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Exhibit B-1

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

FIRST SUPPLEMENT TO:
PRELIMINARY SAFETY ANALYSIS REPORT

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PREFACE

Supplement 6 to the Indian Point Nuclear Generating Plant No. 3 PSAR consists of additional information submitted to supersede or augment information presented in Supplements 1 and 5. Topics of information presented include electrical and instrumentation systems, nuclear fuel data as a function of core life and the reactor coolant system Quality assurance program. The areas where additional information is added to existing material is marked either in the margin or in a sentence within the text. Those pages that are changed should be removed from the text and replaced with the pages in this supplement.

The replacement pages for Supplement 1 are printed on green paper and the replacement pages for Supplement 5 are printed on blue paper.

The material in Supplement 1 that is revised will be found under tabs 2, 10 and 17. The material in Supplement 5 that is revised will be found under tab 15.

PREFACE TO QUESTIONS

Supplement 1 of the Indian Point Unit No. 3 PSAR consists of responses to the Atomic Energy Commission's letter of February 19, 1968, and July 1, 1968.

To facilitate review of the responses to these letters, a list of each item in the February 19 letter has been given a number and is identified in this preface. The July 1, 1968, questions are answered in Item 18 of this Supplement. The "items" are identified by tab numbers 1 through 18.

Attachments to the Commission's letter of February 19, 1968, are discussed as part of the "item" number in which they are referenced.

Item No.	Subject
1	70 General Design Criteria
2	Indian Point Unit No. 2 questions from letters of 2/28/66 and 5/11/66 for Indian Point Unit No. 3
2a	Containment Structural Design including Question 9 from Item 2, letter of 5/11/66 - to be answered in a later Supplement
3	ACRS Letters of 8/16/66 and 12/20/67
4	Research and Development Programs
5	Quality Control
6	Tornado Criterion
7	Gas Pipeline Fire
8	Flooding Debris in Intake for Coolant Water
9	Reactor Design
10	Further information on Reactor Design (Attachment A)
11	Reactor Vessel and Primary System
12	Thermal Shock
13	Engineered Safety Features - Organic Iodine Removal
14	Engineered Safety Features - Fan Cooler and Spray System Design

Item No.	Subject
15	Seismic Design
16	Safety Evaluation
17	Electric Power Distribution, Auxiliary Systems, Steam Systems, and Waste Disposal Systems
18	July 1, 1968 Questions
	a) Indian Point Unit No. 3 - Diablo Canyon Design differences
	b) ACRS letter of 12/20/67 report on Diablo Canyon
	c) Flooding Potential of Indian Point Site
	d) Flooding Protection requirements

ITEM 1

Our review has identified a number of areas in which your application is not complete. For example, amendments to your Preliminary Safety Analysis Report (PSAR) should include a discussion of the design of Indian Point Nuclear Generating Unit No. 3 with respect to the 70 General Design Criteria specified in 10 CFR 50.

ANSWER

I. OVERALL PLANT REQUIREMENTS

CRITERION 1 - QUALITY STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

ANSWER

All systems and components of the facility are classified according to their importance (See Appendix A, Table 1). Those items vital to safe shutdown and isolation of the reactor or whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of excessive amounts of radioactivity are designated Class I. Those items important to power generation but not essential to safe shutdown and isolation of the reactor or control of the release of substantial amounts of radioactivity are designated Class II. Those items not related to power generation or safety are designated Class III.

Class I systems and components are essential to the protection of the health and safety of the public. Consequently, they are designed, fabricated, inspected and erected and the materials selected to the applicable provisions of recognized codes, good nuclear practice and to quality standards that reflect their importance. Discussions of applicable codes and standards, quality assurance programs, test provisions, etc., are given in the sections describing each system.

Reference sections are as follows:

Section Title	Section
Principal Architectural and Engineering Criteria for Design	2.4
Reactor Design	3.2
Reactor Coolant System; Design Bases	4.1, 4.2
Containment System; Design Bases	5.1
Engineered Safety Features; Design Objectives	6.1
Instrumentation and Control; Design Basis	7.1
Electrical System; Design Bases	8.1
Auxiliary and Emergency Systems	9
Waste Disposal and Radiation Protection	11
Seismic Design Criteria	Appendix A
Quality Control	Appendix B
Supplement 1	Item 5
Supplement 1	Item 11
Supplement 1	Item 15

CRITERION 2 - PERFORMANCE STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

ANSWER

All systems and components designated Class I are designed so that there is no loss of function in the event of the maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously. The stresses for both Class I and Class II items are kept within code allowable values for the design earthquake. Similarly, measures are taken in the plant design to protect against possible effects of tornados, sudden barometric pressure changes, seiches, and other natural phenomena. The reactor containment is designed as a Class I structure. Its structural members have sufficient capacity to accept a combination of normal operation loads, functional loads due to a loss of coolant accident, and the loadings imposed by the maximum wind velocity and tornado driven missiles or those due to design earthquake, whichever is the larger.

The diesel-generator emergency power sources are designed so that their integrity is not affected by the maximum potential earthquake, wind storms, floods or disturbances on the external electrical power system grid. Power, control and instrument cabling, motors and other electrical equipment required for operation of the engineered safety features is suitably protected against the effects of either a nuclear system accident or of severe external weather conditions, to obtain a high degree of confidence in the operability of these components in the event that they should be required.

The entire plant is designed so that the required critical components and systems will retain their integrity under design natural phenomena. The implementation of this criterion varies depending on whether a phenomenon affects the entire plant (for example, an earthquake) or whether the phenomenon affects only a single component (for example, a tornado-driven missile).

Reference sections are as follows:

Site (Design Natural Phenomena)	Section 1
Reactor Design	Section 3.2
Reactor Coolant System; Design Bases	Section 4.1
Containment System; Design Bases	Section 5.1
Engineered Safety Features	Section 6
Instrumentation & Control Systems	Section 7
Emergency Power Sources	Section 8.4
Auxiliary & Emergency Systems	Section 9
Waste Disposal & Radiation Protection	Section 11
Seismic Design Criteria	Appendix A
Supplement 1 (Tornadoes)	Item 6
Supplement 1 (Reactor Vessel and Primary Systems)	Item 11
Supplement 1 (Seismic Design)	Item 15
Supplement 1 (Floods)	Item 18

CRITERION 3 - FIRE PROTECTION

The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

ANSWER

Primary emphasis is directed at minimizing the risk of fire by use of thermal insulation and adhesives which do not support combustion, flame resistant wiring, adequate overload and short circuit protection, and elimination of combustible trim and furnishings. Fire prevention in all areas of the nuclear 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. The facility is also equipped with a fire protection system for controlling any fires, with particular emphasis for the protection of Class I equipment.

Although some isolated damage to the electrical equipment could result from a fire in the electrical equipment in the control room, because of the multiplicity of the reactor and engineered safety features system trip actuation circuits, and the associated segregation of redundant channels including wiring, the ability of the systems to protect the reactor will remain unimpaired, and the operator will be able to execute a reactor shutdown primarily from the main control room. Due to the fire resistant nature of the structural and finish materials for the control room and control room furnishings, it is considered that the probability of a fire external to the electrical equipment is minimal. In the unlikely event of such a fire, however, the electrical equipment is protected by metal cabinets. Further, the operator has available portable fire extinguishers sized and located in accordance with National Fire Code and National Fire Protection Association specifications.

Normal fire protection is provided by fixed fire-fog system, sprinklers, hose lines, and portable and wheeled extinguishers.

Water for the fire protection system is provided by two centrifugal fire pumps. Sufficient hydrants are strategically located within the plant perimeter.

