The final core cooling systems design meets the core cooling criteria with substantial margin for all cases. It was also concluded from this study that the high head pumps are capable of maintaining core flooding for all break sizes up to approximately the 4 inch connecting pipe. For larger breaks the needed protection is supplied by the accumulators.

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14. K. V. Moore, R. P. Rose, Transaction of ANS, Volume 9, No. 2, pg. 559 1966.

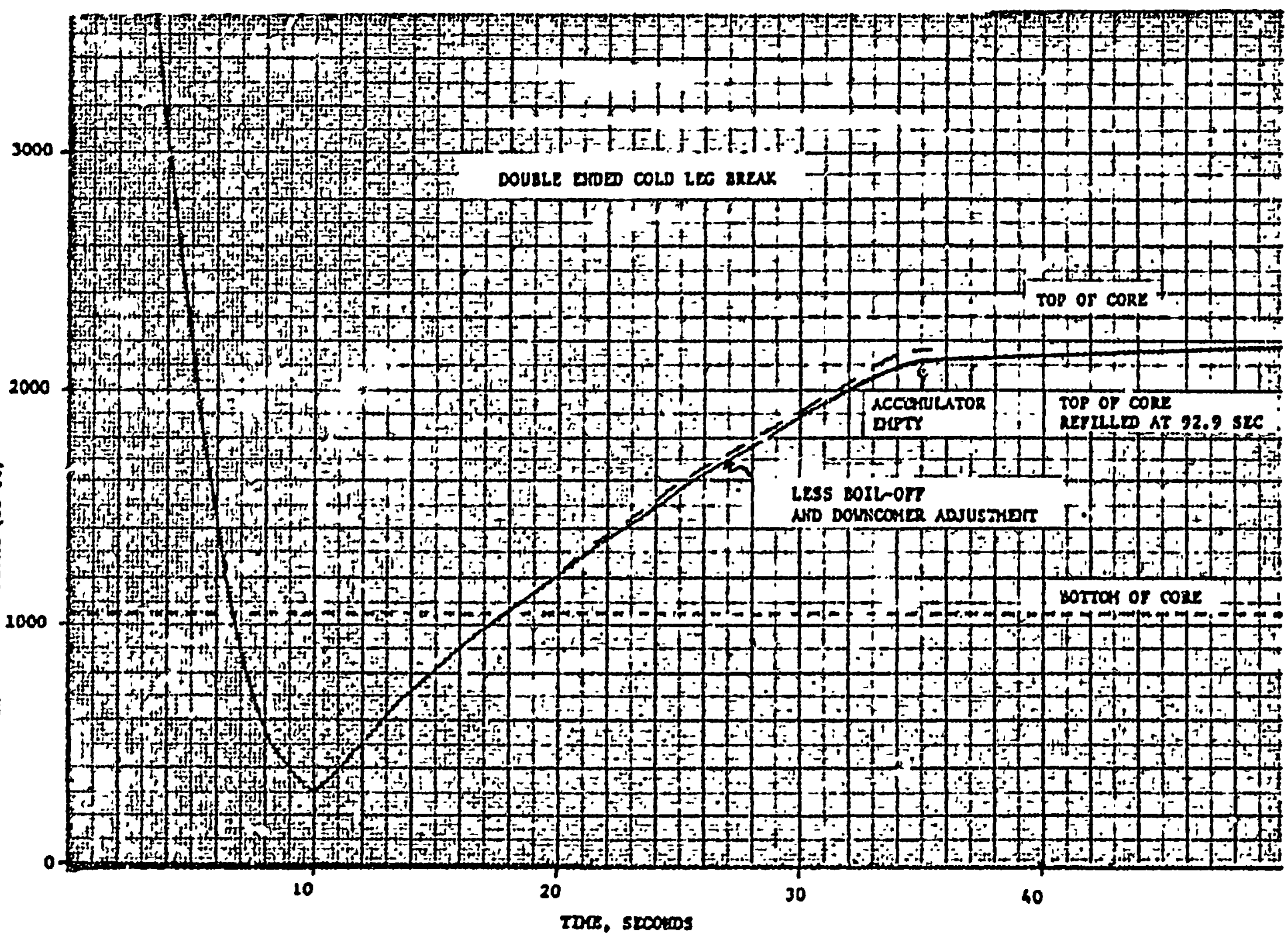


FIGURE 14.3.2-1

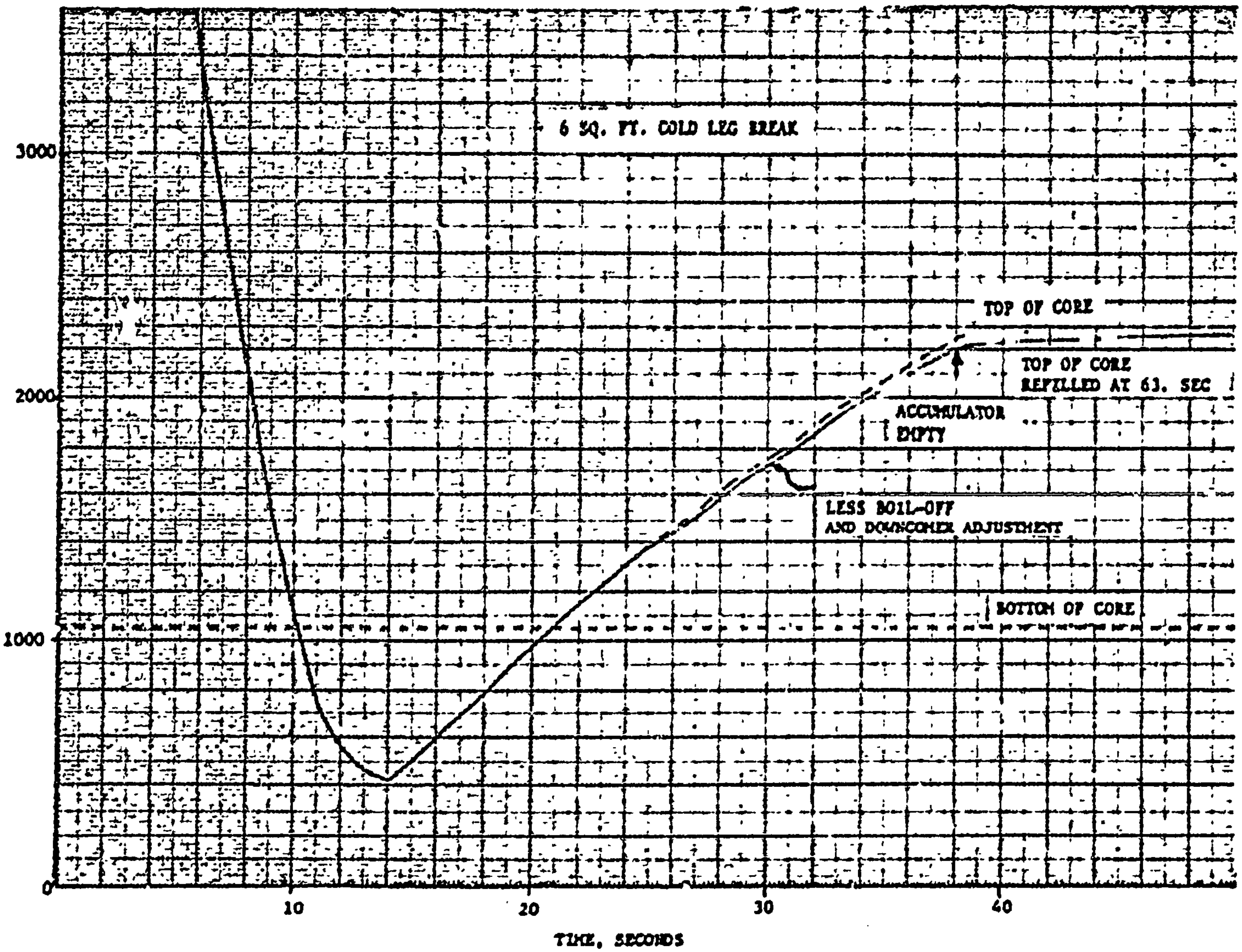


FIGURE 14.3.2-2

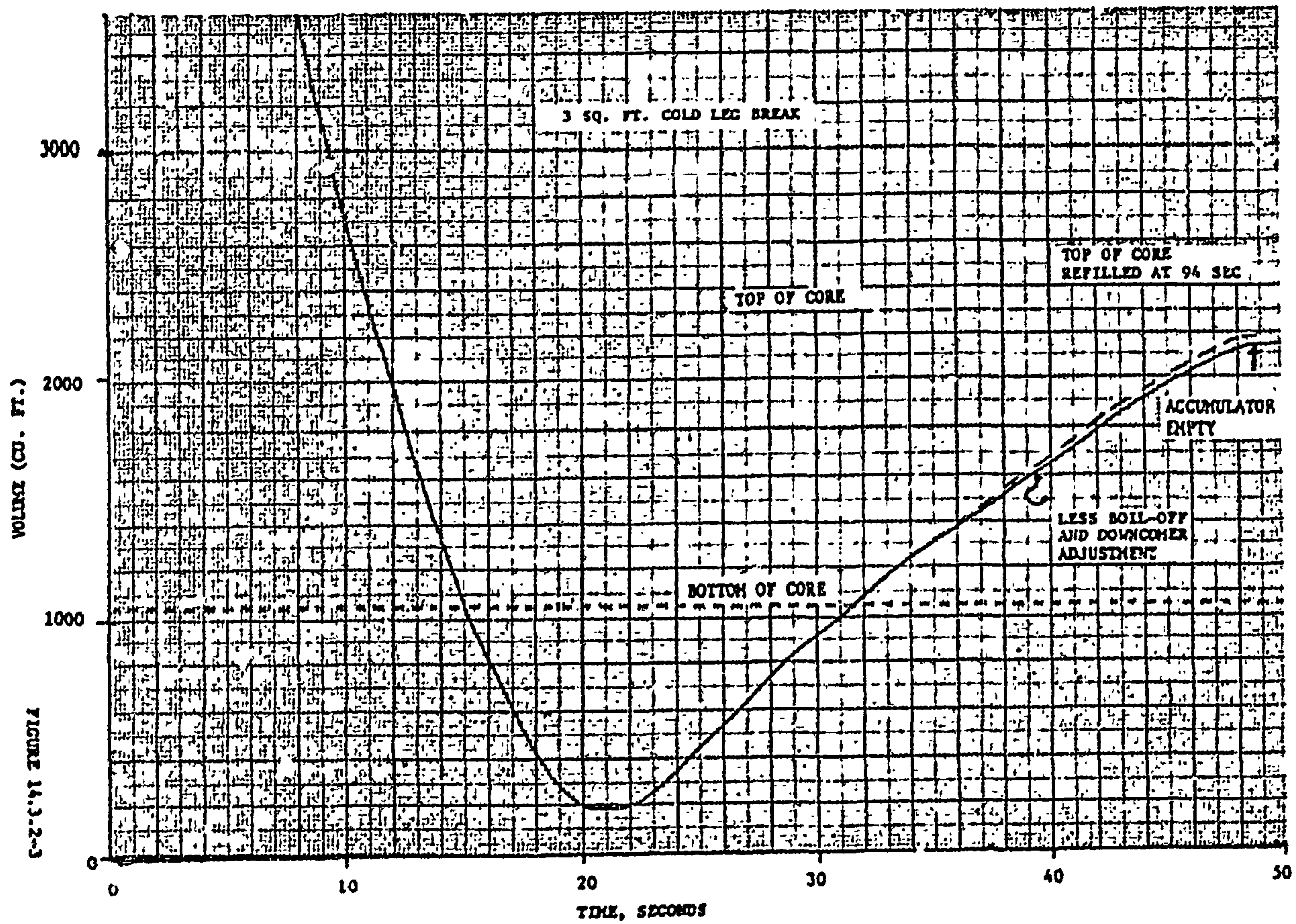


FIGURE 14.3.2-3

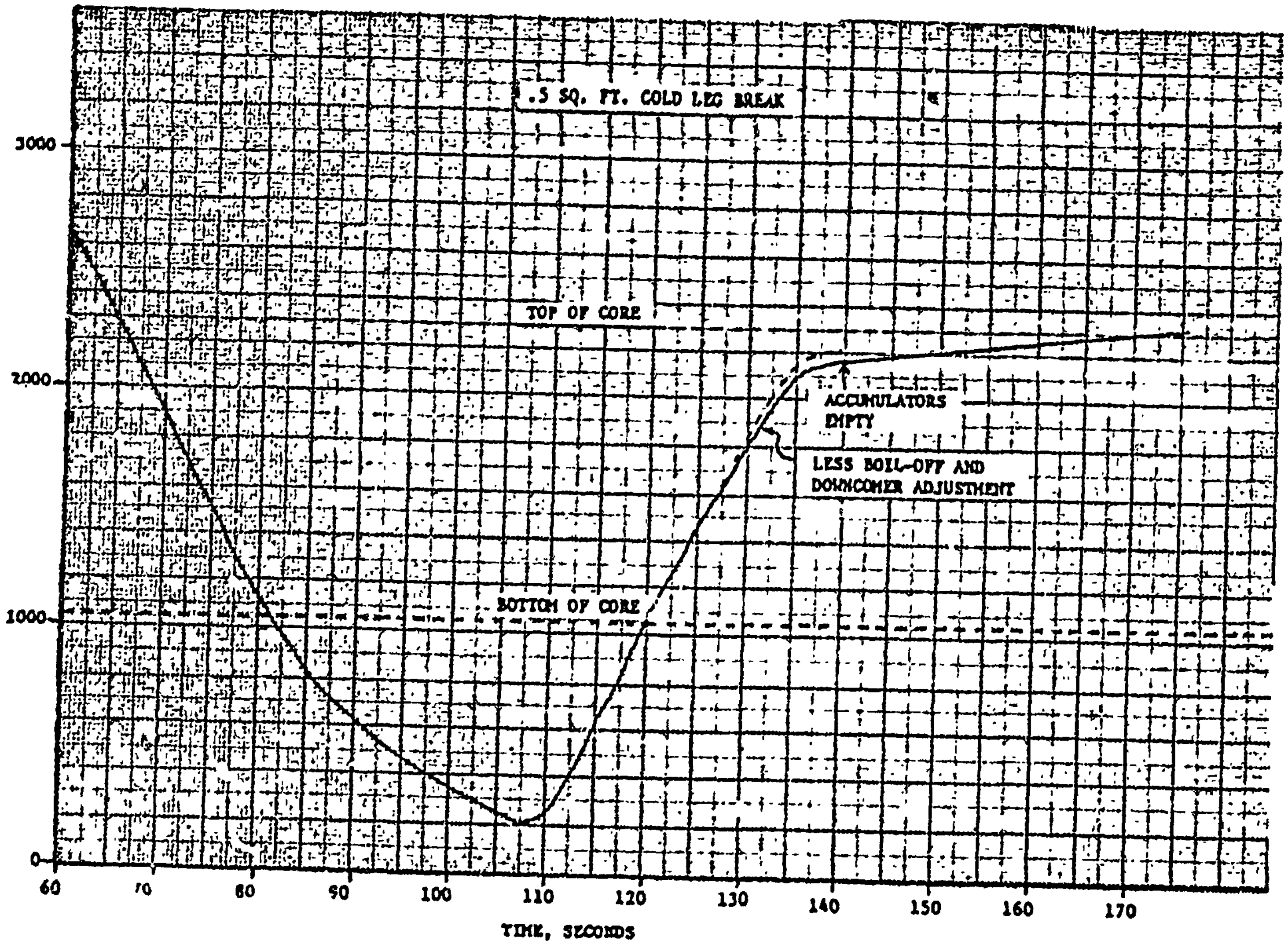


FIGURE 14.3.2-4

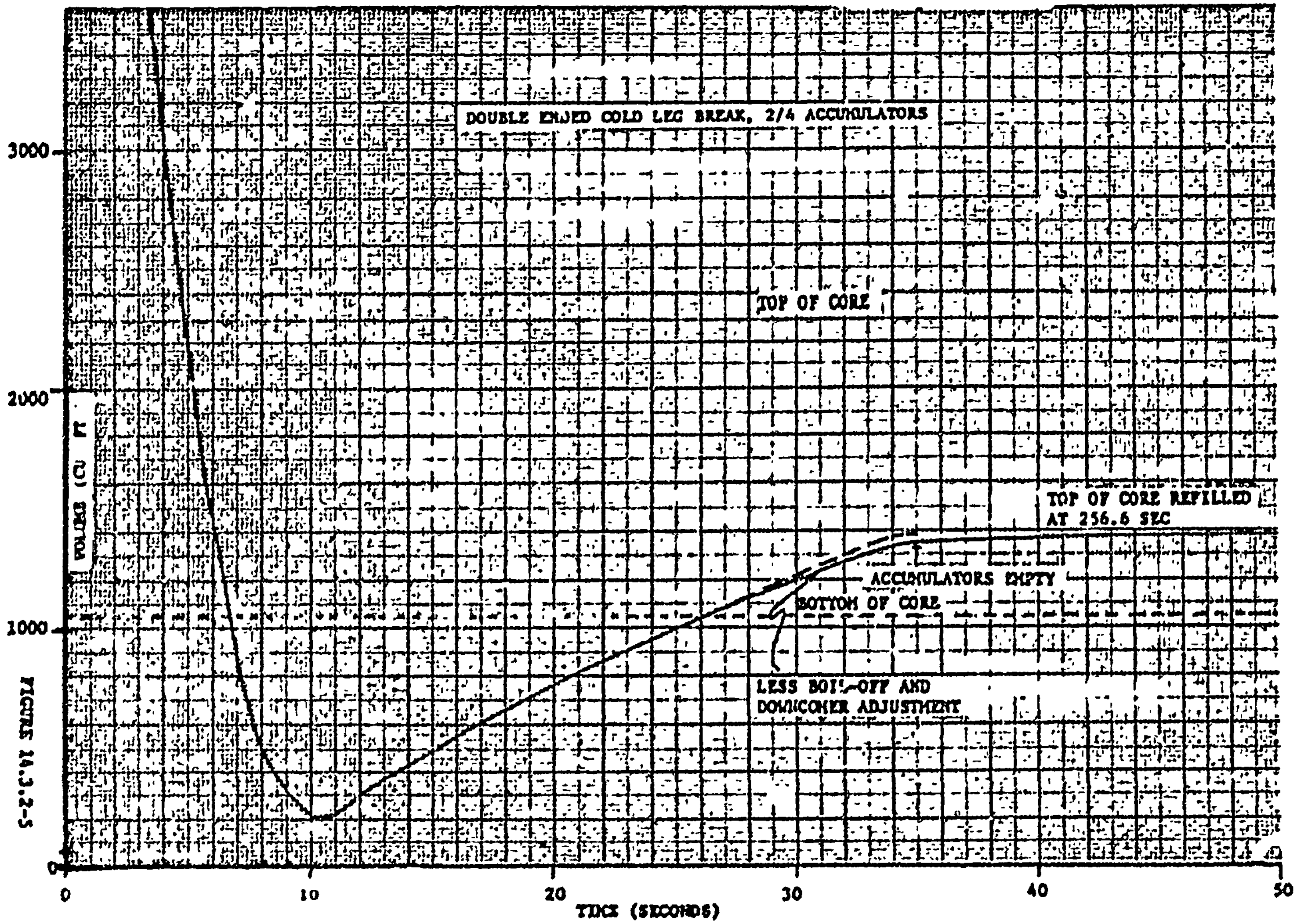


FIGURE 14.3.2-5

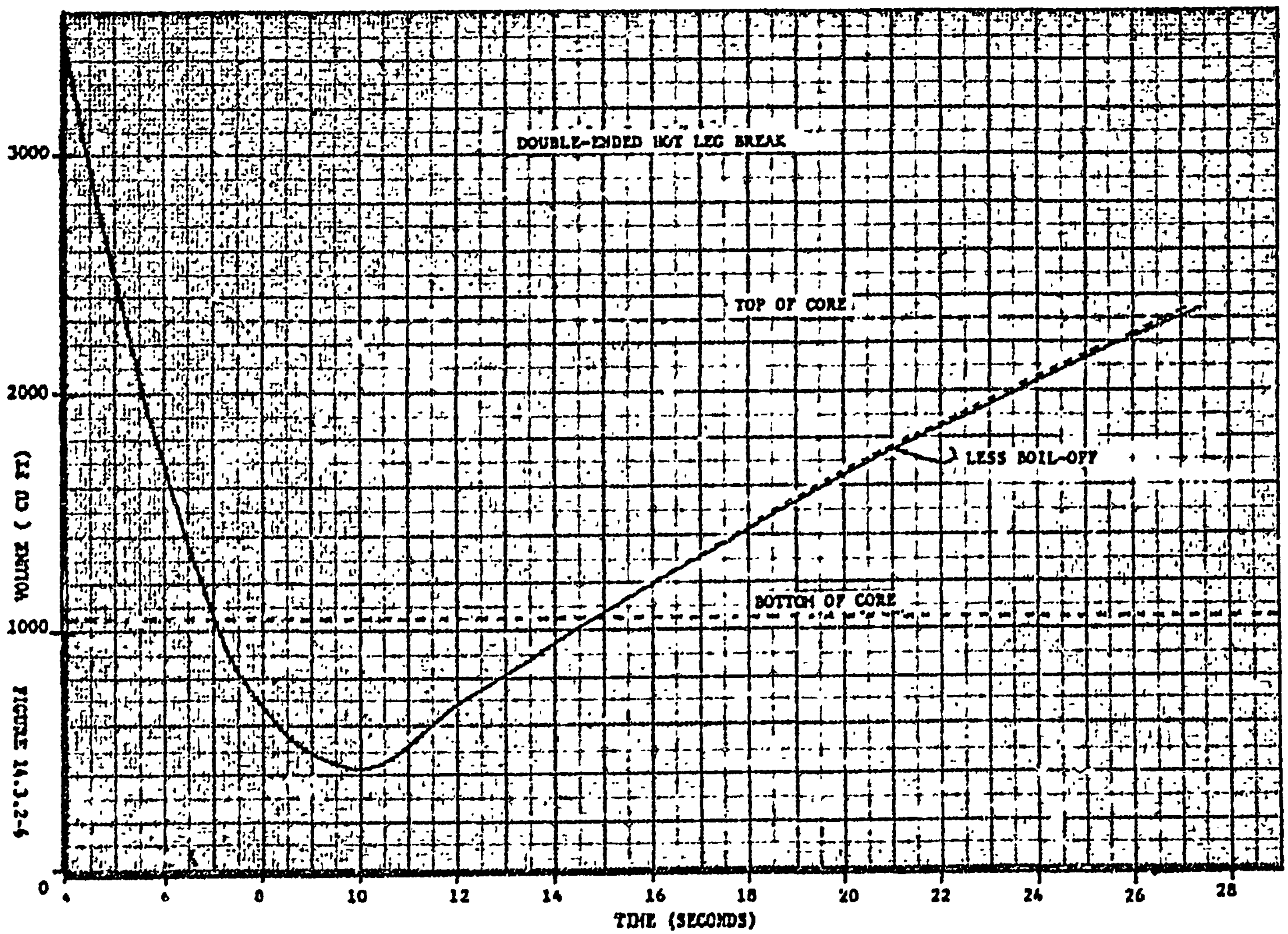


FIGURE 14.3.2-6

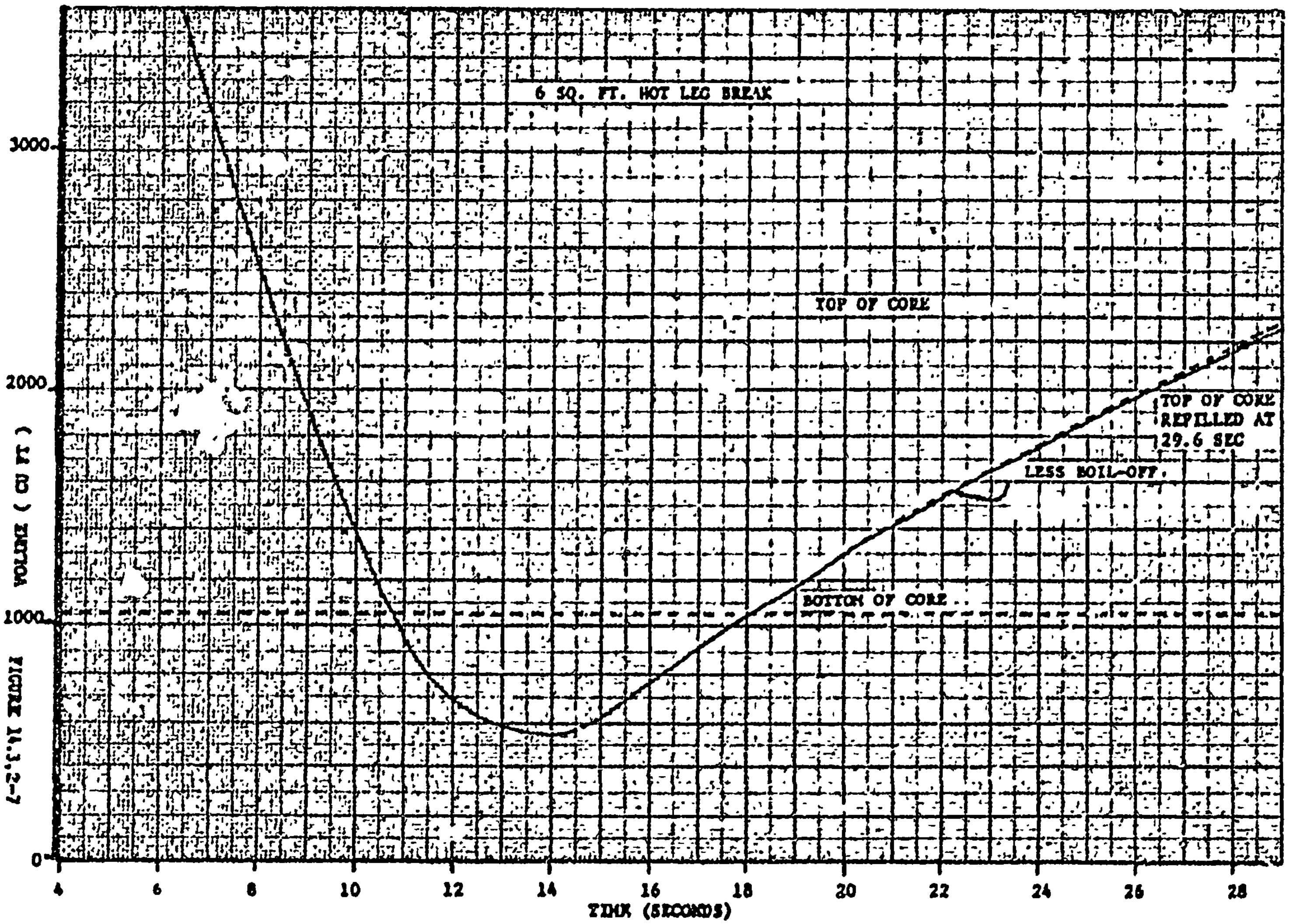


FIGURE 14.3.2-7

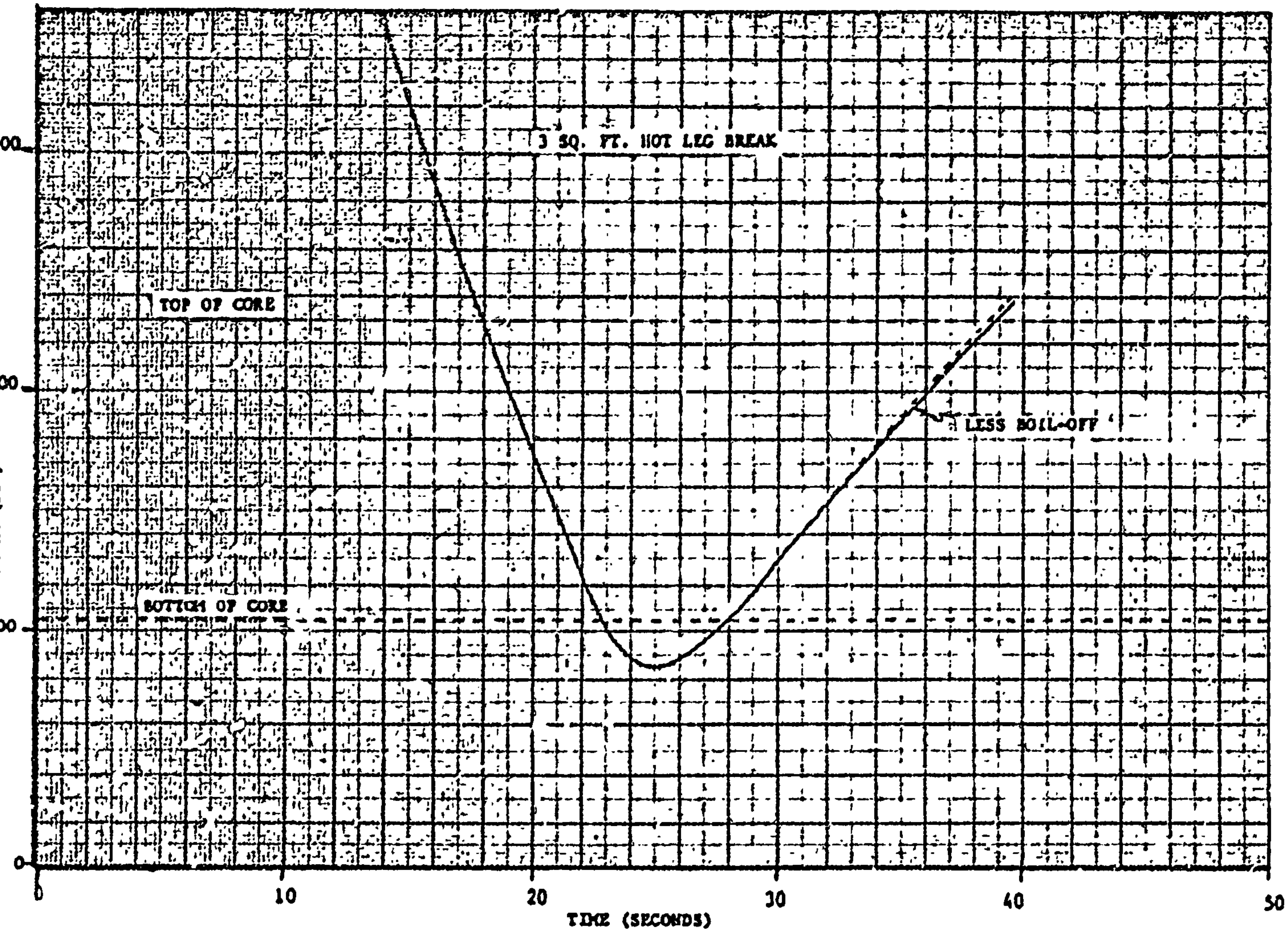


FIGURE 14.3.2-8

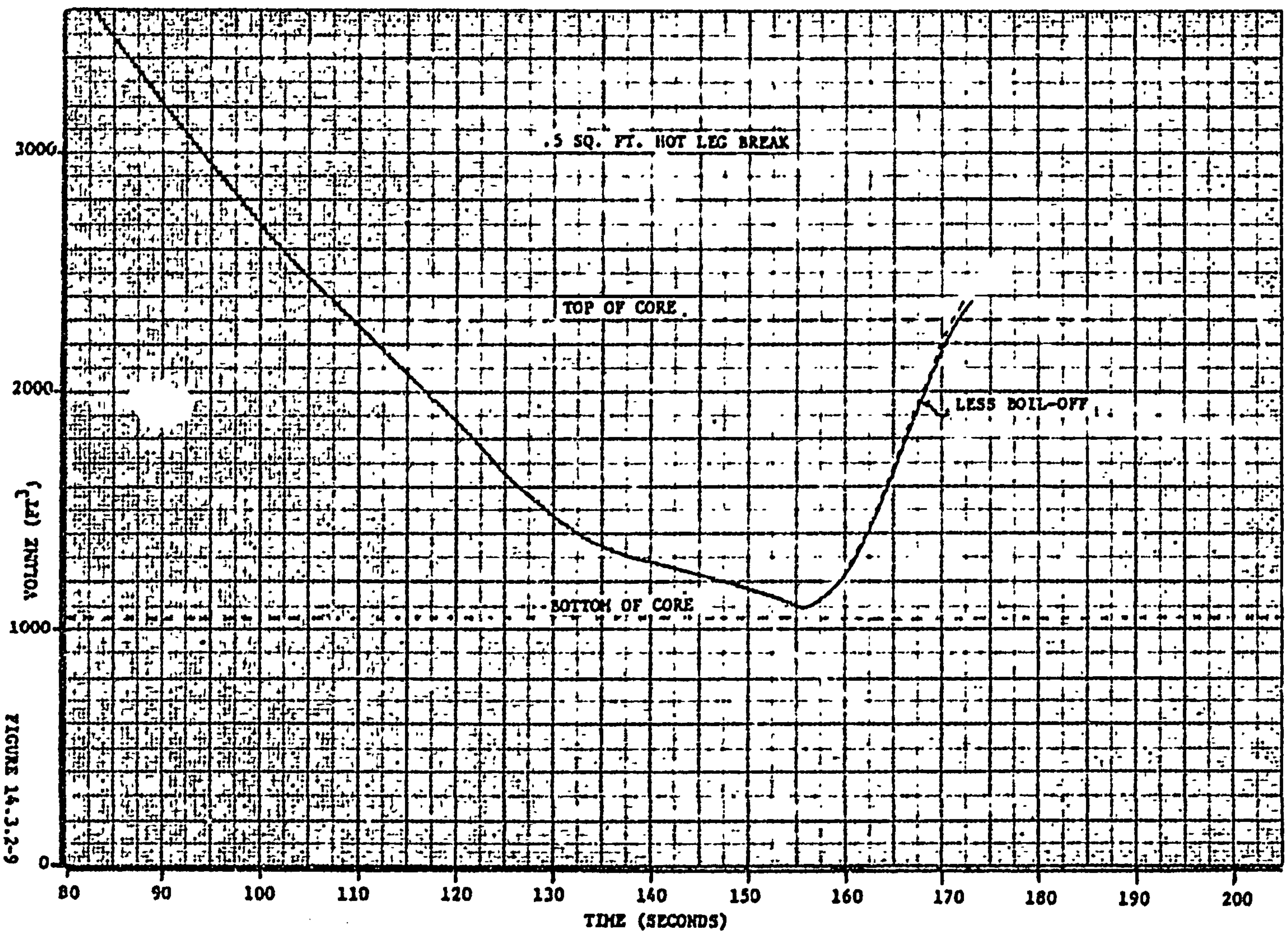


FIGURE 14.3.2-9

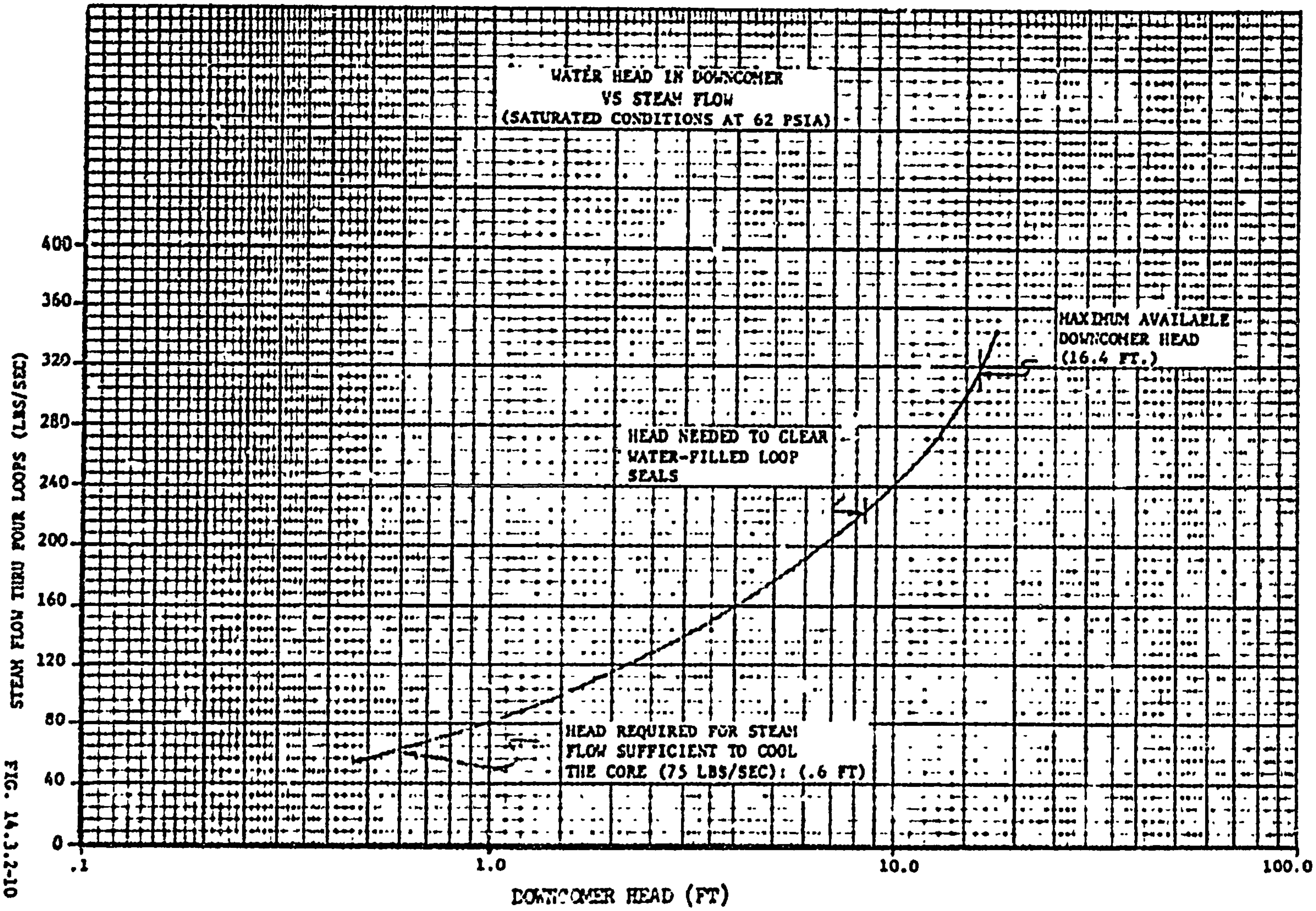
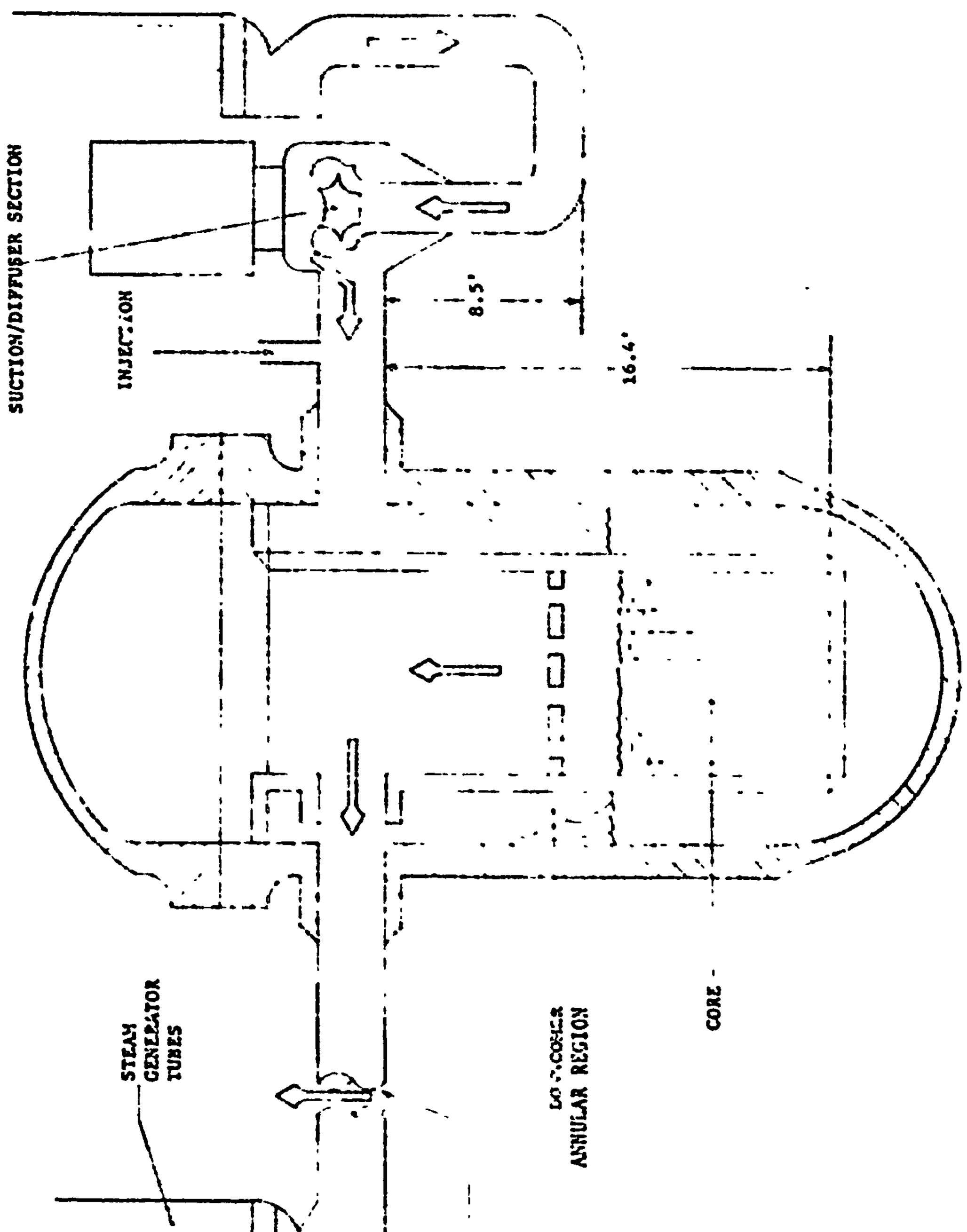
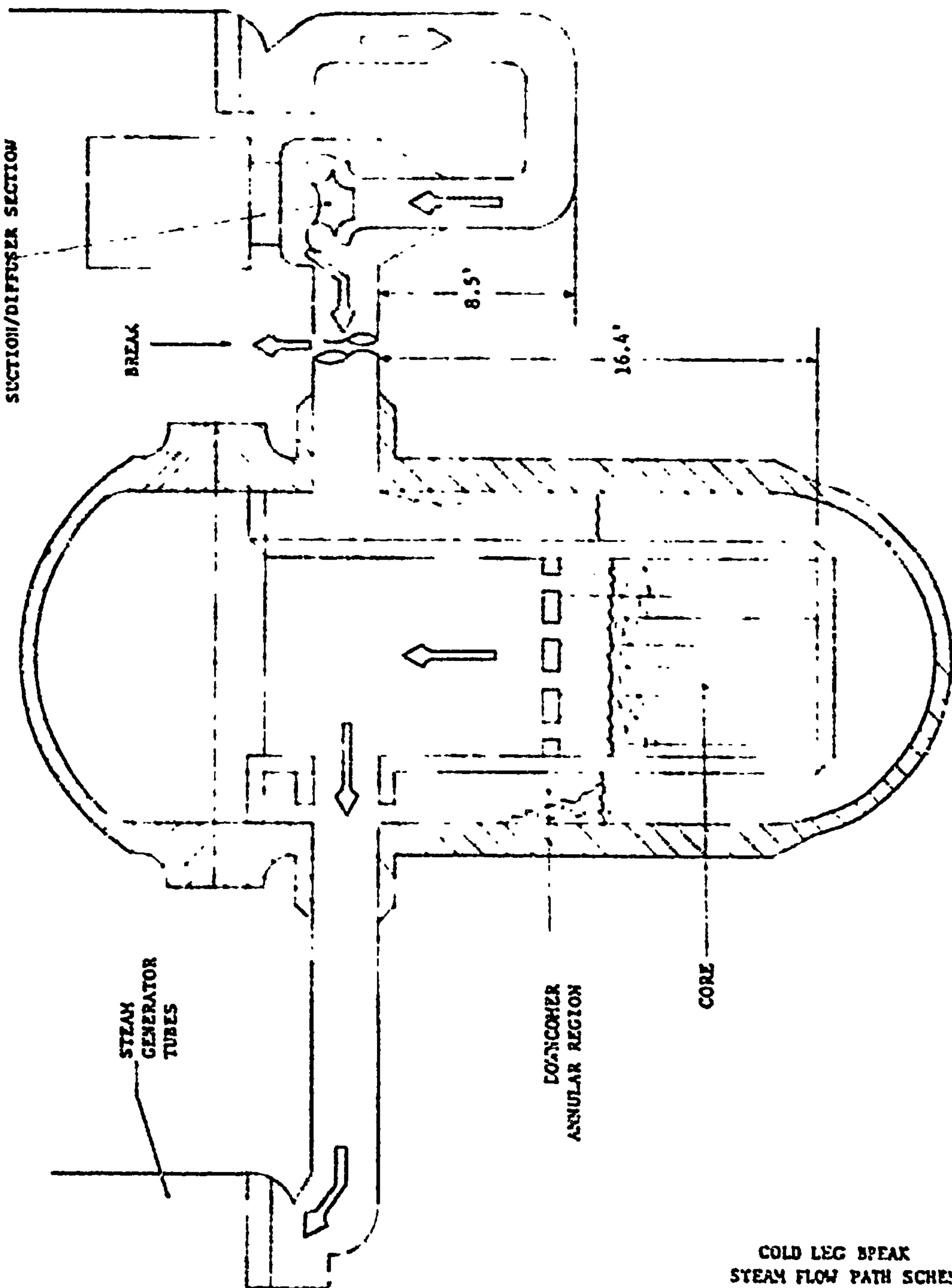

WATER HEAD IN DOWNCOMER
VS STEAM FLOW
(SATURATED CONDITIONS AT 62 PSIA)
MAXIMUM AVAILABLE
DOWNCOMER HEAD
(16.4 FT.)
HEAD NEEDED TO CLEAR
WATER-FILLED LOOP
SEALS
HEAD REQUIRED FOR STEAM
FLOW SUFFICIENT TO COOL
THE CORE (75 LBS/SEC): (.6 FT)
STEAM FLOW THRU FOUR LOOPS (LBS/SEC)
400
360
320
280
240
200
160
120
80
40
0
.1
1.0
10.0
100.0
DOWNCOMER HEAD (FT)