A fire header of sufficient size to deliver an adequate quantity of water throughout the plant at a pressure no less than 75 psig at the highest nozzles is installed. The header system is normally pressurized through the use of a hydro-pneumatic tank using house service air and having an active water capacity of 10,000 gallons.

Loss of header pressure will activate the fire pumps and the alarm system. The Containment and Auxiliary Building Ventilation Systems have indicative lights in the control room. Each ventilation system is provided with smoke or flame (ionization type) detectors as well as fire alarms which alert the control room to the possibility of fire so that early action may be taken to prevent significant damage.

Reference sections are as follows:

Section Title	Section
Containment System	5
Operating Systems	6
Auxiliary and Emergency Systems	9
Supplement #1	Item 7

CRITERION 4 - SHARING OF SYSTEMS (Category A)

Reactor facilities shall not share systems or components unless it it shown safety is not imparied by the sharing.

ANSWER

There is no sharing of safety systems and components between units; however, there are certain electrical ties between units which are provided as a backup to the multiple power sources already available.

Reference sections are as follows:

Section Title	Section
Introduction	2.1
Engineered Safety Features	6
Electrical System	8
Auxiliary and Emergency Systems	9
Waste Disposal and Radiation Protection	11

CRITERION 5 - RECORDS REQUIREMENTS

Records of the design, fabrication and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

ANSWER

Records of the design, fabrication, construction and testing of major Class I components of the plant are maintained throughout the plant life by the licensee.

Reference sections are as follows:

Section Title	Section
Engineered Safety Features	6
Supplement 1	Item 5

CRITERION 6 - REACTOR CORE DESIGN (Category A)

The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all off-site power.

ANSWER

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

The reactor control and protection instrumentation is designed to actuate a reactor trip for any anticipated combination of plant conditions to ensure a minimum DNB ratio equal to or greater than 1.30 and fuel center temperatures below the melting point of UO_2.

Reference sections are as follows:

Section Title	Section
Reactor; Design Bases and Reactor Design	3.1, 3.2
Instrumentation and Control; Protection System	7.1
Safety Analyses	12

CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (Category B)

The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

ANSWER

The design of the reactor core and related protection systems ensures that power oscillations which could cause fuel damage in excess of acceptable limits are not possible or can be readily suppressed.

The potential for possible spatial oscillations of power distribution for this core has been reviewed. In summary, it is concluded that the only potential spatial instability is the xenon-induced axial instability which may be a nearly free running oscillation with little or no inherent damping. Part length control rods are provided to suppress these oscillations if they occur. Out-of-core instrumentation is provided to obtain necessary information concerning axial and azimuthal distributions. This instrumentation is adequate to enable the operator to monitor and control xenon induced oscillations. In-core instrumentation is used to periodically calibrate and verify the information provided by the out-of-core instrumentation.

Reference sections are as follows:

Section Title	Section
Reactor; Nuclear Design and Evaluation	3.2.1
Instrumentation and Control; Protection System	7.1
Supplement 1	Item 4
Supplement 1	Item 9

CRITERION 8 - OVERALL POWER COEFFICIENT (Category B)

The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

ANSWER

The overall power coefficient in the power operating range is maintained negative.

Reference sections are as follows:

Section Title	Section
Reactor Design	3.2
Supplement 1	Item 4
Supplement 1	Item 9

CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (Category A)

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

ANSWER

The Reactor Coolant System in conjunction with its control and protection provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions and maintain the stresses within applicable code limits. The design specifications are drawn up based on analytical studies on both the plant steady state and transient characteristics. The methods used and subsequent specifications have been adequately proved and supported by previous experience.

Fabrication of the components which constitute the pressure retaining boundary of the Reactor Coolant System is carried out in strict accordance with the applicable codes. In addition, there are areas where equipment specifications for Reactor Coolant System components go beyond the applicable codes. Before operation all systems and components are pressure and leak tested as dictated by their design specification and the code requirements. Finally the overall plant is operated for some period at normal operating pressure and temperature before fuel loading.

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

System conditions resulting from anticipated transients or malfunctions are monitored and appropriate action automatically initiated to maintain the required cooling capability and to limit the system condition so that continued safe operation is possible. The system is protected from over pressure by means of pressure receiving devices as required by Section III of the ASME Boiler and Pressure Vessel Code.

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

Reference sections are as follows:

Section Title	Section
Reactor Coolant System; Design Bases	4.1
Supplement 1	Item 2

CRITERION 10 - CONTAINMENT

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

ANSWER

The reactor containment is a reinforced concrete structure with a steel liner which is capable of withstanding the pressure build-up due to a major loss-of-coolant accident. There is substantial margin between the highest pressure peak following any major loss-of-coolant accident and the design pressure of the containment structure which provides assurance that containment will maintain its integrity and capability to protect the public.

The containment system together with the containment spray system and the air recirculation cooling system, provide the necessary containment function for so long as may be necessary following a loss-of-coolant accident. The design is such that the containment design pressure is not exceeded during any pressure transient following a loss-of-coolant accident considering the combined effects of heat sources such as core residual heat and metal water reactions and the minimal operation of the emergency core cooling systems and the containment cooling systems.

The reference sections are as follows:

Section Title	Section
Containment System Structures	5.1
Engineered Safety Features; General Description	6.2
Safety Evaluations	12

CRITERION 11 - CONTROL ROOM

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 access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

ANSWER

The plant is equipped with a control room which contains all controls and instrumentation necessary for operation of the reactor and turbine generator under normal or accident conditons. The control room will be designed and equipped to assure continued occupancy under all operating and accident conditions.

Sufficient shielding, distance, and containment integrity are provided to assure that control room personnel during occupancy of the control room are not subject to doses under postulated accident conditions with engineered safeguards operating at design capacity, which in the agregate would exceed limits specified in 10 CFR 20. With minimum operation of the engineered safeguards, the corresponding doses would not in the agregate exceed ten percent of the suggested limits in 10 CFR 100.

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

Provisions will be made to permit bringing the plant to, and maintaining it in, a hot shutdown condition for an extended period of time from locations outside the control room. During the extended period of hot shutdown there is no reason to believe that access to the control room is not regained. In addition, adequate measures can be taken to attain the cold shutdown condition, over a reasonable period of time, from locations outside the control room, if required.

The necessity of abandoning the control room due to fire is not considered credible. The employment of non-combustible and fire retardant materials in the construction of the control room contained equipment and furnishing, the limitation of combustible supplies to the minimum consistent with safe and efficient operation of the plant, the location of fire fighting equipment in the control room, and the continuous presence of an operator trained in fire fighting techniques minimize the probability of a fire within the control room. In addition the control room ventilation system is designed to keep the control room at a positive pressure and can be operated in a recirculating mode to prevent fire originating outside the control room from spreading to the control area.

In the unlikely event that the control room must be abandoned, there is no reason to assume that any significant damage is sustained in the control room.

The reference sections are as follows:

Section Title	Section
Instrumentation and Control	7.6
Radiation Protection	11.2
Safety Analyses	12
Supplement 1	Item 16

CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (Category B)

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

ANSWER

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

The non-nuclear process instrumentation measures temperatures, pressure, flow, and levels in the Reactor Coolant System, Steam System, Containment and other 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 which is a controlled access area. The quantity and types of process instrumentation provided are adequate for safe and orderly operation of all systems and processes over the full operating range of the plant.

Reference section as follows:

Section Title	Section
Instrumentation and Control	7

CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS (Category B)

Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

ANSWER

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 of 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, the relative reactivity status is continuosly 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 activated from either channel if the neutron source becomes excessive.