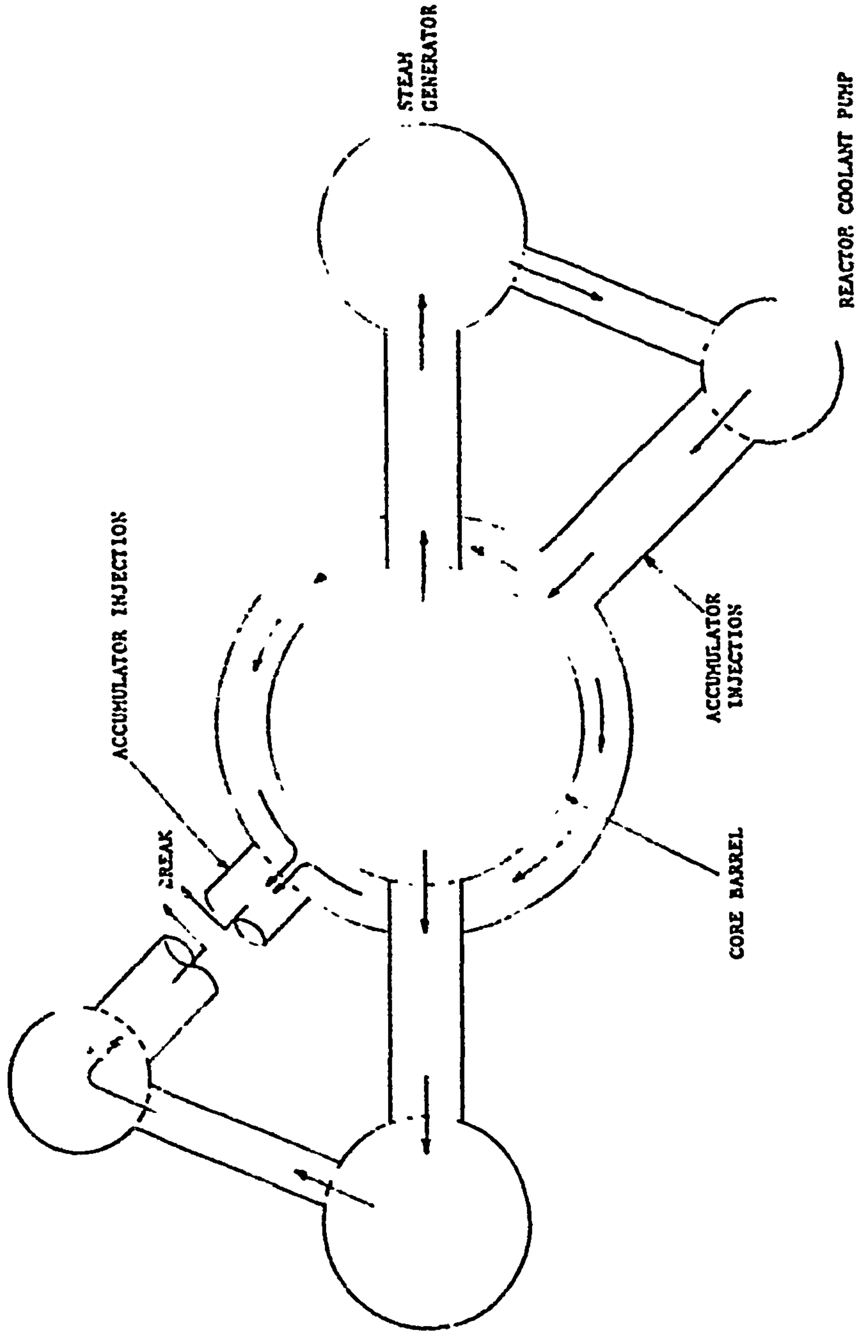
FIG. 14.3.2-10



HOT LEG BREAK
STEAM FLOW PATH SCHEMATIC
FIGURE 14.3.2-10a



COLD LEG BREAK
STEAM FLOW PATH SCHEMATIC
FIGURE 14.3.2-106



COLD LEG BREAK STEAM FLOW PATH
FIGURE 14.3.2-10c

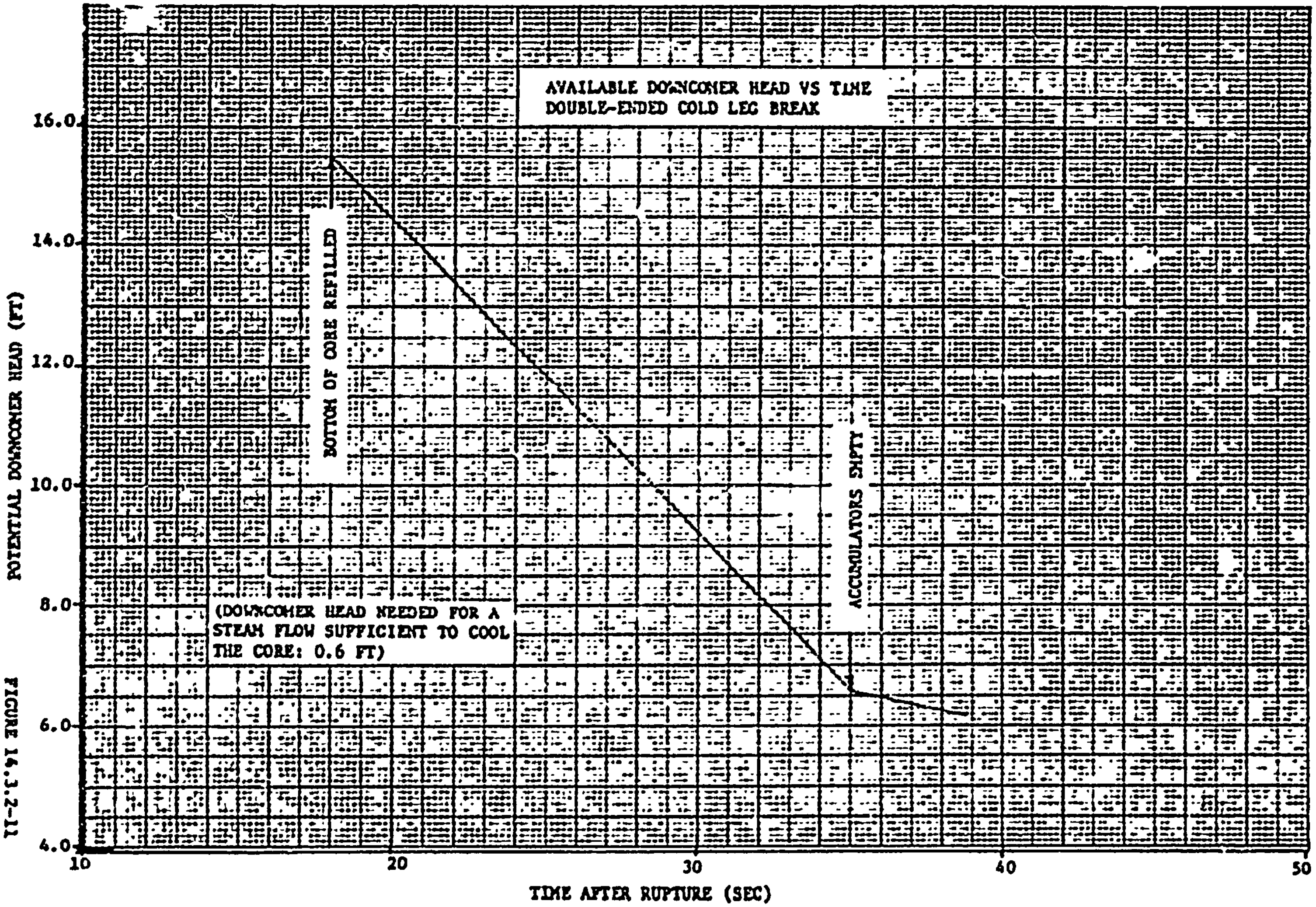
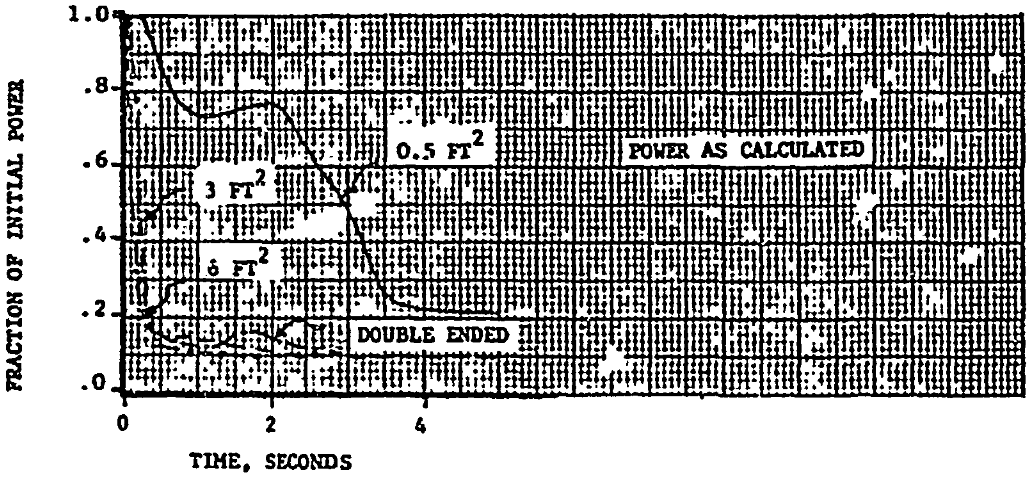
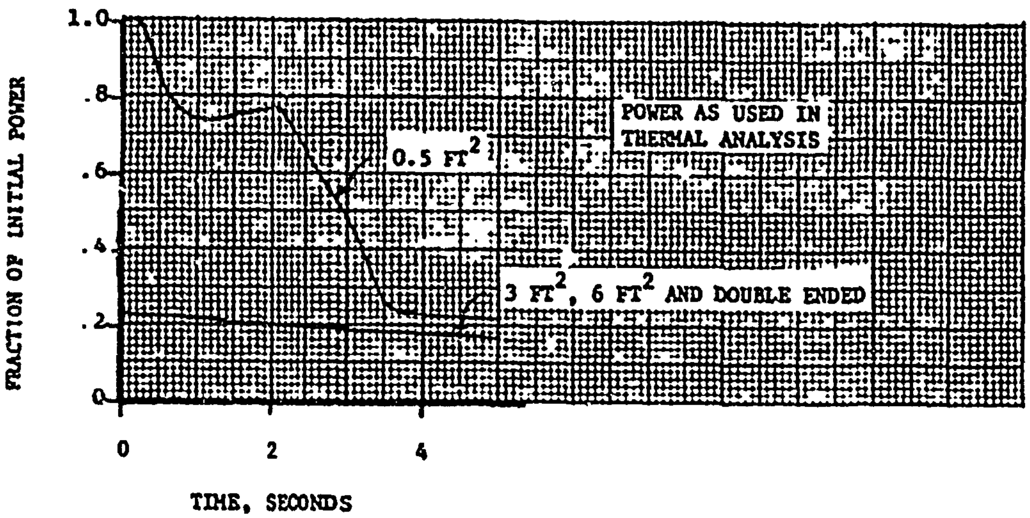


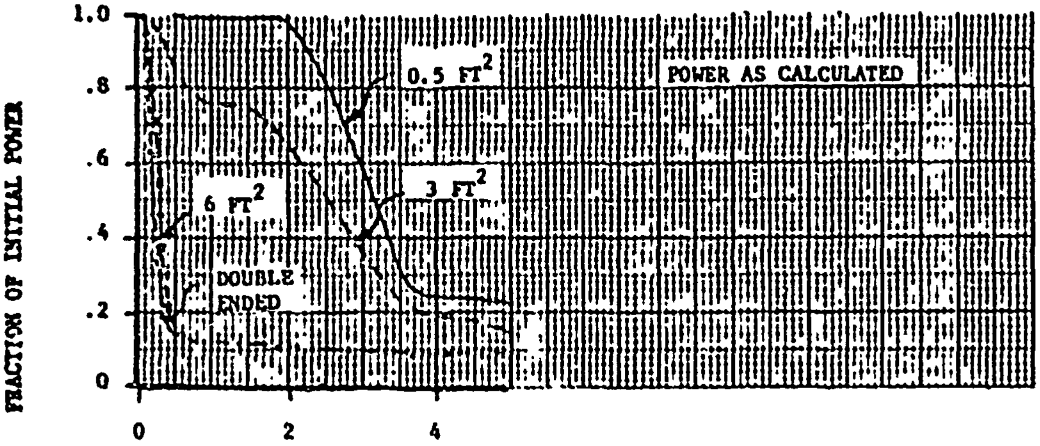
FIGURE 14.3.2-11

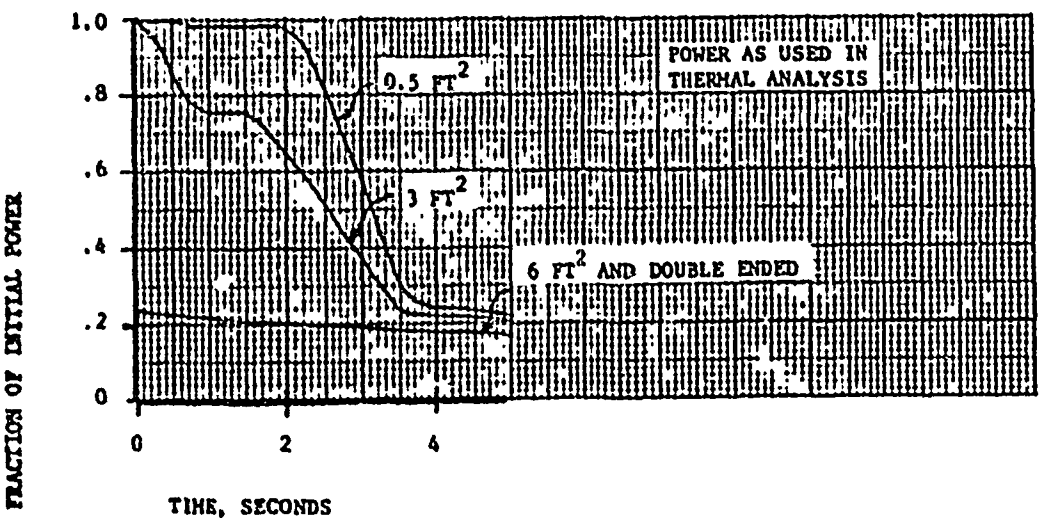




POWER TRANSIENT DURING BLOWDOWN
COLD LEG BREAKS

FIGURE 14.3.2-12





POWER TRANSIENT DURING BLOWDOWN
HOT LEG BREAKS

FIGURE 14.3.2-13

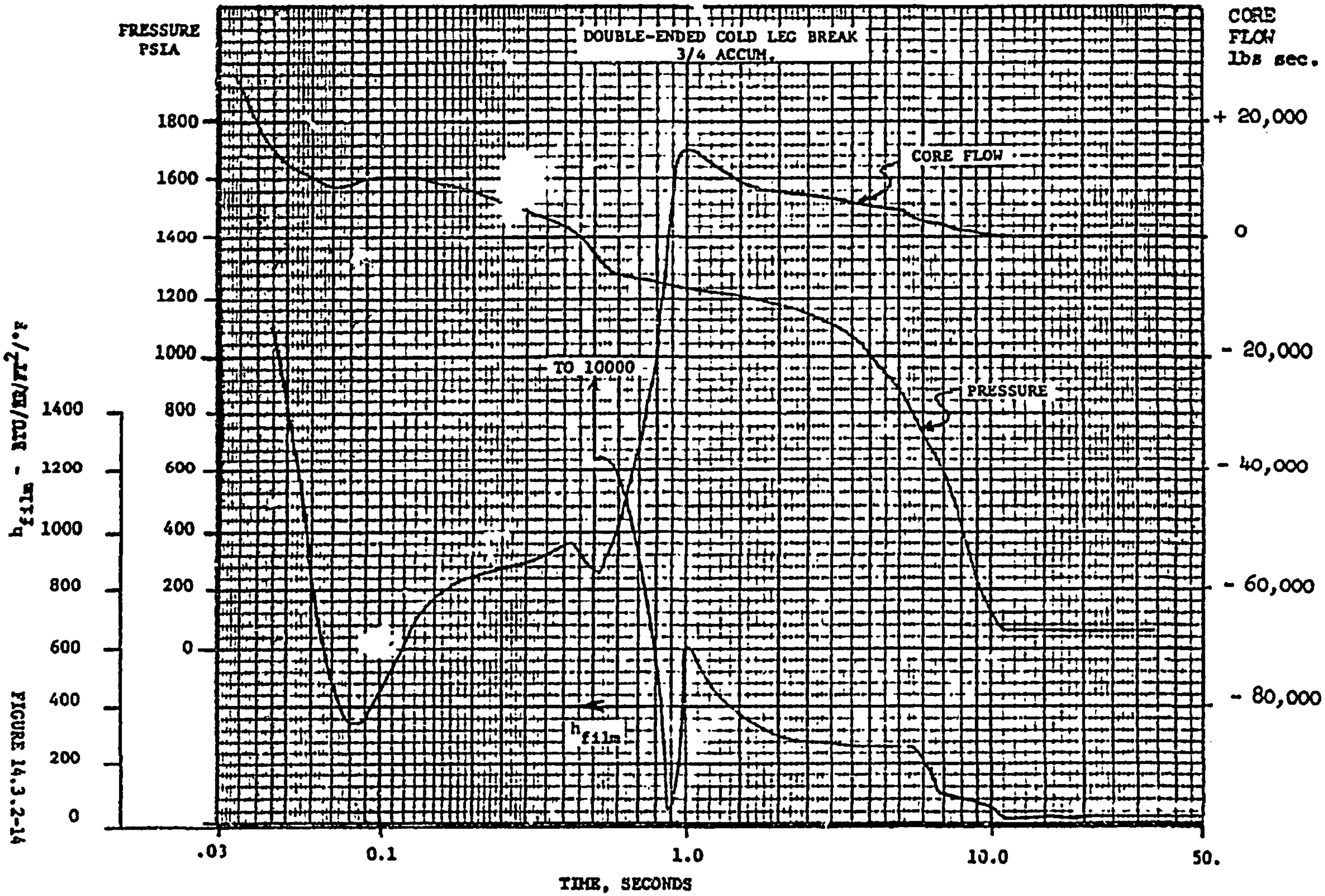


FIGURE 14.3.2-14

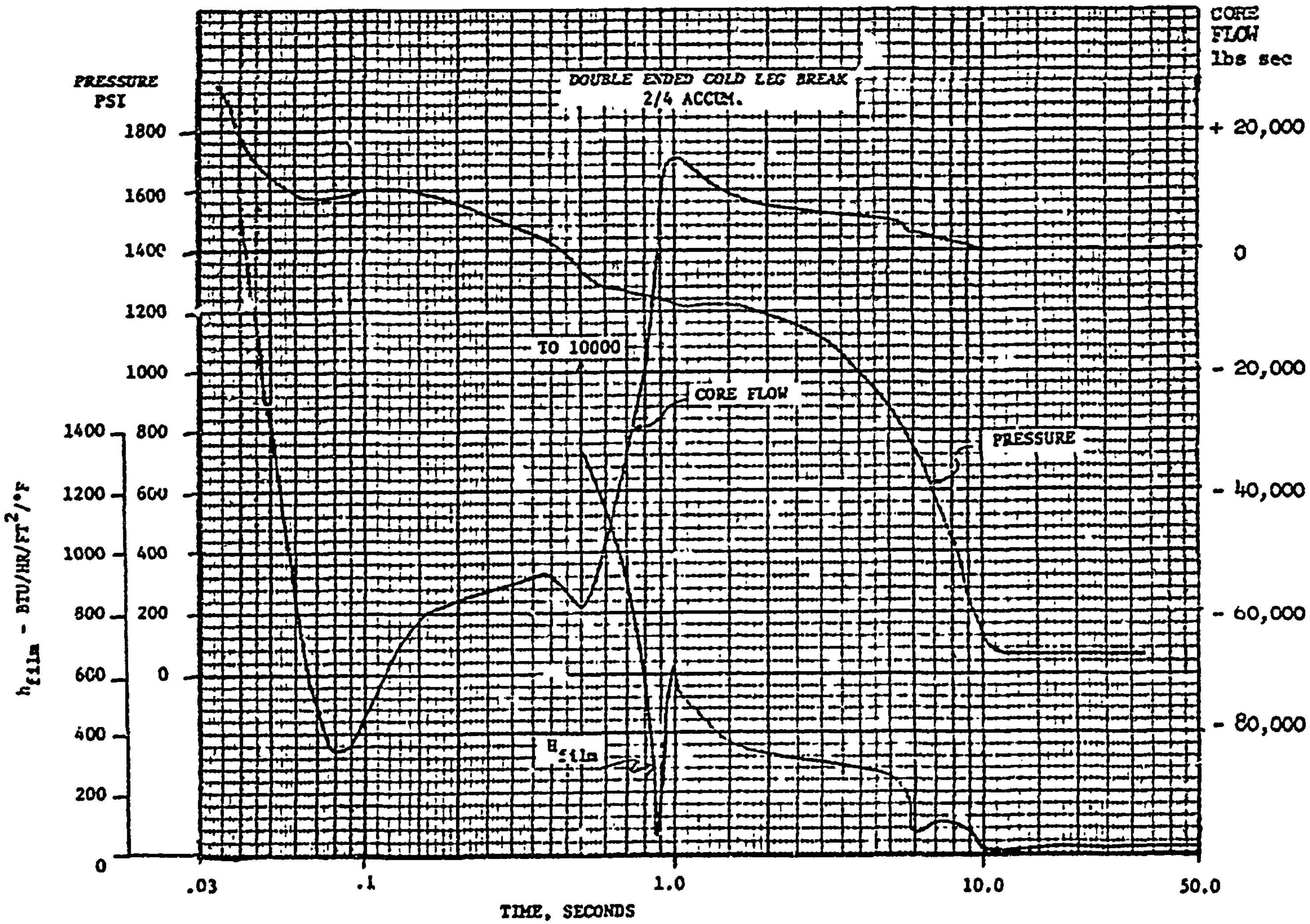


FIGURE 14.3.2-15

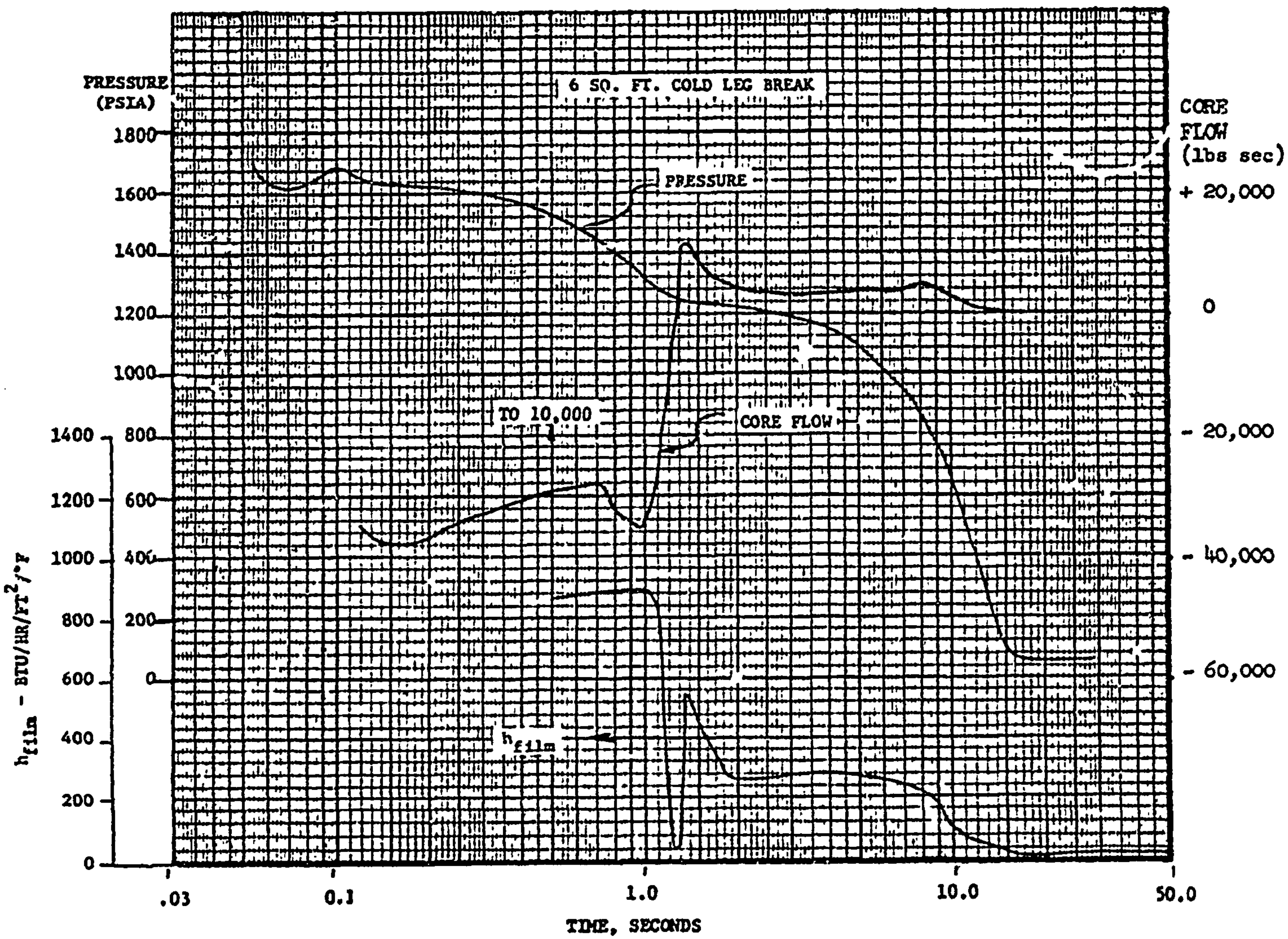


FIGURE 14.3.2-16

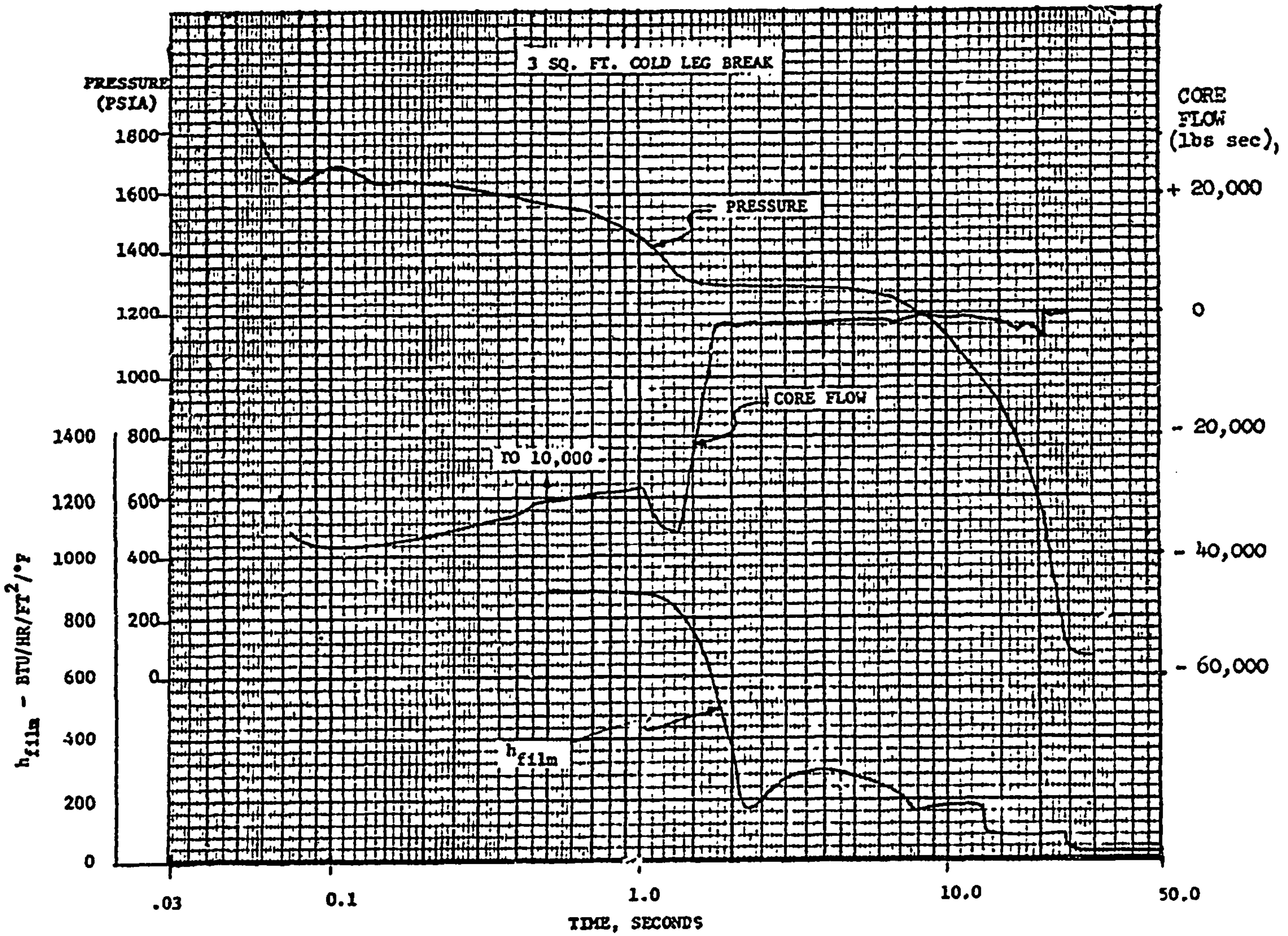


FIGURE 14.3.2-17

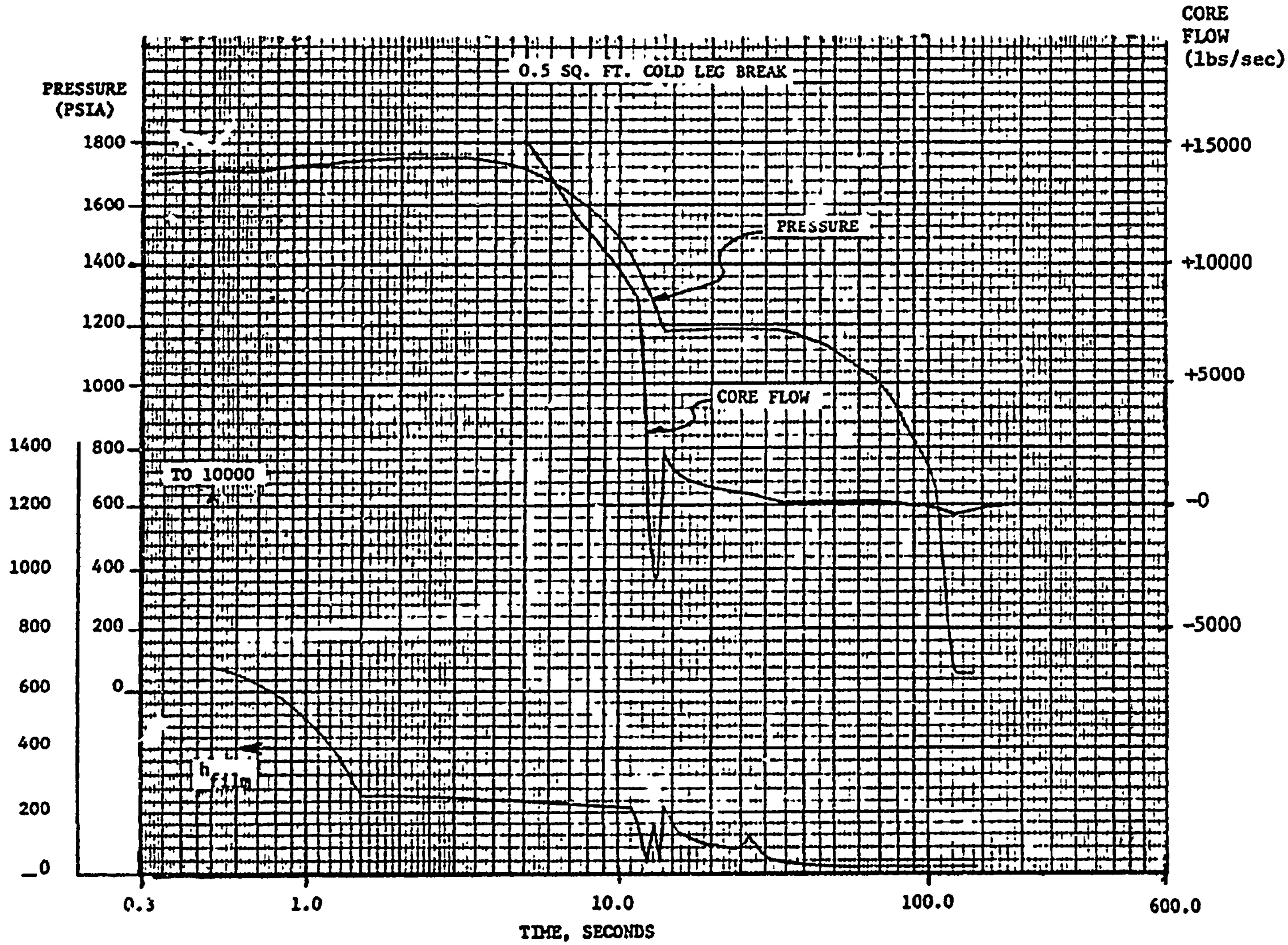

0.5 SQ. FT. COLD LEG BREAK
PRESSURE (PSIA)
CORE FLOW (lbs/sec)
1800
1600
1400
1200
1000
800
600
400
200
0
1400
1200
1000
800
600
400
200
0
+15000
+10000
+5000
-0
-5000
PRESSURE
CORE FLOW
TO 10000
h_{film}
h_{film} - BTU/HR/FT^2/°F
0.3
1.0
10.0
100.0
600.0
TIME, SECONDS