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 evaluation of boric acid concentration provides a long term means of following reactivity status.

Reference sections are as follows:

Section Title	Section
Instrumentation and Control System	
Nuclear Instrumentation System	7.3
In-Core Instrumentation	7.5

CRITERION 14 - CORE PROTECTION SYSTEMS (Category B)

Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

ANSWER

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 closure of containment isolation valves and to initiate the operation of the appropriate engineered safety features.

During reactor operation in the startup or power modes redundant protection signals automatically actuate two reactor trip breakers which are connected in series with the rod drive mechanism coils. This action would interrupt power and initiate reactor trip. This criterion as applied to the Reactor Protection System is discussed more fully under criterion 26, Protection Systems Fail-Safe Design.

Reference sections are as follows:

Section Title	Section
Instrumentation and Control; Protection System	7.1
Supplement 1	Item 16

CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS (Category B)

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

ANSWER

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.

The engineered safety features systems are actuated by the engineered safety features actuation channels. Each coincidence network energizes an engineered safety 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 does not defeat the channel function. The Engineered Safety Features Instrumentation System actuates (depending on the severity of the condition) the Safety Injection System, the Containment Isolation System, and 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 obtained from redundant low pressurizer pressure in coincidence with low pressurizer water level.

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 thorugh 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.

Reference section as follows:

Section Title	Section
Engineered Safety Features; General Description	6.2

CRITERION 16 - MONITORING REACTOR COOLANT PRESSURE BOUNDARY (Category B)

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

ANSWER

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

Reference sections are as follows:

Section Title	Section
In-Service Reactor Coolant System Leakage Detection	4.6
Containment	5
Engineered Safety Features; Safety Injection System	6.2.1

CRITERION 17 - MONITORING RADIOACTIVITY RELEASE

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

ANSWER

A radiation and monitoring system is provided to detect, compute, indicate, annunciate, and record the radiation at selected locations inside and outside the reactor plant.

The containment atmosphere, the plant vent, service water discharge and the Waste Disposal System liquid effluent are monitored for radioactivity concentration during all operations from anticipated transients and from accidental conditions.

All gaseous effluent (1) from possible sources of accidental releases of radioactivity external to the reactor containment (e.g., the spent fuel pit and waste handling equipment) are exhausted from the plant vent which is monitored.

All accidental spills of liquids are maintained within the auxiliary building and collected in a tank. Any contaminated liquid effluent discharged to the condenser circulating water 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 control room and the environmental radiation monitoring system provides adequate monitoring of accident releases.

Reference sections are as follows:

Section Title	Section
Radioactive Waste and Radiation Protection	11
Safety Analysis	12

(1) (Except for the steam generator air ejector exhaust which has its own radioactivity monitor)

CRITERION 18 - MONITORING FUEL AND WASTE STORAGE

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

ANSWER

Monitoring and alarm instrumentation is provided for fuel and waste storage and handling areas 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 is flow monitored to detect inadequate cooling and to assure proper operation.

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.

Reference sections are as follows:

Section Title	Section
Auxiliary Coolant System	9.2
Radiation Protection	11.2

CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (Category B)

Protection systems shall be designed for both high functional reliability and in-service testability necessary to avoid undue risk to the health and safety of the public.

The reactor uses the Westinghouse magnetic-type control rod drive mechanisms which are similar to those used in the San Onofre, Indian Point No. 2 and Connecticut Yankee plants. Upon a loss of power to the coils, the rod cluster control assembly is released and falls by gravity into the core.

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

Reference section as follows:

Section Title	Section
Instrumentation and Control; Protection System	7.1

CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE

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 for each protection function to be served.

Two reactor trip breakers are provided to interrupt power to the rod drive mechanisms. The breaker main contacts are connected in series with the mechanism coils. Opening either breaker interrupts power to all mechanisms causing them to release all rods to fall by gravity into the core. Each breaker is opened through an undervoltage trip coil. Each protection channel actuates one reactor trip breaker undervoltage trip coil. The protection system is thus inherently safe in the event of a loss of rod control power.

The initiation of the engineered safety features provided for loss-of-coolant accidents (e.g. emergency core cooling system pumps, and containment cooling systems) is accomplished from redundant signals derived from Reactor Coolant System and containment instrumentation. Channel independence is carried throughout the system from the sensors to the signal output relays including the power supplies for the channels. The initiation signal for containment spray comes from coincident high containment pressure signals.

The reference section is as follows:

Section Title	Section
Instrumentation and Control; Protection System	7.1

CRITERION 21 - SINGLE FAILURE DEFINITION (Category B)

Multiple failures resulting from a single event shall be treated as a single failure.

ANSWER

This definition of a single failure has been incorporated into the design of the protection systems and the engineered safety features so that a single event cannot result in multiple failure resulting in loss of protective action.

Reference section as follows:

Section Title	Section
Safety Analysis	12

CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS (Category B)

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

ANSWER

The reactor protection systems are designed using the proposed IEEE "Standards for Nuclear Power Plant Protection Systems" as a guide.

The coincident trip philosophy is used to provide protection for single failure causing a spurious trip or from defeating the function of the channel.

Each reactor trip circuit is designed so that trip occurs when the circuit is deenergized; an open circuit or loss of channel power therefore causes that channel to go into its trip mode. Reliability and independence is obtained by redundancy within each channel. Channel independence is carried throughout the system extending from the sensor to the relay providing the logic. In most cases, the safety and control functions are combined only at the sensor. Both functions are fully isolated in the remaining part of the channel; control being derived from the primary safety signal path through an isolation amplifier. As such, a failure in the control circuitry cannot adversely affect the safety channel.

Reference sections are as follows:

Section Title	Title
Engineered Safety Features; General Description	6.2
Instrumentation and Control; Protection System	7.1
Supplement 1	Item 18

CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS (Category B)

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.

ANSWER

The components of the protection systems are designed and arranged so that the mechanical and environmental conditions accompanying any emergency situation in which the components are required to function does not interfere with that function.

Reference sections are as follows:

Section Title	Section
Engineered Safety Features	6
Instrumentation and Control; Protection System	7.1

CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS

In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

ANSWER

The plant control and instrumentation as required for the plant protection system is supplied with power from onsite redundant 125 volt battery sources, thus is not dependent on offsite power. In the event of loss of offsite power, the power source which is required for the appropriate operation of the protection system is supplied from onsite redundant emergency diesels. Power requirements for the protection system are less than that required for the operation of the engineered safety features. (Criterion 39)

There is a gas turbine available at the site as an alternate source of emergency power.

Reference sections are as follows:

Section Title	Section
Engineered Safety Features; General Description	6.2
Supplement 1	Item 17

CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS (Category B)

Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

ANSWER

Each protection channel in service at power is capable of being calibrated and tripped independently by simulated signals for test purposes to verify its operation and that no failure or loss of redundancy has occurred. This includes a checking through to the trip breakers which forms the trip logic. Thus, the operablility of each trip channel can be determined conveniently and without ambiguity.

The emergency diesel generators can be periodically started synchronized to the system and loaded to demonstrate their capability.

The ability of the units to start automatically within the prescribed time and to carry intended loads can be checked during refueling shutdown.

Reference sections as follows:

Section Title	Section
Instrumentation and Control; Protection System	7.1

CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN (Category B)

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., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

ANSWER

Each reactor trip circuit is designed so that trip occurs when the circuit is de-energized; an open circuit or loss of channel power therefore 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 energized by separate electrical busses. 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 each drive, allowing the rod clusters to insert by gravity. The protection system is thus inherently safe in the event of a loss of vital power.