FIG. 14.3.2-18

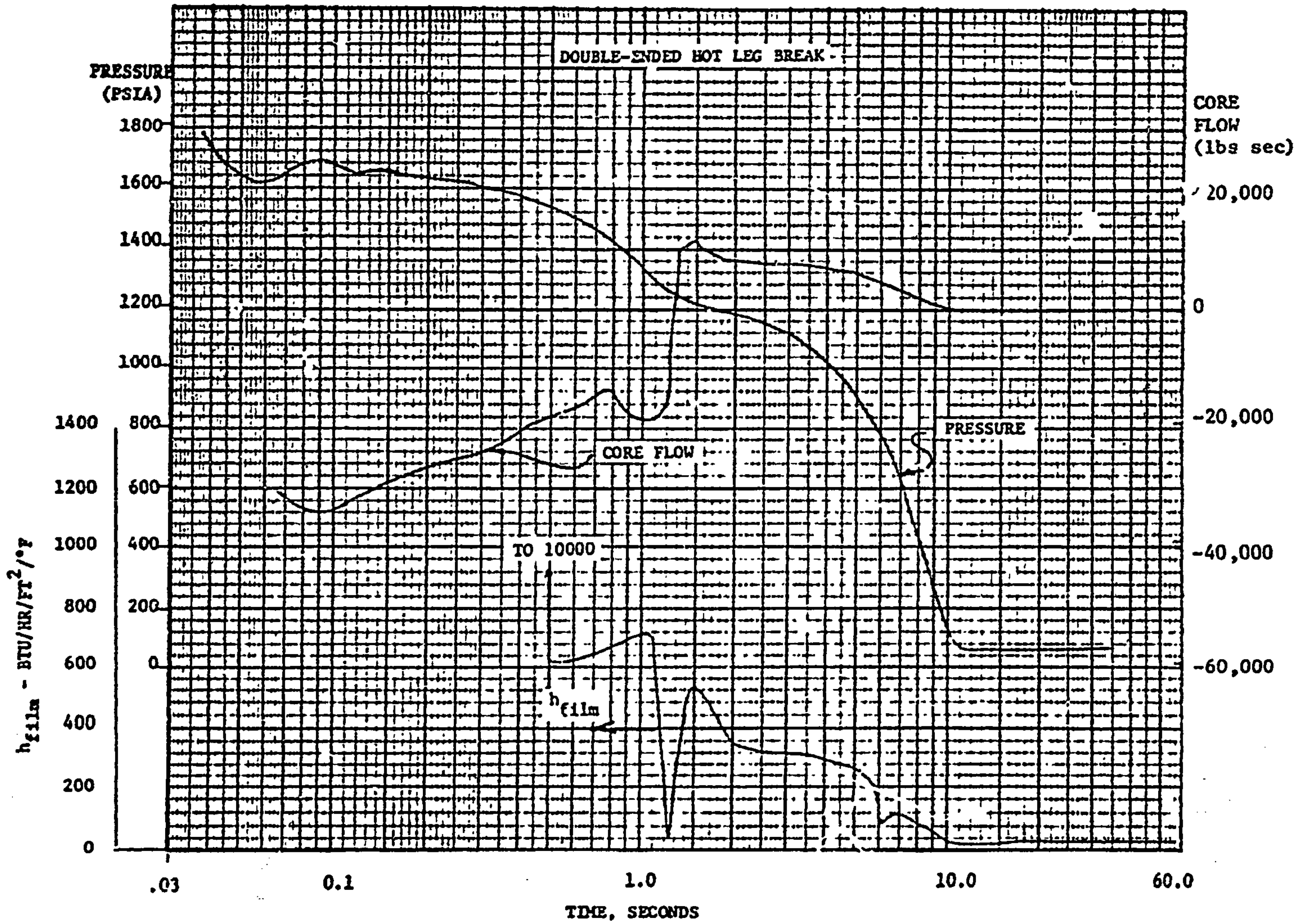


FIGURE 14.3.2-19

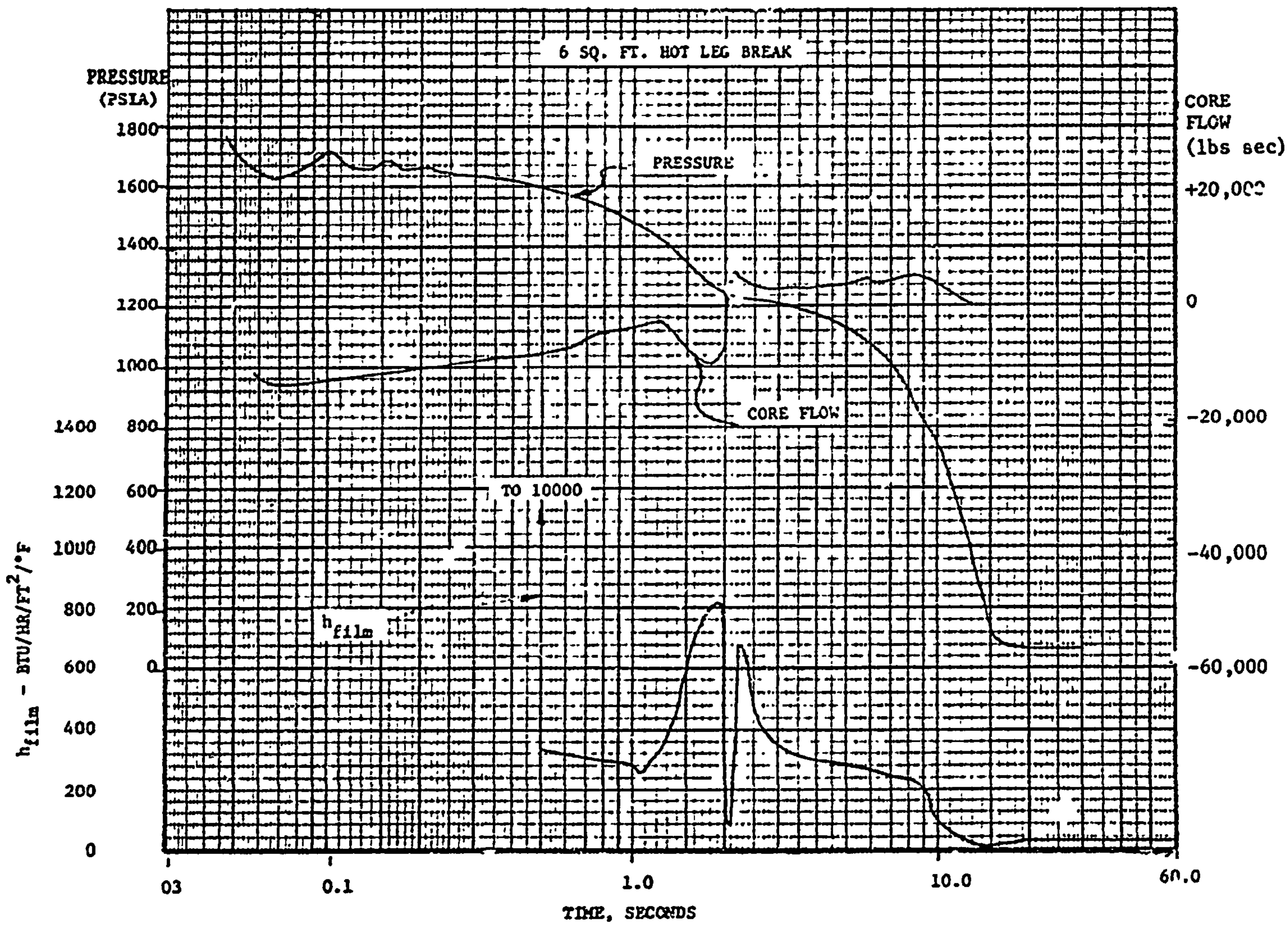


FIGURE 14.3.2-20

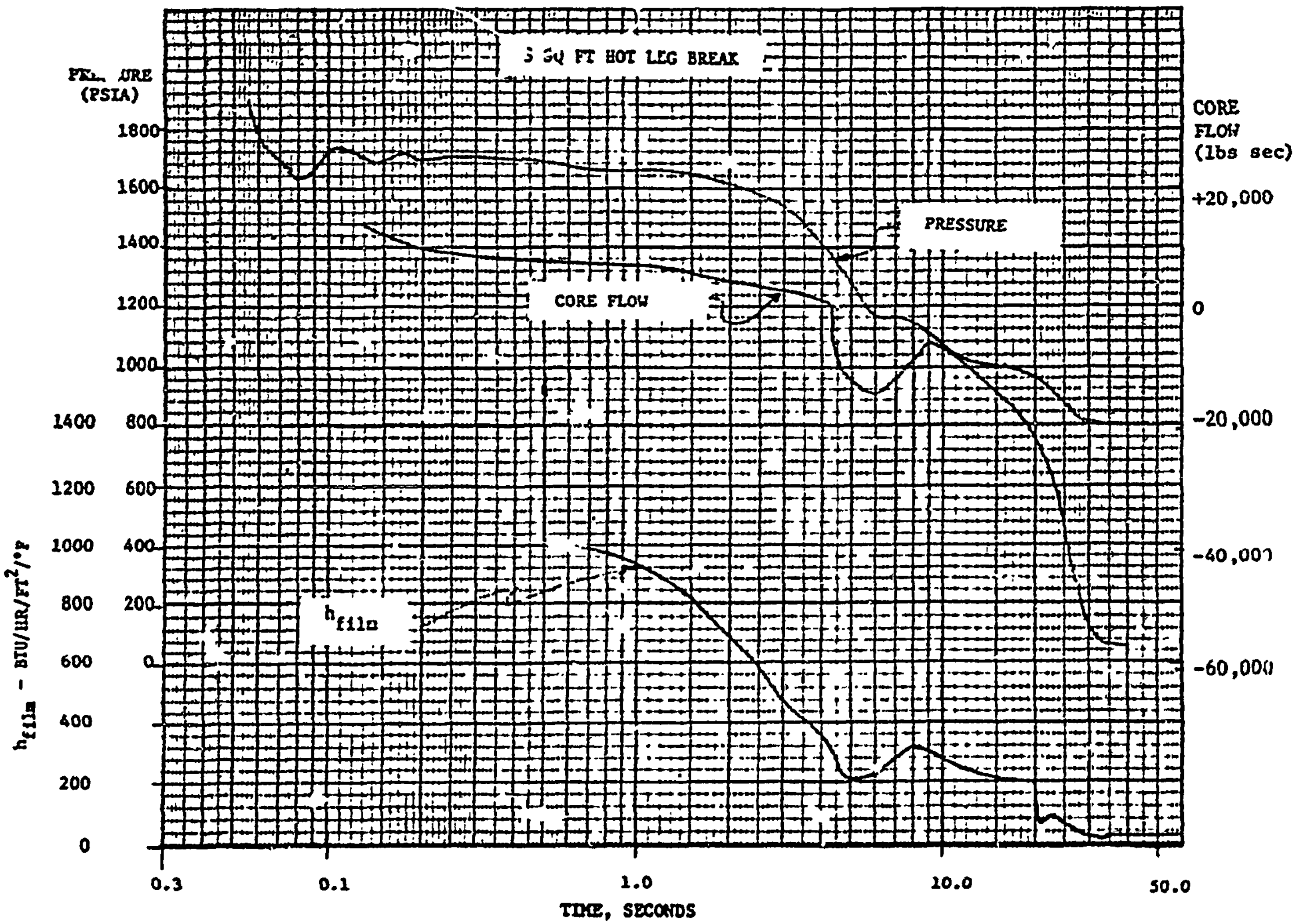


FIGURE 14.3.2-21

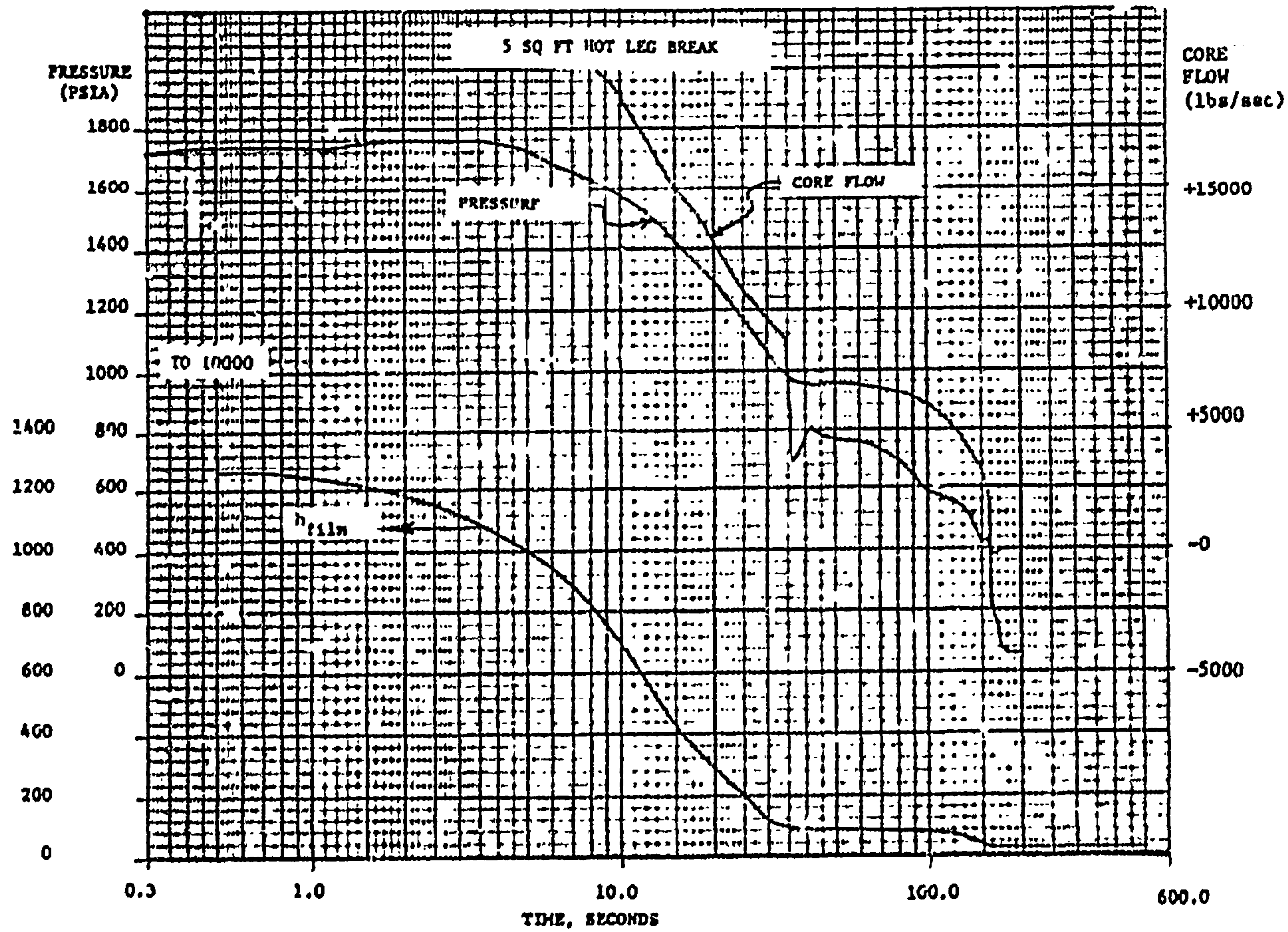


FIGURE 14.3.2-22

DOUBLE ENDED COLD LEG BREAK - 2758 MWt

CLAD TEMPERATURE TRANSIENT V.S. TIME AFTER BREAK

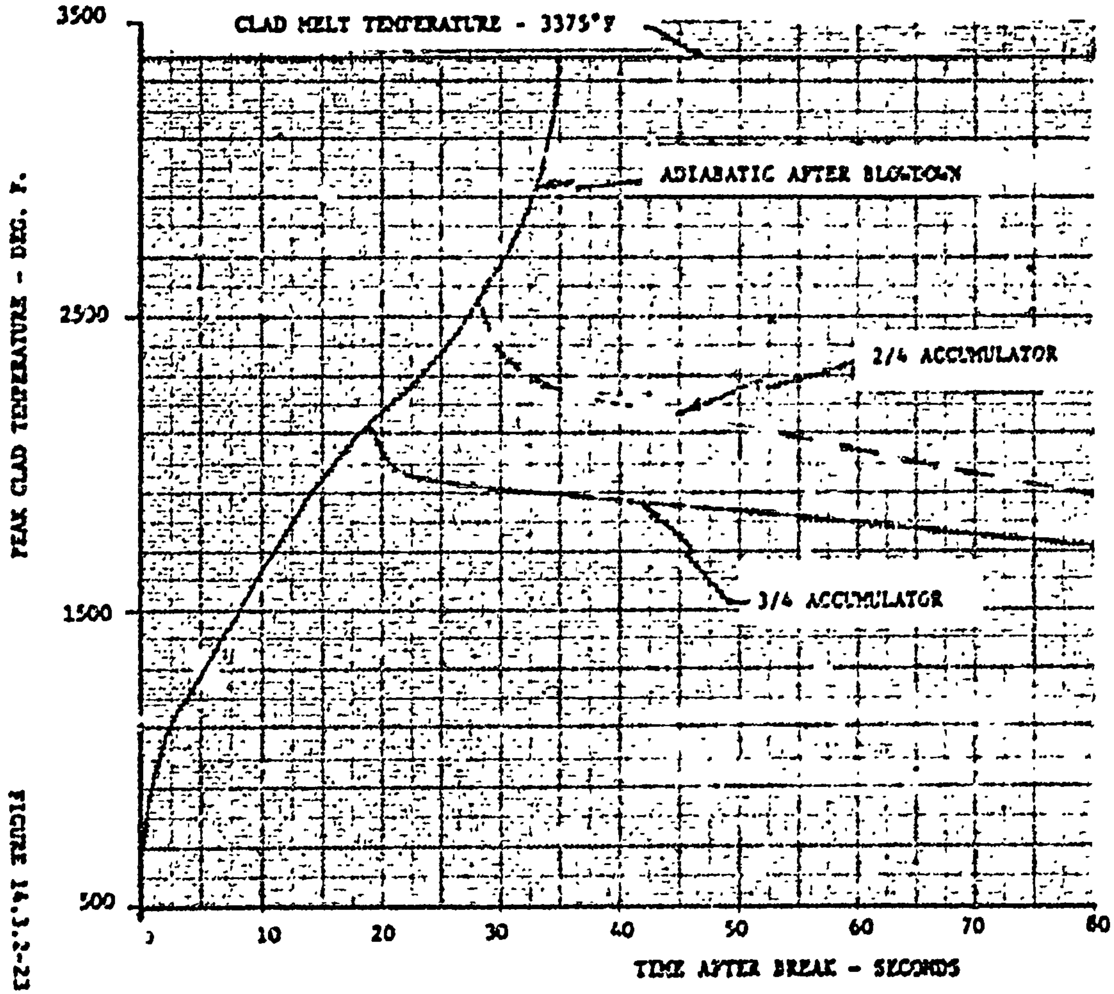


FIGURE 14.3.2-23

6 FT^2 COLD LEG BREAK - 2758 MWt
CLAD TEMPERATURE TRANSIENT
VS TEMPERATURE AFTER BREAK

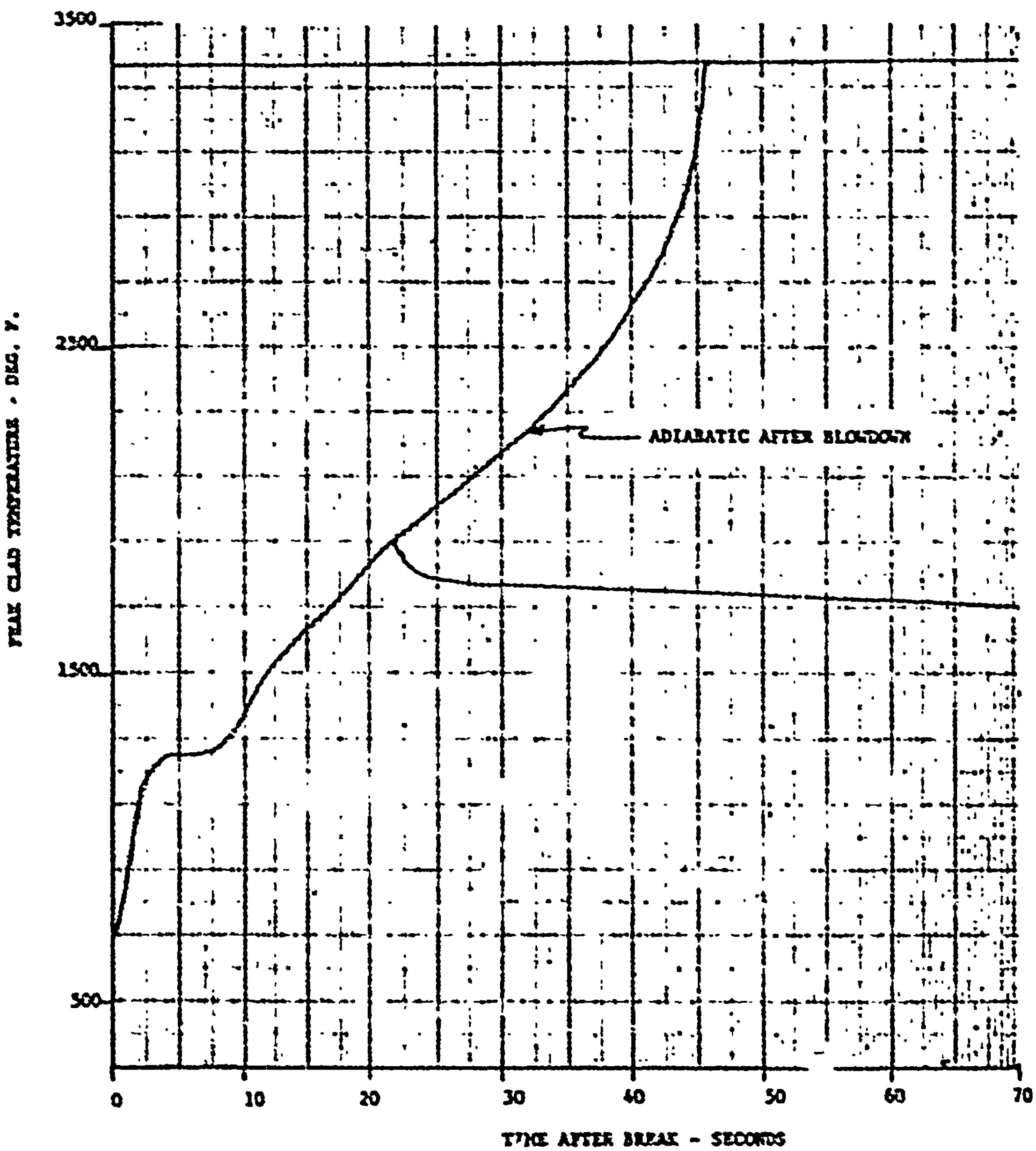


FIGURE 14.3.2-24

3FT^2 COLD LEG BREAK - 2758 MWt
CLAD TEMPERATURE TRANSIENT VS
TIME AFTER BREAK

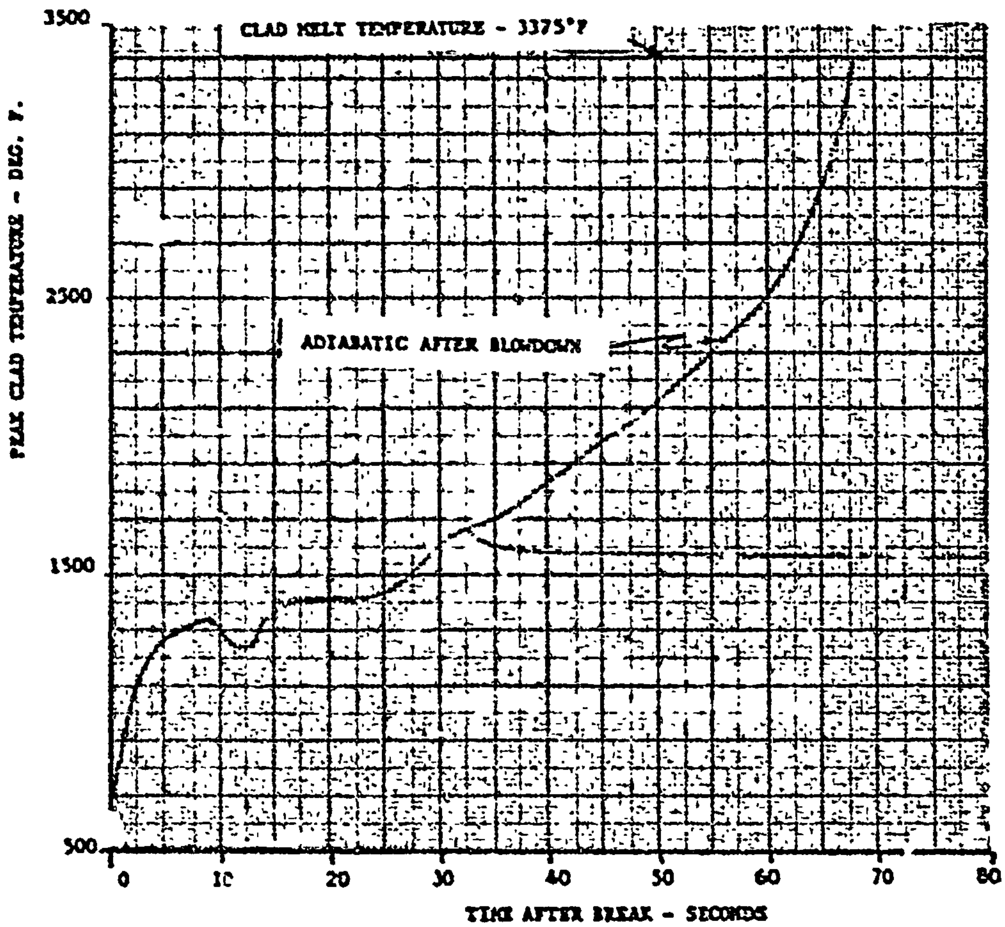


FIGURE 14.3.2-25

.5 FT^2 COLD LEG BREAK - 2758 MWt

CLAD TEMPERATURE TRANSIENT VS TIME AFTER BREAK

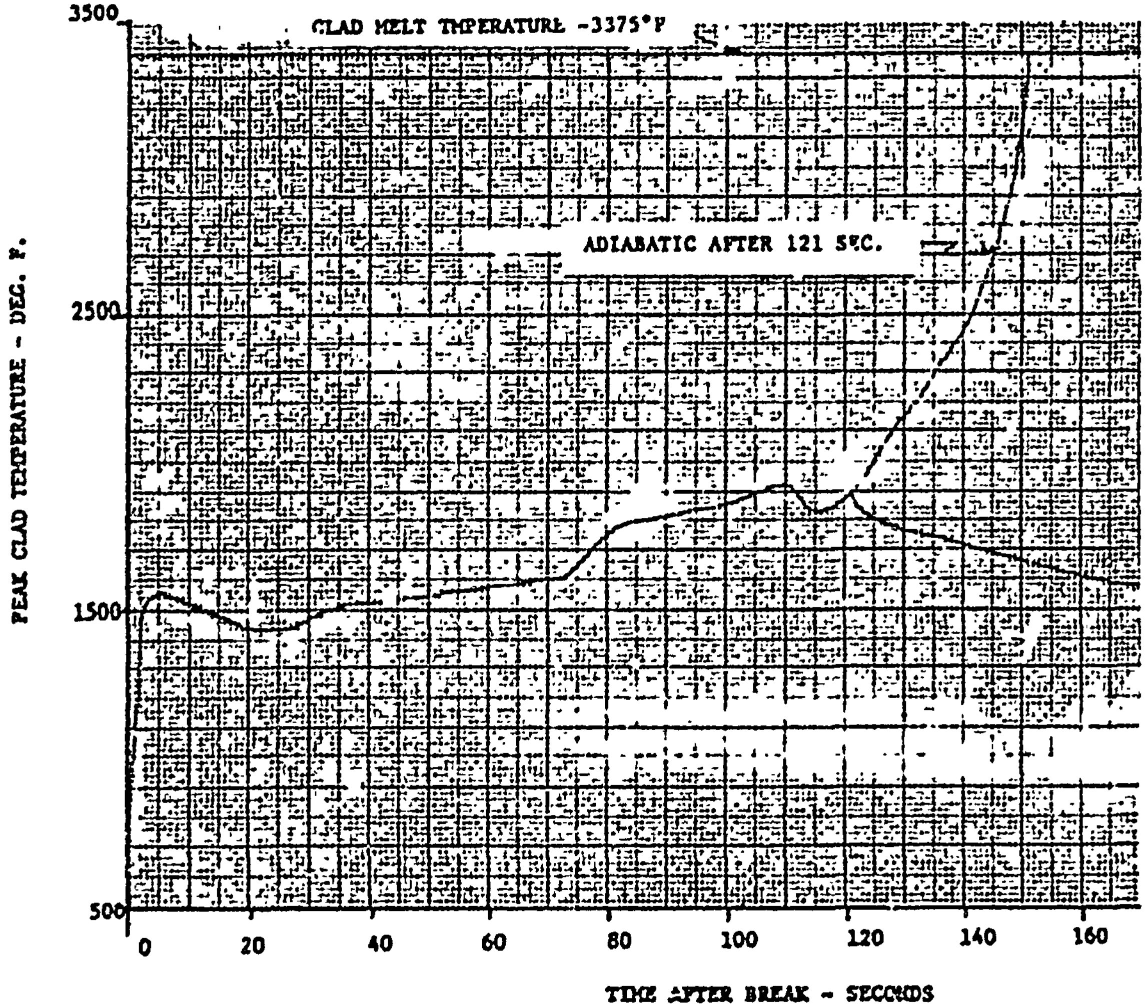


FIGURE 14.3.2-26

CLAD TEMPERATURE TRANSIENT VS TIME AFTER BREAK

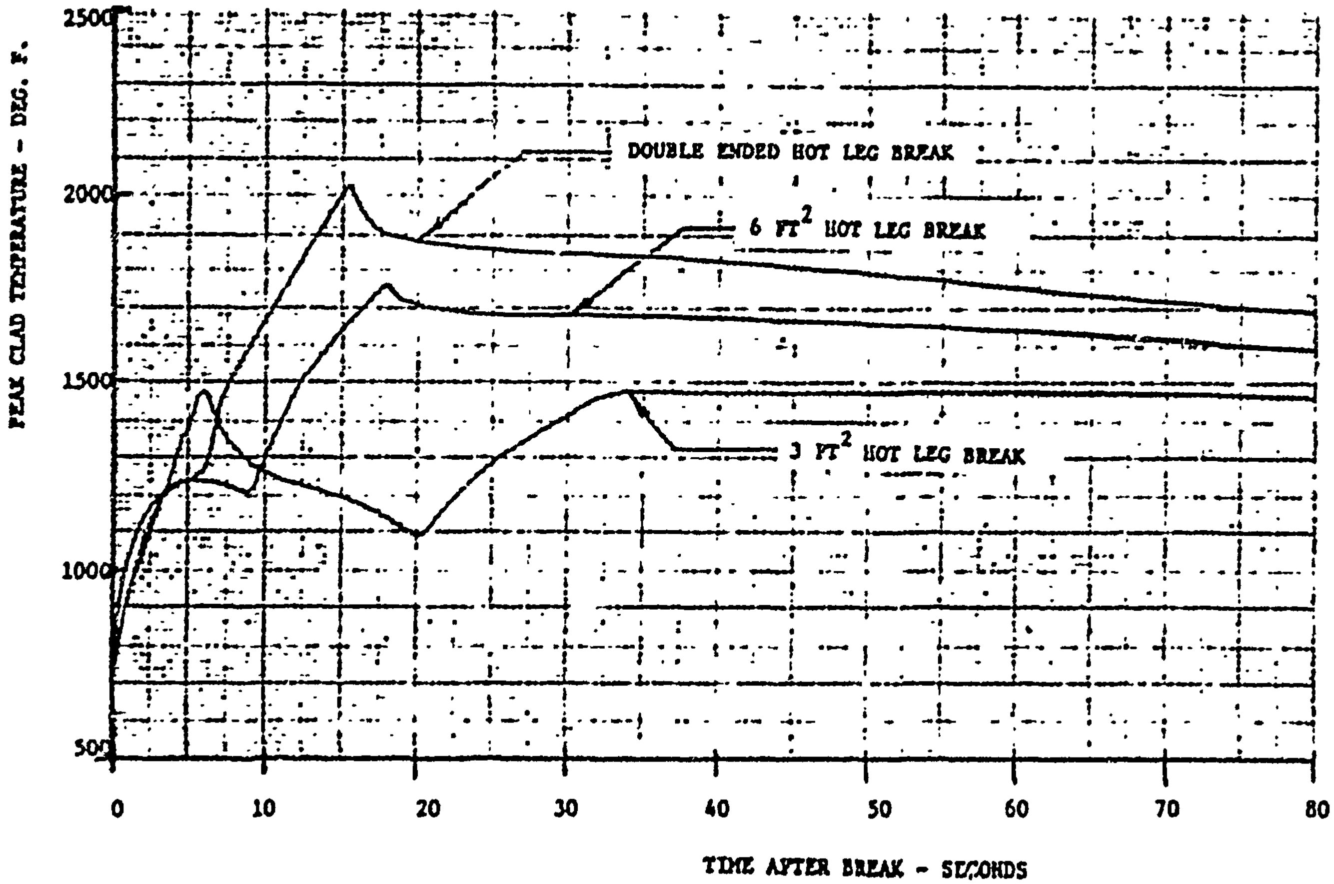


FIGURE 14.3.2-27

.5 FT^2 HOT LEG BREAK -2758 MWt

CLAD TEMPERATURE TRANSIENT VS TIME AFTER BREAK

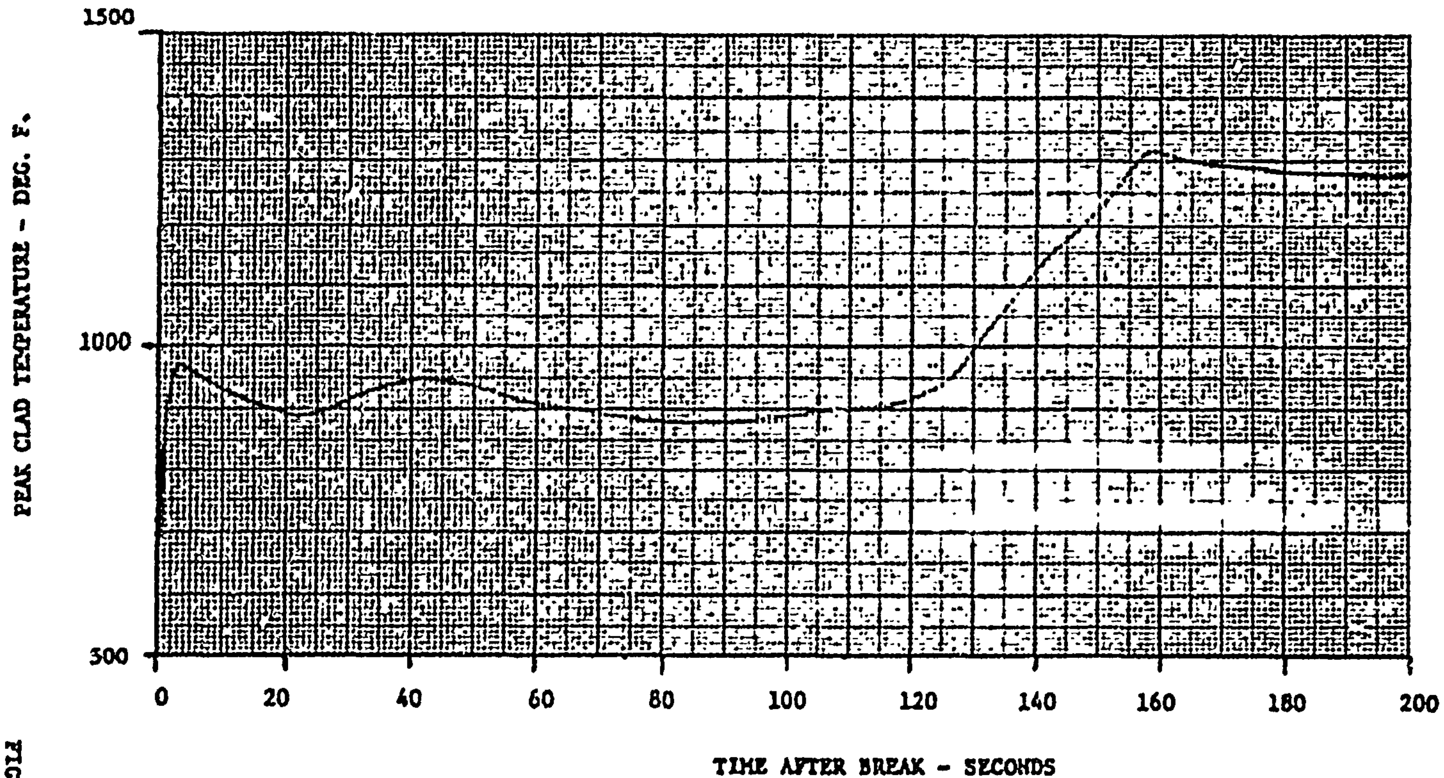


FIGURE 14.3.2-28

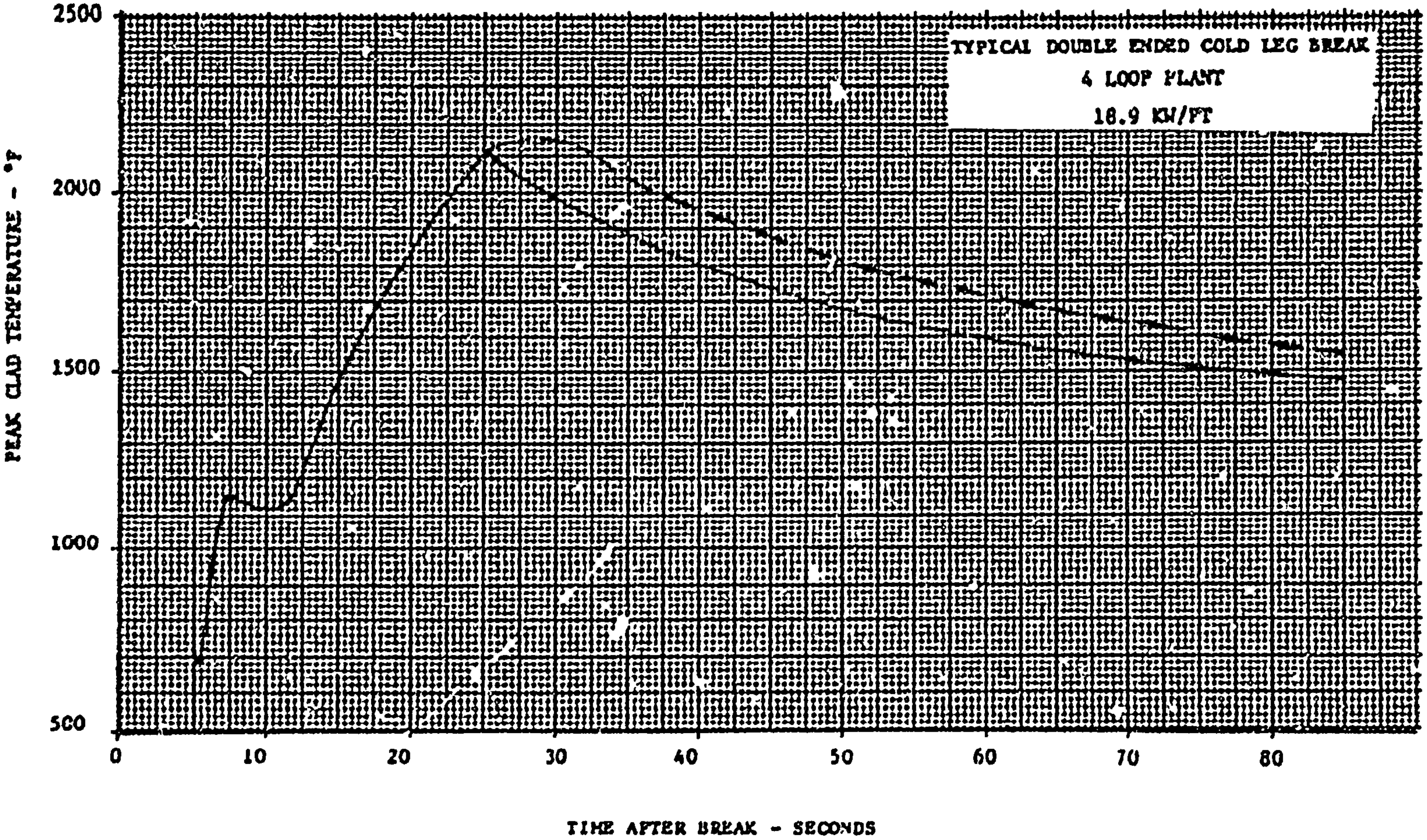


FIGURE 14.3.2-28a

SAFETY INJECTION PUMP CURVES

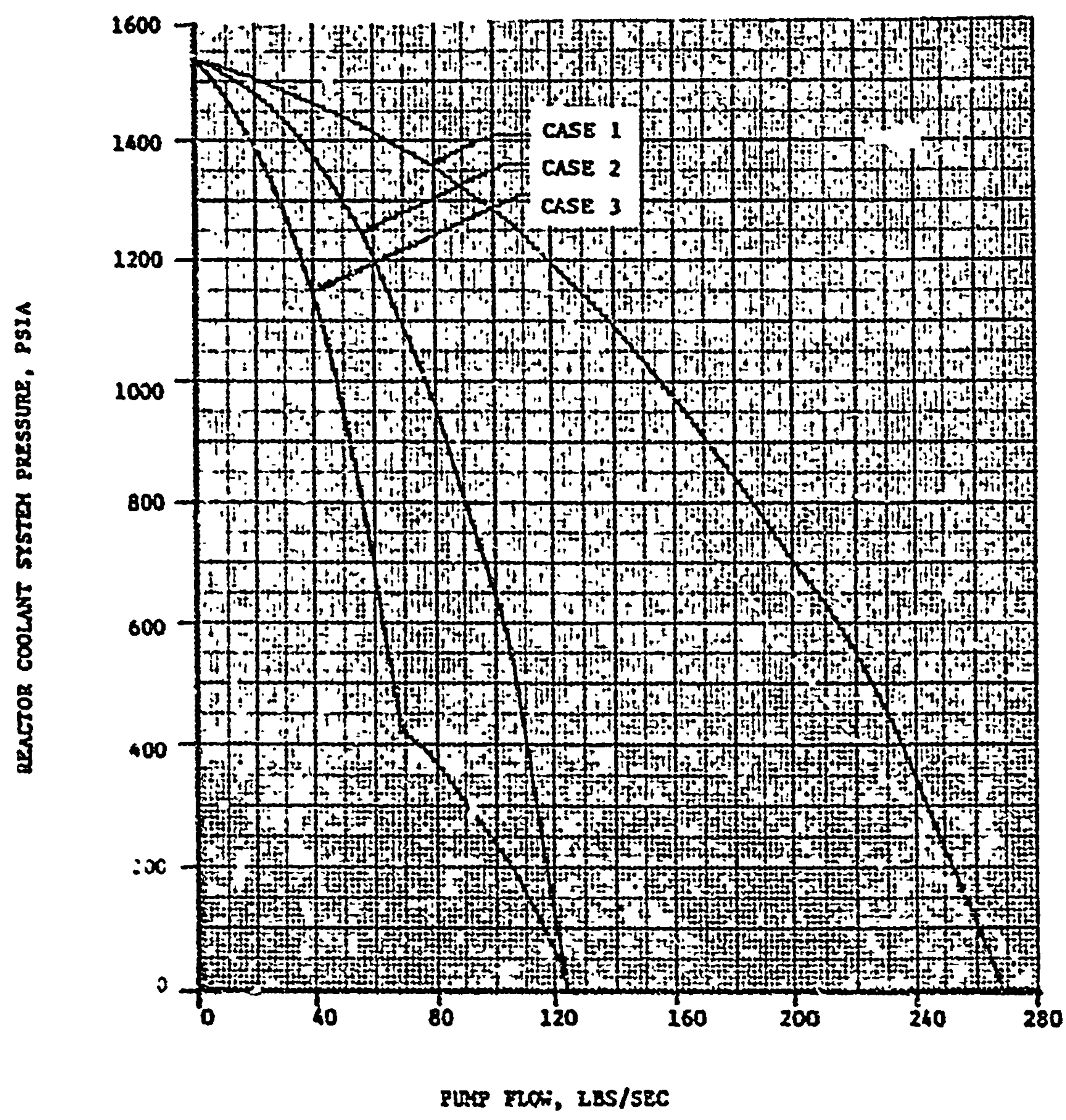


FIGURE 14.3.2-29

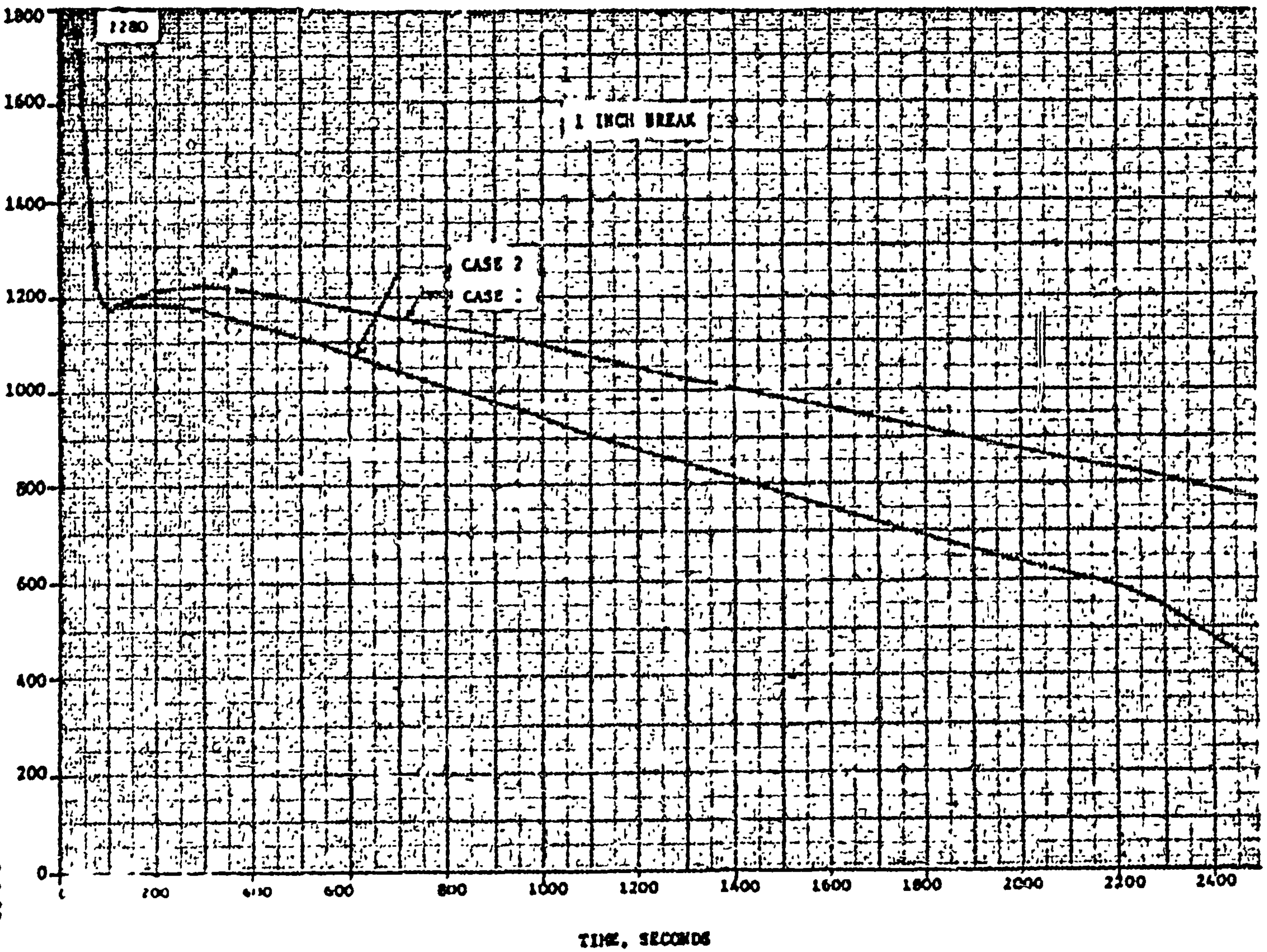


FIGURE 14.3.2-30

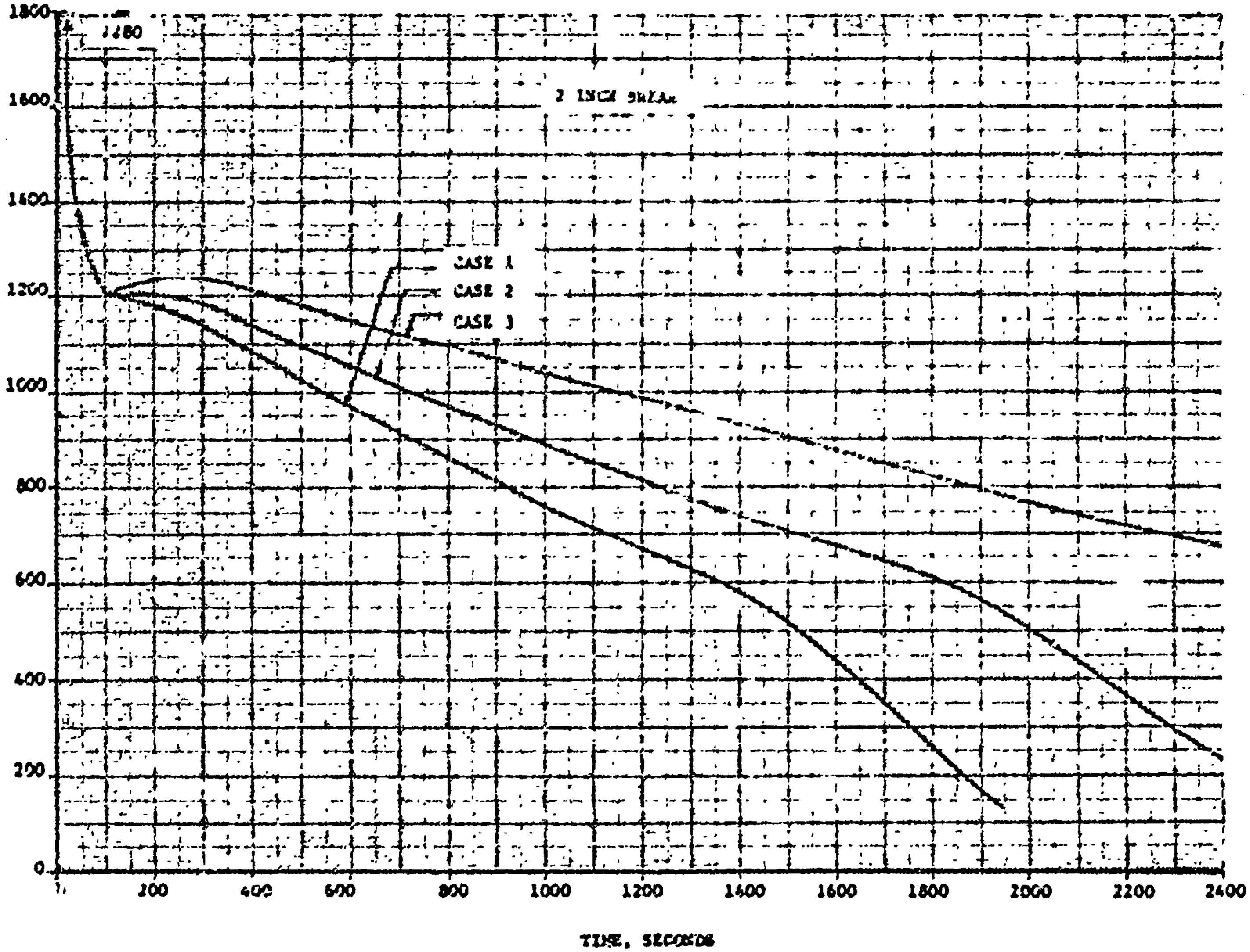


FIGURE 14.3.2-31

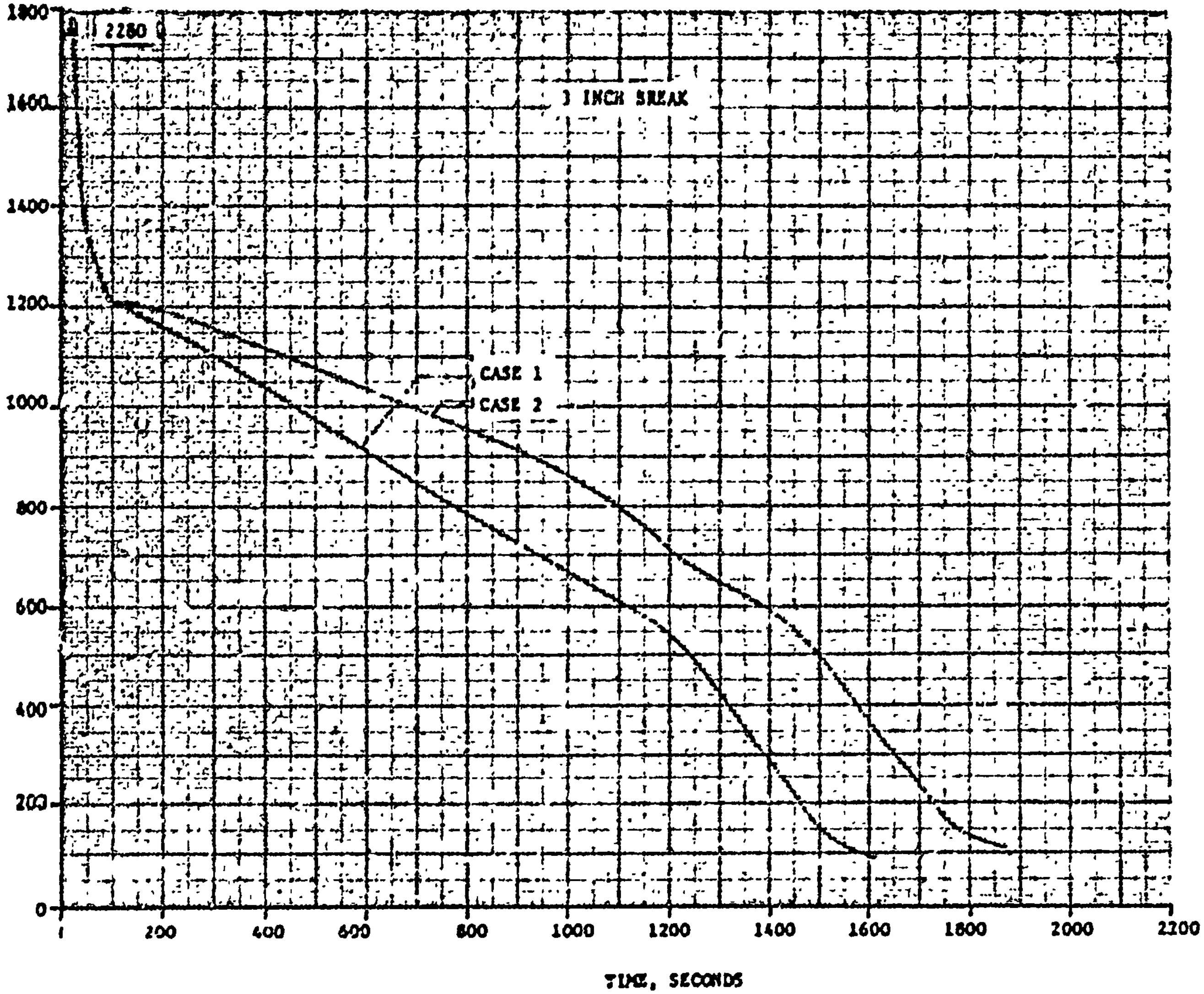


FIGURE 14.3.2-32

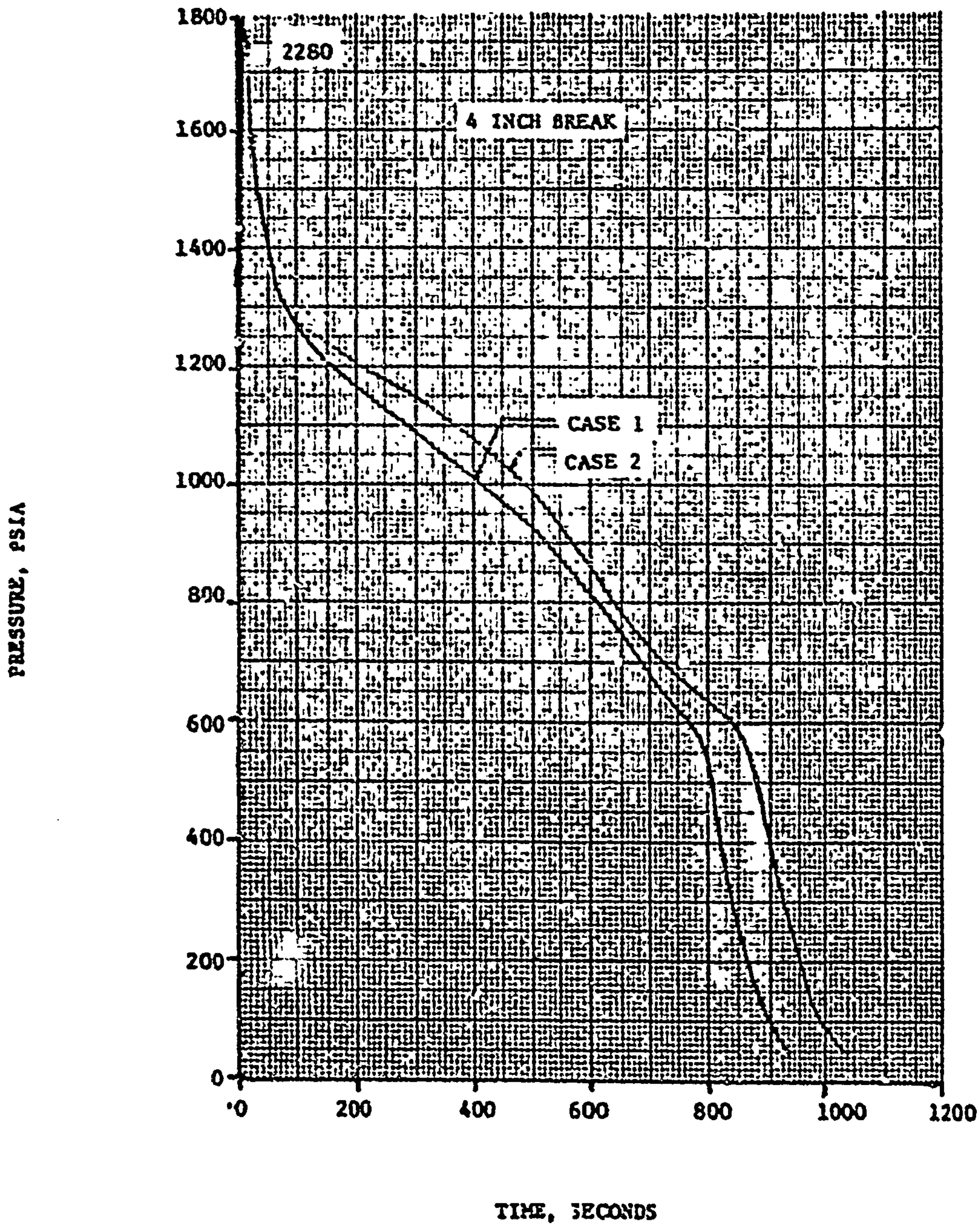


FIGURE 14.3.2-33

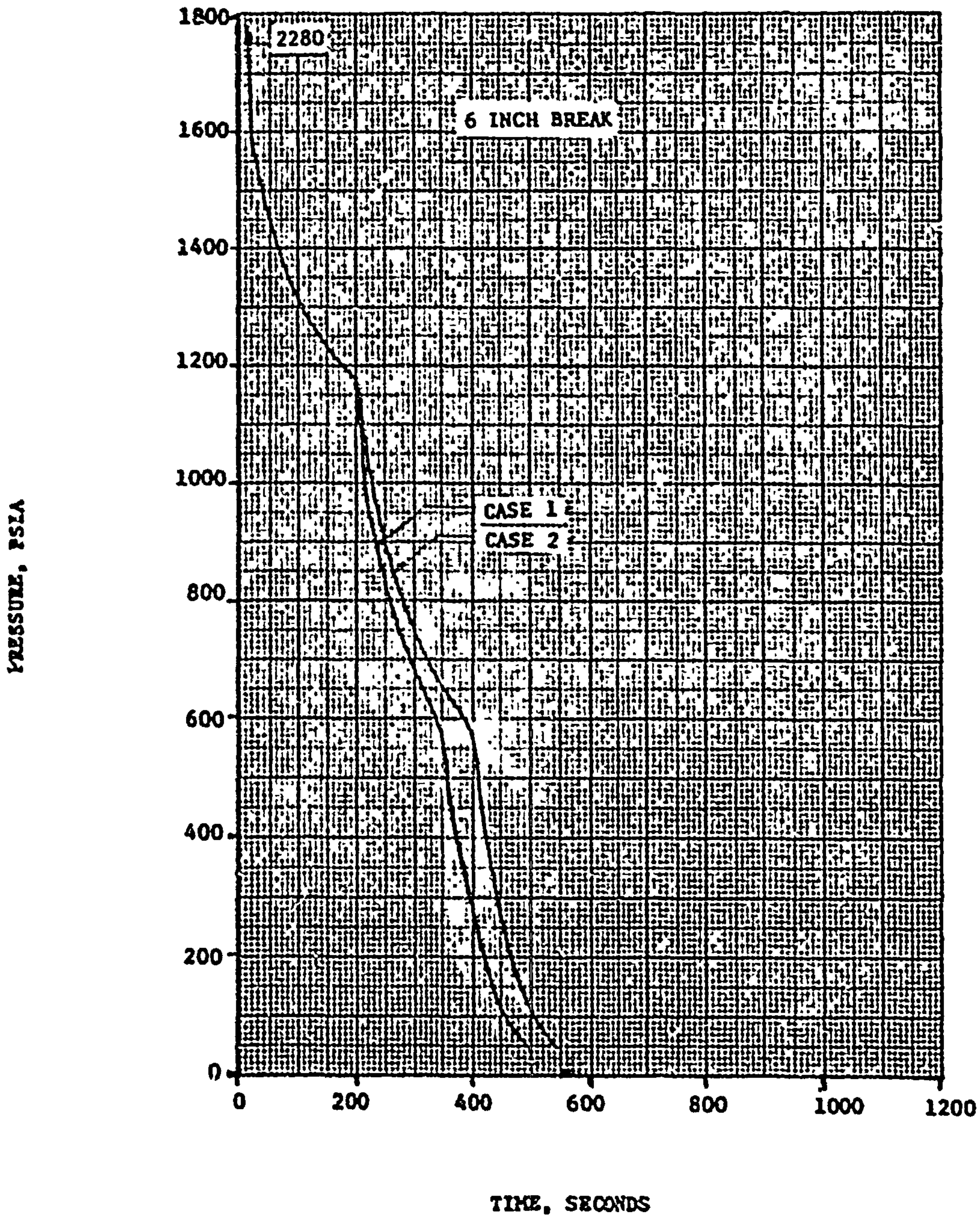


FIGURE 14.3.2-34

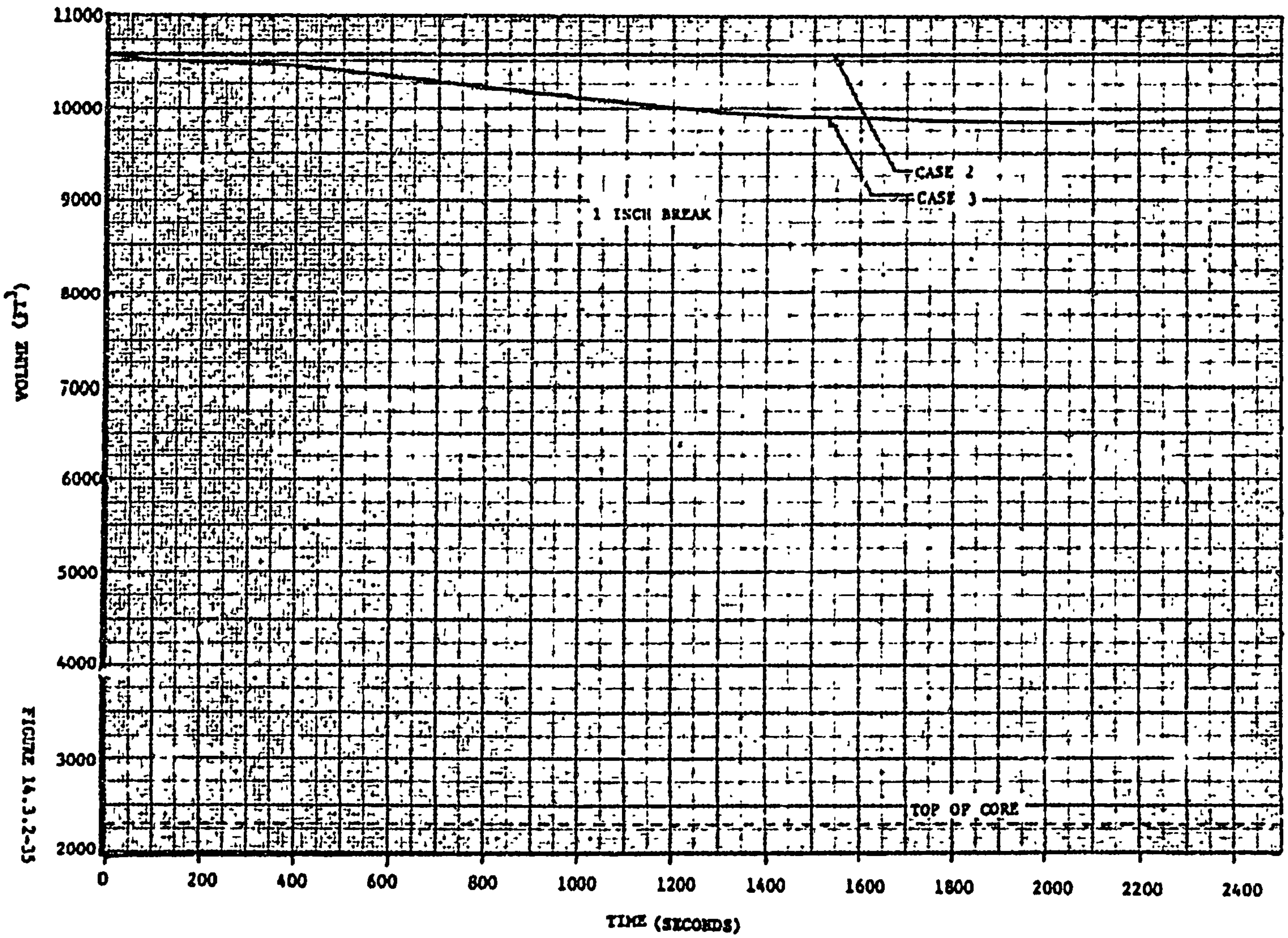


FIGURE 14.3.2-15

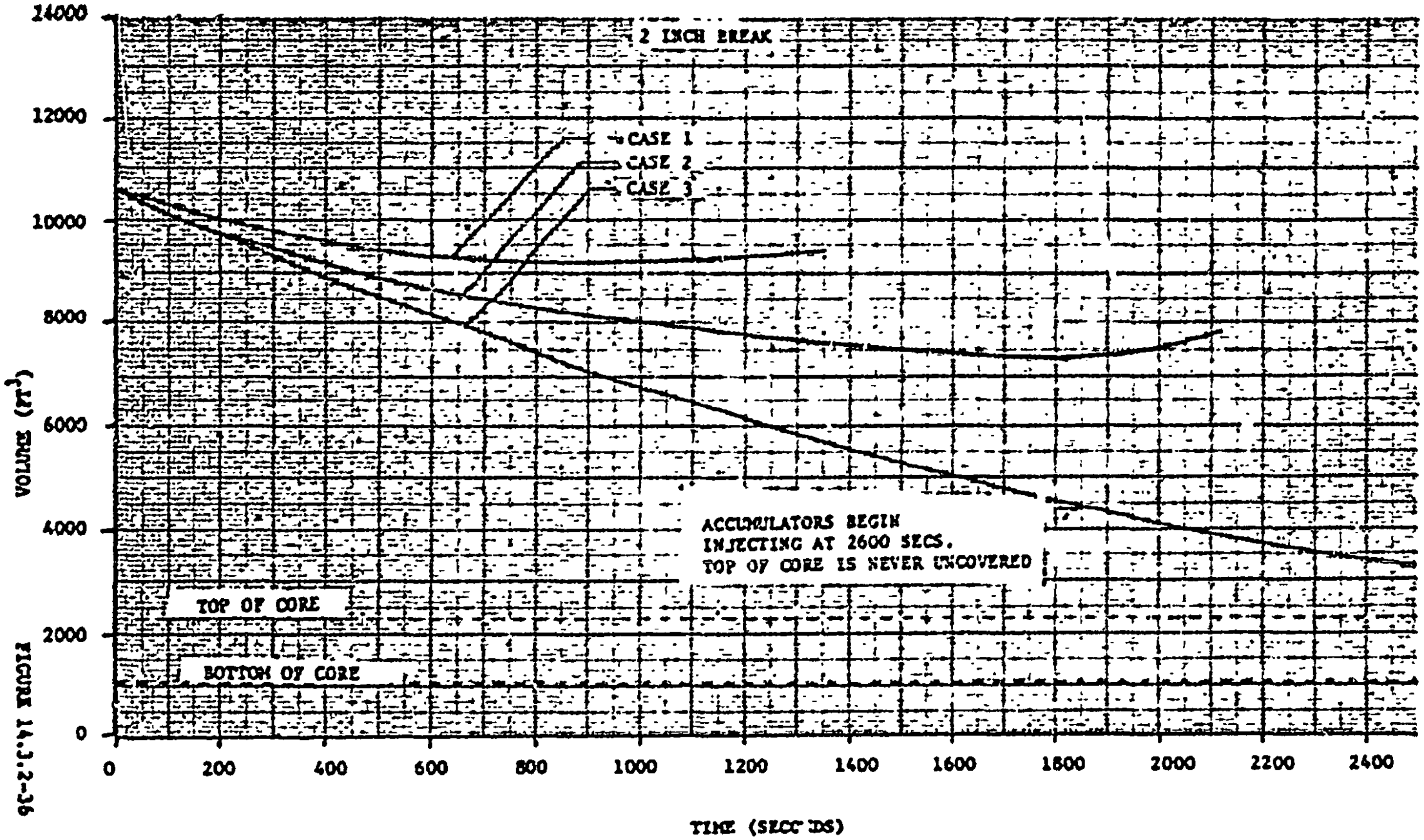


FIGURE 14.3.2-36

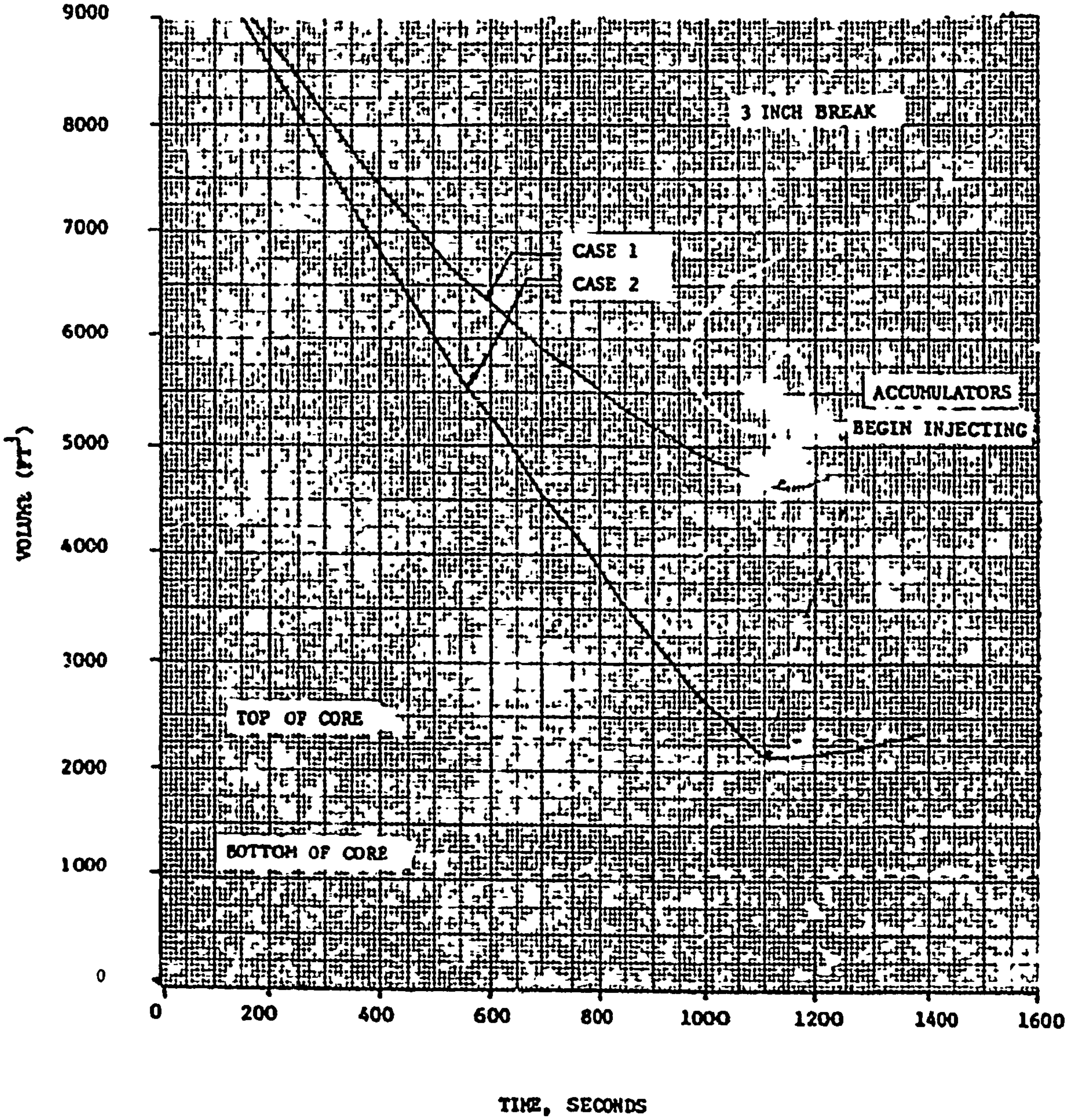


FIGURE 14.3.2-37

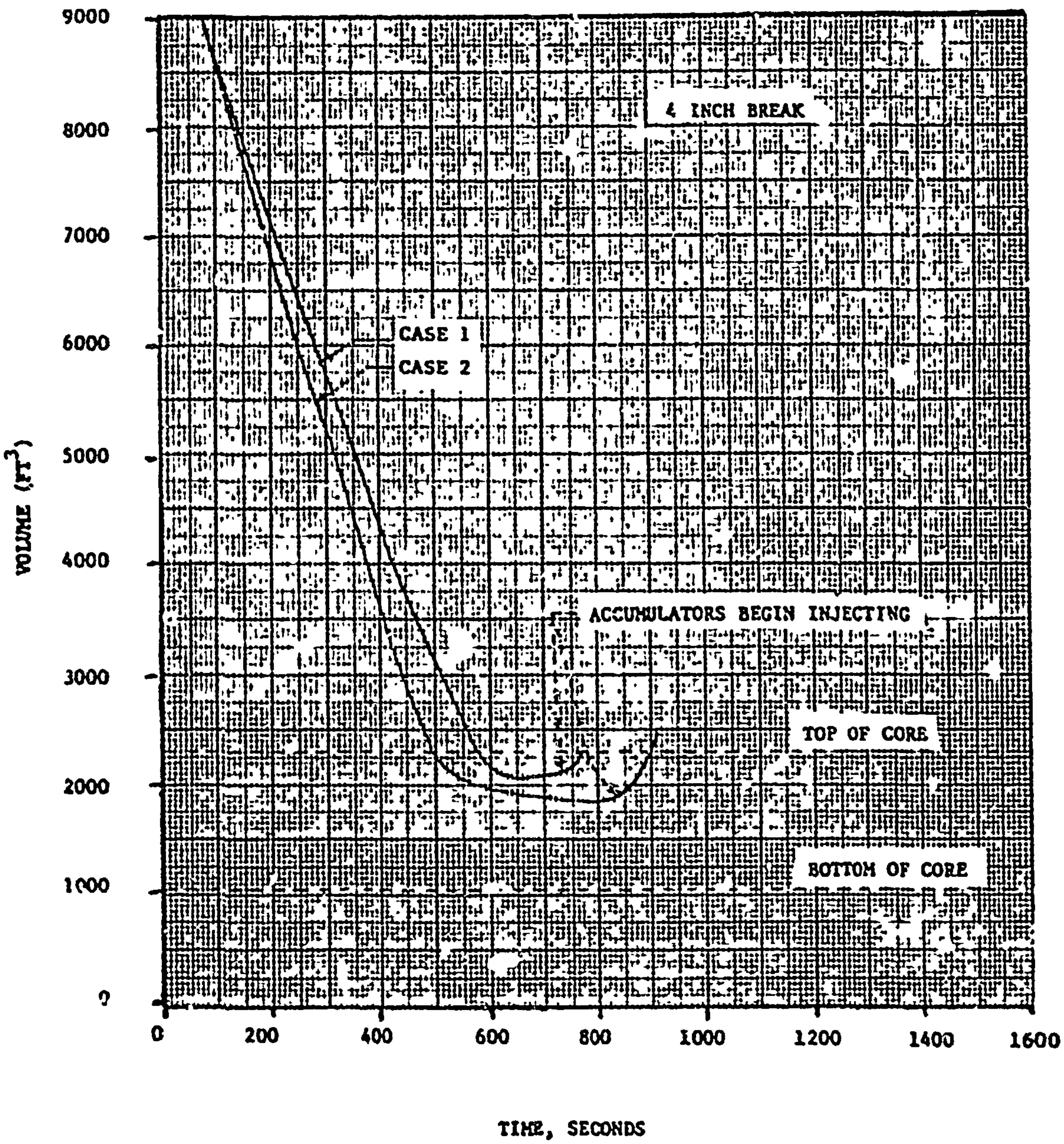


FIGURE 14.3.2-38

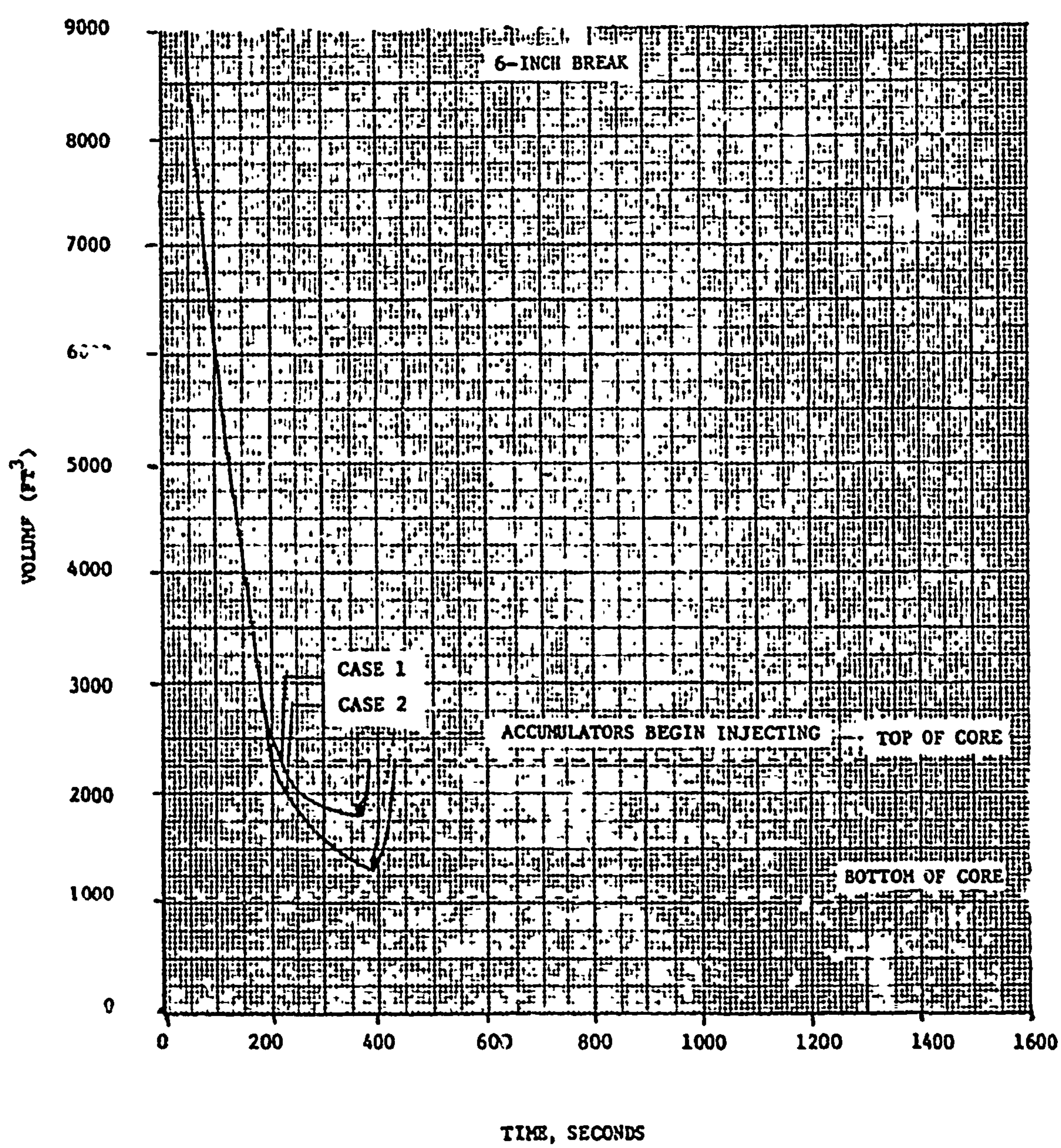


FIGURE 14.3.2-39

6 INCH BREAK

SAFETY INJECTION
TWO HIGH HEAD PUMPS, 3/4 FLOW

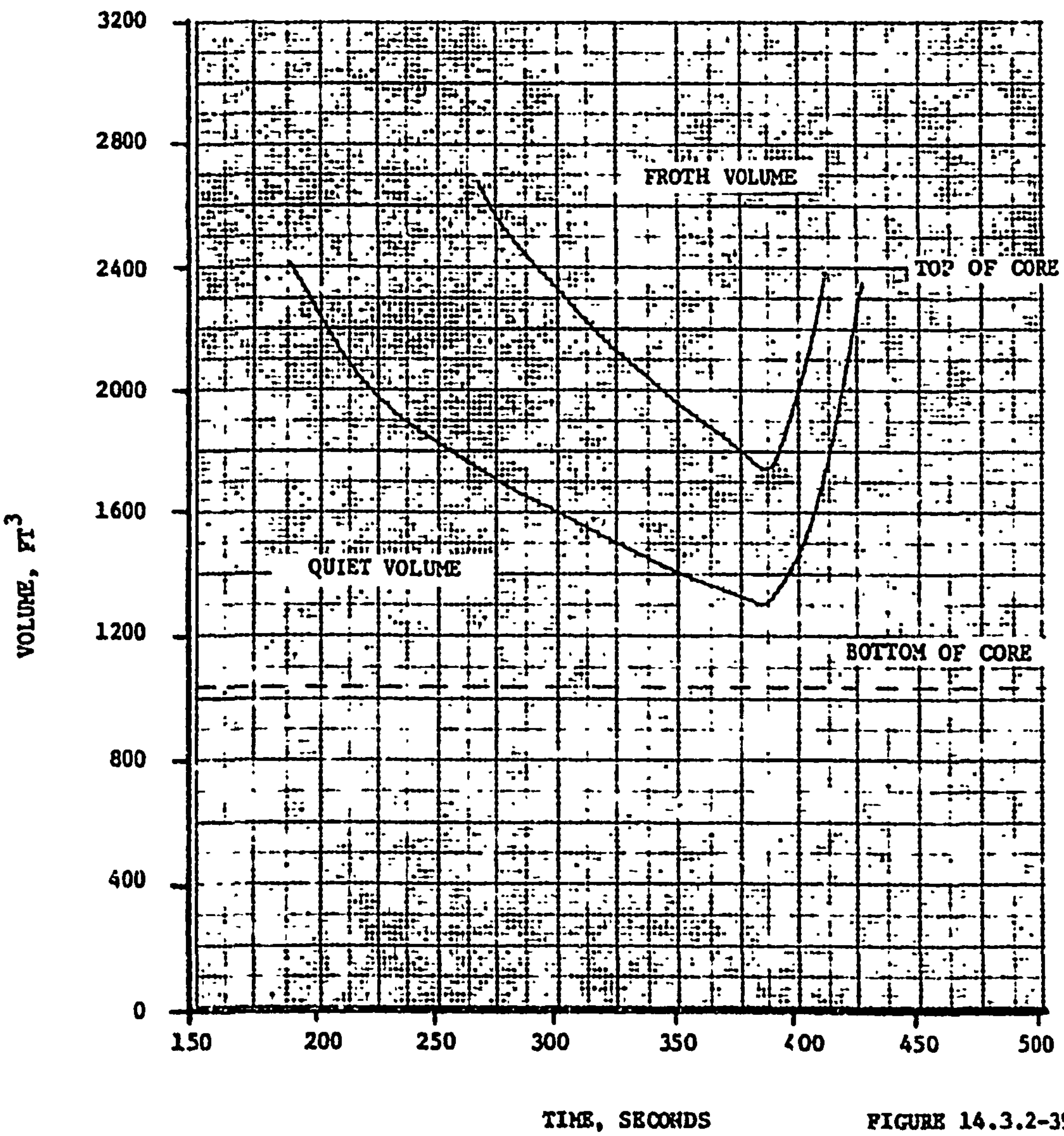


FIGURE 14.3.2-39A

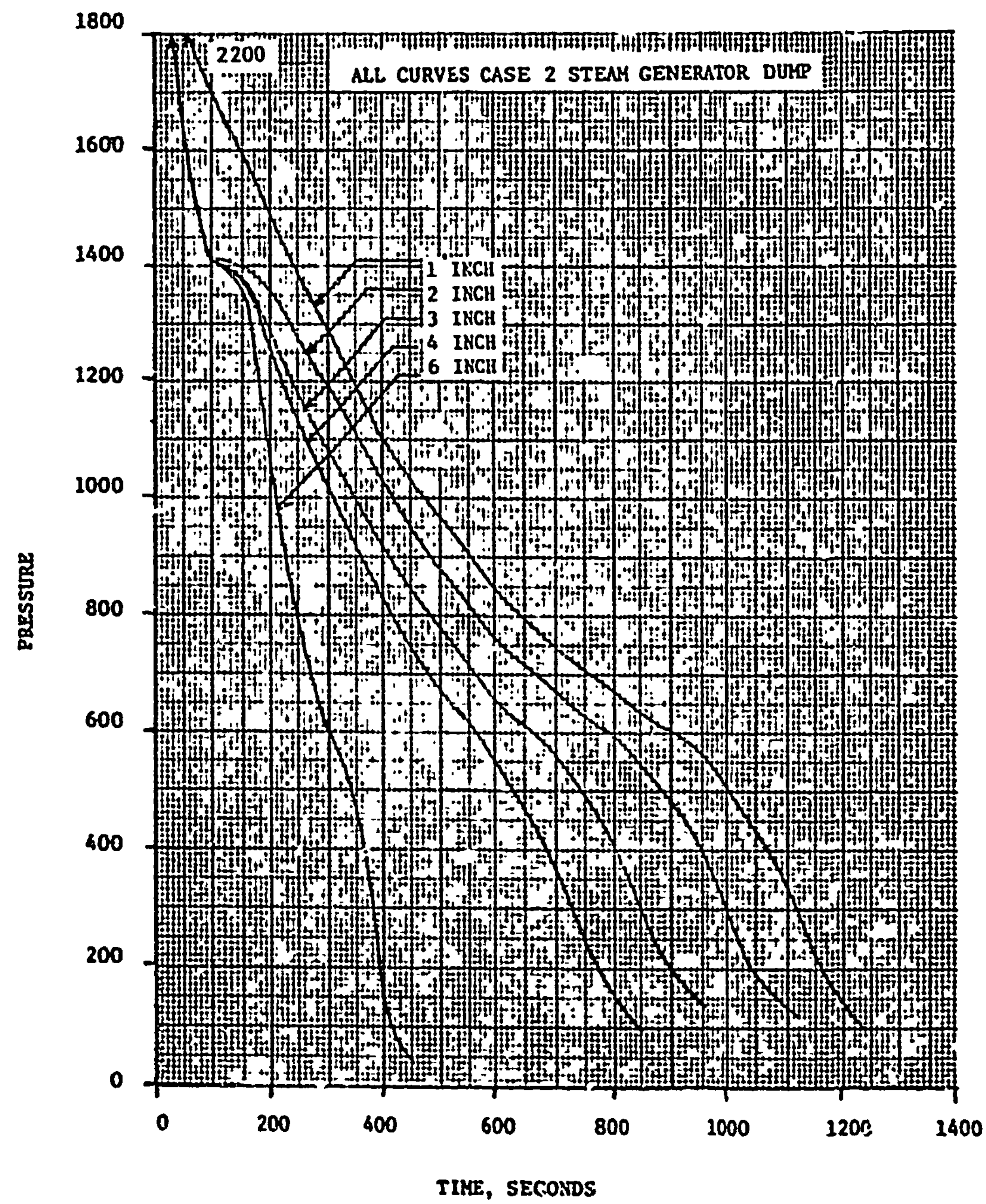


FIGURE 14.3.2-40

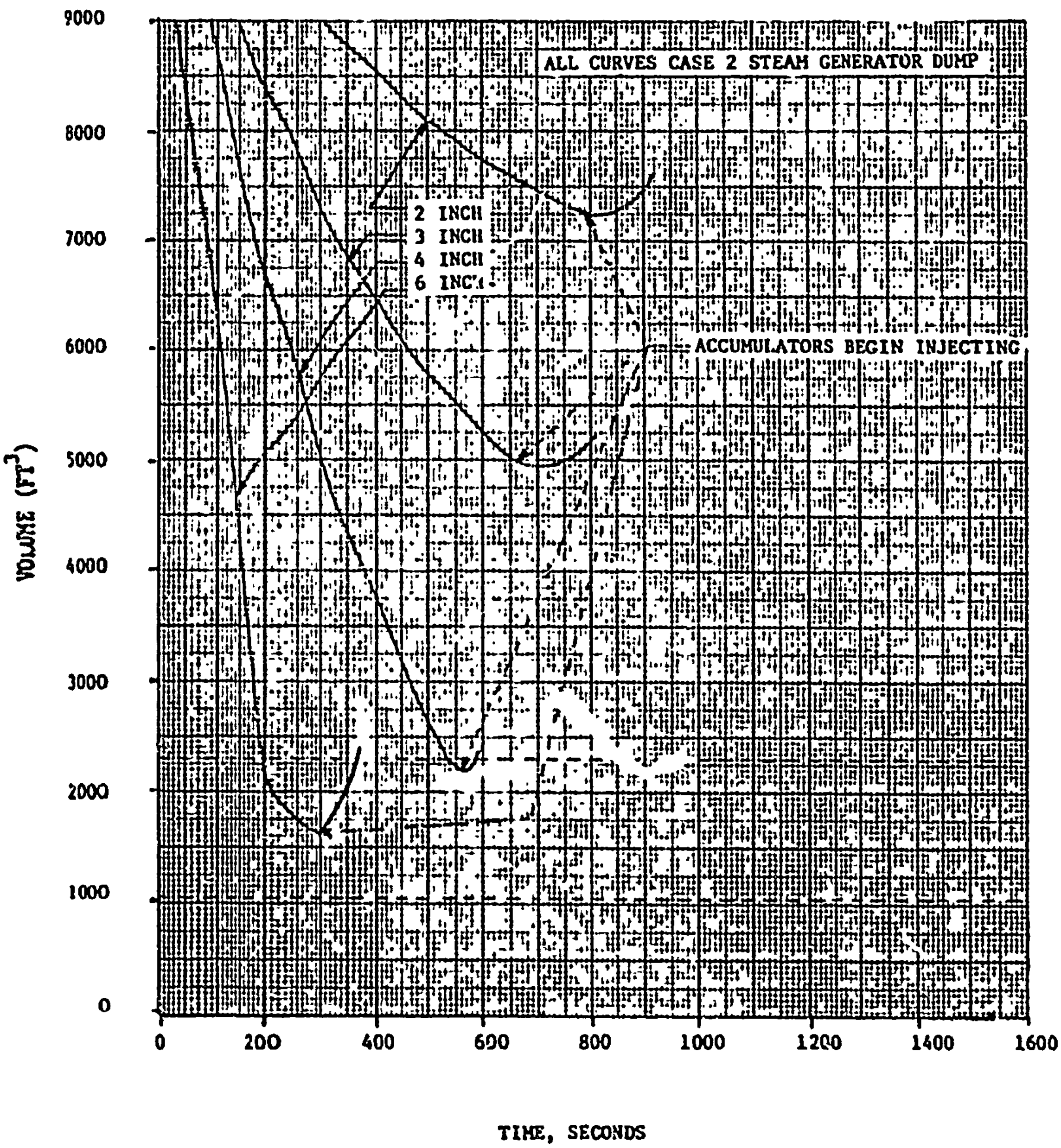


FIGURE 14.3.2-41

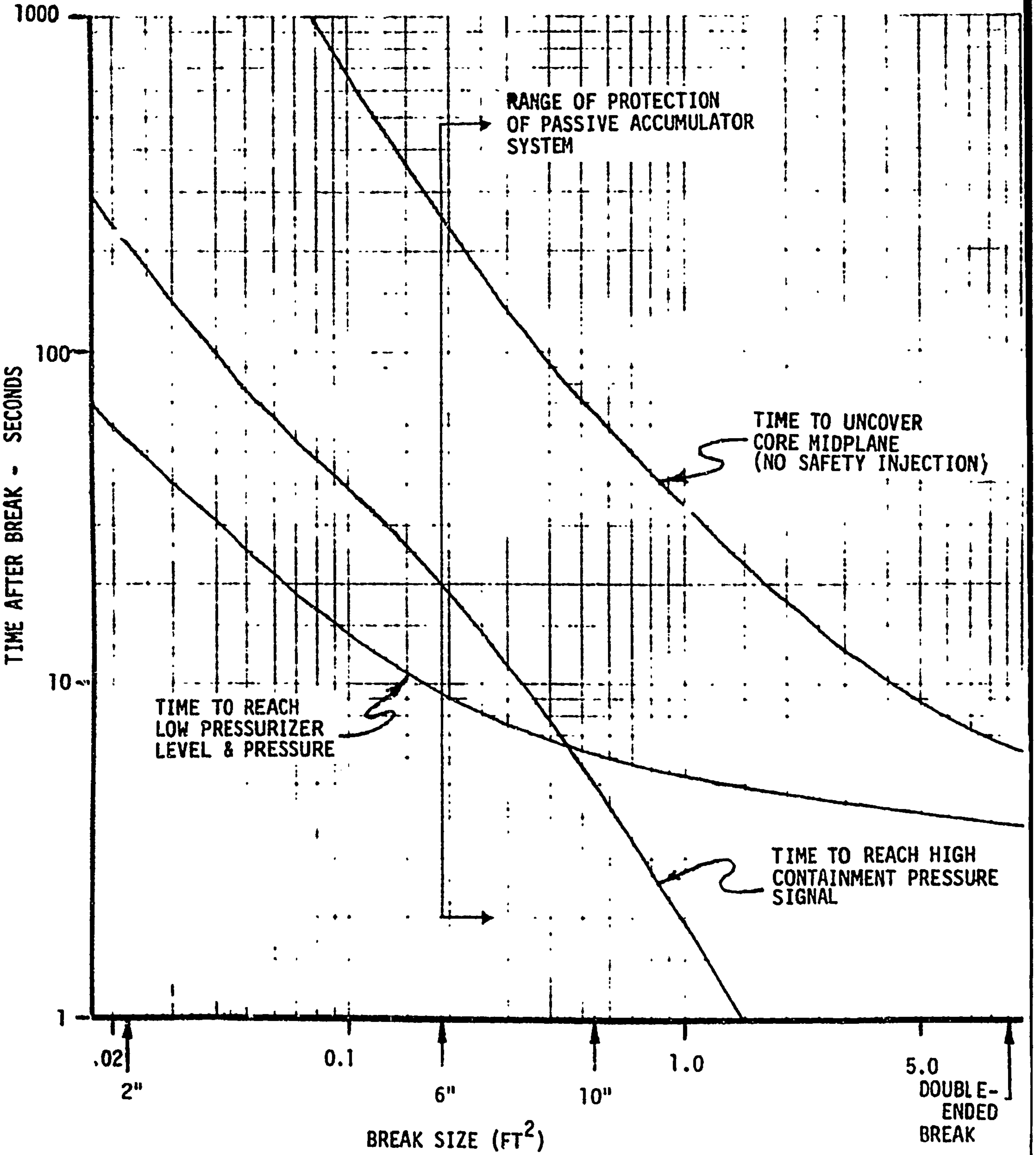


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FIGURE 14.3.2-42
TIME TO INITIATE SAFETY INJECTION

14.3.3 CORE AND INTERNALS INTEGRITY ANALYSIS

Internals Evaluation

The forces exerted on reactor internals and core, following a loss-of-coolant accident, are computed by employing the BLODWN-1 digital computer program developed for the space-time-dependent analysis of multi-loop PWR plants.

Design Criteria

The basic requirement of any loss-of-coolant accident, including the double-ended severance of a reactor coolant pipe, is that sufficient integrity be maintained to permit the safe and orderly shutdown of the reactor. This implies that the core must remain essentially intack and deformation of internals must be sufficiently small so that primary loop flow, and particularly, adequate safety injection flow is not impeded.

The ability to insert control rods to the extent necessary to provide shutdown following the accident must be maintained. Maximum allowable deflection limitations are established for those regions of the internals that are critical for plant shutdown.

The allowable and no loss of function deflection limits under dead loads plus the maximum potential earthquake and/or blowdown excitation loads are presented in Table 14.3.3-1. These limits have been established by correlating experimental and analytical results.

Blowdown and Force Analysis

BLODWN-1 is a digital computer program for calculation of pressure, velocity, and force transients in reactor primary coolant systems during the subcooled portion of blowdown caused by a loss-of-coolant accident.

During this phase of the accident, large amplitude rarefaction waves are propagated through the system with the velocity of sound causing large differences in local pressures. As local pressures drop below saturation, causing formation of steam, the amplitudes and velocities of these waves drastically decrease. Therefore, the largest forces across the reactor internals due to wave propagation occur during the subcooled portion of blowdown.

Blowdown Model

The analytical model used in BLODWN-1 is the same as that of the WHAM computer program developed by Kaiser Engineers for the LOFT program.[5]

The program utilizes the exact solutions to the time dependent, one dimensional, compressible fluid flow equations in which the velocity of propagation of acoustic waves greatly exceeds the fluid velocity. Analytic solutions for the interior points of conduits of uniform flow-passage area are well known.[1,2] They predict the existence of compression and rarefaction waves which travel through the fluid with the velocity of sound. Fluid pressures and velocities at any given point in space are proportional to the local sums and differences, respectively, of the magnitude of the waves which travel in opposite directions.

Solutions at the boundaries of these uniform flow area conduits (which for convenience will be referred to as "legs") are obtained through application of the mass and energy conservation laws. The latter, in the case of orifices, bends, and sudden changes of flow area, accounts for hydraulic losses. Hydraulic losses due to friction are represented by equivalent orifices.

The boundary condition at the location of the system's rupture is in the form of a discharge flow equation. The discharge flow equation incorporates the best available fit to known data[3,4] on metastable flow of that fluid through short pipes and/or orifices, depending on the postulated rupture type.

A time-dependent rupture flow area is specified and approximated by a sequence of stepwise changes. Each step increase in the exit flow area generates a rarefaction wave as the compressed fluid escapes through the rupture. A train of waves is thus sequentially generated and sent upstream. When the waves encounter abrupt changes of flow passage area or branches to other "legs", they are both transmitted through and reflected from, such junctions with modified amplitudes. When reflected compression waves reach the rupture location they affect the discharge flow and generate new waves because of the change in the local pressure just upstream of the rupture.

Apart from calculations involving boundary conditions, BLODWN-1 assigns exact solutions to local fluid pressures and velocities throughout the system. Therefore, it does not suffer from the propagation of truncation errors and from numerical instabilities associated with the methods of analysis in which the time dependent differential equations representing the conservation laws are solved simultaneously by finite difference approximations.

BLODWN-1 utilizes the technique of braching on a one-dimensional flow system to approximate the actual three-dimensional conditions. This is accomplished by using ficticious tees at all junctions of the one-dimensional network of legs. For example, if local histories of fluid pressure on both sides of the thermal shield and the core barrel, as functions of distance from the inlet nozzle in both the axial and the circumferential direction are desired, a hydraulic network of circumferential and vertical legs is used to represent this annular flow region.

Comparison With Experimental Data