Automatic starting of emergency diesel-generators is initiated by undervoltage relays on the normal power busses. Engine cranking is accomplished by a stored energy system supplied soley for the associated diesel-generator.

Essential equipment is selected to withstand the most adverse environmental conditions to which it might be subjected including post accident conditions within the containment.

Reference section as follows:

Section Title	Section
Instrumentation and Control; Protection System	7.1

CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (Category A)

At least two independent reactivity control systems, preferably of different principles, shall be provided.

ANSWER

Two independent reactivity control systems of different principles are provided. One of the two reactivity control systems employs rod cluster control assemblies to regulate the position of neutron absorber within the reactor core.

The other reactivity control systems employ the Chemical and Volume Control System to regulate the concentration of boric acid solution neutron absorber in the Reactor Coolant System.

Reference sections are as follows:

Section Title	Section
Reactor Design Bases	3.1
Chemical and Volume Control System	9.1

CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (Category A)

At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power change, sufficiently fast to prevent exceeding acceptable fuel damage limits.

ANSWER

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 control and shutdown groups.

The control group used in combination with chemical shim control 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 as well as those due to fuel depletion and fission product buildup, and decay.

Reference sections are as follows:

Section Title	Section
Reactor	3
Auxiliary and Emergency Systems	9

CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (Cagegory A)

At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently that to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

ANSWER

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

The shutdown groups are provided to supplement the control 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 combination of control rods and automatic boron addition via the emergency core cooling 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.

Reference sections are as follows:

Section Title	Section
Reactor; Nuclear Design and Evaluation	3.2.1
Engineered Safety Features; General Description	6.2
Chemical and Volume Control System	9.1
Supplement 1	Item 9
Supplement 1	Item 10

CRITERION 30 - REACTIVITY HOLDDOWN CAPABILITY

At least, one of the reactivity control systems shall be capable of making and holding the core subcritical under any condition with appropriate margin for contingencies.

ANSWER

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 the quality of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

Boric acid can be injected by one 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 15 hours after shutdown. Additional boric acid is employed if it is desired to bring the reactor to cold shutdown conditions. 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 primary power.

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.

Reference sections are as follows:

Section Title	Section
Reactor; Design Basis	3.1
Engineered Safety Features; General Description	6.2
Chemical and Volume Control System	9.1
Supplement 1	Item 9

CRITERION 31 - REACTIVITY CONTROL SYSTEM MALFUNCTION

The reactivity control systems shall be capable of sustaining any single malfunction, such as, unplanned continuous withdrawal (not ejection) of a control rod without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

ANSWER

The reactor protection systems are designed to limit reactivity transients to DNBR$\geq$1.3 due to any single malfunction such as in the reactor make-up controls.

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

Reference sections are as follows:

Section Title	Title
Instrumentation and Control	
Protection System	7.1
Control and Regulating System	7.2
Chemical and Volume Control System	9.1
Safety Analysis; Core and Coolant Boundary	
Protection Analysis	12.1
Supplement 1	Item 10

CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (Category A)

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

ANSWER

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

The Reactor Control System employs control rod clusters, at least half of which are fully withdrawn during power operation, serving as shutdown rods. The remaining rods comprise the controlling groups which are used to control load and reactor coolant temperature and are partially withdrawn during full power operation. The rod cluster drive mechanisms are wired into preselected groups, and are therefore prevented from being withdrawn in other than their respective groups. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming two of the highest worth groups to be accidentally withdrawn at maximum speed, yielding reactivity insertion rates which are well within the capability of the overpower-overtemperature protection circuits to prevent core damage.

No credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than its mechanical limit.

Reference sections are as follows:

Section Title	Section
Reactor Design Basis	3.1
Instrumentation and Control; Control and Regulating System	7.2
Safety Analysis; Core and Coolant Boundary Protection Analysis	12.1
Supplement 1	Item 10

CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY

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

ANSWER

The reactor coolant boundary is shown to be capable of accommodating without rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection.

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

Reference sections are as follows:

Section Title	Section
Reactor Coolant System Design	4.2
Reactor Coolant System Operation	4.3
Safety Analysis; Core and Coolant Boundary Protection Analysis	12.1
Supplement 1	Item 12

CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (Category A)

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

ANSWER

The reactor coolant pressure boundary is designed to minimize the probability of a rapidly propagating type failure.

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

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

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

Reference sections as follows:

Section Title	Section
Reactor Coolant System Design	4.2
Reactor Coolant System Operation	4.3
Reactor Coolant System Components	4.4
Supplement 1	Item 5
Supplement 1	Item 12

CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (Category A)

Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120°F above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60°F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

ANSWER

A discussion of how this criterion is met is presented in response to criterion 34.

CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (Category A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

ANSWER

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

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

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

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

Reference Sections are as follows:

Section Title	Section
Reactor Coolant System Design	4.2
Reactor Coolant System Operation	4.3

CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN (Category A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

ANSWER

The details of the Basis for Design for each of the engineered safety features is discussed for each individual system under the appropriate criteria.

The design, fabrication, testing and inspection of the core, reactor coolant pipe and their protection systems gives assurance of safe and reliable operation under all anticipated normal, transient, and accident conditions. However, engineered safety features are provided as backup to these components. These engineered safety features have been designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends, and to cope with any steam or feedwater line breakup to and including the main steam or feedwater headers.

Limiting the release of fission products from the reactor fuel is accomplished by the Emergency Core Cooling System (Safety Injection System) which, by cooling the core, keeps the fuel in place and substantially intact and limits the metal-water reaction to an insignificant amount.

The Safety Injection System consists of high head and low head centrifugal pumps driven by electric motors, and passive accumulator tanks which are self energized and which act independently of any actuation signal or power source.

The release of fission product from the containment is limited in three ways:

1. Blocking the potential leakage paths from the containment. This is accomplished by:

a) A reinforced concrete containment with a welded steel liner attached to the inside face of the concrete shell and continuously pressurized penetrations, which forms a virtually leak-tight barrier to the escape of fission products should a loss-of-coolant accident occur.

b) Isolation of process lines by the Containment Isolation System which impose double barriers in each line which penetrates the containment. An isolation valve seal water system provides a water seal at the valves thus sealing pipes penetrating the containment.

2. Reducing the fission product concentration in the containment atmosphere. This is accomplished by the containment spray system with chemical additives.

3. Reducing the containment pressure and thereby limiting the driving potential for fission product leakage. This is accomplished by cooling the containment atmosphere by the following independent system of equal heat removal capacity:

   a) Containment Spray System

   b) Containment Air Recirculation Cooling System

Reference sections are as follows:

Containment; System Structures	5.1
Engineered Safety Features; General Description	6.2
Supplement 1	Item 2
Supplement 1	Item 13
Supplement 1	Item 16

CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

ANSWER

All engineered safety features are designed to provide high functional reliability and ready testability. A comprehensive program of plant testing is performed for all equipment vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturers' shops, an integrated test of the system as a whole, and periodic tests of the activation circuitry and mechanical components to assure reliable performance upon demand throughout the plant life time. All active components can be individually tested during plant operation.

The initial tests of the individual components and the integrated test of the system as a whole complement each other to assure performance of the system as designed and to prove proper operation of the actuation circuitry.

Reference section as follows:

Section Title	Section
Engineered Safety Features; General Description	6.2

CRITERION 39 - EMERGENCY POWER FOR ENGINEERED SAFETY FEATURES

Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active compoent in each power system.

ANSWER

The power requirements for the engineered safety features can be supplied from a number of off-site sources or standby power.

Each supply is so designed and arranged that it is not dependent on the other source for successfuel operation. Each of the three diesels (the on-site source) has its own stored source of starting energy, and actuates automatically and independently on loss of outside power. Each diesel control equipment includes provision for manual synchronization so the units can be test loaded at any time.

Offsite and standby onsite power are independent.

Redundancy of all active parts of the system assures adequate power for the essential engineered safety features.

Three diesel generators are installed, any two of which have sufficient capacity to supply power for the essential engineered safety features operation in the event of hypothetical accident coincident with loss of off-site power. As a basis of system design, transformers and buses are not considered active components.

After installation at the site the diesel units together with sequencing equipment, pumps, and other active components are tested to assure that the desired starting sequence and loading of the engineered safety features is successfully accomplished.

Reference sections are as follows:

Section Title	Section
Engineered Safety Features; General Description	6.2
Emergency Power Sources	8.4
Supplement 1	Item 17

CRITERION 40 - MISSILE PROTECTION (Category A)

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

ANSWER

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

For such engineered safety features as are required to ensure safety in the unlikely event of a loss-of-coolant accident or other equipment failure, protection from these dynamic effects or missiles is considered in the layout of plant equipment and missile barriers. Fluid and mechanical driving forces are calculated, and consideration is given to the possibility of damage due to fluid jets and missiles which might be produced by the action of such jets.

Consideration is given during the design of the plant to the following potential sources of missiles: valve stems and bonnets, instrument thimbles including installed sensors, bolts, complete control rod drive shafts and/or mechanisms.

Layout and structural design specifically protect injection paths leading to unbroken reactor coolant loops against damage as a result of the maximum reactor coolant pipe rupture. Injection lines penetrate the main missile barrier, which is the crane wall, and the injection headers are located in the missile-protected area between the crane wall and the containment outside wall. Individual injection lines, connected to the injection header, pass through the barrier and then connect to the loops. Separation of the individual injection lines is provided to the maximum extent practicable. Movement of the injection line, associated with rupture of a reactor coolant loop, is accommodated by line flexibility and by the design of the pipe supports such that no damage outside the missile barrier is possible.

The containment structure is capable of withstanding the effects of potential missiles originating outside the containment and which might be directed toward it so that no loss-of-coolant accident can result.

Limited plastic response of structures is considered at points of impact during dynamic load transients and in connection with response of redundant members of statically indeterminate structures.

Reference sections are as follows:

Section Title	Section
Reactor Coolant System	4
Containment Systems Structures	5.1
Engineered Safety Features; General Description	6.2
Safety Analysis	
Core and Coolant Boundary Protection Analysis	12.1
Major Rupture of a Reactor Coolant Pipe	12.3
Supplement 1	Item 6

CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY (Category A)

Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

ANSWER

Each engineered safety feature provides sufficient performance capability to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public; namely, to limit the maximum off-site exposure to the levels outlined in 10 CFR 100. The accident condition considered as the basis is the hypothetical case of release of fission products per TID-14844. Also, the total loss of all outside power is assumed concurrently in this analysis.

Under the above accident conditions, the containment Air Recirculation Cooling and Filtration Systems and the containment spray system are sized so that both systems, each operating with partial effectiveness, are able to supply the necessary post-accident iodine removal and cooling capacity to assure maintenance of containment integrity, that is, keeping pressure below design pressures at all times assuming the core residual heat is released to the containment atmosphere as steam. Partial effectiveness is defined as operation of a system with one active component failure.

Reference sections are as follows:

Section Title	Section
Containment System Structures	5.1
Engineered Safety Features; General Description	6.2
Auxiliary and Emergency System	
Chemical and Volume Control System	9.1
Auxiliary Coolant System	9.2
Supplement 1	Item 14

CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY (Category A)

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

ANSWER

All components of the engineered safety features are designed and laid out so that the mechanical environment accompanying emergency situations in which the components are required to function does not interfer with that function.

The design, fabrication, inspection, and testing of the various engineered safety features components have as a prime objective safe and reliable operation under all anticipated conditions including the accident.

Instrumentation pumps, fans, filter, cooling unit, valves, motors, cables, penetrations and equipment of the engineered safety features located inside the containment are selected to meet the most adverse expected ambient conditions to which they are subjected as an accident consequence. Items are either protected from containment ambient conditions or are designed to withstand, without failure, direct immersion in the worst combination of temperature, pressure, and humidity expected during the required operational period.

The emergency core cooling system pipes serving each loop are anchored at the missile barrier in each loop area to restrict potential accident damage to the portion of piping beyond this point. The anchorage is designed to withstand, without failure, the thrust force of any branch line severed from the reactor coolant pipe and discharging fluid to the atmosphere, and to withstand a bending moment equivalent to that which produces failure of the piping under the action of free end discharge to atmosphere or motion of the broken reactor coolant pipe to which the emergency core cooling pipes are connected. This prevents possible failure at any point upstream from the support point including the branch line connection into the piping header.

(Reference) sections are as follows:

Section Title	Section
Reactor Coolant	4
Containment	5
Emergency System Features	6
Auxiliary and Emergency System	9

CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (Category A)

Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

ANSWER

For any action of the engineered safety features which might accentuate the adverse after-effects of a loss of normal cooling, appropriate preventative features are included in the primary coolant boundary and engineered safety features system design.

The reactor is maintained subcritical following a primary system pipe rupture accident. Introduction of borated cooling into the core, results in a net negative reactivity addition. The control rods insert and remain inserted. The supply of water by the safety injection to cool core cladding does not produce significant metal-water reaction.

The delivery of cold emergency core cooling water to the reactor vessel following accidental expulsion of reactor coolant does not cause further loss of integrity to the Reactor Coolant System boundary.

Reference section as follows:

Section Title	Section
Engineered Safety Features; General Description	6.2

CRITERION 44 - EMERGENCY CORE COOLING SYSTEMS CAPABILITY (Category A)

At least two emergency core cooling systems, preferable of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform it is required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

ANSWER

Adequate emergency core cooling is provided by the Safety Injection System (which constitutes the Emergency Core Cooling System) whose components operate in three modes. These modes are delineated as passive accumulator injection, active safety injection and residual heat removal recirculation.

The primary purpose of the Safety Injection System is to automatically deliver cooling water to the reactor core in the event of a loss-of-coolant accident. This limits the fuel clad temperature and thereby ensures that the core will remain intact and in place, with its essential heat transfer geometry preserved. This protection is afforded for:

a) All pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop, assuming unobstructed discharge from both ends.

b) A loss of coolant associated with the rod ejection accident.

c) A steam generator tube rupture.

The basic design criteria for loss of coolant accident evaluations are:

1. The cladding temperature is to be less than:
   a. The melting temperature of Zircaloy-4.
   b. The temperature at which gross core geometry distortion, including clad fragmentation may be expected.
2. The total core metal-water reaction will be limited to less than 1 percent.

These criteria will assure that the core geometry remains in place and substantially intact to such an extent that effective cooling of the core is not impaired.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the Safety Injection System adds shutdown reactivity so that with a stuck rod, no off-site power and minimum engineered safety features, there is no consequential damage to the Reactor Coolant System and the core remains in place and intact.

Redundancy and segregation of instrumentation and components is incorporated to assure that postulated malfunctions will not impair the ability of the system to meet the design objectives. The system is effective in the event of loss of normal station auxiliary power coincident with the loss of coolant, and is tolerant of failures of any single component or instrument channel to respond actively in the system. During the recirculation phase of a loss of coolant, the system is tolerant of a loss of any part of the flow path since back up alternative flow path capability is provided.