BLODWN-1 is an evolution of the program WHAM.[5] The only changes made were to provide graphical output and storage of results and incorporate a detailed treatment of a double-ended pipe rupture. The comparison of WHAM results with tests obtained by Phillips Petroleum Company during their semi-scale blowdown experiments is shown in Figure 14.3.3-1 and 14.3.3-2 which are reproduced from Reference 6. Since no changes have been made in the analysis, this comparison is equally valid for BLODWN-1.

Force Model

BLODWN-1 evaluates the pressure and velocity transients for a maximum of 4000 locations throughout the system. These pressure and velocity transients are stored as a permanent tape file and are made available to the program FORCE which utilizes a detailed geometric description in evaluating the loadings on the reactor internals.

Each reactor component for which force calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of:

1. The pressure differential across the element
2. Flow stagnation on, and unrecovered orifice losses across the element
3. Friction losses along the element

Input to the code, in addition to the BLODWN-1 pressure and velocity transients, includes the effective area of each element on which acts the force due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

RESPONSES OF REACTOR INTERNALS TO BLOWDOWN FORCES

Vertical Excitation

Structural Model and Method of Analysis

The response of reactor internals components due to an excitation produced by complete severance of a primary loop pipe is analyzed. Assuming a double-end pipe break occurs in a very short period of time, the rapid drop of pressure at the break produces a disturbance which propagates along the primary loop and excites the internal structure. The characteristics of the hydraulic excitation, combined with those of the structures affected, presents a unique dynamic problem.

The internal structure is simulated by a multi-mass system connected with springs and dashpots representing the viscous damping due to structural and impact losses. The gaps between various components, as well as Coulomb type of friction, is also incorporated into the overall model. Since the fuel elements in the fuel assemblies are kept in position by friction forces originating from the preloaded fuel assembly grid fingers, any sliding that occurs between the fuel rods and assembly is considered as Coulbomb type of friction. A series of mechanical modes of local structures were developed and analyzed so that certain basic nonlinear phenomena previously mentioned could be understood. Using the results of these models, a final eleven-mass model is adopted to represent the internals structure under vertical excitation. Figure 14.3.3-3 is a schematic representation of the internals structures. The eleven-mass model is shown in Figure 14.3.3-4. A comparison between Figure 14.3.3-3 and 14.3.3-4 shows the parallel between the plant and the mode. The modeling is conducted in such a way that uniform masses are lumped into easily identifiable discrete masses while elastic elements are represented by springs. A legend for the different masses is given in Table 14.3.3-2. The masses are readily recognized as Items W1 through W11. The core barrel and the lower package are easily discernable. The fuel assemblies have been segregated into two groups. The majority of the fuel mass, W4, is indirectly connected to the deep beam structure represented by mass W8. There is also a portion of the fuel mass, W6, which connects through the long columns to the top plate. The stiffness of the top plate panels is represented by K10 and the stiffness of the upper core plate panels is represented by K8. The hold down spring, K1, is bolted-up between the flange of the deep beam structure and the core barrel flange with the preload, P1. After preloading the hold down spring, a clearance, G1, exists between the core barrel flange and the solid height of the hold down spring. Within the fuel assemblies, the fuel elements W4 and W6 are held in place by frictional contact with the grid spring fingers. Coulomb damping is provided in the analysis to represent this frictional restraint.

The analytical model is also provided with viscous terms to represent the structural damping of the elastic elements. The viscous dampers are represented by C1 through C11.

Restrictions are placed on the displacement amplitudes by specifying the free travel available to the dynamic masses. Available displacements are designated by symbols G1 through G8.

The displacements are tested during the solution of the problem to see if the available travel has been achieved. When the limit of travel has been attained, stops are engaged to arrest further motion of the dynamic masses. The stops or snubbers are designated by the symbols S1 through S11.

Contact with the snubbers results in some damping of the motion of the model. The impact damping of the snubbers is represented by the devices D1 through D11.

During the assembly of the reactor, bolt-up of the closure head presets the spring loading of the core barr[illegible] the spring loading on the fuel assemblies. Since the fuel assemblies in the [illegible]l have been segregated into two groups, two preloads values are provided in the analysis. Preload values P1, P3, and P5 represent the hold down spring preload on the core barrel and the top nozzle springs preload values on the fuel assemblies.

The formulation of the transient motion response problem and digital computer programming were performed. The effects of an earthquake vertical excitation are also incorporated into the program.

In order to program the multi-mass system, the appropriate spring rates, weights, and forcing function for the various masses were determined. The spring rates and weights of the reactor components are calculated separately for each plant. The forcing functions for the masses are obtained from the FORCE program described in the previous section. It calculates the transient forces on reactor internals during blowdown using transient pressures and fluid velocities.

For the blowdown analysis the forcing functions are applied directly to the various internal masses.

For the earthquake analysis of the reactor internals, the forcing function, which is simulated earthquake response, is applied to the multi-mass system at the ground connections (the reactor vessel). Therefore, the external excitation is transmitted to the internals through the springs at the ground connections.

Results

Analysis is being performed for 1 msec, 5 msec, and 20 msec hot leg and cold leg breaks. The response of the structure to this type of excitation indicates that the vertical motion is irregular with peaks of very short duration. The deflections and motion of some of the reactor components are limited by the solid height of springs as is the case of the hold down spring located above the barrel flange.

The internals behave as a highly nonlinear system during the vertical oscillations produced by the blowdown forces. The nonlinearities are due to the Coulomb frictional forces between grids and rods, and to gaps between components causing dicontinuities in force transmission. The frequency response is consequently a function not only of the exciting frequencies in the system, but also to the amplitude. Different break conditions excite different frequencies in the system. This situation can be seen clearly when the response under blowdown forces is compared with the one due to vertical seismic acceleration. Under seismic excitation, the system behaves practically linearly because the component's motion is not sufficient to cause closing of the various gaps in the structure or slippage in the fuel rods.