Reference sections are as follows:

Section Title	Section
Engineered Safety Feature; Safety Injection System	6.2.1
Safety Analysis; Core and Coolant Boundary Protection Analysis	12.1

CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

ANSWER

Design provisions are made to the extent practical to facilitate access to the critical parts of the reactor vessel internals, pipes, valves and pumps for visual or boroscopic inspection for erosion, corrosion and vibration wear evidence, and for non-destructive test inspection where such techniques are desirable and appropriate.

Reference sections are as follows:

Section Title	Section
Reactor Design Basis	3.1
Reactor Coolant System Design	4.2.1
Engineered Safety Features Design Objectives	6.1

CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEMS COMPONENTS (Category A)

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

ANSWER

Design provisions are made so that active components of the Emergency Core cooling System (Safety Injection System) can be tested periodically for operability and functional performance.

Power sources are arranged to permit individual activation of each active component of the Emergency Core Cooling System (Safety Injection System) during plant operation.

The safety injection pumps and the residual heat removal pumps can be tested periodically during plant operation using the minimum flow recirculation lines provided. The residual heat removal pumps are used every time the residual heat removal loop is put into operation. All remote operated valves can be exercised and actuation circuits can be tested periodically during plant operation or routine maintenance.

Reference section as follows:

Section Title	Section
Engineered Safety Features; General Description	6.2

CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

ANSWER

Design provisions include special instrumentation testing and sampling lines to perform test during plant shutdown to demonstrate proper automatic operation of Emergency Core Cooling System. (Safety Injection System).

An integrated system test can be performed when the plant is cooled down. This test would not introduce flow into the reactor coolant system but would demonstrate operation of the valves, pump circuit breakers, and automatic circuitry upon initiation of safety injection.

The accumulator tank pressure and level are continuously monitored during plant operation and flow from the tanks can be checked at any time using test lines.

Reference sections are as follows:

Section Title	Section
Engineered Safety Features; General Description	6.2
Electrical System	8

CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGANCY CORE COOLING SYSTEM (Category A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

ANSWER

The design provides for capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the Emergency Core Cooling System to demonstrate the state of readiness and capacity for the system.

This functional test is performed with the water level below the safety injection set point in the pressurizer and with the Reactor Coolant System initially cold and at low pressure. The Emergency Core Cooling system valving is set to initially simulate the system alignment for plant power operation.

The functioning of the accumulator tanks is checked by closing the stop valve, raising the pressure in the tank and then opening the stop valve and observing the rising pressurizer level. This rising water level in the pressurizer provides indication of system delivery.

This functional test provides information to confirm valve operating times, pump motor starting times, the proper automatic sequencing of load addition to the diesel generators, and delivery rates of injection water to the Reactor Coolant System.

Reference section as follows:

Section Title	Section
Engineered Safety Features; General Description	6.2

CRITERION 49 - CONTAINMENT DESIGN BASIS

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

ANSWER

The reactor containment structure utilizes continuously pressurized penetrations and containment heat removal systems, to limit below 10 CFR 100 values, doses resulting from leakage of radioactive fission products from the containment under those conditions that would result from the largest credible energy release following a loss-of-coolant accident including a margin to cover the effects of metal-water reaction or other undefined energy source.

Reference section as follows:

Section Title	Section
Containment System Structures	5.1

CRITERION 50 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL

Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less that 30° F above nil ductility transition (NDT) temperature.

ANSWER

The selection and use of containment materials comply with the applicable codes and standards tabulated in Section 5.

The concrete containment is not susceptible to a low temperature brittle fracture.

The containment liner is enclosed within the containment and thus is not exposed to the temperature extremes of the environs. The containment ambient temperature during operation is between 50 and 120° F which is expected to be well above the NDT temperature + 30°F for the liner materials. Containment penetrations which can be exposed to the environment are also designed to the NDT + 30°F Criterion.

Reference section as follows:

Section Title	Section
Containment Systems Structure Design	5.1.2

CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT
(Category A)

If part of the reactor coolant pressure boundary is outside the containment, appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriatencess of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

ANSWER

The reactor coolant pressure boundary does not extend outside the containment.

CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

ANSWER

Two separate, full capacity engineered safety features provide adequate heat removal capability for the containment following an accident: the containment spray system and the containment air recirculation cooling and filtration system.

Any of the following combination of equipment provide sufficient heat removal capability ot maintain the post-accident containment pressure below design value, assuming that the core residual heat is released to the containment as steam:

1. Both containment spray pumps
2. All five containment cooling fans
3. One containment spray pump and three of the five containment cooling fans.

Reference section as follows:

Section Title	Section
Engineered Safety Features; General Description	6.2

CRITERION 53 - CONTAINMENT ISOLATION VALVES

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

ANSWER

Isolation valves for all fluid system lines penetrating the containment provide at least two barriers for redundance against leakage of radioactive fluids to the environment in the event of a loss-of-coolant accident. These barriers, in the form of isolation valves or closed systems, are defined on an individual line basis. In addition to satisfying containment isolation criteria, the valving is designed to facilitate normal operation and maintenance of the systems and to ensure reliable operation of other engineered safety features.

With respect to numbers and locations of isolation valves, the criteria applied are generally described in Section 6.

Reference section as follows:

Section Title	Section
Containment Penetrations	5.1.4

CRITERION 54 - CONTAINMENT LEAKAGE RATE TESTING

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verigy its conformance with required performance.

ANSWER

After completion of the containment structure and installation of all penetration and weld channels, and initial integrated leakage rate test is conducted at the peak calculated accident pressure, maintained for a minimum of 24 hours, to verify that the leakage rate is not greater than 0.1 per cent by weight of the containment volume per day.

Reference section as follows:

Section Title	Section
Containment Pre-operational Tests	5.2.2

CRITERION 55 - CONTAINMENT PERIODIC LEAKAGE RATE TESTING

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

ANSWER

With the monitoring provisions for the pressurized double penetrations and weld seam channels there is no need to repeat the integrated leak rate test, unless major maintenance or modifications are made.

A leak rate test at the peak calculated accident pressure using the same method as the initial leak rate test can be performed at any time during the life of the plant, provided the plant is not in operation and precautions are taken to protect instruments and equipment from damage.

Reference section as follows:

Section Title	Section
Containment Post-operational Tests	5.2.3

CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leaktightness to be demonstrated at design pressure at any time.

ANSWER

A permanently piped monitoring system is provided such that all penetrations may be checked for leaktight integrity at any time throughout the operating life of the plant.

Penetrations are designed with double seals so as to permit countinuous pressurization of the interior of the penetration. The large access openings such as the equipment hatch and personnel air locks are equipped with double seals with the space between the seals connected to the pressurizing system. The system utilizes a supply of clean, dry, compressed air which places all the penetrations under an internal pressure equal to the peak calculated accident pressure.

Leakage from the system is checked by measurement of the integrated makeup air flow. In the event excessive leakage is discovered, each penetration can then be checked separately.

Reference section as follows:

Section Title	Section
Containment Penetrations	5.1.4

CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATION VALVES

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valves leakage does not exceed acceptable limits.

ANSWER

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus during periods of reactor shutdown.

Initiation of containment isolation employs coincidence circuits which allow checking of the operability and calibration of one channel at a time. Removal or bypass of one signal channel places that circuit in the partially-tripped mode.

Hydrostatic tests of isolation valves in series are performed by first testing the upstream valve with the second valve open, then opening the upstream valve and closing the second valve, so that each valve has an indepentent test.

The main steam and feedwater barriers and isolation valves in systems which connect to the Reactor Coolant System are hyrostatically tested to measure leakage.

Valves in the Residual Heat Removal and Safety Injection Systems are not considered to be isolating valves in the usual sense inasmuch as the system would be in operation under accident conditions.

Field and operational inspection and testing have been divided into three phases:

a) That taking place during erection of the containment building liner; construction tests.

b) That taking place after the containment structure is erected and all penetrations are complete and installed; pre-operational tests.

c) Monitoring during reactor operation; post-operational tests.

Reference section as follows:

Section Title	Section
Containment System Structure - Inspection and Testing	5.2

CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

ANSWER

Design provisions are made to facilitate periodic visual inspections of active components and other important equipment of the containment pressure reducing systems.

Reference sections are as follows:

Section Title	Section
Containment System Structures	5.1
Containment System Structure - Inspection and Testing	5.2
Engineered Safety Features; General Description	6.2

CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEMS COMPONENTS

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

ANSWER

The containment spray system is designed to the extent practical so that the spray pumps, spray injection valves, spray nozzles and additive injection valves can be tested periodically and after any component maintenance for operability and functional performance.

The air recirculating and cooling units, and the service water pumps that supply the cooling units are in operation on a relatively continuous schedule during plant operation, and no additional periodic tests are required.

Reference section as follows:

Section Title	Section
Engineered Safety Features; General Description	6.2

CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS

A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

ANSWER

Permanent test lines for the containment spray loops are located so that all components up to the isolation valve at the spray nozzles may be tested. These isolation valves are checked separately.

The air test lines for checking that spray nozzles are not obstructed connects upstream of the isolation valve. Air flow through the nozzles is monitored by use of a smoke generator.

Reference sections are as follows:

Section Title	Section
Containment System Structures	5.1
Engineered Safety Features; General Description	6.2

CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEMS

A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

ANSWER

Capability is provided to test initially to the extent practical the operational startup sequence beginning with transfer to alternate power sources and ending with near design conditions for the Containment Spray and Containment Air Recirculation Cooling and Filtration Systems.

Reference section as follows:

Section Title	Title
Containment System Structures	5.1

CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS

Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as, ducts, filters, fans, and dampers.

ANSWER

Access is available for visual inspection of the Containment Spray System Components.

Reference section as follows:

Section Title	Section
Engineered Safety Features	6

CRITERION 63 - TESTING OF AIR CLEANUP SYSTEMS COMPONENTS

Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

ANSWER

All active components of the Containment Spray System are adequately tested both in pre-operational performance tests in the manufacturer's shop and in place testing after installation. Thereafter, periodic tests are also performed after component maintenance.

Reference section as follows:

Section Title	Section
Engineered Safety Features	6

CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS

A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

ANSWER

Permanent test lines are provided for the containment spray headers and located so that all components up to the isolation valve at the containment may be tested. These isolation valves are checked separately. Air test lines for checking the spray nozzles are connected downstream of the isolation valves. Air flow through the nozzles is monitored by a smoke generator or tell tales.

Reference section as follows:

Section Title	Section
Engineered Safety Features	6

CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

ANSWER

Means are provided to test initially under conditions as close to design as is practical, the full operational sequence that would bring the Containment Spray System into action, including transfer to the emergency diesel-generator power source.

Reference sections are follows:

Section Title	Section
Engineered Safety Features	6

CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

ANSWER

During reactor vessel head removal and while loading and unloading fuel from the reactor, the 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. 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 were 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.

Reference section as follows:

Section Title	Section
Fuel Handling System	9.4

CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

ANSWER

All fuel storage and waste handling facilities are contained in the auxiliary building and equipment is designed such that accidental releases of radioactivity directly to the environment are not possible.

The spent fuel storage pit is designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor. It is designed to accommodate a total of 1 1/3 cores and a fuel shipping cask. The spent fuel storage pit is constructed of reinforced concrete. The entire interior basin face and transfer canal are lined with stainless steel plate.

The spent fuel pit cooling loop removes the residual heat from the spent fuel pit. The loop is normally required to handle the heat load from 1/3 of a freshly unloaded core but it can safely accommodate the heat load from 1 1/3 cores for which there is available storage space. Water removed from the storage pit must be pumped out as there are no gravity drains. Therefore, loss of coolant by inadvertent draining is not possible.

The spent fuel pit cooling loop consists of a pump, heat exchanger, filter, demineralizer, piping, associated valves and instrumentation. The pump draws water from the pit, circulates it through the heat exchanger, and returns it to the pit. Component coolant 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. However, in the event of failure of the spent fuel pump alternate cooling connections are provided in the spent fuel pit loop for connecting a temporary pump to the spent fuel pit loop.

Reference sections are as follows:

Section Title	Section
Fuel Handling System	9.4
Supplement 1	Item 17

CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHEILDING

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.

ANSWER

The fuel handling shielding facilitates removal and transfer of spent fuel assemblies and control rod clusters from the reactor vessel to the refueling canal. It is designed to attenuate radiation from the spent fuel, control clusters, and reactor vessel internals to tolerable levels.

The reactor cavity, flooded 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 feet above the reactor vessel flange. This height insures a minimum of 10 feet of water above a withdrawn fuel assembly and permits a visual control of the operation at all times while maintaining low radiation levels (less than 20 mr/hr) for periodic occupancy of the area by operating personnel.

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 feet 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 feet thick shields the spent fuel transfer tube. This shielding is designed to protect personnel from radiation during the time the spent fuel assemblies pass through the main concrete support of the reactor, containment and the transfer tube.

The concrete walls of the reactor cavity and the refueling canal are designed to shield personnel from radiation emanating radially from the spent fuel assemblies and control rod clusters during refueling. The sections of wall are four and six feet thick. Water in the spent fuel pit protects personnel who are above the fuel assemblies during handling operations. The water level above the pit floor insures sufficient shielding above the fuel assemblies and control rods at all times during storage and handling.

The design provides proper shielding for radiation protection in the spent fuel and waste storage areas to meet 10 CFR 20 requirements.

Reference section as follows:

Section Title	Section
Fuel Handling System	9.4

CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

ANSWER

The fuel handling system is designed to provide 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-radiation cooling. The system is designed to minimize the possibility of mishandling or maloperations that would cause fuel damage and potential fission product release. The refueling operation will follow a detailed procedure which is established to provide a safe, efficient refueling operation.

Fuel and waste storage areas are maintained in the auxiliary building. A ventilation system is provided which gives maximum safety for operating personnel and prevent undue amounts of radioactivity being released to public environs.

All fuel storage facilities are contained and equipment designed to that accidental releases of radioactivity directly to the atmosphere are monitored and will not exceed the guidelines of 10 CFR 100.

The reactor cavity, refueling canal and spent fuel storage pit are reinforced concrete structures. Canal walls and floor linnings are stainless steel. The entire interior basin face of the spent fuel storage pit is lined with stainless steel. These structures are designed to withstand the anticipated earthquake loadings as Class I structures so that the liner should prevent 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 Class I equipment.

All waste handling and storage facilities are contained and equipment designed so that accidental release directly to the atmosphere is monitored and will not exceed the limits of 10 CFR 100.

Reference sections are as follows:

Section Title	Section
Fuel Handling System	9.4
Waste Disposal System	11.1
Radiation Protection	11.2
Supplement 1	Item 17

CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT

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 shall 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 occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

ANSWER

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, 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 excessive releases.

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 reused 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 environmental conditions do not restrict the release of radioactive effluents to the atmospehre.

Liquid wastes are processed to remove most of the radioactive materials. 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.