Under certain blowdown excitation conditions, the core moves upward, touches the core plate, and falls down on the lower structure causing oscillations in all the components. During the time that the oscillations occur and depending on its initial position, the fuel rods slide on the fuel assembly. The response shows that the case could be represented as two large vibrating masses (the core and the barrel) and the rest of the system oscillating at an average frequency of 114 cps with respect to the barrel and the core. The lower structure is oscillating at an average frequency of 114 cps with respect to the barrel and the core. The upper flange, with respect to the barrel, oscillates at a frequency of 69 cps. The same structure under seismic excitation shows a "natural" frequency of approximately 25 cps; in this case, the difference can be explained because after a hot leg break the upper plenum of the reactor compresses the flange downward increasing the stiffness of the structure. The lower structure shows "natural" frequencies of approximately 105 cps when the core is in contact and approximately 120 cps when the core is lifted. From the several cases analyzed, a frequency of 300 cps has been seen present very frequently in the upper package after the fuel assembly touches the upper core plate. The same structure shows a "natural" frequency of 25 cps when the structure is excited by the seismic accelerations.

The effect of damping has also been considered and it can be seen that the higher frequencies disappear rapidly after each impact or slippage.

The results of the computer program give not only the frequency response of the components, but also the maximum impact force and deflections. From these results, the stresses are computed using the standard "Strength of Material" formulas. The impact stresses are obtained in an analogous manner using the maximum forces seen by the various structures during impact.

Upper Package and Guide Tubes

The most severe case, represented by a hot leg break, shows the core lifting and contacting the upper core plate after displacing 0.36 inch. The local deformation of the upper core plate between the support columns is of the order of 0.026 inch and is caused by a 0.002 sec. contact between fuel assemblies and upper core plate structure. This deformation is insufficient to cause the plate to contact the guide tubes, since the clearance between plate and guide tube is 0.100 inch.

Fuel Assembly Thimbles

When the core moves vertically touching the upper and lower structures, the thimbles are subjected to impact stresses. These stresses are obtained from the maximum dynamic impact forces on the fuel assemblies. The results are compared with the buckling loads to assure that the cross section distortion doesn't exceed the allowable limits. Preliminary results show that control rod insertion is not jeopardized. Final results, will be submitted when available.

Transverse Excitation

Core Barrel

The hydraulic pressure transients caused by a loss of coolant accident with the break occurring in the hot leg are calculated for 1, 5, and 20 milisecond break times. The resulting loading on the upper core barrel is represented by a dynamic uniformly distributed compressive pressure wave.

The dynamic stability of the upper core barrel is analyzed. The maximum compressive pressure wave is well below the critical value to produce buckling of the upper core barrel. This is shown by the comparison between $P_{o\ max}$ and P_{crit} in Table 14.3.3-3

In addition, the quantitative dynamic response of the upper core barrel is studied for the worst blowdown break time and found to be neglibible. The maximum deflections, stresses, and pressure pulses for the break times are also shown in Table 14.3.3-3.

The response to the initial pressure wave was obtained by taking the pressure as a periodic function of time from the hydraulic analysis with a period equal to twice the impulse time and using one half cycle. The initial response thus is an upper bound since the actual pressure oscillations occur at essentially the same frequency with decaying amplitude.

Under the transient pressure conditions resulting from a loss of coolant accident in a cold leg initially the reactor core barrel is subjected to non-axisymmetric internal pressure waves. The initial loading condition is followed by oscillating pressure waves on the core barrel which are both time and space dependent.

In general, there are two possible modes of dynamic response of the core barrel. One mode is the beam response mode of the core barrel resulting from the non-self-equilibriating circumferential component of the pressure forcing action. This response mode is analyzed utilizing shear beam theory since the core barrel is a statically determinate elastic system in the beam mode. The beam mode of core barrel response is conservatively analyzed by comparing the excitation frequencies to the natural frequencies of the core barrel to establish the dynamic response amplification. The dynamic response analysis of the beam mode of the core barrel is conservative since the inertia and stiffness of the surrounding fluid media are neglected. In fact, in addition to decreasing the natural frequencies of the elastic system, the fluid environment acts as a damper and prohibits the formation of standing wave vibration modes of the elastic system. In the absence of standing wave vibration modes, the response of the core barrel at resonant conditions is finite as opposed to theoretically infinite as predicted by the in-vacuo equations without damping.

The second possible response mode of the core barrel is as a shell, predominantly in the ring modes with the formation of only one axial wave. The "ring" modes of shell vibration involve both the membrane and bending components of loads on the shell, with bending becoming predominant as the number of circumferential waves increases. Thus, the ring vibration modes are analyzed including both bending and membrane terms. The dynamic response is then determined by comparing the pressure loading oscillation frequencies to the natural frequencies as a shell.

As for the beam mode, the shell analysis of the core barrel is conservative by virtue of neglecting the effect of the water in limiting the response of the core barrel. The analysis is still in progress and the results will be submitted when available.

Guide Tubes

The guide tubes are studied applying the blowdown forces to the structures and calculating the resulting deflections. The guide tubes are considered as being the elastically supported at the upper plate and simply supported at the lower end with variable cross section. Consideration is given to the frequencies and amplitudes of the forcing function and the response is computed to assure that the deflections do not prevent shutdown of the reactor.

The results of the analysis will be submitted when available.

<u>Analysis of Effects of Loss of Coolant and Safety Injection on the Reactor Vessel</u>

The analysis of the effects of injecting safety injection water into the reactor coolant system following a postulated loss of coolant accident are being incorporated into a WCAP report to be submitted to the AEC.

For the reactor vessel, three modes of failure are considered including the ductile mode, brittle mode and fatigue mode.

a) Ductile Mode - the failure criterion used for this evaluation is that there shall be no gross yielding across the vessel wall using the material yield stress specified in Section III of the ASME Boiler and Pressure Vessel Code. The combined pressure and thermal stresses during injection through the vessel thickness as a function of time have been calculated and compared to the material yield stress at the times during the safety injection transient.

   The results of the analyses showed that local yielding may occur in approximately the inner 12 per cent of the base metal and in the cladding.

b) Brittle Mode - the possibility of a brittle fracture of the irradiated core region has been considered from both a transition temperature approach and a fracture mechanics approach.

The failure criteria used for the transition temperature evaluation is that a local flaw cannot propagate beyond any given point where the applied stress will remain below the critical propagation stress at the applicable temperature at that point.

The results of the transition temperature analysis showed that the stress-temperature condition in the outer 65 per cent of the base metal wall thickness remains in the crack arrest region at all times during the safety injection transient. Therefore, if a defect were present in the most detrimental location and orientation (i.e., a crack on the inside surface and circumferentially directed), it could not propagate any further than approximately 35 per cent of the wall thickness, even considering the worst case assumptions used in this analysis.

The results of the fracture mechanics analysis, considering the effects of water temperature, heat transfer coefficients and fracture toughness of the material as a function of time, temperature and irradiation will be included in the report. Both a local crack effect and a continuous crack effect have been considered with the latter requiring the use of a rigorous finite element axisymetric code.

c) Fatigue Mode - the failure criterion used for the failure analysis was the one presented in Section III of the ASME Boiler and Pressure Vessel Code. In this method the piece is assumed to fail once the combined usage factor at the most critical location for all transients applied to the vessel exceeds the code allowable usage factor of one.

The results of this analysis showed that the combined usage factor never exceeded 0.2, even after assuming that the safey injection transient occurred at the end of plant life.

In order to promote a fatigue failure during the safety injection transient at the end of plant life, it has been estimated that a wall temperature of approximately 1100°F is needed at the most critical area of the vessel (instrumentation tube welds in the bottom head).

The design basis of the Safety Injection System ensures that the maximum Zircaloy cladding temperature does not exceed the Zircaloy-4 melt temperature. This is achieved by prompt recovery of the core through flooding, with the passive accumulator and the injection systems. Under these conditions a vessel temperature of 1100°F is not considered a credible possibility and the evaluation of the vessel under such elevated temperatures is for a hypothetical case.

For the ductile failure mode, such hypothetical rise in the wall temperature would increase the depth of local yielding in the vessel wall.

The results of these analyses show that the integrity of the reactor vessel is never violated.

The safety injection nozzles have been designed to withstand ten postulated safety injection transients without failure. This design and associated analytical evaluation was made in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code.

The maximum calculated pressure plus thermal stress in the safety injection nozzle during the safety injection transient was calculated to be approximately 50,900 psi. This value compares favorably with the code allowable stress of 80,000 psi.

These ten safety injection transients are considered along with all the other design transients for the vessel in the fatigue analysis of the nozzles. This analysis showed the usage factor for the safety injection nozzles was 0.[illegible] which is well below the code allowable value of 1.0.

The safety injection nozzles are not in the highly irradiated region of the vessel and thus they are considered ductile during the safety injection transient.

The effect of the safety injection water on the fuel assembly grid springs has been evaluated and due to the fact that the springs have a large surface area to volume ratio, being in the form of thin strips, and are expected to follow the coolant temperature transient with very little lag hence, no thermal shock is expected and the core cooling is not compromised.

Evaluations of the core barrel and thermal shield have also shown that core cooling is not jeopardized under the postulated accident conditions.

TABLE 14.3.3-1

INTERNALS DEFLECTIONS UNDER ABNORMAL OPERATION
(Inches)

	Allowable Limit	No Loss-of-Function Limit
Upper Barrel, expansion/compression (to assure sufficient inlet flow areas/and to prevent the barrel from touching any guide tube to avoid disturbing the RCC guide structure).	3	6
Upper Package, axial deflection (to maintain the control rod guide structure geometry).	1	2
RCC Guide Tube, cross section distortion (to avoid interference between the RCC elements and the guides).	0.035	0.072
RCC Guide Tube, deflection as a beam (to be consistent with conditions under which ability to trip has been tested).	1.0	1.5
Fuel Assembly Thimbles, cross section distortion (to avoid interference between the control rods and the guides).	0.035	0.072

The allowable limit deflection values given above correspond to stress levels for the internals structure well below the limiting criteria given by the collapse curves in WCAP-5890, Rev. 1. Consequently, for the internals the geometric limitations established to assure safe shutdown capability are more restrictive than those given by the failure stress criteria.

TABLE 14.3.3-2

W1 - Core Barrel
W2 - Lower Package
W3 - Fuel Assemblies Major
W4 - Fuel Rods Major
W5 - Fuel Assemblies Minor
W6 - Fuel Rods Minor
W7 - Core Plate & Short Column
W8 - Deep Beam
W9 - Core Plate & Long Columns
W10 - Top Plate (Ctr.)
W11 - Core Barrel

K1 - Hold Down Spring
K2 - Lower Package Major
K3 - Top Nozzle Springs Major
K5 - Top Nozzle Springs Minor
K7 - Short Columns
K8 - Upper Core Plate
K9 - Long Columns
K10 - Top Plate
K11 - Core Barrel

Snubbers

S1 - Core Barrel Flange
S2 - Hold Down Spring
S3 - Top Nozzles Bars, Major
S4 - Pedestal Bars, Major
S5 - Top Nozzles Bars, Minor
S6 - Pedestal Bars, Minor
S7 - Top Nozzle Bumpers, Major
S8 - Top Nozzle Bumpers, Minor
S9 - Pedestals, Major
S10 - Pedestals, Minor
S11 - Deep Beam Flange

Impact Dampers

D1 - Barrel Flange
D2 - Hold Down Spring
D3 - Top Nozzle Bars, Major
D4 - Pedestal Bars, Major
D5 - Top Nozzle Bars, Minor
D6 - Pedestal Bars, Minor
D7 - Top Nozzles, Major
D8 - Top Nozzles, Minor
D9 - Pedestal, Major
D10 - Pedestal, Minor
D11 - Deep Beam Flange

Structural Dampers

C1 - Hold Down Springs
C2 - Lower Package
C3 - Top Nozzle, Major
C5 - Top Nozzle, Minor
C7 - Short Columns
C8 - Upper Core Plate
C9 - Long Columns
C10 - Top Plate
C11 - Core Barrel

Clearances

G1 - Hold Down Spring
G3 - Fuel Rod Top, Major
G4 - Fuel Rod Bottom, Major
G5 - Fuel Rod Top, Minor
G6 - Fuel Rod Bottom, Minor
G7 - Fuel Assembly Major
G8 - Fuel Assembly Minor

Preloads

P1 - Hold Down Spring
P3 - Top Nozzle Springs Major
P5 - Top Nozzle Springs Minor

TABLE 14.3.3-3

CRITICAL BUCKLING PRESSURES AND DYNAMIC RESPONSE OF UPPER CORE BARREL FOR HOT LEG SUBCOOLED BLOWDOWN

Break Time	Max. Uniform Pressure Wave*	Pulse Time**	Critical++ Buckling Press.	Maximum Stress+	Max. Radial Deflection
MS	PSI	MS	PSI	PSI	MILS
1	360	25	2290	11,844	± 35.6
5	360	26	2990	11,844	± 35.6
20	354	25	2990	11,646	± 35.0

* Pressure wave is uniformly distributed over upper core barrel both axially and circumferentially.

** Rise plus decay time of initial pressure wave.

\+ Membrane hoop stress - compressive

++ The critical buckling pressure is calculated based on two conservative assumptions:

1. The stiffening effect of the fluid environment is neglected.
2. The upper core barrel is assumed simply supported.

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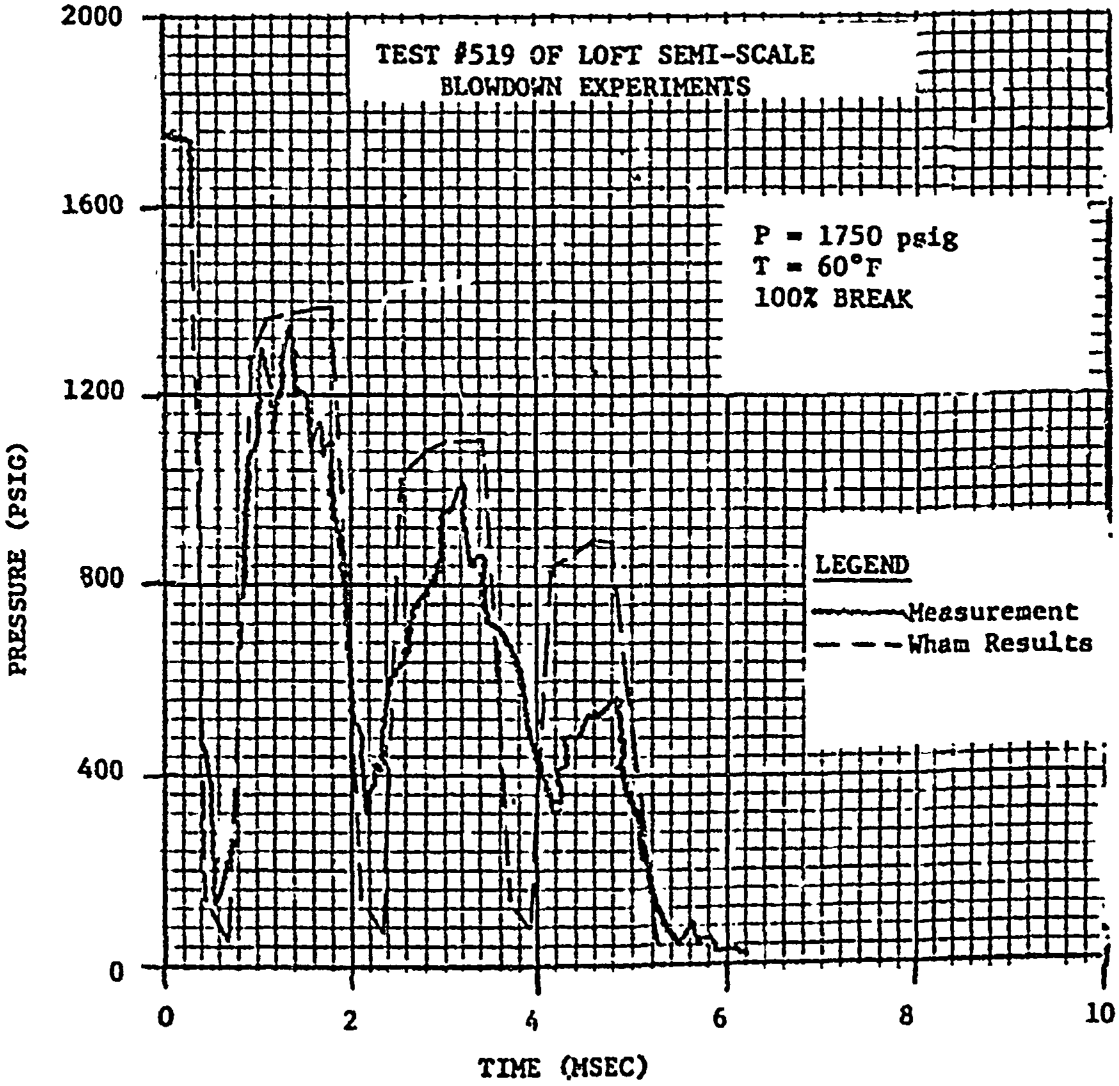


FIGURE 14.3.3-1

TEST #560 OF LOFT SEMI-SCALE BLOWDOWN EXPERIMENTS

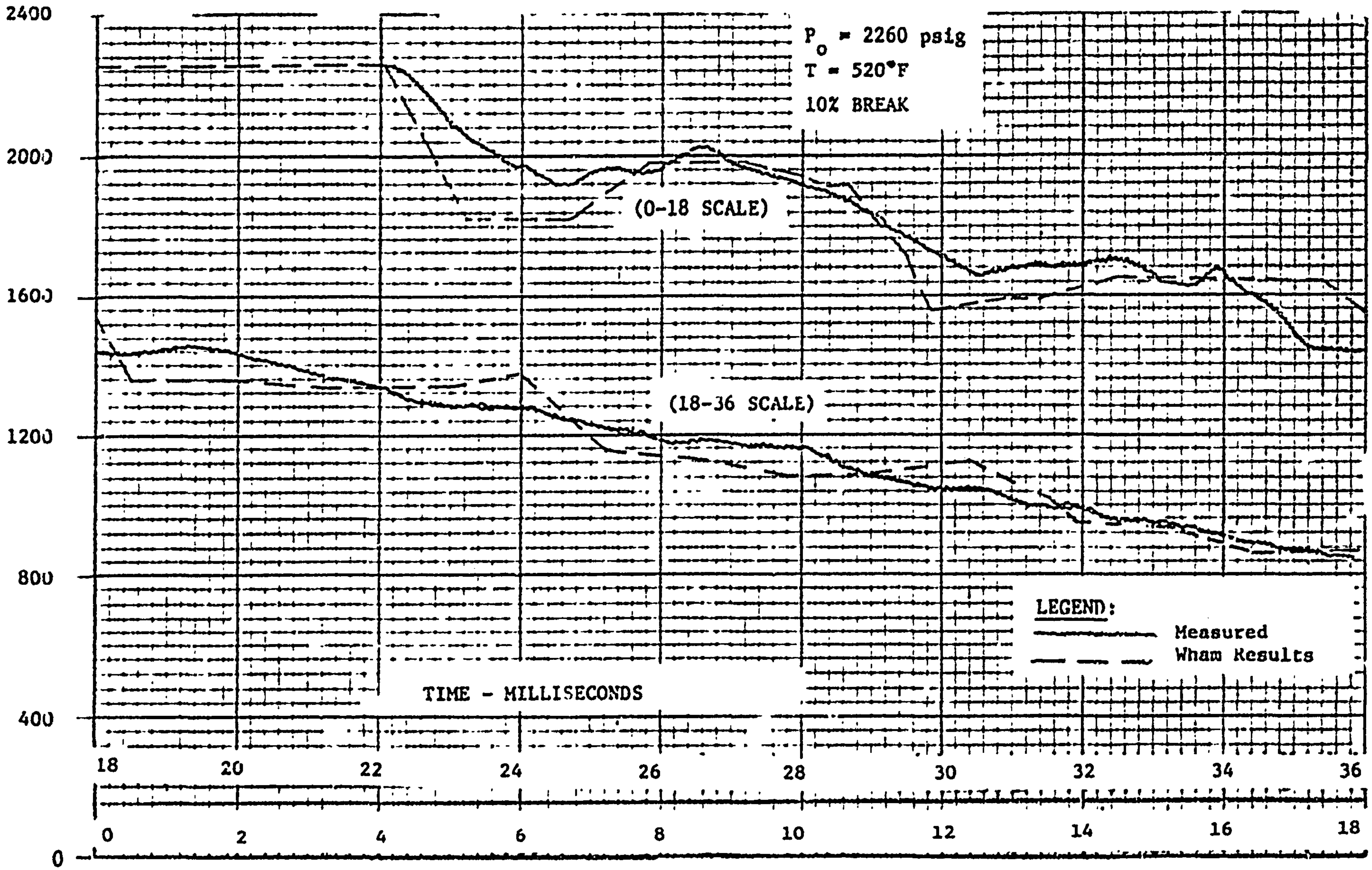
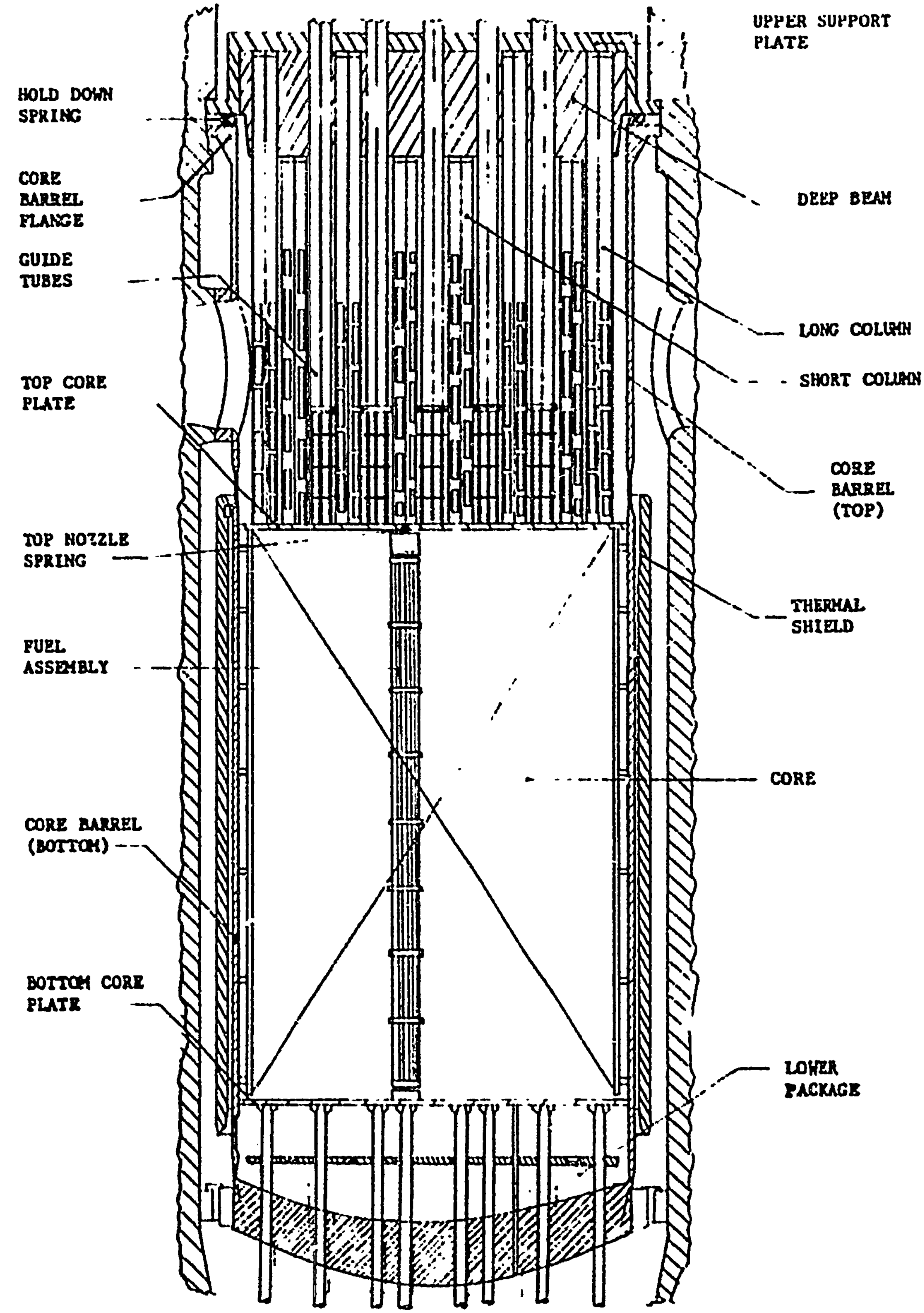


FIGURE 14.3.3-2



REACTOR VESSEL INTERNALS

FIGURE 14.3.3-3

939-3

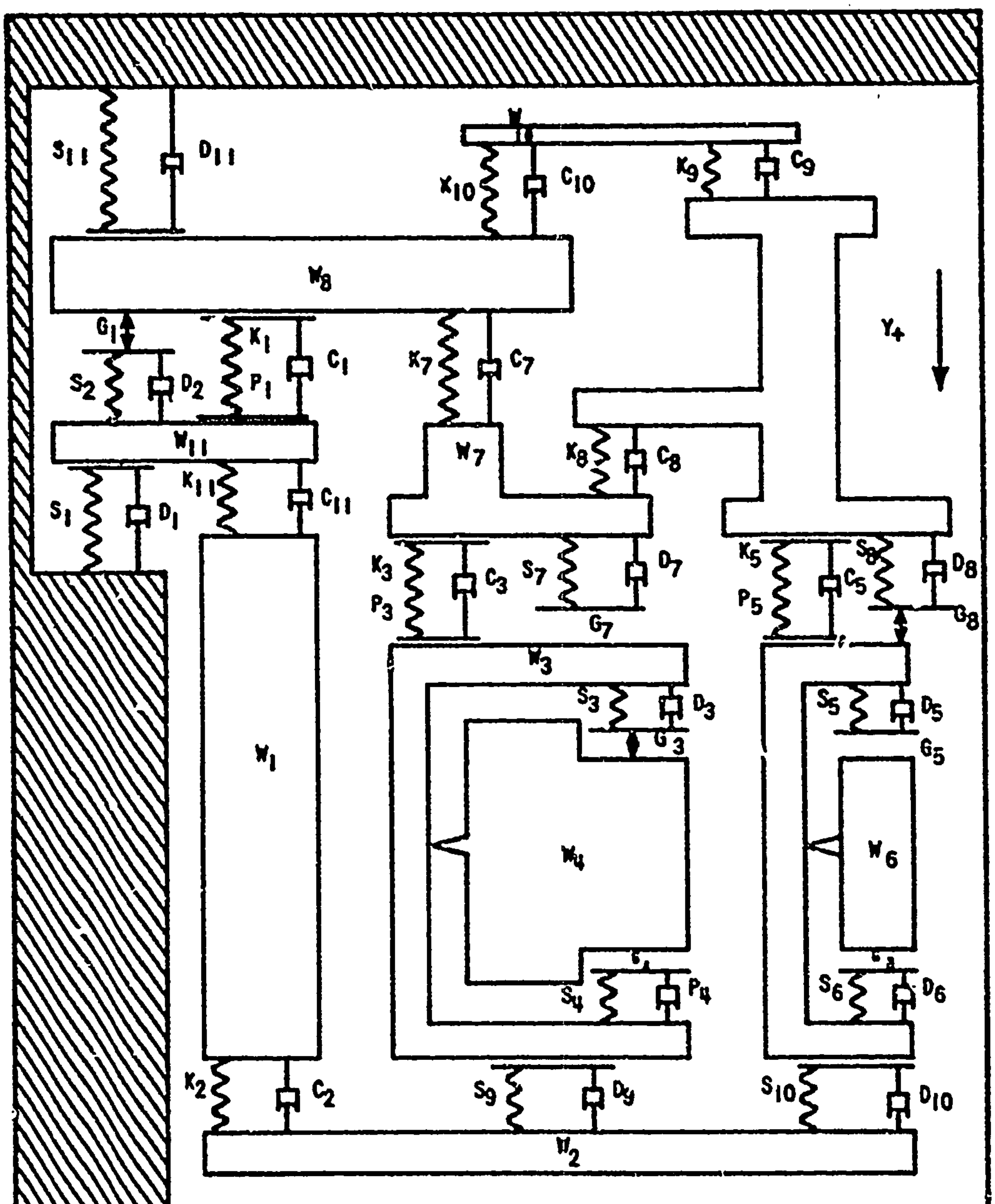


Figure 14.3.3-4. Multi-Mass Vibrational Model

14.3.4 CONTAINMENT INTEGRITY EVALUATION

Method of Analysis

Calculation of containment pressure and temperature transients is accomplished by use of the digital computer code, COCO. The analytical model is restricted to the containment volume and structure. Transient phenomena within the Reactor Coolant System affect containment conditions by means of convective mass and energy transport through the pipe break.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into two systems. The first system consists of the air-steam phase, while the second is the water phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. As thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for all possible cases of superheated or saturated steam, and subcooled or saturated water. Switching between states is handled automatically by the code. The following are the major assumptions made in the analysis:

a) Discharge mass and energy flow rates through the Reactor Coolant System break are established from the coolant blowdown and core thermal transient analysis (described in the preceding paragraphs).

b) At the break point, the discharge flow separates into steam and water phases. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment.

c) Homogeneous mixing is assumed. The steam-air mixture and the water phase have uniform properties. More specifically, thermal equilibrium between the air and steam is assumed. This does not imply thermal equilibrium between the steam-air mixture and the water phase.

d) Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

During the transient, there is energy transfer from the steam-air and water systems to the internal structures and equipment within the shell.

Provision is made in the computer analysis for the effects of several engineered safeguards, including internal spray, fan coolers, and recirculation of sump water. The heat removal from containment steam-air phase by internal spray is determined by allowing the spray water temperature to rise to the steam-air temperature.

Energy Sources

The amount of mass and energy carried into the containment during blowdown is calculated by the FLASH computer code. The following is a summary of all the energy sources potentially available for transfer to the containment for a loss-of-coolant accident.

a) Reactor Coolant Energy
b) Accumulator Energy (Mixes with Reactor Coolant System)
c) Initial Core Stored Energy
d) Core Internals Metal Energy
e) Reactor Vessel Metal (below vessel nozzles)
f) Core Power Generation (Shut down energy and decay heat)
g) Zr - H_2O reaction

All the power generated by the core during blowdown is transferred to the coolant, and reaches the containment. The initial core stored and metal sensible energy is transferred to the coolant by a time dependent temperature difference calculation. It should be emphasized that the energy transferred from the core to the coolant for the containment evaluation far exceeds that transferred from the core thermal evaluation. That is to say a conservatively high core heat transfer coefficient is used for the containment evaluation, while a conservatively low coefficient is used during the core thermal evaluation. Between the end of blowdown and the beginning of core reflooding there is no energy entering the containment. While the core is being reflooded the remaining stored energy in the core and internals causes a portion of the accumulator water to be boiled, and this energy is transferred to the containment.

Any energy addition resulting from a $Zr-H_2O$ reaction is also considered. The reaction energy reaches the containment by transfer to coolant, while the recombination energy of the H_2 generated in the reaction is added directly to the steam-air mixture in the containment. The hydrogen is assumed to burn as it is produced.

Finally, hot metal surfaces not cooled by safety injection water (reactor vessel above nozzles and steam generator tubes) are simulated as hot walls in contact with the containment steam-air mixture. A small heat transfer coefficient is employed to reflect actual conditions since these surfaces are covered by stagnant steam inside the reactor coolant system.

The following are some additional conservative assumptions used in the analysis:

a) The reactor power is based on operation at the maximum calculated power of 3216 MWt which is 16.6% greater than the application at 2758 MWt.

b) The decay heat is based on power operation for an infinite time.

c) Coolant temperatures are the maximum levels attained in steady state operation, including allowance for instrument error and deadband.

d) Gross system volumes are calculated from component dimensions, to which is added a 3% margin.

e) Pressurizer liquid inventory at the nominal full power level plus an appropriate margin for instrument error and deadband.

Energy Sinks

Containment Structures

Provision is made in the containment pressure transient analysis for heat transfer though, and heat storage in, both interior and exterior walls. Every wall is divided into a large number of nodes. For each node, a conservation of energy equation expressed in finite difference form accounts for transient conduction into and out of the node and temperature rise of the node. Table 14.3.4-1 is a summary of the containment structural heat sinks used in the analysis.

The heat transfer coefficient to the containment surface is calculated by the code based primarily on the work of Tagami[1]. From this work it was determined that the value of the heat transfer coefficient increases parabolically to peak value at the end of blowdown and then decreased exponentially to a stagnant heat transfer coefficient which is a function of steam to air weight ratio.

It should be noted that this method is different than that presented in the Preliminary Facility Description and Safety Analysis Report. In that report the heat transfer coefficients were based on the work of Koflat[2]. The revised method of calculation results in decreased heat transfer to the containment structure during blowdown.

Tagami presents a plot of the maximum value of h as a function of "coolant energy transfer speed," defined as:

$$\frac{\text{total coolant energy transferred into containment}}{\text{(containment vessel volume)} \quad \text{(time interval to peak pressure)}}$$

From this the maximum of h for steel is calculated:

$$h_{max} = 75 \quad \frac{E}{t_p V} \quad 0.60$$

h_{max} = maximum value of h (Btu/hr ft^2°F)

t_p = time from start of accident to end of blowdown

V = containment volume (Ft^3)

E = Initial coolant energy (Btu)

The parabolic increase to the peak value is given by:

$$h_s = h_{max}\sqrt{\frac{t}{t_p}} \qquad 0 \leq t \leq t_p \qquad (2)$$

h_s = heat transfer coefficient for steel (Btu/hr ft^2 °F)

t = time from start of accident (sec)

The exponential decrease of the heat transfer coefficent is given by:

$$h_s = h_{stag} + (h_{max} - h_{stag})\, e^{-.05(t-t_p)} \qquad t > t_p \qquad (3)$$

where

$$h_{stag} = 2 + 50x \qquad 0 \leq x \leq 1.4 \qquad (4)$$

h_{stag} = h for stagnant conditions (Btu/hr ft^2 °F)

x = steam to air weight ratio in containment

For concrete the heat transfer coefficient is taken as 40% of the value calculated for steel.

Air-Recirculation Fan-Coolers

The ability of the containment air recirculation coolers to function properly in the accident environment is demonstrated by the Westinghouse computer code "HECO". The code determines the plate-fin cooling coil heat removal rate when operating in a saturated steam-air mixture.

In the code a mass flow rate of cooling water is first established. This determines the tube inside film coefficient. Next the resistance to heat transfer between the cooling water and the outside of the fin collars is computed, including inside film coefficient, fouling factor*, tube radial conduction, fin-collar interface resistance, and conduction across the fin collars. The analysis now becomes iterative. One assumes an overall heat transfer rate Q_{tot} and the temperature at the outside of the fin collars is determined from Q_{tot} and the sum of the resistances cited above.

A second iterative procedure is now established. The variable whose value is assumed is the effective film coefficient between the fins and the gas stream, which involves the effect of convective heat transfer and mass transfer. With this valueo of $h_{effective}$, one can determine fin efficiency and the fin temperature distribution. It is assumed that a condensate film exists on the vertical fins. An analysis is performed which relates this film thickness to the rate of removal due to gravity and shear, and the rate of addition of condensate by mass transfer from the bulk gas. In the process, from an energy balance, one determines the temperature of the interface between the bulk gas and the condensate;

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* A fouling factor of .001 hr-ft^2-°F/Btu, under both normal and design basis accident conditions, has been assumed for cooling coil design purposes. This value is conventionally used in sizing heat exchangers cooled by river water at 125°F or less and with tube water velocity greater than 3 ft/sec (Reference 5), and is considered sufficiently conservative for this application. Computer analysis of the coils selected shows that the required post-accident heat removal rate can be achieved even with a slight increase in fouling.

this is necessary for determining the mass transfer rate from the gas. Now that the thickness of the condensate film is known, the value of the assumed $h_{effective}$ is checked from the relation $h_{eff} = K\ water/_{\delta film}$. If the assumed and computed values are not the same, a new guess is made and calculations repeated until the assumed and computed values are equal.

When this occurs, the heat transfer rate from the fins and fin collar is computed, using the standard equations for fin and fin collar heat transfer and the values of $h_{effective}$ and film-bulk gas interface temperature. If this value is not the same as Q_{tot}, initially assumed in order to determine fin collar temperature, the whole analysis is repeated with a new estimate of Q_{tot}. When, finally, the heat transfer rate to the cooling water from the fin collar equals the resulting computed rate to the fin collar and fins from the gas, the effect of this heat transfer rate on the cooling water is computed. The water exit temperature is established and this value is used as the inlet temperature for the next heat exchanger pass. Also, the effect of convective heat transfer and condensate mass transfer are determined relative to the gas composition and thermodynamic state. The updated gas state is used as inlet conditions for the next pass. The process is repeated for the second, third, etc. passes until the gas exits the heat exchanger.

The mass transfer coefficients used in the "HECO" code were derived from analyses and reports of experimental data contained in references 3, 4, and 5. From reference 3 the mass flow rate of condensate is defined by

$$\dot{m} = \bar{h}_D (\rho_{sg} - \rho_{sw}) \qquad (1)$$

From ref. (3), pp. 471-473, experimental data for mass and heat transfer correlate will by the expression

$$\frac{\bar{h}_D}{u_s}(Sc)^{-2/3} = \bar{S}t\,(Pr)^{-2/3}$$

as shown in Figure 16-10 of ref. (3). Thus

$$\bar{h}_D = u_s \cdot St\left(\frac{Sc}{Pr}\right)^{2/3} \qquad (2)$$

$$\bar{h}_D = \frac{u_s \cdot h}{\rho C\, u_s}\left(\frac{Sc}{Pr}\right)^{2/3}$$

As reference (3) points out, for large partial pressures of the condensing components, equation (2) must be corrected by a factor P_t/P_{am}. Thus h_D is defined by

$$\bar{h}_D = \frac{h}{\rho C}\,\frac{P_t}{P_{am}}\left(\frac{Sc}{Pr}\right)^{2/3} \qquad (3)$$

This is essentially the same result as reported by ref. (4) pg. 343 and reference (6).

Reference 3 states that experiments show equation (1) to be valid when the Schmidt number does not differ greatly from 1.0. Equations (1) and (3) are combined to give the mass transfer rate, which is

$$\dot{m} = \frac{h}{\rho C}\,\frac{P_t}{P_{am}}\left(\frac{Sc}{Pr}\right)^{2/3}(\rho_{sg} - \rho_{sw})$$

An approximation was made in assuming that $\left(\frac{Sc}{Pr}\right)^{2/3} \cong 1.0$ thus the local mass transfer rate was computed from

$$\dot{m} = \frac{h}{\rho C}\,\frac{P_t}{P_{am}}(\rho_{sg} - \rho_{sw})$$

The heat transfer rate due to condensation is computed from

$$q_1 = \frac{\dot{m}\ \lambda hP}{\rho CP_{am}t}\ (\rho_{sg} - \rho_{sw})$$

where ρ_{sg} is evaluated at the local bulk gas temperature

ρ_{sw} is evaluated at the local gas-condensate interface temp.

λ is evaluated at the local gas-condensate interface temp.

P and C are evaluated at the local bulk gas temperature

The heat trasnfer coefficient, h, was determined from experiments on W plate-fin coils which are the same geometry as would be used in this application.

The heat transfer rate, locally, is computed from

$$q_2 = h\ (T_g - T_i)$$

The basis for selecting these values is that the authorities cited as references have shown, through analyses and through cited experiments, that the methods used are accurate.

The air side pressure drop across the cooling coils under design basis accident condition is estimated to be approximately 2.1 in. H_2O, or .076 psi. This will have negligible effect on the heat removal capability of the cooling coils.

The pressure of non-condensible gases are taken into consideration by virtue of the fact that the theory behind the analysis assumed that the condensable vapor must diffuse through a non-condensible gas.

Application of this method results in the fan-cooler heat removal rate per fan as presented in Figure 14.3.4-1.

A testing program is in progress to confirm the validity of the above design procedures and the performance of the cooling coils under post-accident conditions. Details of the testing program are proprietary to Westinghouse, and therefore, will be presented to the Atomic Energy Commission by separate correspondence.

<u>Nomenclature</u>

$\dot{m}$ mass flow rate of condensate, $lbm/hr\text{-}ft^2$

$\bar{h}_D$ mass transfer coefficient, ft/hr.

ρ_{sg} density of saturated steam at local bulk gas temp., lbm/ft^3

ρ_{sw} density of saturated steam at local condensate-gas interface temp., lbm/ft^3

u_s free steam gas velocity, ft/min.

Sc Schmidt number, $M/\rho D$, dimensionless

μ viscosity of bulk gas, lbm/ft-hr.

ρ bulk gas density, lbm/ft^3

D Gas-air diffusion coefficient, $\frac{ft^2}{hr}$

St Stanton number, $h/\rho c u_s$, dimensionless

h convective heat transfer coefficient, $Btu/hr\text{-}ft^2\text{-}°F$

C Specific heat of bulk gas, Btu/lbm-°F

P_r Prandtl number, $\mu c/k$, dimensionless

k thermal conductivity of bulk gas, Btu/hr-ft-°F

P_t total gas pressure, lbf/ft^2

P_{am} air log-mean $\dfrac{P_{aw} - P_{ag}}{\ln \dfrac{P_{aw}}{P_{ag}}}$, lbf/ft^2

P_{aw} partial pressure of air at the local gas-condensate interface, lbf/ft^2

P_{ag} Partial pressure of air at the local bulk gas temperature, lbf/ft^2

λ Latent heat of vaporization (or condensation) at the local gas condensate interface temperature, Btu/lbm

q_1 local heat transfer rate due to condensation, $Btu/hr\text{-}ft^2$

q_2 local heat transfer rate due to convection, $Btu/hr\text{-}ft^2$

T_g local bulk gas temperature, °F

T_i local gas-condensate interface temperature, °F.

Containment Spray

When a spray drop enters . hot saturated steam-air environment, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the drop. This mass flow will carry energy to the drop. Simultaneously the temperature difference between the atmosphere and the drop will cause a heat flow to the drop. Both of these mechanisms will cause the drop temperature and vapor pressure to rise. The vapor pressure of the drop will eventually become equal to the partial pressure of the steam and the condensation will cease. The temperature of the drop will be essentially equal to the temperature of the steam-air mixture.

The terminal velocity of the drop can be calculated using the formula given by Weinberg[7] where the drag coefficient C_D is a function of the Reynolds number:*

$$V^2 = \frac{4Dg(\rho - \rho_m)}{3C_D\rho_m} \tag{1}$$

For the 700 micron drop size expected from the nozzles, the terminal velocity is less than 7 ' /sec. For a 1000 micron drop, the velocity would be less than 10 ft/sec. The Nusselt number for heat transfer, $\underline{Nu}$, and the Nusselt number for mass transfer, $\underline{Nu}'$ (Sherwood Number), can be calculated from the empirical relations given by Ranz and Marshall.[8]

$$\underline{Nu} = 2 + 0.6\,(\underline{Re})^{1/2}\,(\underline{Pr})^{1/3} \tag{2}$$

$$\underline{Nu}' = 2 + 0.6\,(\underline{Re})^{1/2}\,(\underline{Sc})^{1/3} \tag{3}$$

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* Nomenclature used is given at the end of this discussion.

The Prandtl number and the Schmidt number for the conditions assumed are approximately 0.7and 0.6 respectively. Both of these are sufficiently independent of pressure, temperature and composition to be assumed constant under containment conditions.[9,10] The coefficients of heat transfer (h_c) and mass transfer (k_G) are calculated from Nu and Nu' respectively. The equations describing the temperature rise of a falling drop are:

$$\frac{d}{dt}(Mu) = mh_g + q \qquad (4)$$

$$\frac{d}{dt}(M) = m \qquad (5)$$

where

$$q = h_c A (T_s - T) \qquad (6)$$

$$m = k_G A (P_s - P_v) \qquad (7)$$

These equations can be integrated numerically to find the internal energy and mass of the drop as a function of time as it falls through the atmosphere. Analysis shows that the liquid drop temperature rises to the steam-air mixture temperature in less than 0.5 seconds, which occurs before the drop has fallen 5 feet. These results demonstrate that the spray will be 100% effective in removing heat from the atmosphere.

Nomenclature

A	area
C_D	drag coefficient
D	droplet diameter
g	acceleration of gravity
h_c	coefficient of heat transfer
h_s	steam enthalpy
k_G	coefficient of mass transfer
M	droplet mass
m	diffusion rate
$\underline{Nu}$	Nusselt number for heat transfer
$\underline{Nu}'$	Nusselt number for mass transfer
P_s	steam partial pressure
P_v	droplet vapor pressure
$\underline{Pr}$	Prandtl number
q	heat flow rate
$\underline{Re}$	Reynolds number
$\underline{Sc}$	Schmidt number
T	droplet temperature
T_s	steam temperature
t	time
u	droplet internal energy
V	velocity
ρ	droplet density
ρ_m	ste[illegible]r mixture density

Containment Pressure Transients

The containment pressure was calculated for a range of large area ruptures of the Reactor Coolant System. The - pture sizes considered were:

a. Double Ended Rupture
b. 6 ft^2 break
c. 3 ft^2 break
d. .5 ft^2 break

Figure 14.3.4-2 presents the results of the transients. Fo- all cases a pressure peak of less than 40 psig was calculated. Si e design pressure for the IPP #2 plant is 47 psig, a margin more than ' .e the conservative value of the blow down peaks, is available.

In the transients, one spray pump and three fans starting at 60 seconds were assumed. These acted to quickly reduce the pressure after the peak pressures were reached.

The following paragraphs are a summary of the energy sources and sinks used in the calculation.

Energy Sources

The energy sources presented in Table 14.3.4-2 are potentially available to be transferred to the containment during the blowdown time.

In the above energy summation all sensible energy sources are referenced to the datum of saturated water at containment design pressure, which is the maximum amount of energy that can be transferred from the metal to the coolant.

The integrated energy balance at the end of blowdown is presented in the Table 14.3.4-3 The values were determined by the FLASH Code.

In this calculation all energy generated by the core during blowdown is transferred to the coolant as it is generated. The sensible energy sources are transferred to the coolant as a function of time, and for longer blowdown times more sensible energy is absorbed. For the very large breaks very little energy is transferred to the steam generators, because of the rapid uncovering of the tubes, while for smaller breaks the tubes do not uncover as rapidly and significant heat transfer results.

A negligible amount of energy is transferred from the react. essel during the relatively fast blowdown.

Energy Sink

Figure 14.3.4-3 presents the energy absorption capability within the 2.6×10^6 ft^3 free volume of the IPP containment. Thus the internal energy of the steam-air mixture must be increased to 306×10^6 Btu for the containment pressure to reach the design pressure of 47 psig.

The integrated containment energy balance at the end of blowdown is given by:

$$U_f = U_i + \Sigma (mh)_{in} + \Sigma Q_{in} - \Sigma Q_{out}$$

Where

U_f = Final internal energy in the containment

U_i = Initial external energy in the containment

$\Sigma (mh)_{in}$ = Enthalpy added by blowdown sources

ΣQ_{in} = Energy added directly to containment atmosphere by hydrogen-oxygen recombination

ΣQ_{out} = Heat removal by containment structure and cooling system.

The internal energy is made up of three sources: air, steam, and sump water. Only the air-steam mixture with their respective partial pressures contribute to the containment total pressure. The internal energy for the initial assumed containment conditions, 120°F and 15 psia, is as follows:

Steam (m) (u) = (2260) (1077)	$= 2.43 \times 10^6$ Btu
Air (m) (C_v) (T) = (178,205) (0.172) (120)	$= 3.67 \times 10^6$ Btu
Sump (m) (u) = (12,343) (87.9)	$= 1.08 \times 10^6$ Btu
	7.18×10^6 Btu

The internal energy balance at the end of blowdown is given in the Table 14.3.4-6. All entries are in millions of Btu's.

The difference between the internal energies given by the energy balance equation and by the coco program represents an error of less than ± 1% in the calculation.

Figure 14.3.4-4 shows the heat transfer coefficient calculated for the various break sizes.

Containment Margin Evaluation

Evaluation of the capability of the reactor containment and containment cooling systems to absorb energy additions without exceeding the containment design pressure requires consideration of two periods of time following a postulated large area rupture of the reactor coolant system.

The first period is the blowdown phase. Since blowdown occurs too rapidly for the containment cooling systems to be activated, there must be sufficient energy absorption capability in the free volume of the containment (with due credit for energy absorption in the containment structures) to limit the resulting pressure below design.

The second period is the post-blowdown period where the containment cooling systems must be able to absorb any postulated post-blowdown energy additions and continue to limit the containment pressure below design.

Margin - Blowdown Peak to Design Pressure

Point A in Figure 14.3.4-5 corresponds to the internal energy at the end of a 3 ft^2 break blowdown, 258 x 10^6 Btu. In order for the pressure to increase to design pressure (47 psig) the internal energy must be increased to 306 x 10^6 Btu (Point B). The allowed energy addition is therefore 48 x 10^6 Btu. Since energy transferred to the containment from the core is in the form of steam the total transferred core energy corresponding to allowed energy addition is as follows:

$$Q_{core} = \frac{h_{fg}}{hg} Q_{Allowed} = 48 \times 10^6 \times \frac{914.7}{1177.9} = 37.2 \times 10^6 \text{ Btu}$$

This allowable value of energy which could be transferred from the core to the containment without increasing the transient containment pressure to design pressure can be compared to the energy stored in the reactor vessel and transferred to the steam generator during blowdown for the double ended break. The thick metal of the reactor vessel was not considered since a negligible amount of this energy can be transferred in the short blowdown time.

Stored in the core	17.3 x 10^6 Btu
Core internals Metal	3.7 x 10^6 Btu
Transferred to Steam Generators	8.4 x 10^6 Btu
	29.4 x 10^6 Btu

Thus, the containment has the capability to limit containment pressure below design even if all of the available energy sources were transferred to the containment at the end of blowdown. This would also include no credit for energy adsorption in the steam generator. For this to occur an extremely high core to coolant heat transfer coefficient is necessary. This would result in the core and internals being completely subcooled and limit the potential for release of fission products.

Additional Energy Added As Superheat

Line A to C on Figure 14.3.4-5 represents a constant mass line extended into the superheated region. Comparison of the energy addition allowable for the superheated case relative to the saturated case shows a lesser ability of the containment to absorb an equivalent amount of energy as superheat. An addition of 13×10^6 Btu of energy after blow down would cause the containment pressure to increase to design. The recombination of hydrogen and oxygen from a 14.6% $Zr-H_2O$ reaction completed before the end of blow down would be required to generate 13×10^6 Btu s of energy. For the case analyzed, the core was assumed to be in a subcooled state, and no $Zr-H_2O$ reaction would be possible. In order for $Zr-H_2O$ reaction to occur before the end of blowdown all of the stored initial energy must remain in the core. If this occurred a blow down peak containment pressure of only 34.2 psig would be reached instead of 39.5 psig in the case analyzed. Lines D and E on Figure 14.3.4-5 represent the superheat energy addition required to increase the pressure to the design pressure and this corresponds to the hydrogen oxygen recombination energy from a 24.6% $Zr-H_2O$ reaction.

It is, therefore, concluded that the containment has the capability to absorb the maximum energy addition from any loss-of-coolant accident without reliance on the containment cooling system. In addition, a substantial margin exists for energy additions from arbitrary energy sources much greater than any possible.

Margin - Post Blowdown Energy Additions

The Safety Injection System is designed to rapidly subcool the core and stop the addition of mass and energy to the containment. Thus it is expected that there will not be any significant energy addition to the containment following blowdown. However, the following cases are presented to demonstrate the capability of the containment to withstand post accident energy additions without credit for core cooling.

Case 1. Blowdown from a large area rupture with continued addition of the core residual energy and hot metal energy to the containment as steam.

Case 2. Same as Case 1 but with the energy addition from a maximum Zirconium - water reaction.

Figure 14.3.4-6 presents the containment pressure transient for Case 1. For this case the decay heat generated for a 3216 MWt core operated for an infinite time is conservatively assumed. This decay heat is added to the containment in the form of steam by the boiling off of water in the reactor vessel. For this case injection water merely serves as a mechanism to transfer the residual energy to the containment as it is produced. Injection water is in effect throttled at the required rate.

In addition, all the stored energy in the core and internals which is calculated to remain at the end of blow down is added in the same way during the time interval between 26 and 49 seconds (corresponds to accumulator injection time). Also all the sensible heat of the reactor vessel is added as steam exponentially over 2000 seconds time interval.

The containment cooling system capabilit assumed in the analysis was one of two available containment spray pumps and three of five available containment fan coolers. This is the minimum equipment available considering the single failure criterion in the emergency power system, the spray system and the fan cooler system.

The containment heat removal capability started at 60 seconds exceeds the energy addition rate and the pressure does not exceed the initial blowdown value. An extended depressurization time results due to the increased heat load on the containment coolers.

It should be emphasized that this situation is highly unrealistic in that continued addition of steam to the containment after blowdown could not occur. The accumulator and Safety Injection System acts to rapidly reflood and subcool the core.

Figure 14.3.4-7 presents the containment pressure transient for Case 2. To realistically account for the energy necessary to cause a metal-water reaction, sufficient energy must be stored in the core. Storing the energy in the core rather than transferring it to the coolant causes a decrease in the blowdown peak.

The reaction was calculated using the parabolic rate equation developed by Baker and assuming that the clad continues to react until zirconium oxide melting temperature , 4800°F is reached. An additional 10% reaction of the unreacted clad is assumed when the oxide melting temperature is reached. A total reaction of 32.3% has occurred after 1000 seconds. Previous analysis has shown that steam limited reactions could result in a higher total reaction but at a much later time. The reaction provided by the parabolic rate equation therefore, imposes the greatest load on the containment cooling system.

As in Case 2, the residual heat and sensible heat is added to the containment as steam. The energy from the $Zr-H_2O$ reaction is added to the containment as it is produced. The hydrogen was assumed to burn as it entered the containment from the break.

The blowdown peak was reduced to 34.0 psig and a peak pressure of 43.8 psig was reached at 400. seconds. At this time the heat removal capability of the containment cooling system assumed to be operating (one containment spray pump and three fan coolers) exceeded the energy addition from all sources.

For comparison the containment pressure transients for Cases 1, 2 and the double ended blowdown are replotted in Figure 14.3.4-8. It is concluded that operation of the minimum containment cooling system equipment provides the capability of limiting the containment pressure below its design pressure with the addition of all available energy sources and without credit for the cooling effect from the safety injection system.

Discussion of Energy Sources Used in Cases 1 and 2

The following is a summary of the energy sources and the containment heat removal capacities used in the containment capability study. Figure 14.3.4-9 presents the rate of energy addition from core decay heat, $Zr-H_2O$ reaction energy, and the hydrogen-oxygen recombination energy. The heat removal capability for the partial containment cooling (one spray pump and three fan coolers) is also presented. These heat removal values are for operation with the containment at design pressure.

The integrated heat additions and heat removals for Cases 1 and 2 are plotted in Figures 14.3.4-10 and 14.3.4-11, respectively. These curves are presented in a manner that demonstrates the capability of the containment and the cooling systems to absorb energy. The integrated heat removal capacity is started at the internal energy corresponding to design pressure, while the integrated heat additions begin from the internal energy calculated at the end of blowdown for each case. The upper line on each curve is the containment structures and containment cooling systems capability to absorb energy additions without exceeding design pressure. The lower curve for each are the energy addition curves, and since these energy additions are the maximum possible with no credit for core cooling, there is more than adequate capability to absorb arbitrary additions.

The curves in Figures 14.3.4-12 and 14.3.4-13 present the individual contribution of the heat removal and heat addition source, respectively.

Evaluation of Containment Internal Structures

The containment internal structures such as the reactor coolant loop compartments and the reactor shield wall are designed for the pressure buildup that could occur following a loss-of-coolant. If a loss-of-coolant accident were to occur in these relatively small volumes, the pressure would build up at a rate faster than the overall compartments.

A digital computer code, COMCO, was developed to analyze the pressure build-up in the reactor coolant loop compartments. The COMCO code is largely an extension of the COCO Code in that a separation of the two phase blowdown into steam and water is calculated and the pressure build-up of the steam-air mixture in the compartment is determined. Each compartment has a vent opening to the free volume of the containment.

The main calculation performed is a mass energy balance within the control volume of a compartment. The pressure builds up in the compartment until a mass and energy relief through the vent exceeds the mass and energy entering the compartment from the break. The reactor coolant loop compartments are designed for the maximum calculated differential pressure resulting from an instantaneous double ended rupture of the reactor coolant pipe.

There are two reactor coolant loop compartments with two loops in each compartment. The total free volume of each compartment is 113,500 ft^3 with a vent area of 1000 ft^2. The calculated differential pressure across the wall of the compartment is 6.4 psi.

The primary shield around the reactor vessel is designed for a pressure of 1000 psi to provide missile protection against the highly unlikely failure of the reactor vessel by longitudinal splitting or by various modes of circumferential cracking.

Evaluation of Long Term Fan Cooler Capability

The ability of the fan coolers to limit containment pressure following loss of the component cooling system has been examined. If the component cooling loop were lost for any reason during long term recirculation, core subcooling could be lost and boiling in the core would begin. Since the fans cooling units are cooled by service water, the energy from the core would be removed from the containment via the fans. The following table summarizes the maximum pressure the containment could reach for assumed times of component cooling system failure.

	3 Fans	2 Fans
C.C. Failure at 12 hours	12	28
C.C. Failure at 1 day	9	22
C.C. Failure at 1 week	4	8

Radiolytic Hydrogen Formation

In order to assess the potential magnitude of gas accumulation in the containment due to radiolytic decomposition of emergency core cooling water, a conservative analysis was made of the decomposition caused by two sources of radiation: the intact core and the released fission products apt to be associated with the water phase in the containment.

Core radiation was estimated assuming:

1) Prior to the accident the reactor has operated at full rated power for 10,000 hr.

2) Absorption of gamma radiation occurs in an infinite medium composed of a homogeneous mixture of the fuel, support structure, cladding, and the water within the core baffle.

3) Beta radiation is absorbed by the fuel and cladding (not by water).

4) Hydrogen yield is 0.44 molecules per 100 electron volts of energy absorbed by water.[11] No thermal recombination occurs in solution; i.e., H_2 produced is quantitatively transported to the containment atmosphere.

The conservatism of assumption (4) should be noted. Current investigations by WAPD and others show that after cessation of boiling in the core region, significant recombination occurs, apparently governed by mass transport parameters. By neglecting this effect an upper limit result is obtained.

The contribution of dissolved fission products is assess by calculating the yield from 50% of the core halogen fission products when all of the beta and gamma emission is absorbed by water. The value of hydrogen yield assumed for this source is 0.30 molecules per 100 ev, based on irradiation tests reported in WCAP-7153 where the gas/liquid volume ratio was selected to simulate the containment/sump volumes of the full sized plant.

The quantities of radiolytic hydrogen and the associated radiolytic oxygen generated with time after the accident, as calculated by this model, are presented in Figure 14.3.4-14. The pressure effect of these gases is given in Figure 14.3.4-15, where the containment atmosphere is assumed to be cooled to its pre-accident temperature. Figure 14.3.4-16 shows the concentration of radiolytic hydrogen on a steam-free basis.

The results show that the pressure of radiolytic gas is of little concern during the post-accident recovery, being less than 1.5 psi at the end of one month. The concentration of H_2, averaged in the containment volume, reaches 4.1 volume % (the lower limit for flammability) in 18 days.

In view of the potential for receiving a flammable limit, the pessimism of the calculational model is being evaluated, with a view to incorporating the thermal recombination effects which may diminish the net yield of hydrogen by radiolysis in the core region. When results of current tests are analyzed, the decision will be made regarding incorporation of a hydrogen recombiner system. The objectives to be met will be to show that the contents of the containment can be safely vented before reaching a flammable concentration to provide a recombiner system capable of sustaining a concentration well below a flammable limit indefinitely.

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TABLE 14.3.4-1

STRUCTURAL HEAT SINKS

Heat Sink	Material	Area Ft^2	Thickness in	Density lb/ft^3	Heat Capacity BTU/lb°F	Conductivity BTU/HR ft °F
Containment Cylinder	Steel lined concrete	50,000	3/8	511	0.11	26
Containment Dome	Steel lined concrete	61,000	1/2	511	0.11	26
Containment Floor	Unlined concrete	15,000	12	150	0.186	0.08
Refueling Canal	Lined concrete	16,000	3/8	511	0.11	26
Misc. Concrete Structure	Unlined Concrete	61,000	12	150	0.186	0.08
Misc. Steel Structure	Steel					
a. Thin		16,000	1/8	511	0.11	26
b. Thick		18,000	1/2	511	0.11	26

TABLE 14.3.4-2

ENERGY SOURCES

1.	Reactor Coolant System Internal Energy308.9	305.9×10^6 Btu
2.	Accumulator Internal Energy (Three)	11.4×10^6
3.	Initial Core Stored Energy	37.2×10^6
4.	Core Internals Metal Energy	12.5×10^6
5.	Reactor Vessel Me'al (below vessel nozzle)	13.6×10^6
	Sub Total	379.7×10^6
6.	Core Power Generation During Blowdown	
	a. Double ended (12 secs)	6.0×10^6
	b. 6 ft^2 (16.4 secs)	7.5×10^6
	c. 3 ft^3 (27 secs)	10.1×10^6
	d. .5 ft^2 (125 secs)	24.1×10^6
7.	$Zr-H_2O$ reaction	~0.0
	TOTALS a. Double ended	385.7×10^6
	b. 6 ft^6	387.2×10^6
	c. 3 ft^2	389.8×10^6
	d. .5 ft^2	403.8×10^6

TABLE 14.3.4-3

INTEGRATED ENERGY BALANCE

Outside Reactor Coolant System Control Volume	DE	6 ft^2	3 ft^2	.5 ft^2
1. Blowdown Enthalpy	327.3	329.0	327.6	322.4
2. Transferred to Steam Generator	2.3	3.5	8.4	32.7
	329.6	332.5	336.0	355.1
Inside Reactor Coolant System Control Volume				
1. Reactor Coolant Internal Energy (water remaining in vessel plus accumulator addition)	9.2	11.4	11.3	24.4
2. Stored in Core	19.8	15.9	17.3	0.2
3. Core Internal Metal	3.6	3.7	3.7	3.9
4. Reactor Vessel Metal	13.6	13.6	13.6	13.6
5. Internal Energy of Water Remaining in Accumulator (Injection not complete)	8.7	8.6	8.1	6.0
	54.9	53.2	52.0	48.1
	384.5	385.7	390.0	403.2

TABLE 14.3.4-4

INTERNAL ENERGY BALANCE

	Double Ended	6 ft^2	3 ft^2	.5 ft^2
U_i	7.2	7.2	7.2	7.2
$\Sigma\ (mh)_{in}$	327.3	329.0	327.6	323.0
$\Sigma\ Q_{in}$	∿0	∿0	∿0	∿0
$\Sigma\ Q_{out}$ a/structure	-11.9	-12.6	-14.4	-21.2
b/fans	0	0	0	- 3.8
c/sprays	0	0	0	- 3.1
Total U_f	322.6	323.6	320.4	302.1

From Coco the final conditions are:

Steam	251.0	255.0	250.0	229.0
Air	8.0	8.0	8.0	7.9
Sump	64.5	64.0	63.8	69.0
	323.5	327.0	321.8	305.9

FAN COOLER HEAT REMOVAL AS A FUNCTION OF CONTAINMENT PRESSURE

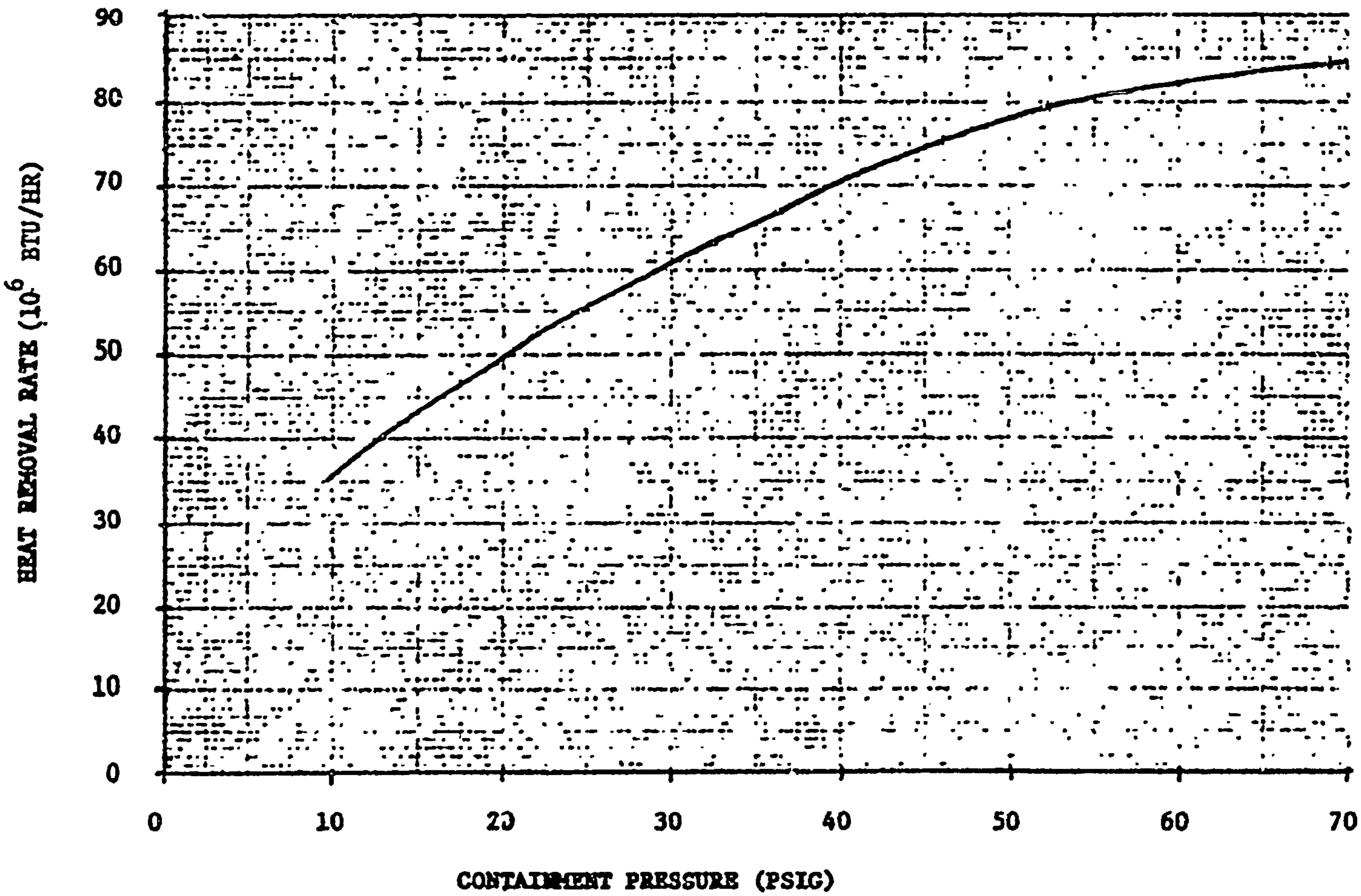

FIGURE 14.3.4-1

Supplement 7
3/70

CONTAINMENT PRESSURE TRANSIENT

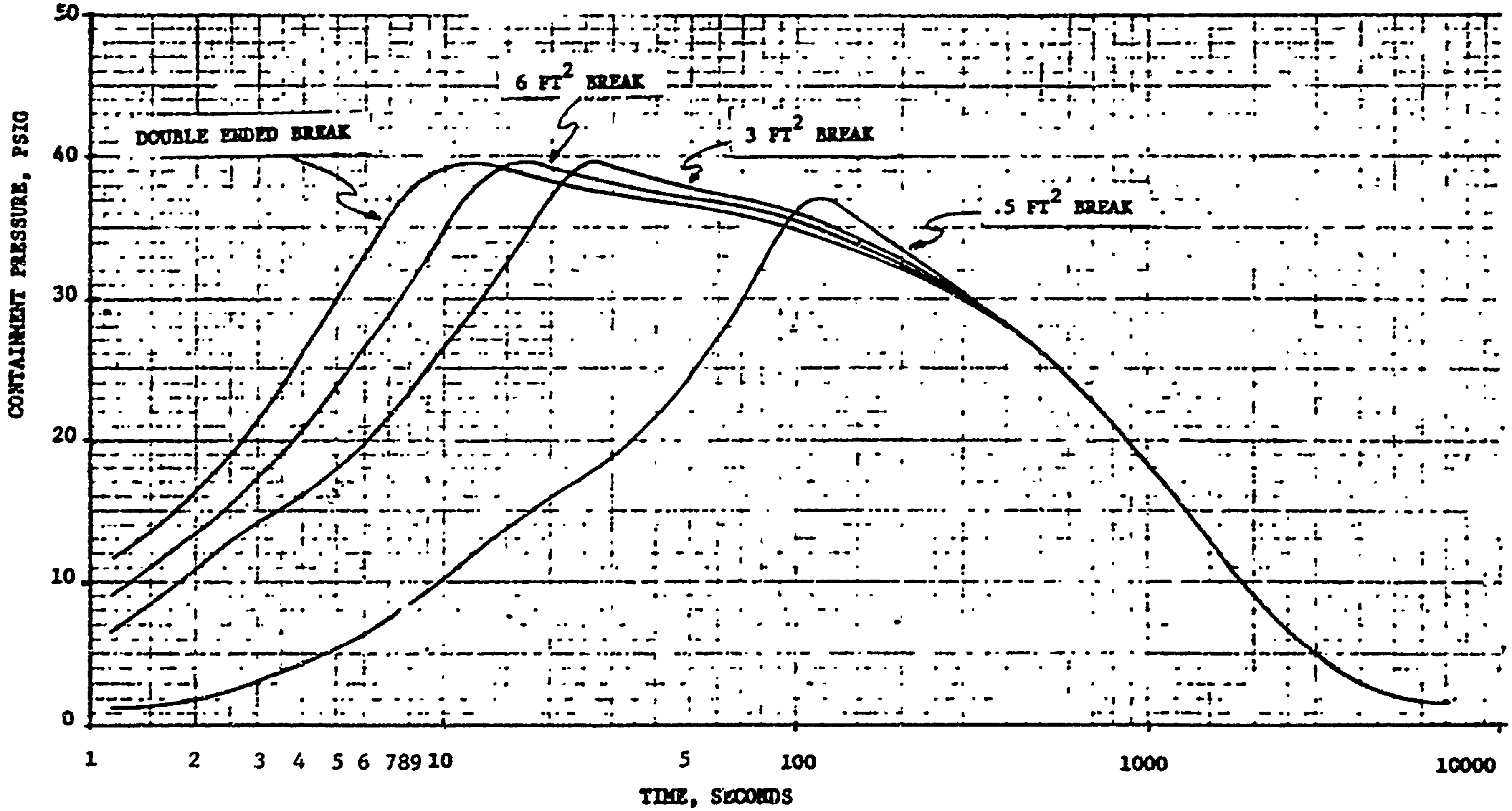

FIGURE 14.3.4-2

CONTAINMENT CAPABILITY STUDY

CONTAINMENT PRESSURE VS STEAM-AIR INTERNAL ENERGY

VOLUME: 2.6×10^6 FT^3

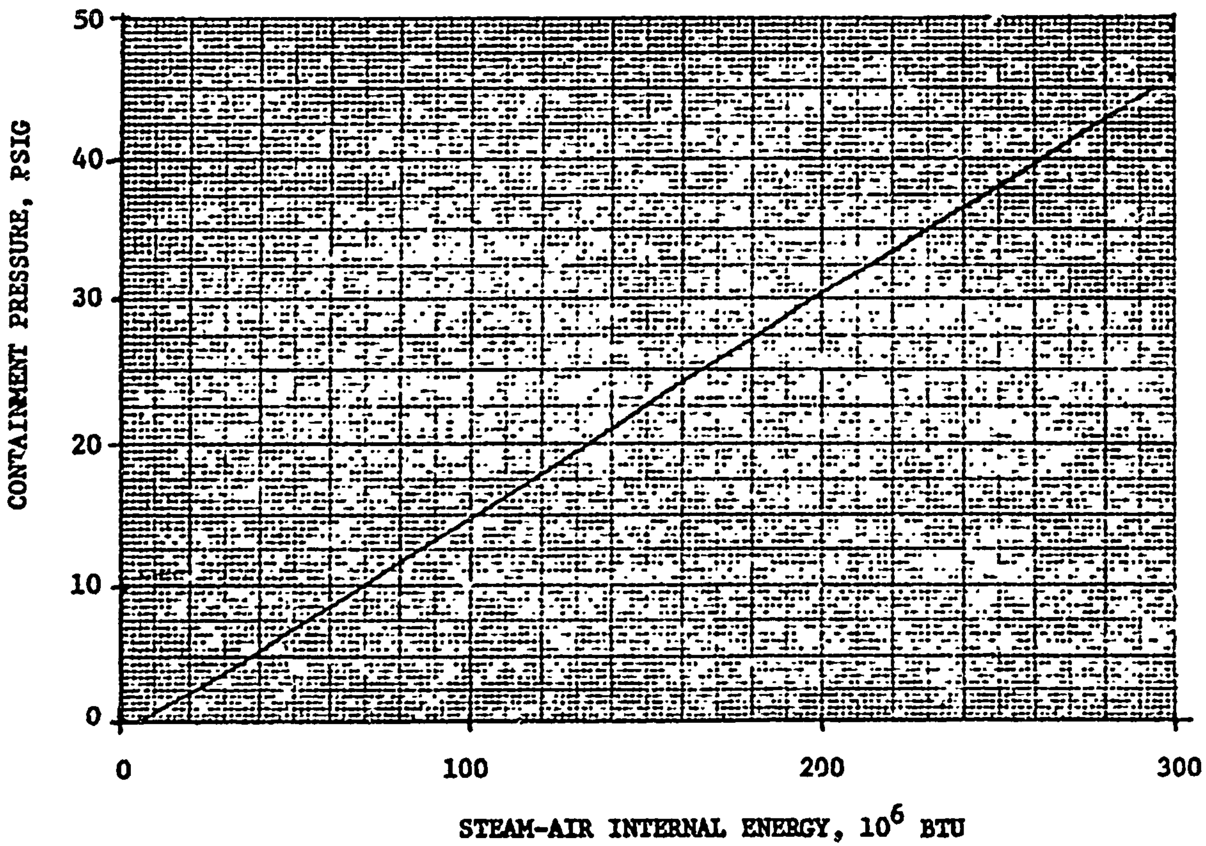

FIGURE 14.3.x-3

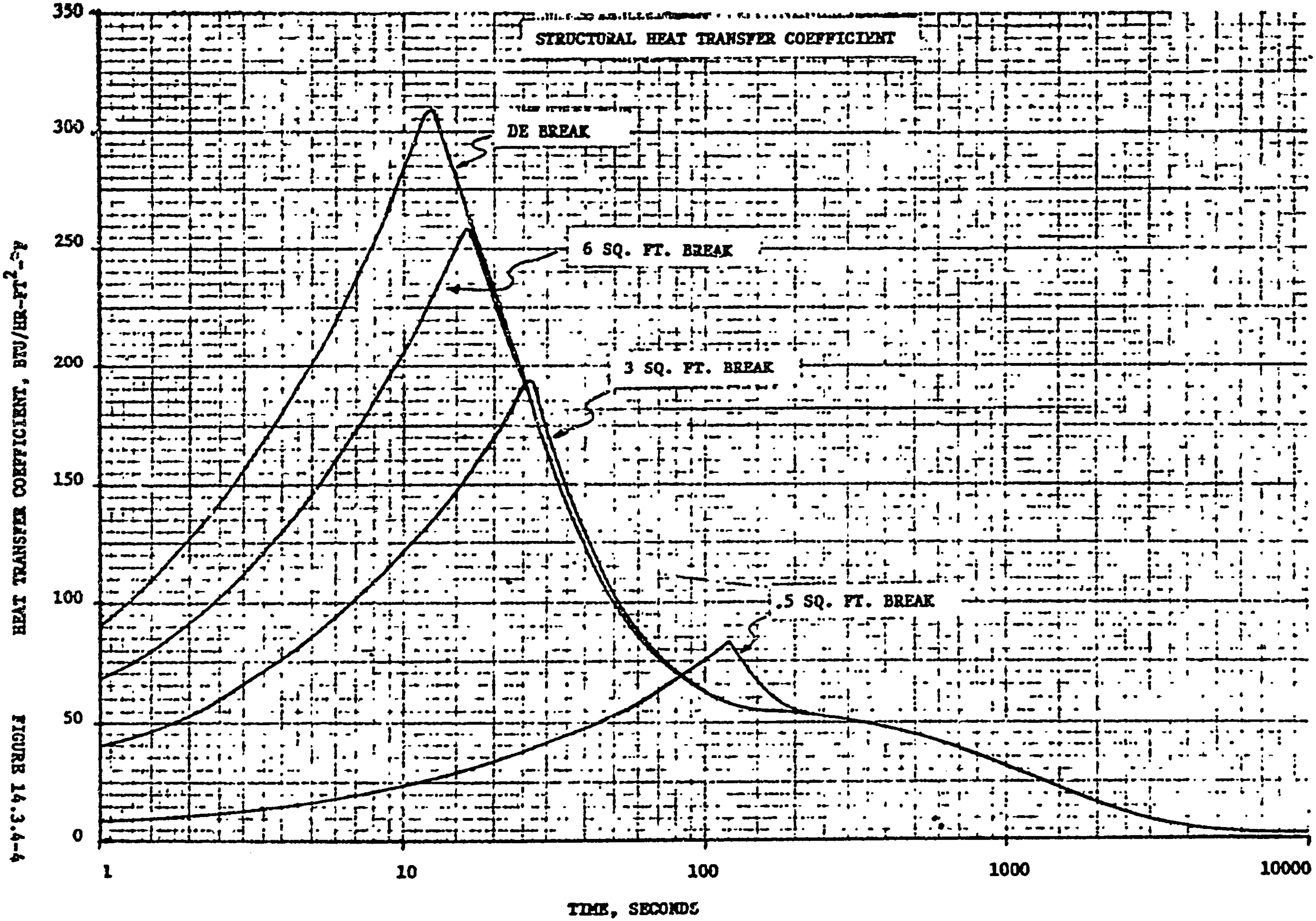

FIGURE 14.3.4-4

CONTAINMENT CAPABILITY STUDY

CONTAINMENT PRESSURE VS STEAM-AIR INTERNAL ENERGY

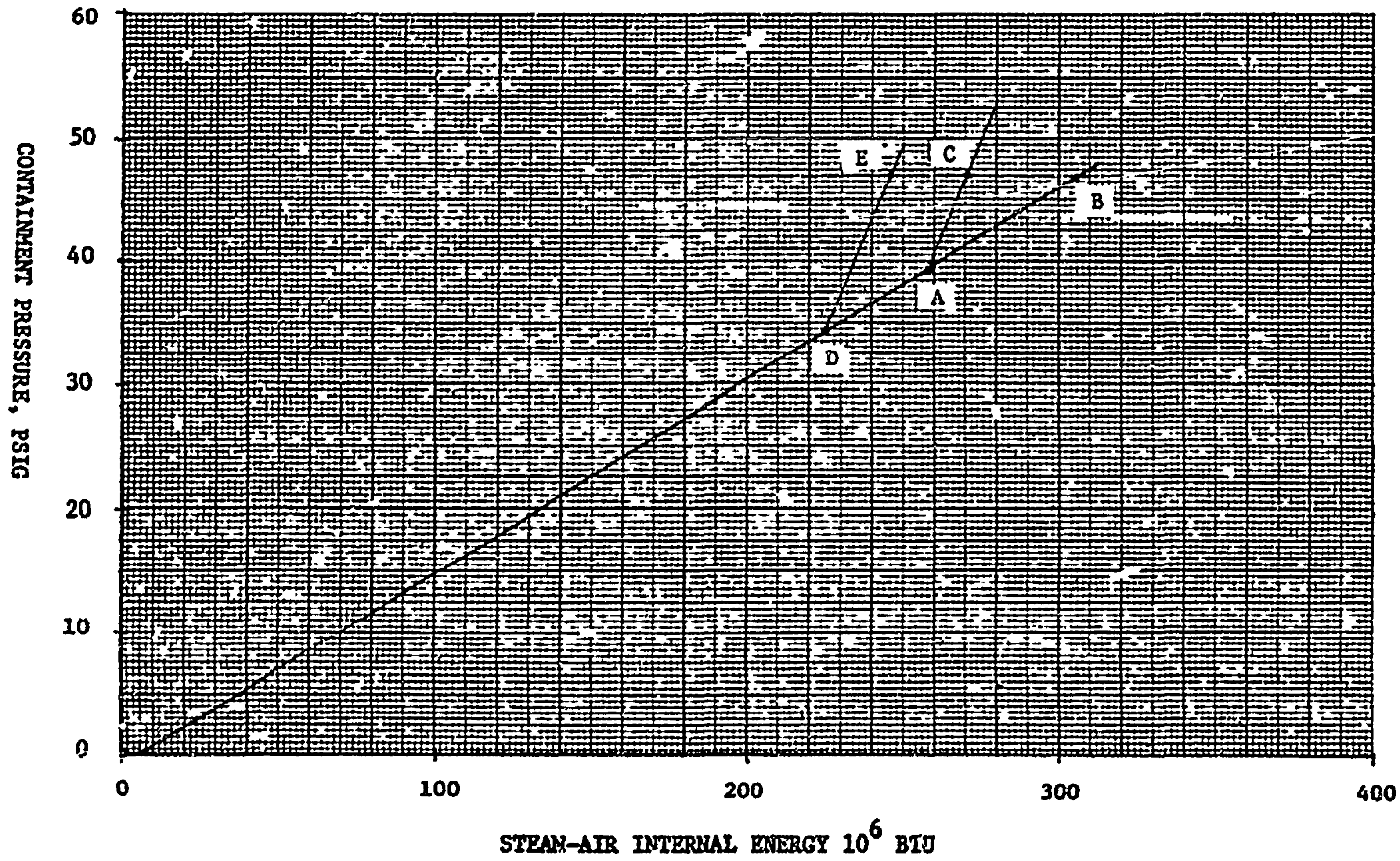

FIGURE 14.3.4-5

CONTAINMENT CAPABILITY STUDY

ALL AVAILABLE ENERGIES

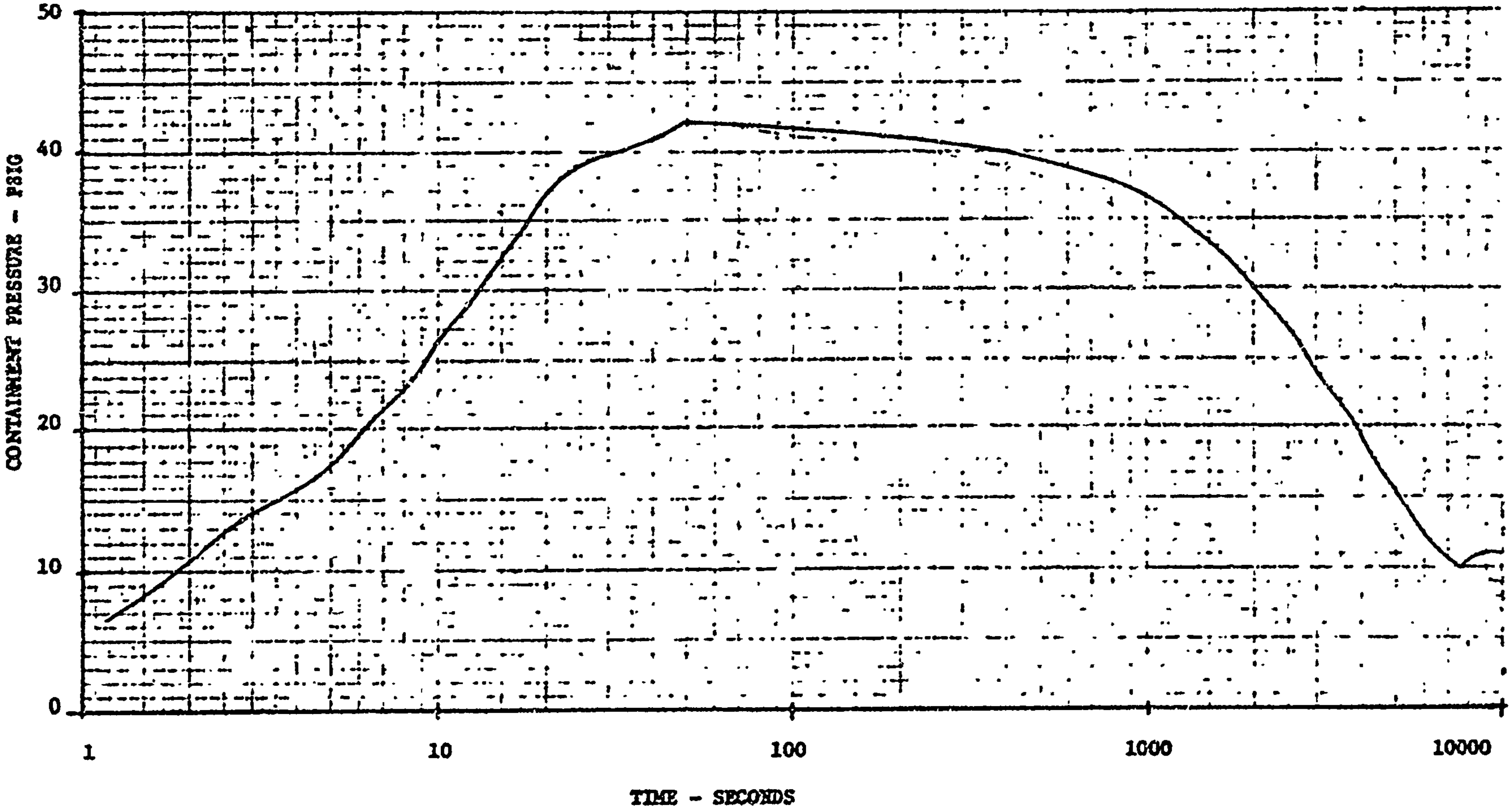

FIGURE 14.3.4-6

CONTAINMENT CAPABILITY STUDY

$ZR-H_2O$ REACTION - (33%)

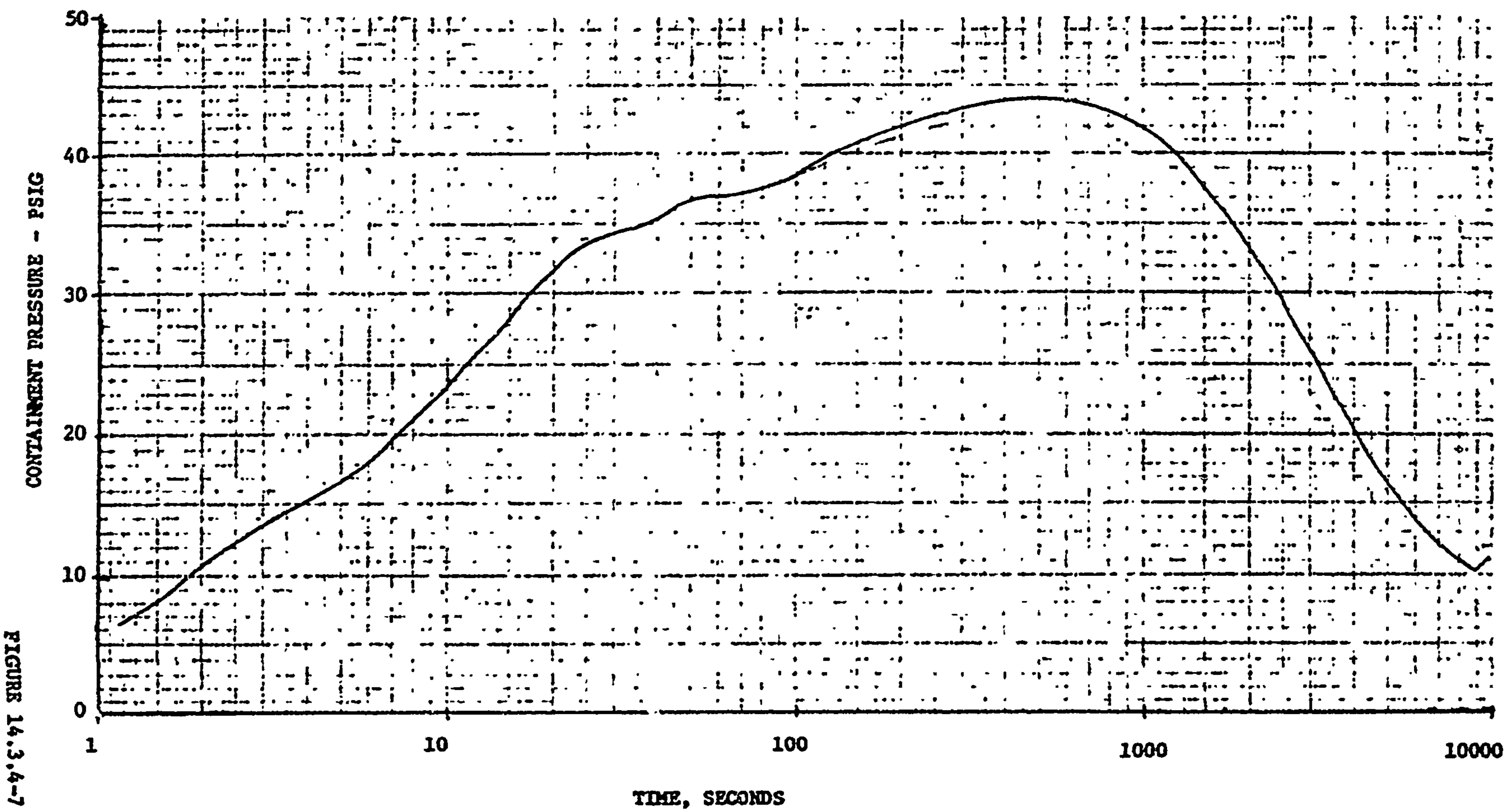

FIGURE 14.3.4-7

CONTAINMENT PRESSURE TRANSIENT

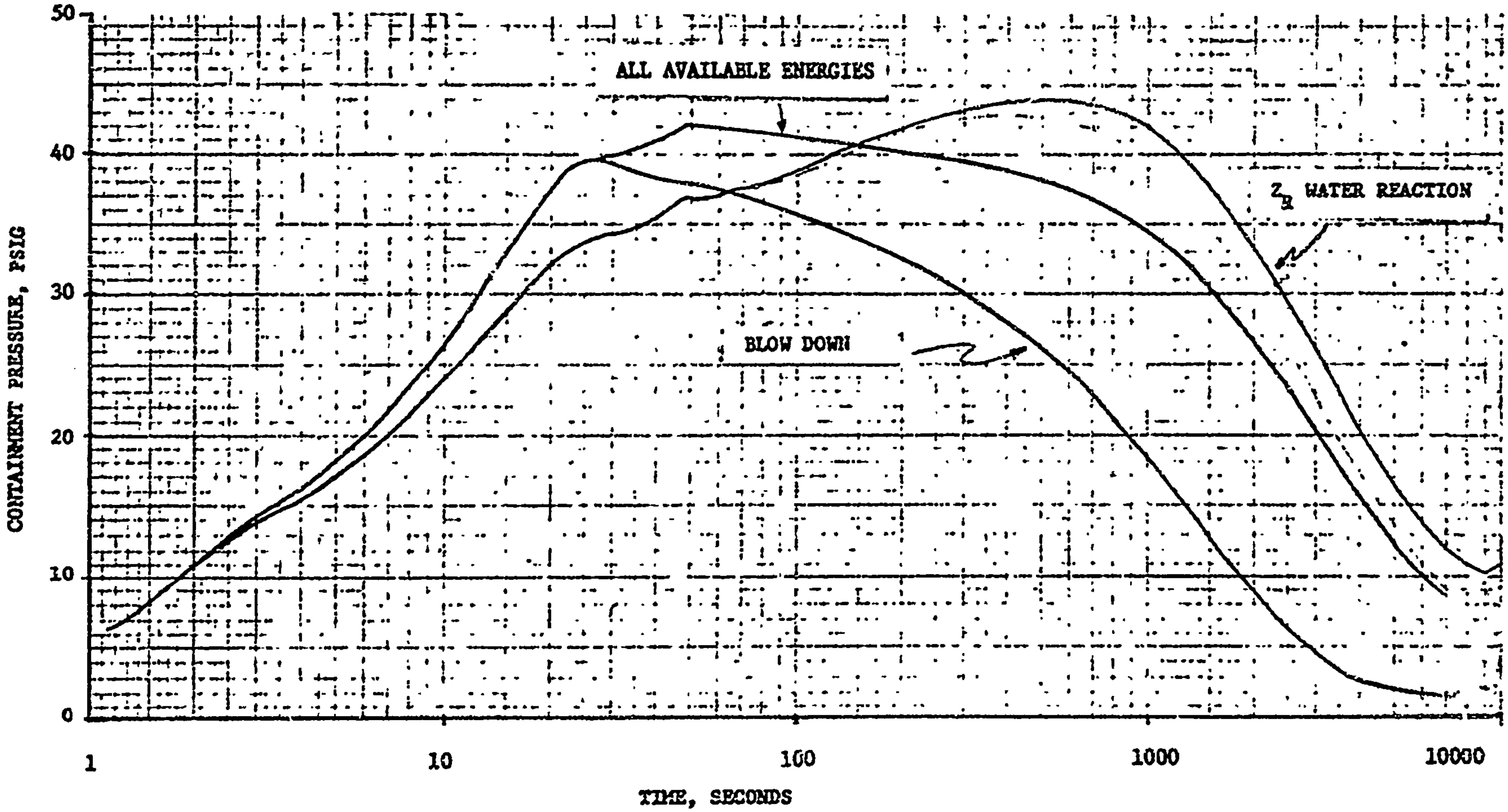

FIGURE 14.3.4-8

CONTAINMENT CAPABILITY STUDY
RATE OF ENERGY ADDITION

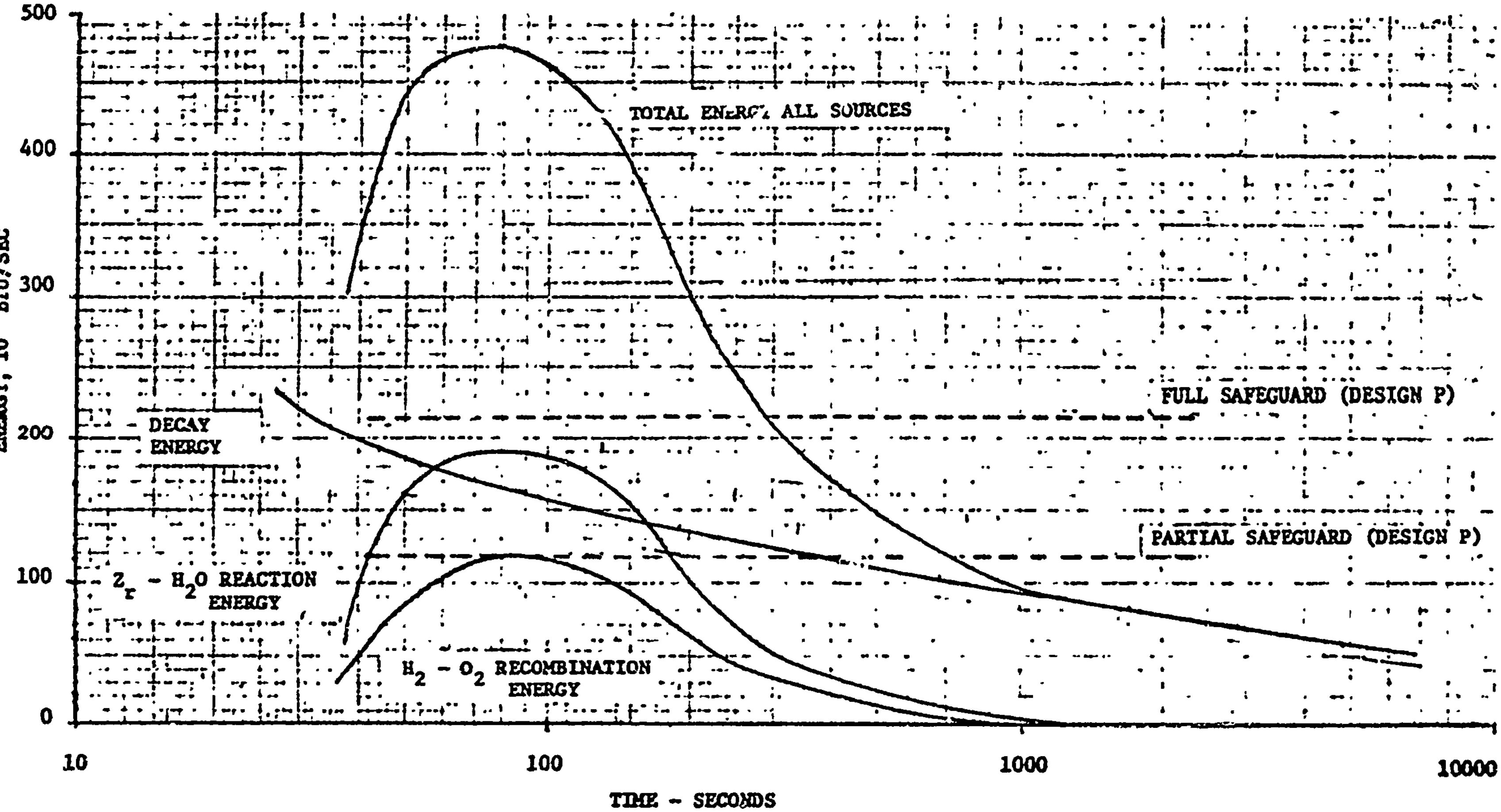

FIGURE 14.3.4-9

CONTAINMENT CAPABILITY

CASE 1

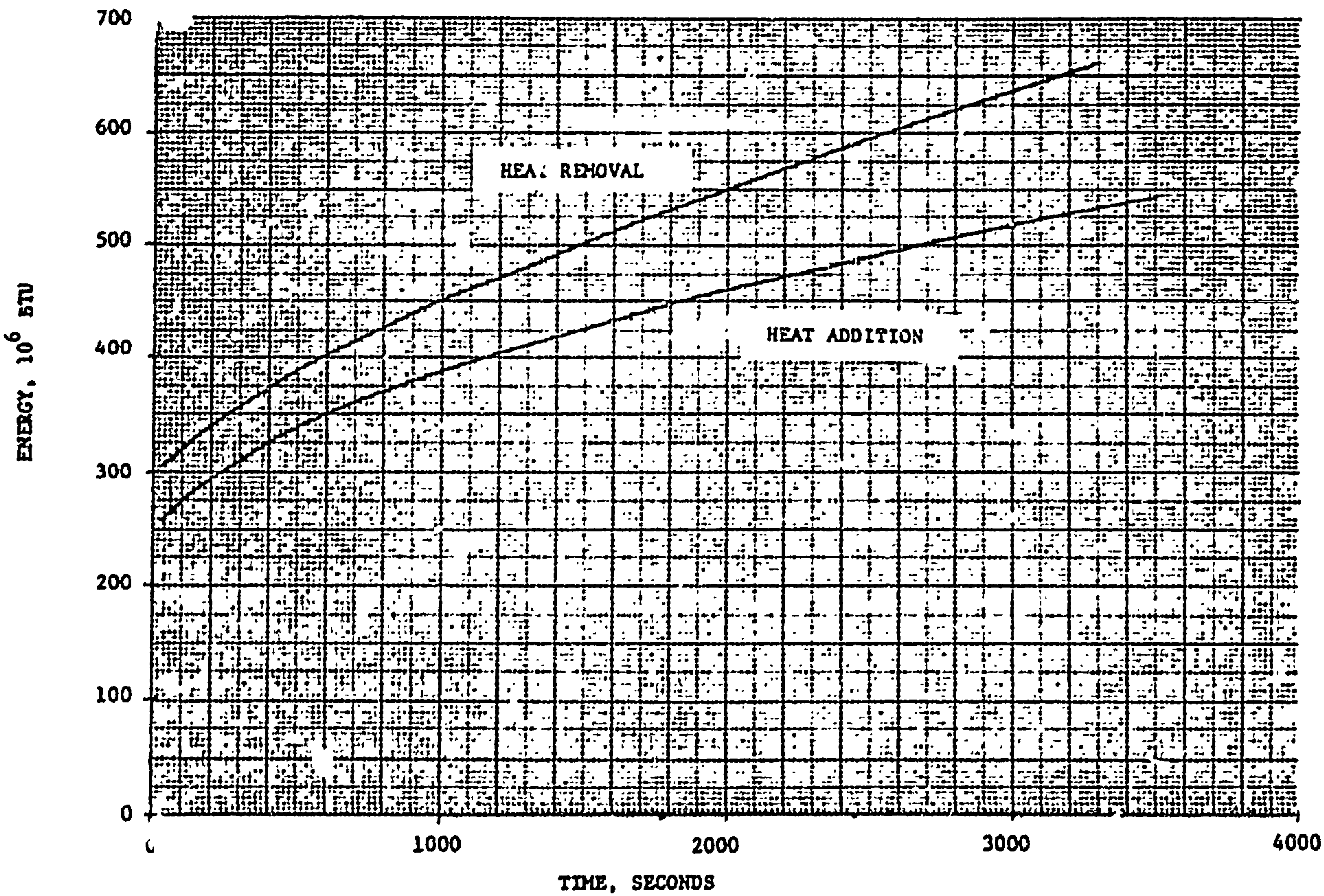

FIGURE 14.3.4-10

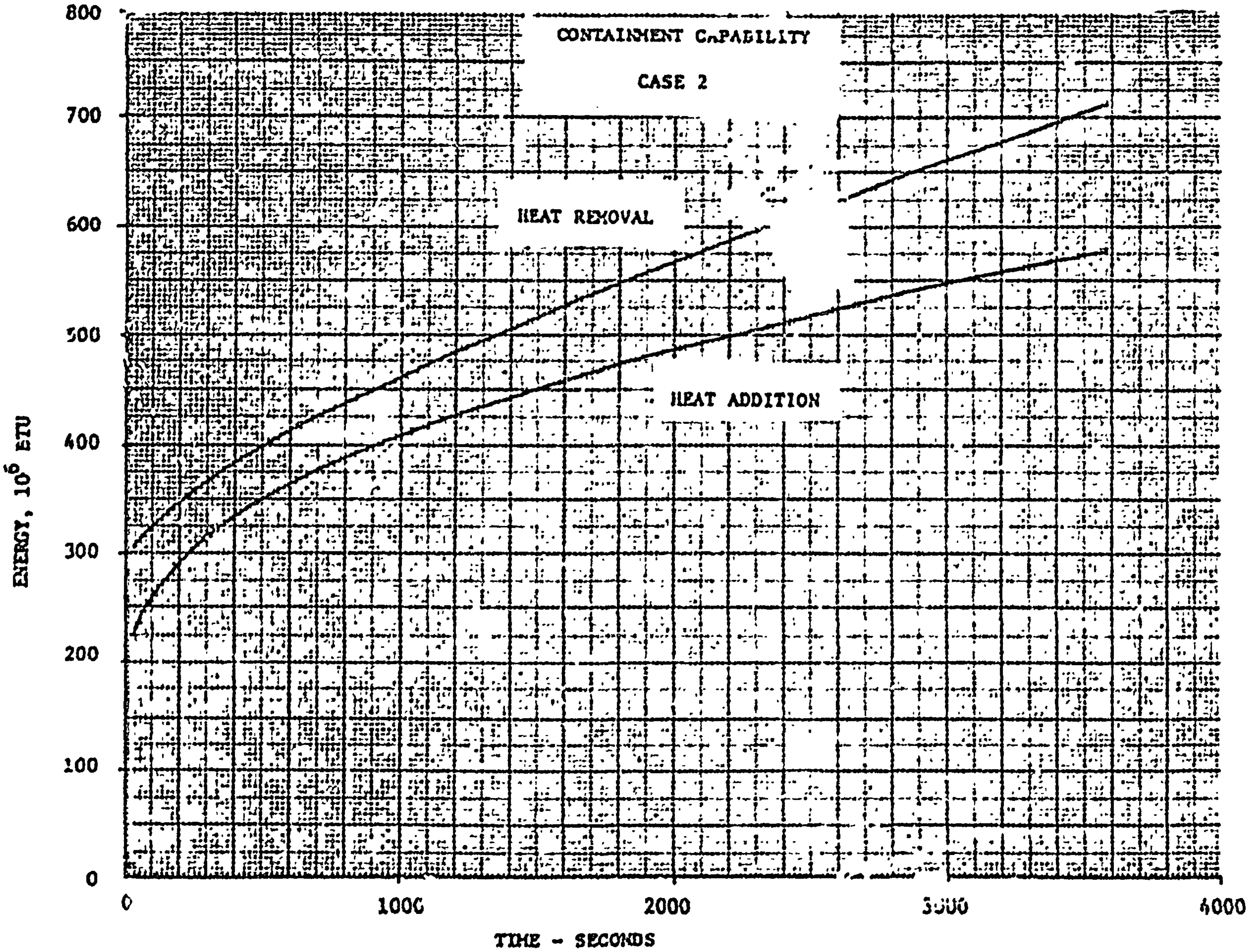

FIGURE 14.3.4-11

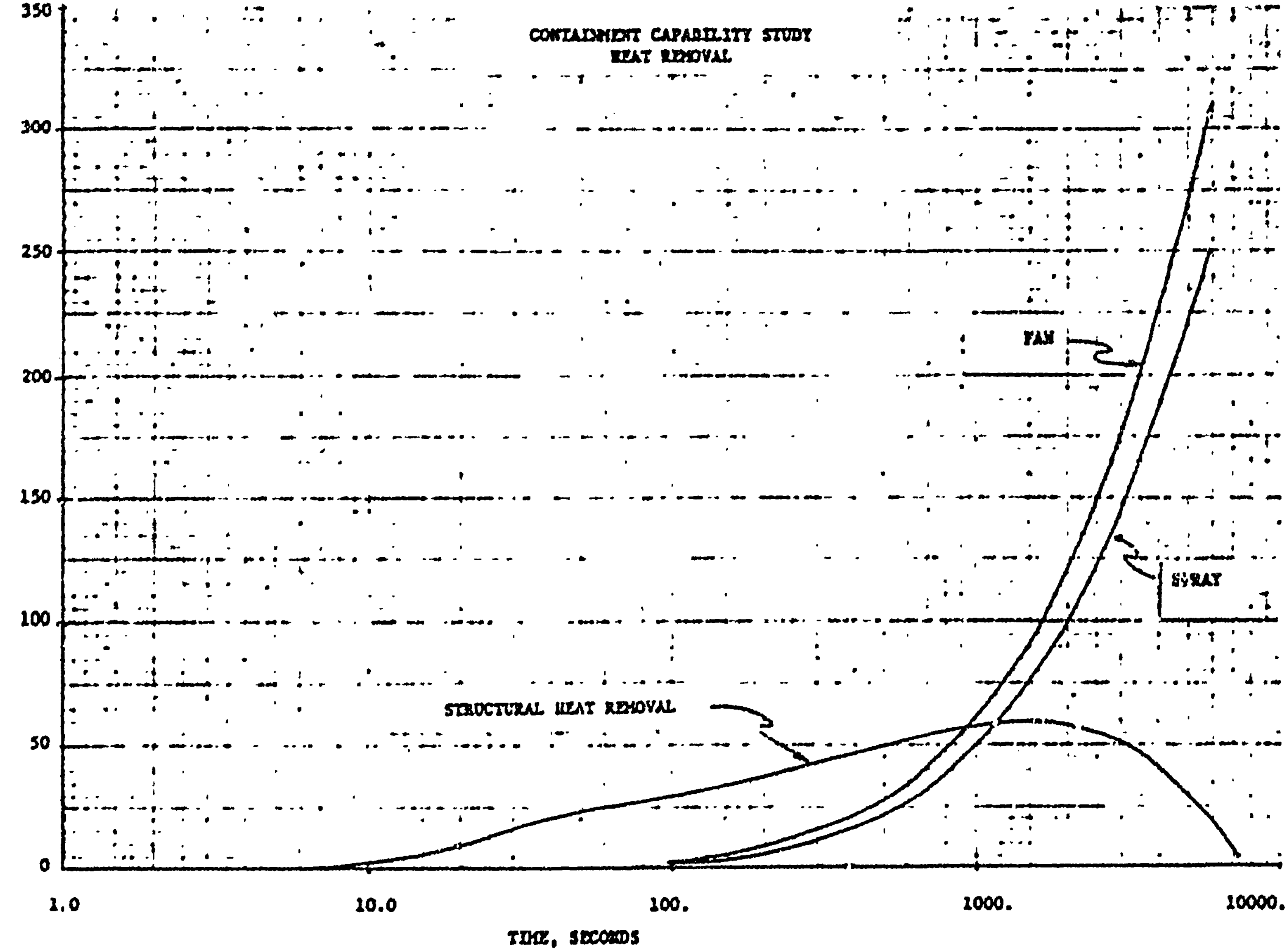

FIGURE 14.3.4-12

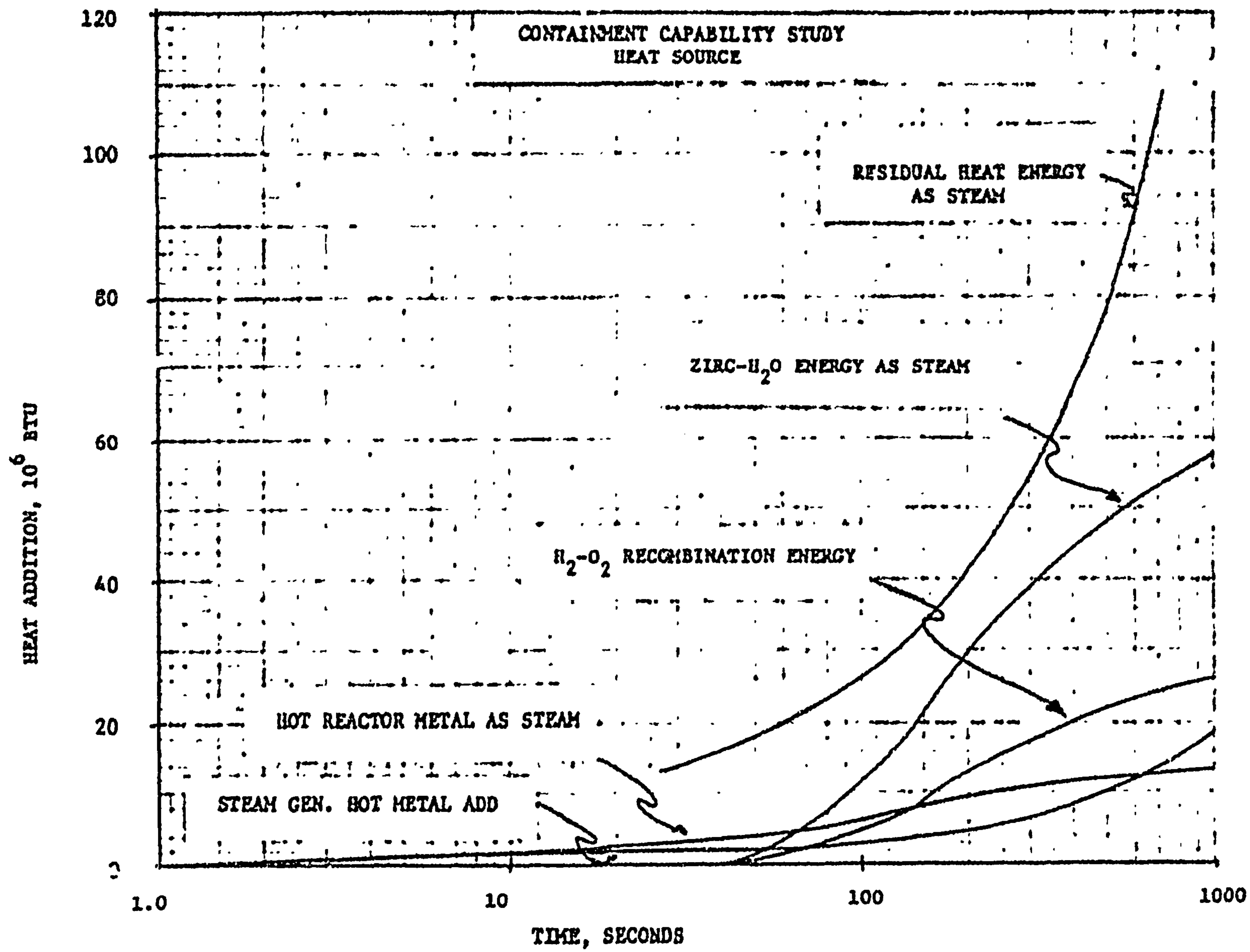

FIGURE 14.3.4-13

HYDROGEN GAS GENERATION

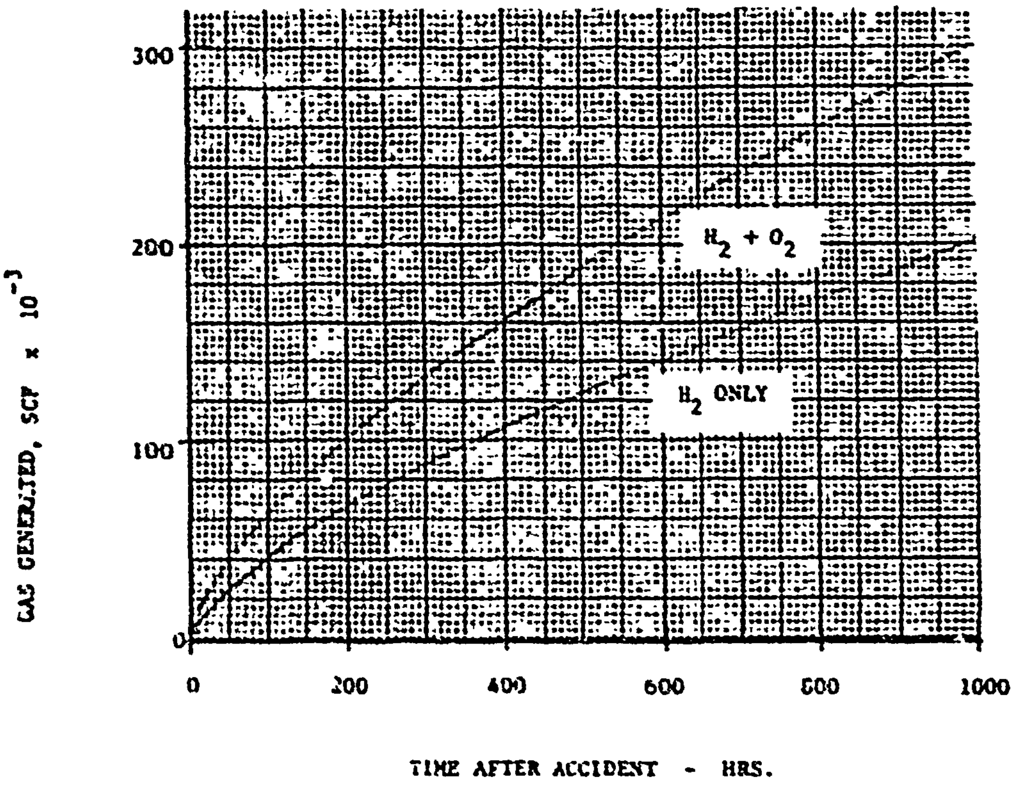

FIGURE 14.3.4-14

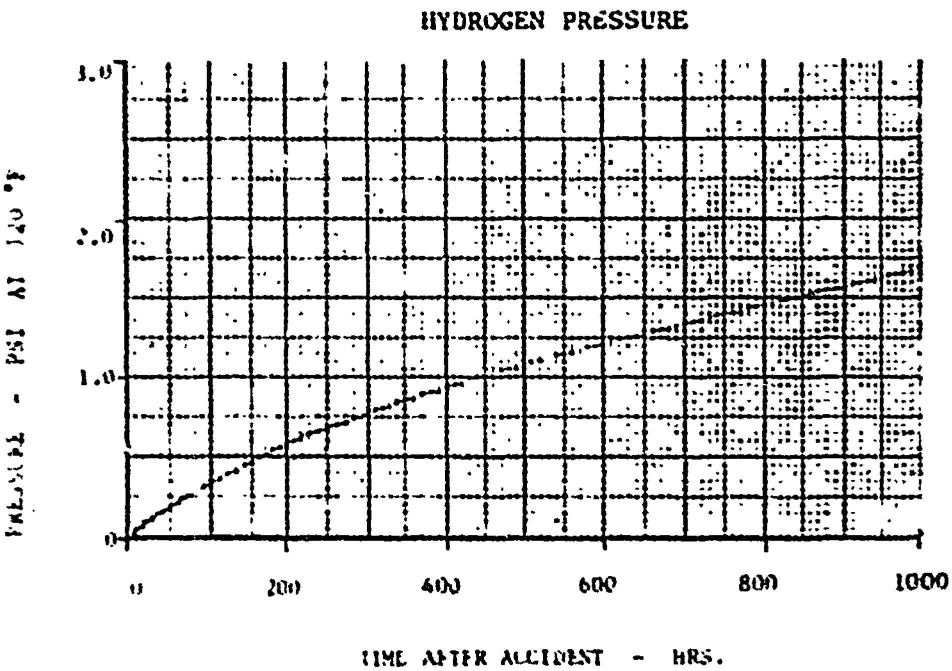

FIGURE 14.3.4-15

VOLUME PERCENT HYDROGEN

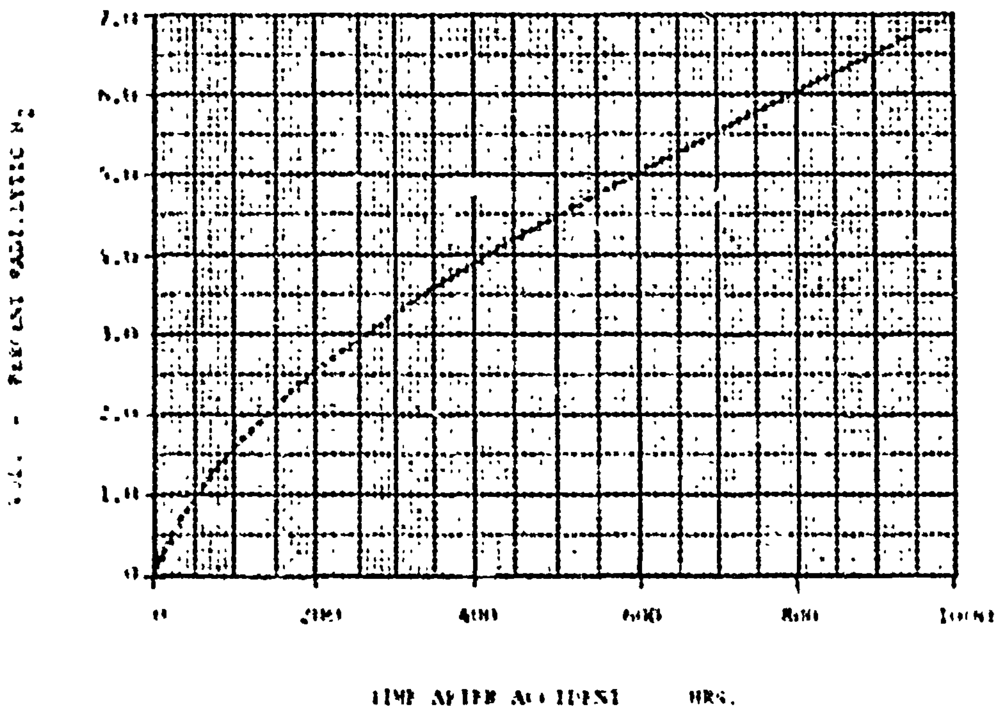

FIGURE 14.3.4-16

14.3.5 ENVIRONMENTAL CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT

General

Chapters 5 and 6 describe the protective systems and features which are specifically designed to limit the consequences of a major loss of coolant accident. The capability of the safety injection system for preventing melting of the fuel clad and the ability of the containment and containment cooling systems to absorb the blowdown resulting from a major loss of coolant have been discussed in Section 14.3.4. The capability of the safeguards in meeting dose limits set in 10 CFR 100 is demonstrated in this section.

Because of the design conservatism and care taken during fabrication and installation of the Reactor Coolant System a break of the system integrity of any size is considered highly unlikely. For break diameters up to four inches clad damage is not expected and hence activity release to the containment would be limited to that contained in the coolant, that is the corrosion product activity and fission product activity diffused to the coolant through assumed fuel defects. For larger break sizes up to and including the hypothetical double ended rupture of a coolant loop clad rupture occurs and a portion of the activity contained in the fuel pellet-clad gap would be released to the containment and would be available for leakage.

For the purpose of evaluating radiation exposure, a double ended rupture of a reactor coolant loop is considered with partial safeguards operating from the diesel-generator power system. As shown in Section 14.3.2 the safety injection system, with diesel-generator power from two of the three units, will maintain clad temperature well below the melting point of Zircaloy-4 and will limit zirconium-water reaction to an insignificant amount. As a result of the cladding temperature increase and the rapid system depressurization, however, cladding failure may result in the hotter regions of the core. Release of the inventory of the volatile fission products in the pellet-cladding gap might follow.

It is assumed that all of the gaseous activity present in the pellet-cladding gap of all the fuel rods is released. The ability of the safeguards to limit environmental activity release and hence whole body and thyroid dose is analyzed.

Effectiveness of Safeguards for Limiting Activity Release

In the following sections the expected iodine removal performance of the spray and fan-cooler systems is described and the containment and isolation features are discussed.

Effectiveness of Containment and Isolation Features in Terminating Activity Release

The reactor containment serves as an activity leakage limiting boundary. The containment is steel lined and designed to withstand internal pressure in excess of that resulting from the design basis loss of coolant accident (Section 5). All weld seams and penetrations are designed with a double barrier to inhibit leakage. In addition the weld channel and penetration pressurization system supplies a pressurized nitrogen seal, at a pressure above the containment design pressure, between the double barriers so that if leakage occurred it would be into the containment (Section 6.5). The Containment Isolation System, Section 5.2, provides a minimum of two barriers in piping penetrating the containment. The Isolation Valve Seal Water System, Section 6.6, provides a water seal at a pressure above containment design in the piping lines that could be a source of leakage. The containment is designed to leak at a rate of less than 0.1 percent per day at design pressure without including the benefit of either the Isolation Valve Seal Water System or the Penetration Pressurization System. The weld seams and penetrations are pressurized continuously during reactor operation causing zero outleakage through these paths. The Isolation Valve Seal Water System is actuated on the containment isolation signal and prevents leakage through the pipelines which could be a leak source. This system would be actuated within one minute to terminate containment leakage.

Effectiveness of Spray System for Iodine Removal

The effectiveness of the spray system for removal of inorganic iodine from the containment atmosphere is evaluated in detail in Appendix 6A, "Iodine Removal Effectiveness Evaluation of the Containment Spray System".

If there is a large excess of chemical reagent to react with the iodine and convert it to a non-volatile form with little or no tendency to return to the gas phase, the elemental iodine removal rate of spray can be expressed by:

$$\frac{dA}{dt} = -\lambda_s A \qquad (1)$$

where:

A = inventory of elemental iodine which is available for leakage at any time, t

λ_s = elemental iodine removal coefficient.

Integration of equation (1) gives:

$$A = A_o e^{-\lambda_s t} \qquad (2)$$

As discussed in Appendix 6A, an elemental iodine removal coefficient of 64 hr^{-1} is expected for the case of both spray pumps operating and 32 hr^{-1} for one spray pump operating. Further analytic work has been performed by Westinghouse and is reproted in WCAP 7499-L, Westinghouse Proprietary. This report verifies the conservatism inherent in the Griffeth single drop model by an assessment of each of the uncertainties associated with the model used to determine the iodine removal coefficient.

Effectiveness of Fan-Cooler Filter System for Iodine Removal

The removal of iodine by the filters installed in the fan-cooler units is exponential as is the removal of any soluble component under conditions of a constant mass transfer rate:

$$A = A_o e^{-\lambda_f t} \qquad (3)$$

The formula has the same form as formula (2) developed in the previous section for the spray system.

Since the filters installed in the fan-cooler units have different efficiencies for removal of elemental and organic iodine, two values of λ_f have to be specified.

$$\lambda_{fi} = \eta_i \frac{F}{V_c} \qquad (4)$$

$$\lambda_{fo} = \eta_o \frac{F}{V_c} \qquad (5)$$

where:

λ_{fi} = elemental iodine removal coefficient due to filters

λ_{fo} = organic iodine removal coefficient due to filters

η_i = elemental iodine filter efficiency

η_o = organic iodine filter efficiency

F = air flow rate through the filters

V_c = containment free volume

Elemental iodine removal efficiency by filters has been studied in a number of tests.(1,2,3,4) Efficiencies of 99.9 percent were measured for impregnated charcoal filters in humid air.(4) Efficiencies were 99.5 percent even when the charcoal was water logged. In this analysis a conservative removal efficiency of 90 percent for inorganic iodine has been assumed.

Organic Iodine Inventory and Removal Effectiveness

It is not expected that significant organic iodine would be liberated from the fuel as a result of a loss-of-coolant accident. This conclusion is based on the absence of indications of such release in in-pile fuel meltdown experiments conducted by Oak Ridge National Laboratory. The fraction expected would be on the order of 0.2 percent or less on the basis that in the region of the fuel rod where conditions would be most favorable for the existence of organic iodine, the rates of thermal radiolytic decomposition would exceed the rate of replenishment.

The more plausible mechanism for organic iodine formation is by reaction of elemental iodine in an absorbed state on organic-contaminated surfaces. Whether limited by diffusion to the surface or by the reaction rate of absorbed iodine, the resulting fractional conversion of airborne iodine per unit time is proportional to the surface to volume ratio of the enclosure. Therefore, observed yields of organic iodine function of aging time in various test enclosures were extrapolated to the Indian Point Unit #2 containment in proportion to the surface/volume ratio. These results, in no case exceeded a calculated conversion rate of 0.0035 percent of the atmospheric iodine per hour.

At this rate, the formation of organic iodine has a negligible effect on the consequences of containment leakage. In short, the mechanisms which are believed to have produced significant amounts of organic iodine in test facilities would be so diminished in effect by the vastly reduced relative surface to volume in the plant containment, that the organic iodine component will be of minor importance.

Recent experiments have shown that the formation of organic forms of iodines is largely dependent on specific conditions of the test such as activity concentration, pressure, temperature, humidity, radiation field level presence of impurities, etc. Since a full scale test has not yet been performed for this purpose, it has been commonly agreed among experimenters on the basis of the data available at the present time, that the amount of iodine which could possibly exist in the reactor containment after the MCA in organic form will not exceed five percent of the total airborne fraction. Removal of the organic fraction has been the subject of many research programs in the past few years and the results show an appreciable success both with the use of impregnated filters and surfactants added to the spray solution.

Experiments have been conducted at ORNL, and reported in ORNL-3864, in which methyl iodine (CH_3I) was exchanged with commercially impregnated charcoals. The test data, obtained at 168°F and 100 percent relative humidity predict an efficiency of >98 percent for a filter bed of the reference dimensions and gas velocity at these conditions.

A test to evaluate the efficiency of activated impregnated charcoal filters for Connecticut Yankee resulted in filter efficiencies of 90 percent for methyl iodine with 100 percent humidity.(5) Although recent tests at ORNL (reported in ORNL-4040 and ORNL-4140) indicated significantly lower efficiencies with high humidity, the results are not representative because the charcoal was partially water logged for those tests. The Indian Point Unit No. 2 moisture separators are designed to remove a minimum of 99.9 percent of the entrained water in the air-steam entering the air handling units following a loss-of-coolant and the moisture separator effluent has essentially zero moisture content.

The data available indicates that a methyl iodine removal efficiency of 70 percent or greater can be expected for the activated and impregnated charcoal filter units. A value of 70 percent has been assumed for the filters in the analysis. No credit has been taken for methyl iodine removal by the sodium hydroxide spray.

Activity in the Fuel Rod Gap

The gap activity is computed based on buildup in the fuel from the fission process and diffusion to the fuel rod gap at rates dependent on the operating temperature. For analysis, the fuel pellets are considered divided into regions, each with a release rate dependent on the mean fuel temperature within that region.

The diffusion coefficient, D', for Xe and Kr in UO_2, varies with temperature through the following expression:

$$D'(T) = D'(1673) \exp\left[-\frac{E}{R}\left(\frac{1}{T} - \frac{1}{1673}\right)\right] \quad (6)$$

where:

E = activation energy

D'(1673) = diffusion coefficient at 1673°K = 1×10^{-11} sec^{-1}

T = temperature in °K

R = gas contant

The above expression is valid for temperatures above 1100°C. Below 1100°C fission gas release occurs mainly by two temperature independent phenomena, recoil and knock-out, and is predicted by using D' at 1100°C. The value used for D' (1673°K), based on data at burnups greater than 10^{19} fissions/cc, accounts for possible fission gas release by other mechanisms and pellet cracking during irradiation.

The diffusion coefficient for iodine isotopes is assumed to be the same as for Xe and Kr. Toner and Scott[6] observed that iodine diffuses in UO_2 at about the same rate as Xe and Kr and has about the same activation energy. Data surveyed and reported by Belle[7] indicates that iodine diffues at slightly slower rates than do Xe and Kr.

For a full core cycle at 3216 MWt, the above analysis results in a halogen pellet-clad gap activity of about three percent of the equilibrium core I-131 inventory. The noble gas activity present in the pellet-clad gap and assumed release to the containment is about 2.5 percent of the core Xe-133 inventory.

The percentage of the total core activity present in the gap for each isotope is also listed in Table 14.3.5-1. The core temperature distribution used in this analysis, based on the design hot channel factors is presented in Table 14.3.5-1 also.

Atmosphere Dispersion

The meteorological dispersion of the leakage from the containment has been calculated using the Sutton dispersion model and dispersion parameters measured at the Indian Point site. The Sutton model has been modified to account for additional dispersion of the leakage due to turbulence in the wake of the containment building. Conservative dispersion characteristics applicable to three time periods were selected and the doses calculated for each period (Section 2.6).

The Sutton equation for the dispersion of a point source at ground level gives the ground level plume concentration as a function of distance.

$$X = \frac{2Q}{\pi \bar{u} C_y C_z x^{2-n}} e^{-(y^2/C_y^2 x^{2-n})} \qquad (7)$$

where $C_y C_z$ and n are the dispersion parameters, $\bar{u}$ is the wind speedy, y is the lateral distance from the plume center line, x is the downwind distance and Q is the point source release term.

In order to take into account building dilution, the Sutton equation is applied to a virtual point source upwind from the containment. The distance of this source from the building is obtained by the requirement that the dispersion factors σ_y and σ_z of the gaussian distribution obey the relationships:

$$4 \sigma_y = \sqrt{A}$$

$$4 \sigma_z = \sqrt{A}$$

where A is the cross sectional area of the containment building. Thus σ_y and σ_z yield each a value for the distance; the geometric average of those values is the distance x_o upwind of the virtual source.

$$x_o = \left(\frac{A}{8\, C_y\, C_z}\right)^{1/(2-n)}$$

The modified Sutton equation becomes:

$$\chi = \frac{2Q}{\pi\, \bar{u}\, C_y C_z (x + x_o)^{2-n}}\; e^{-(y^2/C_y^2)(x + x_o)^{n-2}} \qquad (8)$$

The first and second periods of the dose calculation utilized this modified dispersion formula, a building area of 2000 square meters, and the inversion parameters assumed in TID 14844 which are conservative for the Indian Point Site.

Category	C_y	C_z	n	$\bar{u}$	x_o
Inversion-I	0.4 $m^{n/2}$	0.07 $m^{n/2}$	0.5	1 m/sec	430 m

The first period comprises the first two hours after the accident. The direction of the 1 meter per second wind is assumed to be constant throughout the period.

The second period is the next 22 hours after the accident during which the same inversion condition is assumed to exist, but the average wind speed from the same direction is assumed to be 2 meters per second.

The third period is from 24 hours after the accident to 31 days after the accident. During this period, the meteorological conditions are assumed to be randomly distributed among the categories listed below:

Category i	Fraction Fi	$1/\bar{u}$	C_z	C_y	n
Lapse - L_1	0.137	0.575	0.48	0.6	0.2
Lapse - L_2	0.061	0.191	0.43	0.53	0.3
Neutral - N	0.378	0.358	0.39	0.47	0.4
Inversion - I	0.424	0.493	0.07	0.40	0.5

The parameters, $\bar{u}$, C_y, C_z, n for L_1, L_2 and N are those measured at the site (Section 2.6) and those for I are the TID 14844 assumptions. Because the winds are not expected to be from the same direction throughout the 30-day peri~d, the dispersion formula was modified to account for long term variability of the mean wind direction. The most adverse distribution was assumed to result in a maximum of 35 percent of the winds blowing in one 20^{o} section. The dispersion formula used is:

$$(x/Q)_{30} = \frac{2f}{\beta \sqrt{\pi}} \sum_i \frac{F_i}{\bar{u}_i \, C_{x_i} \, x^{(2-n_i/2)}} \qquad (9)$$

This expression is obtained by integrating the Sutton equation from $-\infty$ to $+\infty$ in the y direction and then averaging the concentration over the desired sector, β, for the appropriate fraction of the time, f. The other parameters have been defined with F_i being the fraction of the time any particular weather category exists. As stated, $\beta = 0.353$ (2 tan 10^{o}) and f + 0.35.

Based on the above data, the dispersion factors listed in Table 14.3.5-2 are obtained. These are also plotted in Figure 14.3.5-1.

Method of Analysis

To evaluate the ability to meet the suggested 10 CFR 100 guideline, the thyroid dose and the whole body dose are calculated as a function of distance from the reactor. Results are presented for off-site exposure at the site boundary and low population zone distance while on site exposure is evaluated in the control room.

Off-Site Inhalation Dose

The thyroid dose resulting from iodine activity leaking from the reactor containment following an accident is computed from:

$$D(X,T) = \int_{t=o}^{t=T} B(t)\, \frac{\chi}{Q}(x,t)\, L(t)\, S(t)\, dt \qquad (10)$$

where

$$S(t) = \sum_{j=1}^{j=5 \text{ isotopes}} \{C_j DCF_j [(1 - \beta) e^{-(\lambda_{si}+\lambda_{fi}+\lambda_j)t} + \beta e^{-(\lambda_{so}+\lambda_{fo}+\lambda_j)t}]\}$$

$D(X,T)$	is the thyroid dose accumulated to time T at X meters from the containment, rem
$B(t)$	is the breathing rate, m^3/hr
$\chi/Q(X,t)$	is the dispersion factor at distance X and time t, sec/m^3
$L(t)$	is the fractional containment leak rate, sec^{-1}
C_j	is the total curies of iodine isotope j released to the containment, curies
DCF_j	is the dose conversion factor for isotope j, rem/curie-inhaled
β	is the fraction of the initial airborne iodine inventory which is in the organic form
λ_{si}	is the rate removal coefficient of the containment spray for inorganic iodine, hr^{-1}
λ_{fi}	is the rate removal coefficient of the charcoal filters for inorganic iodine, hr^{-1}
λ_j	is the natural decay constant for isotope j, hr^{-1}
λ_{so}	is the rate removal coefficient of the containment spray for organic iodine, hr^{-1}
λ_{fo}	is the rate removal coefficient of the charcoal filters for organic iodine, hr^{-1}

The above expression accounts for spray and filter removal of the inorganic and organic forms and for isotopic decay.

Figure 14.3.5-2 is a general plot that can easily be used to obtain dose for various combinations of initial fraction of iodine released and fraction of this in the organic form, and removal coefficients for organic and inorganic iodine. The figure presents the resultant 2 hr thyroid dose at the site boundary and the 30-day thyroid dose at the low population zone as a function of a variable single iodine removal coefficient. The two curves are based on 25 percent iodine release to the containment and include: 1) the effect of iodine natural decay, 2) the variable containment leak rate given in Table 14.3.5-3, 3) the breathing rate changes given in Table 14.3.5-3, and 4) the variation in atmospheric dispersion as given in Table 14.3.5-3. The thyroid doses for any given combination of halogen release fraction, f, λ_i (total inorganic iodine removal coefficient = $\lambda_{si} + \lambda_{fi}$), λ_o (total organic iodine removal coefficient = $\lambda_{so} + \lambda_{fo}$) and β_o(fraction of the initial airborne, iodine inventory in the organic form) are given by:

$$D^{site}_{2\ hr} (\lambda_i,\lambda_o,\beta_o,f) = \frac{f}{0.25} \beta_o D^{site}_{2} (\lambda_o) + \frac{f}{0.25} (1 \quad \beta_o) D^{site}_{2} (\beta_i) \qquad (11)$$

$$D^{LPZ}_{30d} (\lambda_i,\lambda_o,\beta_o,f) = \frac{f}{0.25} \beta_o D^{LPZ}_{30} (\lambda_o) + \frac{f}{0.25} (1 - \beta_o) D^{LPZ}_{30} (\lambda_i) \qquad (12)$$

where f is the fraction available for leakage and the values of D (λ) are taken from the appropriate curve of Figure 14.3.5-2.

As developed in Appendix 6A, the spray removal coefficient for inorganic iodine is 32 hr^{-1} for one spray pump and 64 hr^{-1} for both spray pumps operating. The removal capability of the charcoal filter fan units is evaluated based on the 2.6 x 10^6 ft^3 containment free volume, a recirculation rate of 65,000 CFM per unit, and efficiencies of 90% and 70% for the inorganic and organic iodine forms respectively.

Off-site Whole Body Dose

The gamma dose from the radioactive cloud is calculated using the method of the semi-infinite cylinder. The expression, derived from the Reactor Shielding Design Manual, by T. Rockwell, 1956, has been modified to account for the specific site meteorology as follows:

$$D_\gamma = 327 \sqrt{\frac{\chi/Q}{\bar{u}}} \sum_i Q_i \sum_k \mu_{sk} E_{\gamma k} \eta_k \text{ rem} \quad (13)$$

where the subscript i refers to each released isotope; the subscript k refers to the energy levels.

E_γ = energy per photon, Mev/γ
η_γ = energy relative abundance, $\frac{\gamma}{dis}$
μ_s = total attenuation coefficient in air, cm^{-1}
Q = curies released, curies
$\bar{u}$ = wind velocity, cm/sec
$\frac{\chi}{Q}$ = atmospheric dispersion factor, $\frac{sec}{m^3}$

Dose from β emitters is calculated as follows:

$$D_\beta = 0.27 (\chi/Q) \sum_i Q_i E_{\beta i} \text{ rem} \quad (14)$$

where E_β = Beta energy per disintegration, Mev

Cases Analyzed

1) One Minute Isolation of Containment - Gap Release

With the isolation valve seal water system and the penetration pressurization system provided to block potential leak paths the leakage would terminate within one minute, the approximate actuation time for the above systems. For this case it is assumed that fuel-clad gap activity (equivalent to 3% of the core I-131, 2.5% core X_e-133) is released to the containment and that the containment leaks at its design rate for one minute at which time leakage terminates. The resultant dose beyond the site boundary would be less than 0.7 rem to the thyroid.

The whole body dose from the fraction that would leak out of the containment in the one minute required for the isolation valve seal water system and the penetrating pressurization system to terminate leakage would be 1.2 mr beyond the site boundary. The sources for the gap inventory are listed on table 14.3.5-1.

The exposure of the occupants of the control room has been evaluated using the dispersion factor, $\frac{X}{Q}$, of $2.3 \times 10^{-3} \frac{sec}{3}$ evaluated at 30 meters distance from the containment. Immediately after the accident, the control room ventilation ducts will close under high radiation level alarm. Allowing for 1 minute intake before internal recirculation through filters is started, the thyroid exposure is 1.2 rem inside the control room during the iodine clean up period. Inhalation dose to a person on the ground outside the building for 1 minute during which the cloud is passing by is 2 rem. Whole body exposure is caused by the amount of noble gases trapped in the control room during the one minute interval.

The initial β and λ dose rate from these activities would be less than 0.4 mrem/hour.

The total whole body dose depends on the duration prior to purging the control room.

2) Containment Leaks at Design Rate - Gap Release

The capability of the safeguards systems has also been evaluated without taking credit for the leakage reduction afforded by the by the seal water and penetration pressurization systems. For these cases the integration has been performed over three time intervals with B(t), $\frac{\chi}{Q}$, (x,t) and L(t) constant over each interval. The intervals are:

0 to 2 hours
2 hrs to 24 hrs
1 day to 30 days

The activity available for leakage is the same as in case (1), i.e., the fuel-clad gap activity.

It is assumed that iodine release terminates after 30 days because of pressure suppression and removal of airborne iodine activity inside the containment by the engineered safeguards. The values of B, χ/Q and L used for the three time intervals are given in Table 14.3.5-3. The values of λ_j and DCFj are also given in Table 14.3.5-3. The iodine available for release was assumed to be 2.5% of the total core inventory of each iodine isotope even though the tabulation in Table 14.3.5-1 shows that the gap fraction of the shorter half lived iodine isotopes is less.

Figures 14.3.5-3 and -4 show the effect of four different combinations of fan-filter and spray units operating on: the inhalation dose when the organic iodine inventory varies from zero to 5% of the total initial fraction available for leakage. The dose for the assumption of 25% of the iodine available for leakage (per TID-14844) is also shown for comparison on the same Figures. These cases are:

Case I   5 fans and 2 sprays
λ_i = 70.5 hr^{-1}, λo = 5.25 hr^{-1}

Case II   3 fans and 1 spray
λ_i = 36 hr^{-1}, λo = 3.1 hr^{-1}

Case III 5 fans, no spray

$\lambda_i = 6.75 \text{ hr}^{-1}$, $\lambda o = 5.25 \text{ hr}^{-1}$

Case IV no fans, 1 spray

$\lambda_i = 32 \text{ hr}^{-1}$, $\lambda o = 0.0 \text{ hr}^{-1}$

Figures 14.3.5-5 and -6 show the two hour and 30 day whole body dose from the leakage cloud taking no credit for leakage termination at one minute.

During the initial minute prior to isolating the control room ventilation system some iodine gets sucked into the control room and is cleaned up with filters as in case 1). Inhalation dose and whole body dose rate from trapped activity are the same as for case 1), i.e., 2.2 rem to the thyroid and 0.4 mrem/hr respectively.

In addition the gamma dose from the cloud around the building is listed below for three time intervals:

Time	Wind Speed Assumed	χ/Q to Control Room sec/m^3	Gamma Dose
0-2 hrs.	1 m/sec	2.3×10^{-4}	6 mrem
2-24 hrs.	2 m/sec	1.15×10^{-4}	25 mrem
1-30 days	4 m/sec	5.5×10^{-4}	41 mrem

The total 0-30 day gamma exposure in the control room from the cloud is less than 100 mrem.

3) TID-14844 Release Fraction

For purpose of comparison only, the TID-14844 assumptions are considered taking no credit for the capability of the safety injection system in preventing core meltdown sources from being present in the containment and available for leakage. The release of activity to the containment results in 25% of the halogens, 100% of the noble gases, and 1% of the solids being airborne.

Even with the assumption of core meltdown release sources, in the one minute required for the-containment to isolate (seal water injection), the activity leakage source could be only that in the fuel-clad gap of the fuel rods and off-site and control room doses from the leakage cloud would be the same as for Case 1).

If it is further assumed that the isolation provisions do not completely terminate leakage and that the containment leaks at its design rate, the off-site thyroid dose is as given in Figures 14.3.5-3 and -4 using the left hand ordinate. Off-site whole body doses for this case are presented in Figure 14.3.5-7 and -8.

Isolation of the control room ventilation system in one minute would maintain thyroid dose levels in the control room to the values given in Case 2). With the same assumptions for dispersion to the control room as given in Case 2), the whole body dose from the cloud of gamma emitters surrounding the control room would be 4.4 rem for 30 days.

Figure 14.3.5-9 shows the dose from the activity confined in the containment as a function of distance. It can be seen that the thick concrete containment walls effectively limit dose from this source.

External Recirculation

Indian Point Unit No. 2 has an internal spilled coolant and injection water recirculation system incorporating two pumps for returning water to the reactor core for decay heat removal after a loss of coolant accident. The residual heat removal pumps serve as backup to these pumps. The residual heat removal (RHR) compartment and piping is surrounded by 2 foot thick concrete shield walls. In addition, each RHR compartment is shielded from its adjacent RHR compartment and piping by 2 feet of concrete. Figure 14.3.5-10 shows the results of an evaluation of direct radiation levels surrounding a 14 inch RHR pipe. The evaluation is based on gap activity except noble gases being diluted in the reactor coolant and refueling water volume which is being recirculated through the pipes. With the 24 inches of concrete provided the dose levels would be an order of magnitude less than shown for 12 inches of concrete.

As discussed in Section 6.2, design leakage for the external recirculation system is less than 1000 cc/hr. Westinghouse has performed experiments in which solutions of iodine in sodium hydroxide solutions of pH that would exist in the containment after a loss of coolant were evaporated to dryness. The result was that less than 10^{-3} of the iodine was released. For purpose of conservatism it has been assumed that for a period of one hour 10% of the iodine in the leakage is released to atmosphere. Assuming gap iodine activity immediately after the loss of coolant is present in the sump water being recirculated, the off-site thyroid dose for the period is less than 2 mrem. Protection from inhalation dose in the auxiliary building following an accident can be attained by use of self contained breathing apparatus during those periods when access is required.

Summary and Conclusions

Off-site exposure is summarized in Table 14.3.5-4. It is concluded that even with very pessimistic assumptions that do not take full credit for the safeguards systems provided, both the whole body and thyroid dose after a loss of coolant would be within the 10 CFR 100 suggested guidelines. Further, with the provided leakage prevention systems functioning as designed the off-site dose would be within the 10 CFR 20 allowable yearly dose levels.

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TABLE 14.3.5-1

CORE AND GAP ACTIVITY

Assumptions: Operation at 3216 MWt for 500 days

Isotope	Curies in the Core ($\times 10^7$)	Percent of Core Activity in the Gap	Curies in the Gap ($\times 10^5$)
I-131	7.9	2.3	18
I-132	12.2	0.26	3.13
I-133	18.1	0.79	14.2
I-134	21.8	0.16	3.6
I-135	16.15	0.43	7.1
			1.99
Kr-85	.1163	17	
	18.03		34
Xe-133	4.9	1.9	6.15
Xe-133m	6.85	1.25	44
Xe-135		0.088	

CORE TEMPERATURE DISTRIBUTION

% of Core Fuel Volume Above the Given Temperature	Local Temperature, °F
0.0	4100
0.2	3700
1.8	3300
7.0	2900
15.5	2500

TABLE 14.3.5-2

SITE DISPERSION FACTORS

Distance (meters	(χ/Q) 2 hours (sec/m^3)	(χ/Q 22 hours (sec/m^3)	(χ/Q) 30 days (sec/m^3)
400	9.51×10^{-4}	4.75×10^{-4}	1.03×10^{-4}
700	5.98×10^{-4}	2.99×10^{-4}	3.87×10^{-5}
1,000	4.20×10^{-4}	2.10×10^{-4}	2.07×10^{-5}
2,000	1.90×10^{-4}	9.50×10^{-5}	6.13×10^{-6}
4,000	7.68×10^{-5}	3.84×10^{-5}	1.82×10^{-6}
7,000	3.55×10^{-5}	1.77×10^{-5}	6.79×10^{-7}
10,000	2.14×10^{-5}	1.07×10^{-5}	3.63×10^{-7}
20,000	7.78×10^{-6}	2.89×10^{-6}	1.07×10^{-7}

These are plotted vs distance on Figure 14.3.5-1

TABLE 14.3.5-3

DATA USED IN EVALUATING OFF-SITE INHALATION DOSE

Isotope Dependent Data

Iodine Isotope	Total Core Iodine C_j (curie/MWt)*	DCF_j (rem/curie inhaled)
1-131	2.47×10^4	1.48×10^6
1-132	3.47×10^4	5.29×10^4
1-133	5.63×10^4	3.95×10^5
1-134	6.47×10^4	2.53×10^4
1-135	5.03×10^4	1.23×10^5

* 3216 MWt used in analysis

Time Dependent Data

	Time Period		
	0-2hr	2-24 hr	1 to 30 day
$B(m^3/hr)$	1.25	0.8	0.8
$\frac{X}{Q}$ (site boundary = 520m) $\frac{sec}{m^3}$	7.5×10^{-4}	3.9×10^{-4}	6.5×10^{-5}
$\frac{X}{Q}$ (low polulation zone = 1100m), $\frac{sec}{m^3}$	3.8×10^{-4}	1.9×10^{-4}	1.7×10^{-5}
L (sec^{-1})	1.16×10^{-8}	1.16×10^{-8}	5.2×10^{-9}

TABLE 14.3.5-4

Summary of Offsite Exposure Calculations for Loss of Coolant Accident

	2 hour Exposure at 520 Meters (Min. exclusion Radius)	Total Exposure at 1100 Meters (Min. Low Population Zone Radius)
I. Thyroid Dose (Based on Zero to 5% of Airborn as CH_3I)		
Containment leakage terminated in one minute by Isolation Valve Seal Water System	0.7 rem	0.36
Gap release* - Continuous leakage with 2 Spray Pumps and 5 fan Filters operating	0.8 - 1.45 rem	0.42-0.68 rem
Gap release* - Continuous leakage with one Spray pump and 3 fan filters operating	1.7 - 2.7 rem	0.85 - 1.4 rem
Gap release* - continuous leakage with 5 fan filters operating	8.8 rem	4.8 rem
Gap release* - continuous leakage with 2 spray pumps	0.95 - 6.7 rem	0.55 - 13.9 rem
10CFR 100 Suggested Limit	300 rem	300 rem
II. Whole Body Dose		
Containment leakage terminated in one minute by Isolation Valve Seal Water System	<1mr	<1mr
Gap release - continuous leakage	18mr	68mr
TID - 14844 release - continuous leakage	3.8	4.9
10CFR 100 Suggested Limit	25 rem	25 rem

* TID - 14844 initial iodine leakage inventory of 25% of core equilibrium quantity will result in thyroid dose 10 times value shown.

SITE DISPERSION FACTORS VS DISTANCE

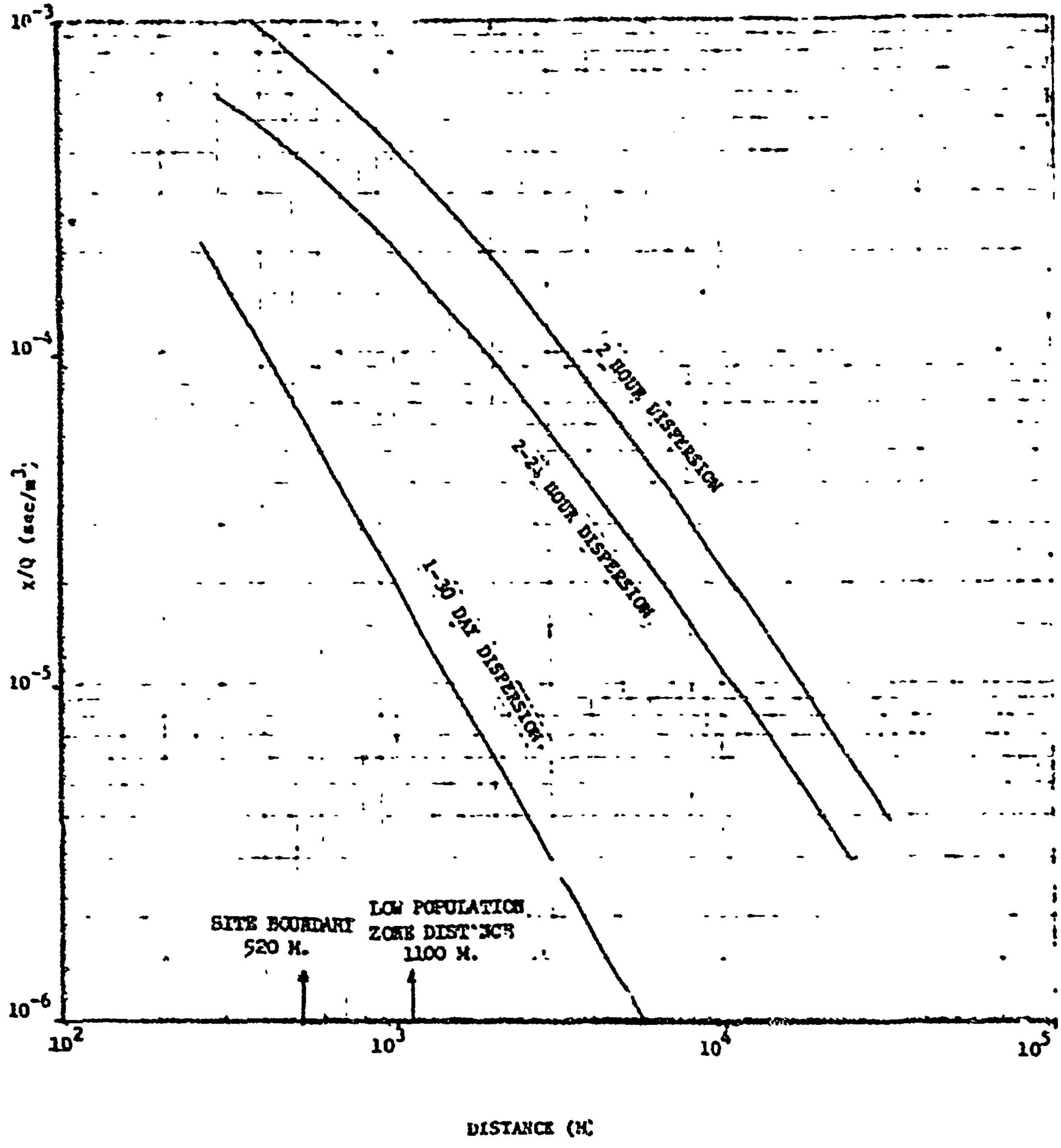


FIGURE 14.3.5-1

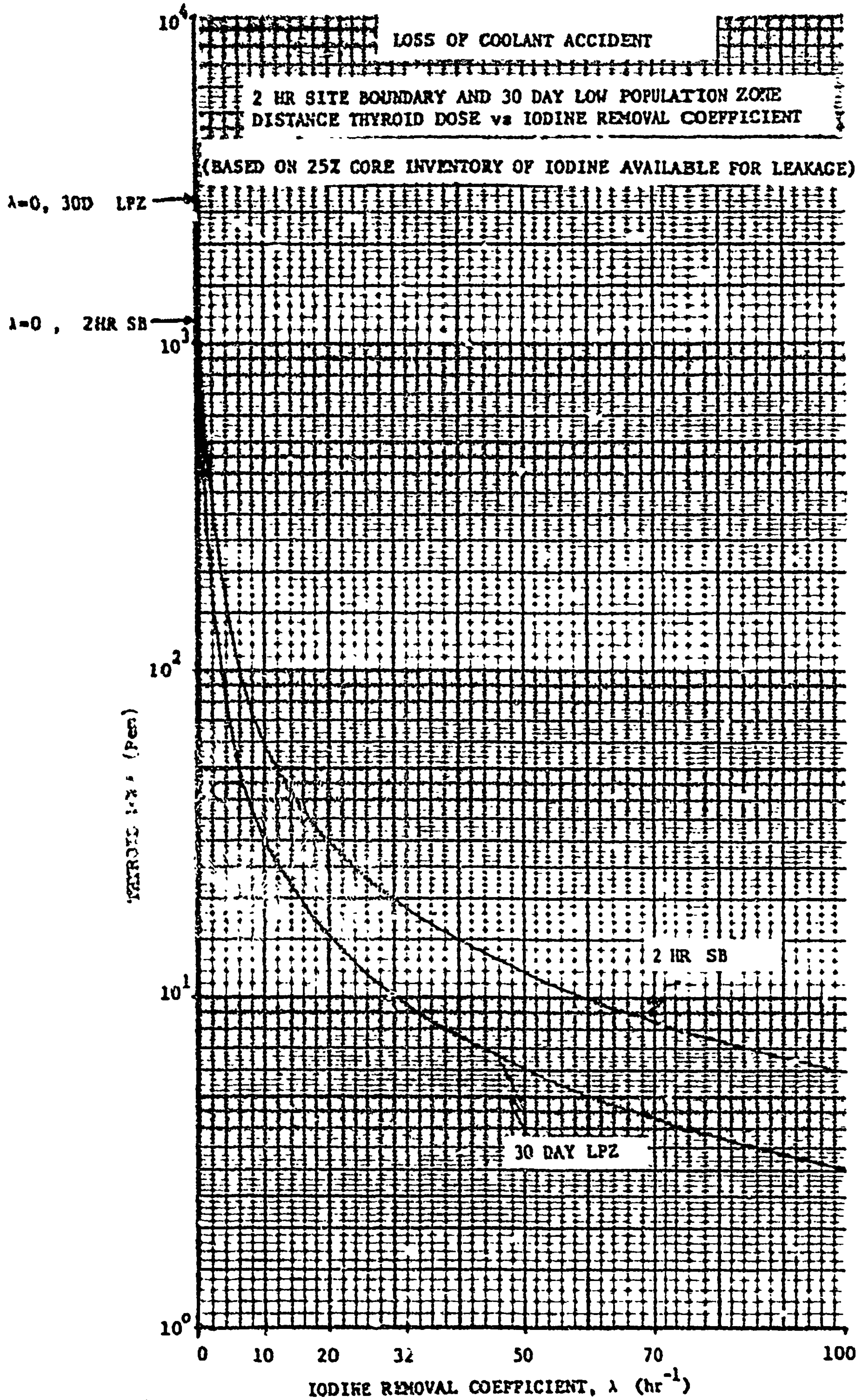
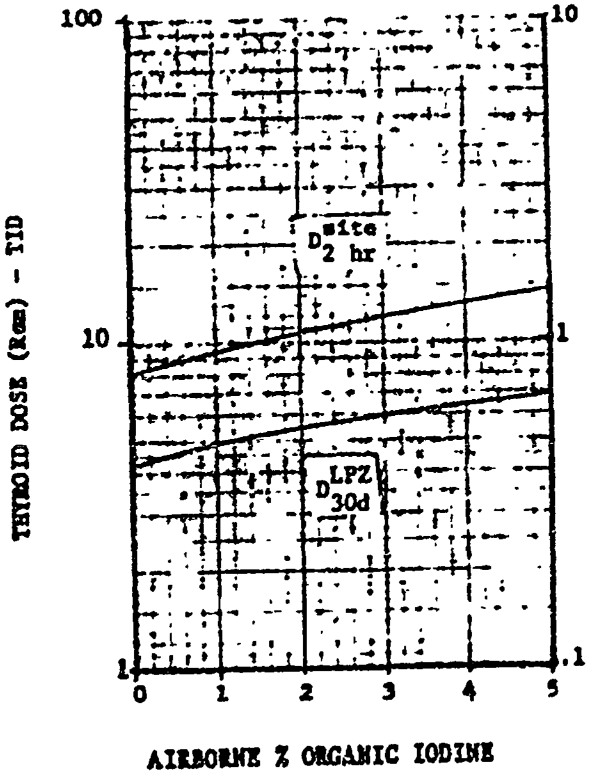


FIGURE 14.3.5-2

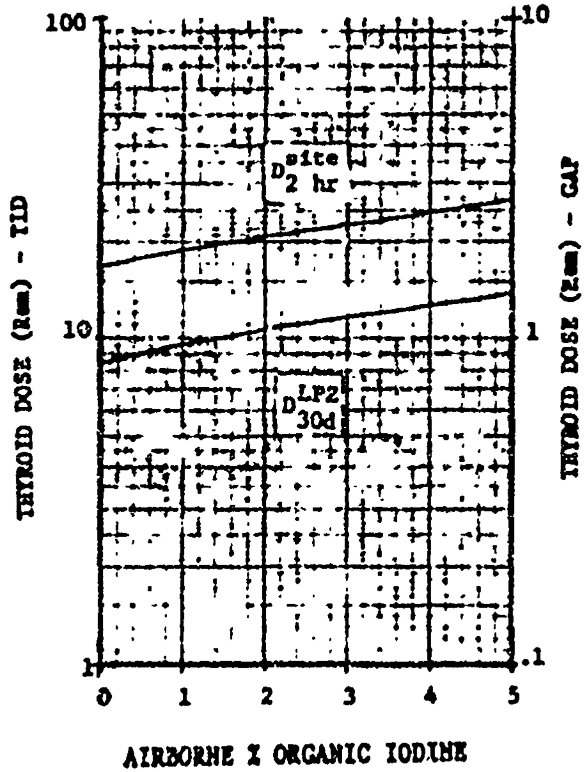
LOSS-OF-ACCIDENT - THYROID DOSE



Case I 5 fans + 2 spray

$\lambda_I = 70.5\ hr^{-1}$

$\lambda_o = 5.25\ hr^{-1}$



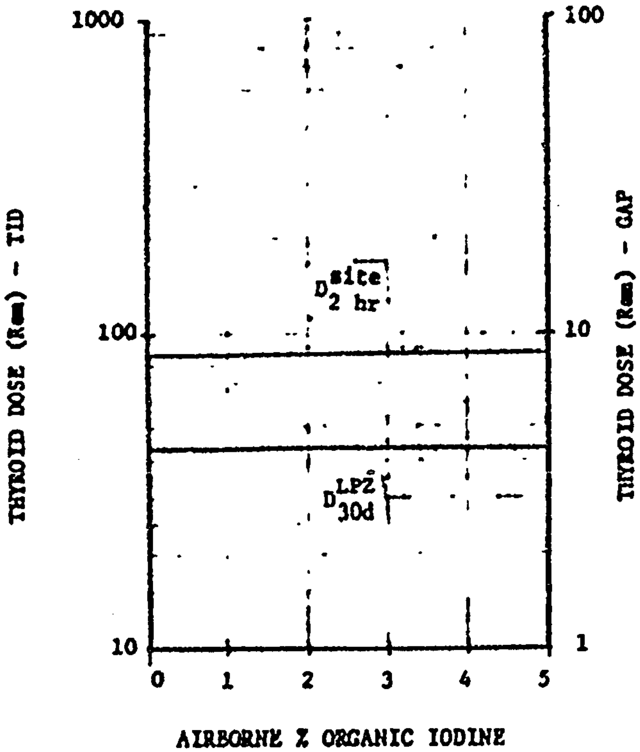
Case II 3 fans + 1 spray

$\lambda_I = 36\ hr^{-1}$

$\lambda_o = 3.1\ hr^{-1}$

FIGURE 14.3.5-3

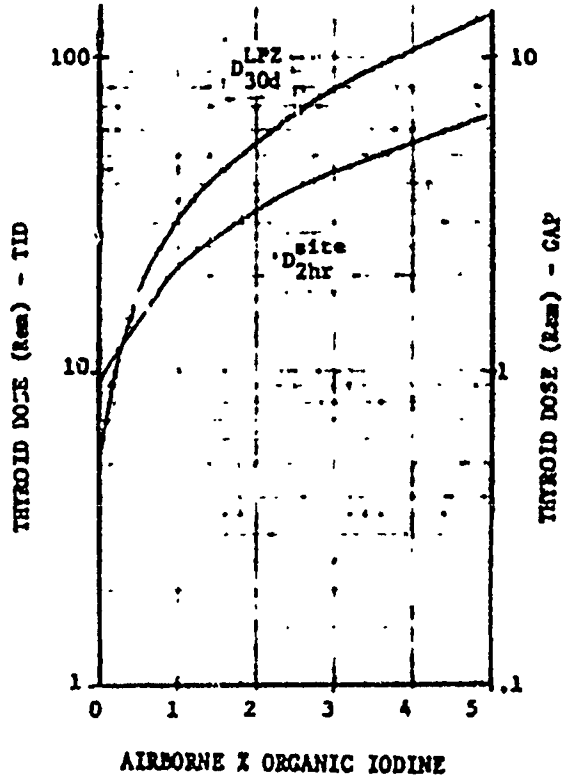
LOSS-OF-COOLANT ACCIDENT - THYROID DOSE



Case III 5 fans + no spray

λ_I = 6.75 hr^{-1}

λ_o = 5.25 hr^{-1}



Case IV 2 spray + no fans

λ_I = 64 hr^{-1}

λ_o = C hr^{-1}

FIGURE 14.3.5-4

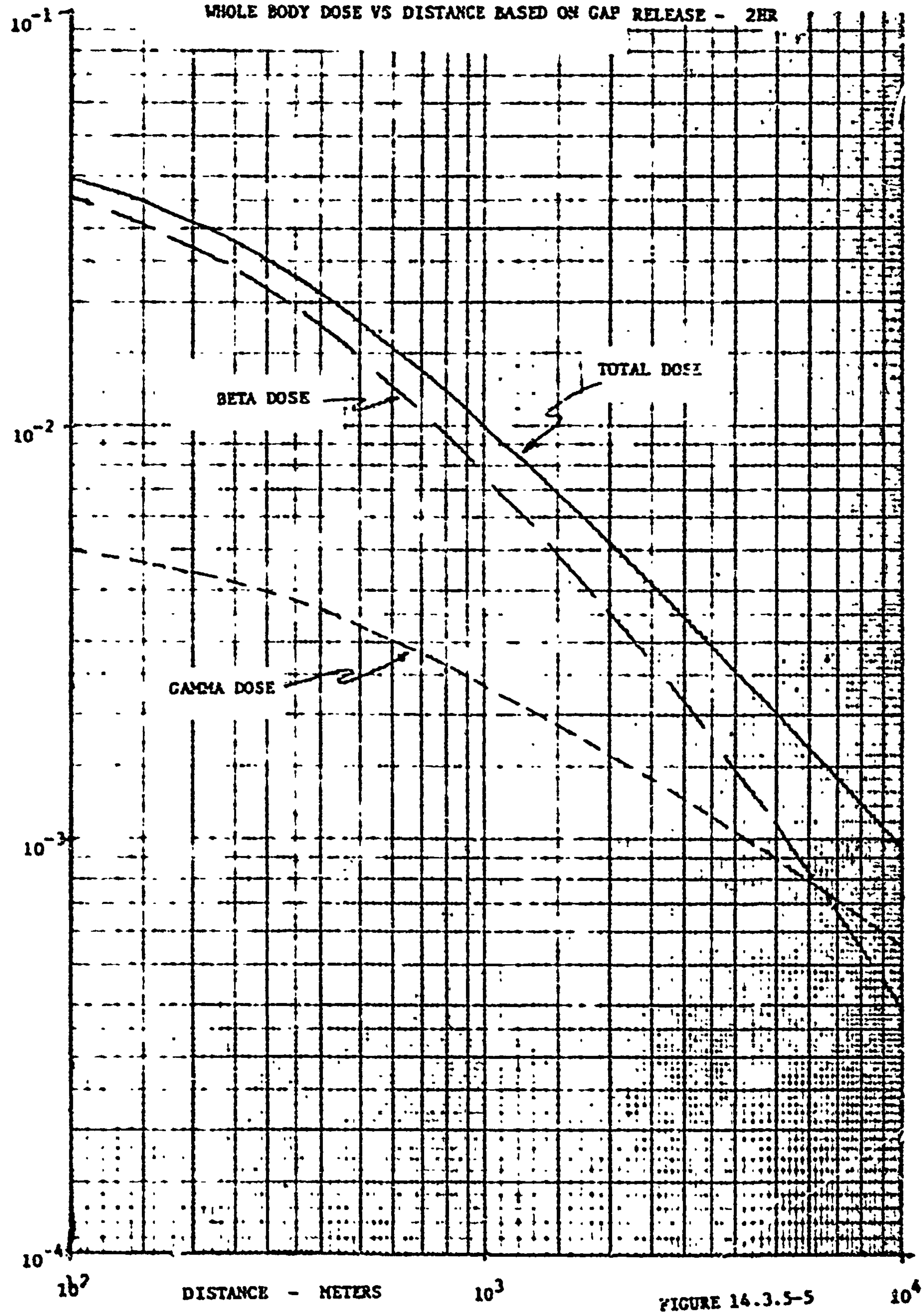


FIGURE 14.3.5-5

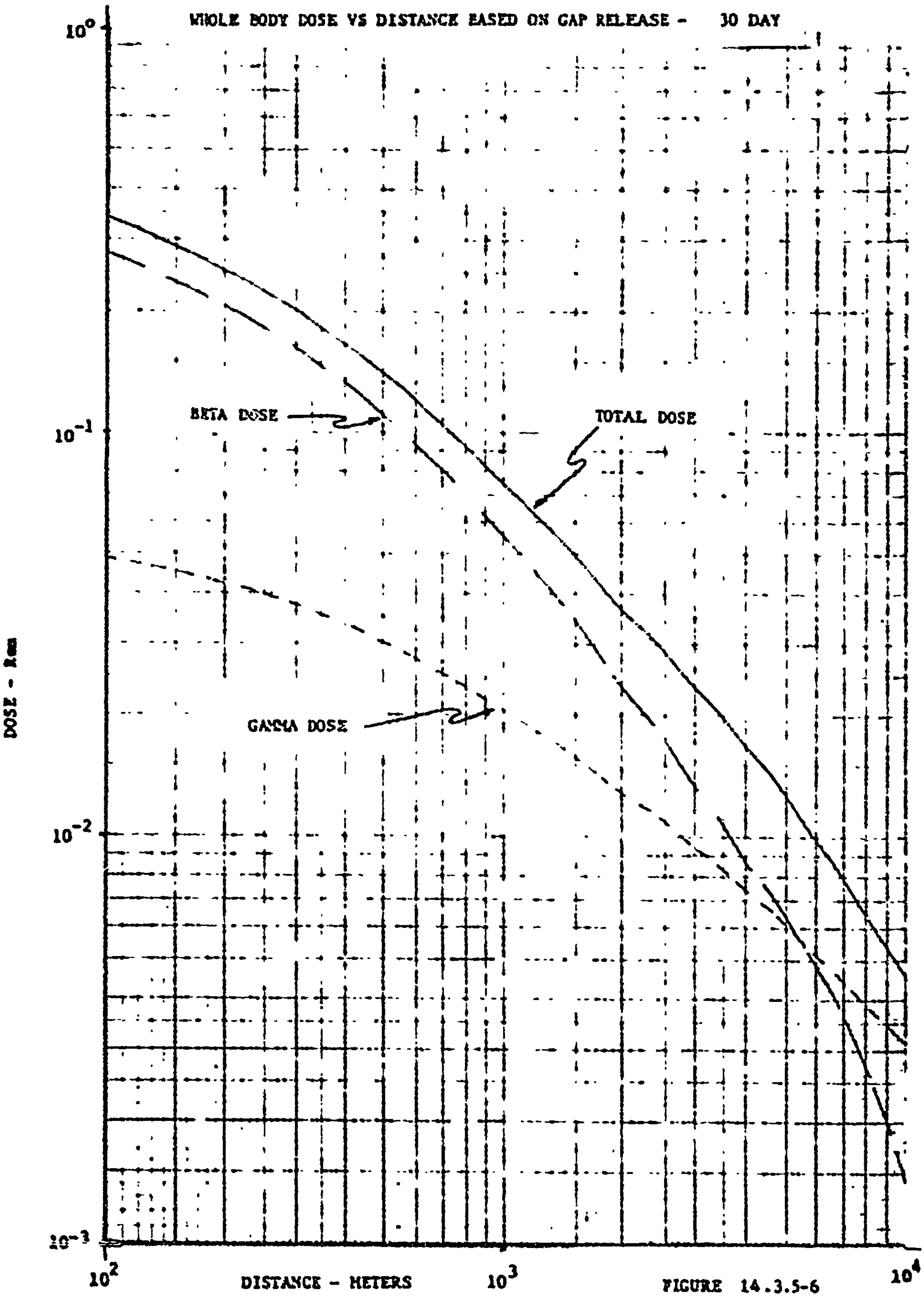


FIGURE 14.3.5-6

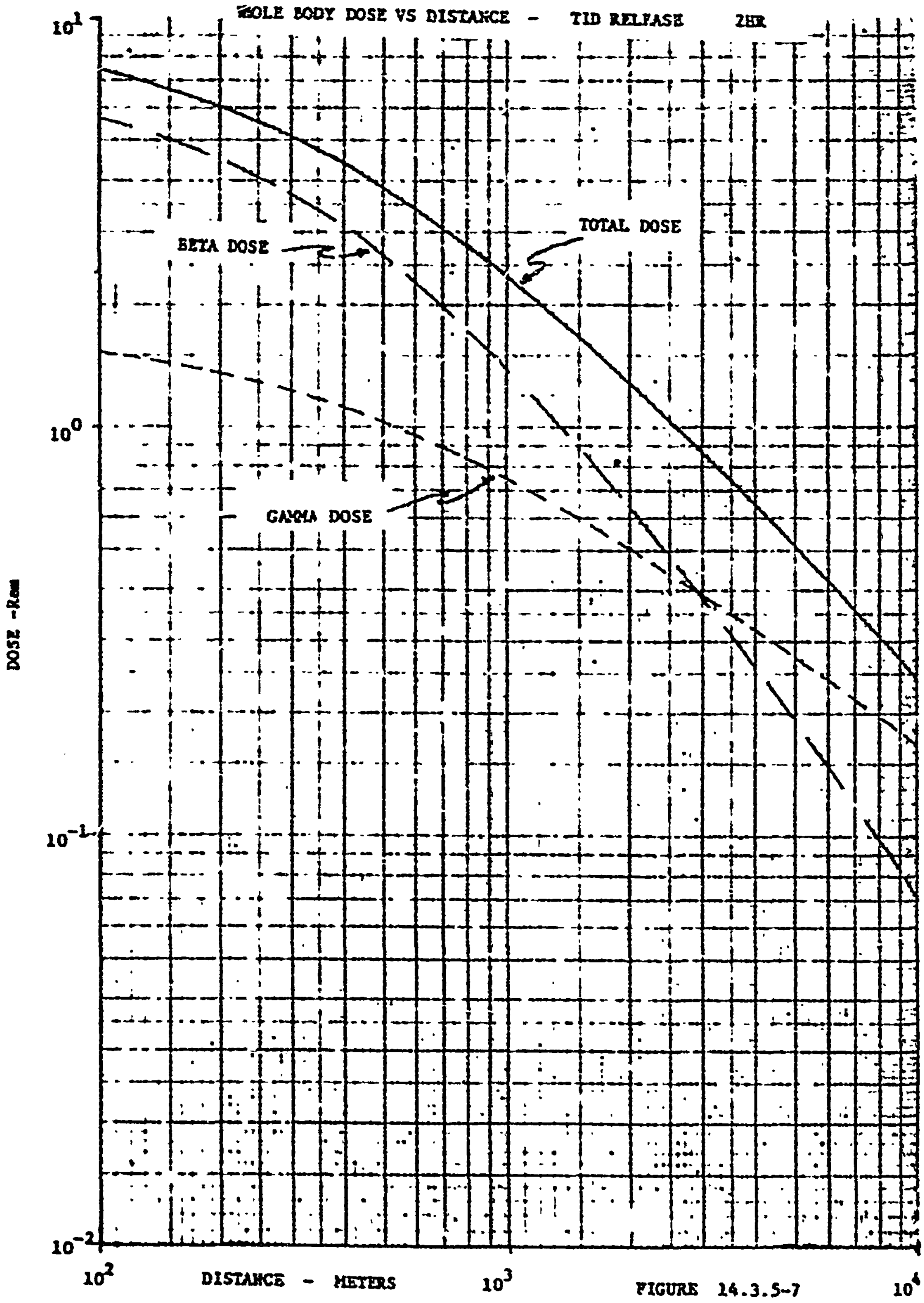


FIGURE 14.3.5-7

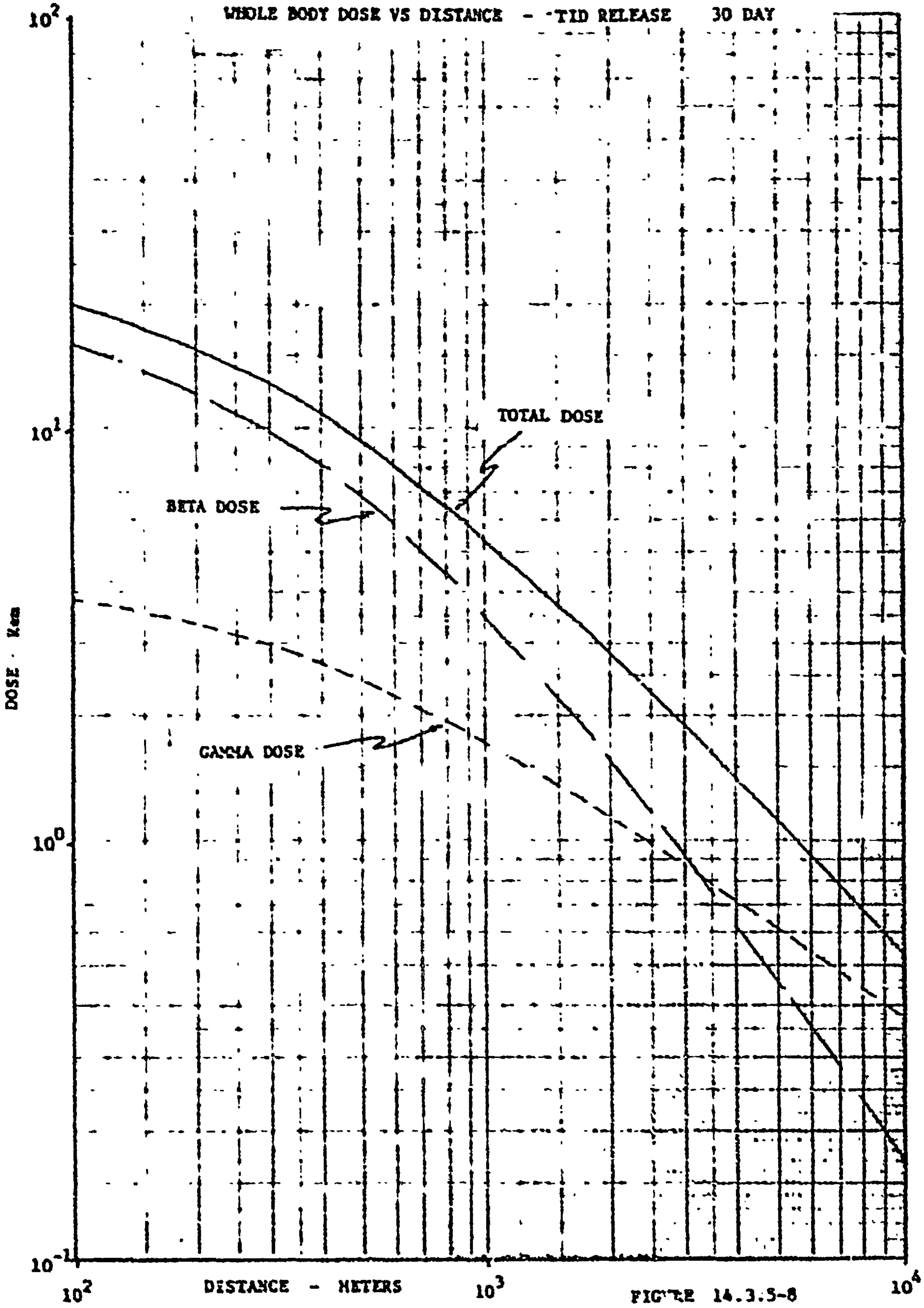


FIGURE 14.3.5-8

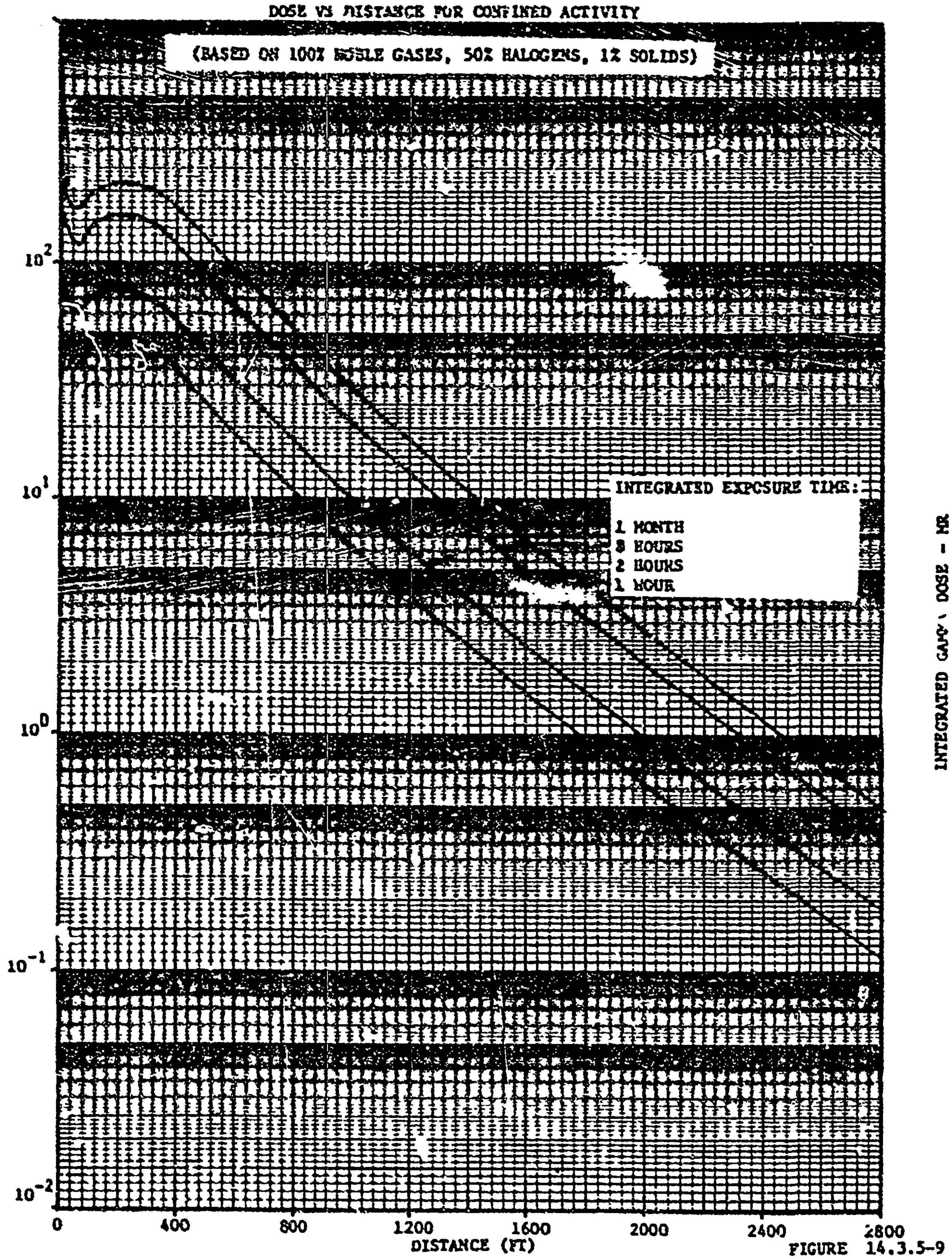


FIGURE 14.3.5-9

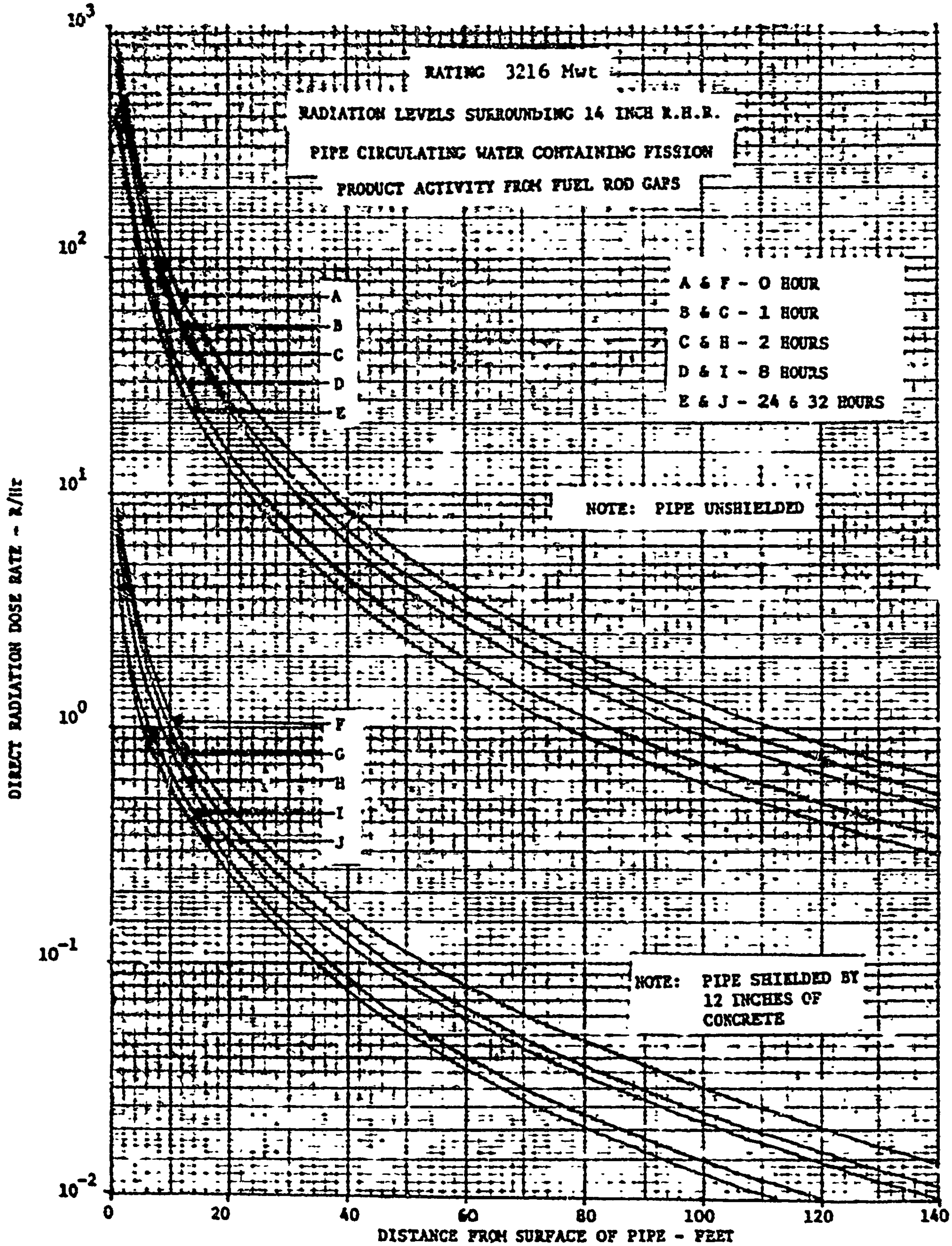


FIGURE 14.3.5-10