Reference sections are as follows:

Section Title	Section
Waste Disposal System	11.1
Supplement 1	Item 17

ITEM 2

As stated in your application, the design of Indian Point Unit No. 3 is essentially the same as Indian Point Unit No. 2. An adequate response to some of the information requested concerning Indian Point Unit No. 2 which reflects updated technology and the availability of more definitive information will be required. Specifically, we find that complete answers to the following questions concerning Unit 2 are not incorporated in the Unit 3 PSAR:

Letter of February 28, 1966 (Letter 1)

Questions: 3(a), (c), (d), (e), (f), (g), (h), (i), (k), (1);
4(a), (b), (c), (d), (e), (f);
5(b), (c), (d), (e), (f);
6; 8; 10; 11; 12; 13;
14(b), (c), (d), (e), (f);
18; and 19(i)

Letter of May 11, 1966 (Letter 2)

Questions: 4, 5, 7, 8.

Letter 1

Item 2 (Letter 1 Question 3)

Your attention is directed to a letter from the ACRS to the Chairman, AEC, dated November 24, 1965, concerning reactor pressure vessels. Please discuss the consideration which has been given in the design of your facility to the recommendations contained in numbered paragraphs 1 and 2 of the ACRS letter. For your guidance in providing a complete answer to this question the following are some of the detailed areas of concern explored recently by an ACRS subcommittee on the proposed Rochester Gas and Electric Corporation Brookwood facility. Your reply should incorporate answers to these questions.

a. Please give details on the best prediction of maximum fast neutron flux dose in the pressure vessel, including uncertainties in prediction.

ANSWER

The maximum integrated fast neutron (E > 1 Mev) exposure of the vessel is computed to be 2.6×10^{19} n/cm^2 for 40 years operation at 80 per cent load factor, at a power level of 3025 MWt.

This calculated neutron exposure exceeds the value of 1.4×10^{19} n/cm^2 (E > 1 Mev) reported in the Preliminary Facility Description and Safety Analysis Report. The reasons for the increase are:

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

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

The calculation of the fast neutron flux at the irradiation samples, including uncertainties is given in the attached Appendix. The fast neutron flux at the irradiation samples always leads that at the vessel since the samples are closer to the reactor core.

APPENDIX TO QUESTION 3(a)

DETERMINATION OF REACTOR PRESSURE

VESSEL NDTT

1. MEASUREMENT OF INTEGRATED FAST NEUTRON (E >1.0 MEV) FLUX AT THE IRRADIATION SAMPLES

The spectrum of neutron fluxes at the irradiaton samples is obtained from the multigroup diffusion code P1MG[1]. Dosimeters including CdO shielded U_{238} and Np_{237}, Co Al, Cu, Ni, Cd shielded Co Al, and Fe from specimens will be contained in the capsule assemblies.

The procedure for measurement of fast neutron flux by the Fe^{54} (n,p) Mn^{54} reaction is described below. The measurement technique for the other dosimeters, which are sensitive to different portions of the neutron spectrum, is similar.

The product of this reaction Mn^{54} has a half life of 314 days and emits gamma rays of 0.84 Mev energy which are easily detected using a Na I scintillator. In irradiated steel samples, chemical separation of the Mn^{54} may be performed to ensure freedom fron interfering activities. This separation is simple and very effective, yielding sources of very pure Mn^{54} activity. In some samples all the interferences may be corrected for by gamma spectrometric methods without any chemical separation. The count data is used to give the specific activity of Mn^{54} per gram of iron. Because of the relatively long half-life of Mn^{54} the flux may be calculated for irradiation periods up to about two years. Beyond this time the dosimeter begins to reflect the later stages of the irradiation. Calculation of total dose is from flux and integrated power output. The burnout of the Mn^{54} produced is not significant until the thermal flux is about 10^{14} neutrons cm^{-2} sec^{-1}.

The analysis of the sample requires that two steps be completed, one the measurement of Mn^{54} disintegration rate per unit mass of sample and second measurement of the iron content of the sample. Having completed these analyses, the calculation of the flux is as follows:

For an irradiation the activity of an activation product is given by:

$$A = \emptyset \, \sigma \, N \, (1 - e^{-\lambda t_i}) \, e^{-\lambda t_d} \qquad (1)$$

Where $\emptyset$ is the neutron flux
σ the cross-section
N number of target atoms
λ decay constant of product
t_i irradiation time
t_d decay time from end of irradiation to counting time

Then for a power reactor operating at various power levels over some long period we should allow for flux changes by dividing the exposure period into several parts and normalizing the flux in each part as that fraction of full power represented. Then for τ periods:

$$A = \emptyset_m \, \sigma N \sum_{1}^{t} (1-e^{-\lambda t_{i_n}}) \, e^{-\lambda t_{d_n}} F_n \qquad (2)$$

Where $\emptyset$ is flux at maximum power

t_{i_n} irradiation time in n^{th} period

t_{d_n} cooling time from end of n^{th} period

F_n flux normalizing factor which is

$$\frac{\text{actual power output in } n^{th} \text{ period}}{\text{maximum possible in } n^{th} \text{ period}}$$

If now we write

$$\emptyset_m \, \sigma N = C \sum_{1}^{55} \emptyset_{PIMG} \, (E,r) \, {}^{s}F_e \, (E) \qquad (3)$$

where the right hand side of equation (3) is the sum of the products of P1MG fluxes and Fe^{54} (n,p) Mn^{54} cross section[2] average over the P1MG energy groups, then the measured neutron flux (E > 1 Mev) is given by

$$\emptyset\ (E > 1\ \text{Mev}) = C \sum_{E=1.0}^{10\ \text{Mev}} \emptyset_{P1MG}(E,r) \qquad (4)$$

where C is a constant.

The error involved in the measurement of the specific activity of the dosimeter after irradiation is estimated to be ±5%.

2. CALCULATION OF INTEGRATED FAST NEUTRON (E > 1.0 MEV) FLUX AT THE IRRADIATION SAMPLES

The method to be described herein is an approximation to the ideal 3 dimensional neutron transport solution but correlations between its predictions and measurements on samples irradiated in the Yankee and Saxton cores indicate good agreement.

The spectrum of neutron fluxes at the capsule location is obtained from the one dimensional multigroup diffusion code P1MG[1] for the array of annular shields surrounding a cylindrical core of infinite high. The cylindrical core has a cross sectional area equal to that of the actual core. The radial source distribution chosen for the core represents the expected average over the life of the plant. The magnitude of the neutron fluxes generated by the P1MG Code, which does not treat transport effects, is adjusted by applicaton of a spatial correction factor. This factor is the ratio of the fast neutron dose rate calculated by the SPIC-1[3] code for an all water medium surrounding a typical Westinghouse PWR to the fast neutron dose rate obtained by P1MG in the identical geometry. The SPIC-1 fast neutron

dose rate calculation uses an empirical fast neutron attenuation kernel in the form of a linear combination of single exponentials which has been fitted to the experimental fast neutron dose rate distribution in pure water.

The axial and aximuthal variations of neutron flux at the capsule location are determined separately. The axial distribution is expressed as the ratio of the normalized results of two calculations using PDQ4,[4] a two dimensional 4 group (r,z) diffusion code. In the first of these an infinitely high equivalent cylindrical core with a fission neutron source strength S_I, per unit height is surrounded by an all water medium containing the capsule location. In the second, the finite height is surrounded by an all water medium. The fixed source option of the PDQ4 code is selected so that the axial variation of source strength in the core represents a good approximation to the average over the core life. The radial distribution is identical to that chosen for P1MG. The ratio,

$$\frac{\phi\ (E,r,z)_F}{S_F} \quad x\ \frac{S_I}{\phi\ (E,r)_I}$$

where subscripts F and I denote finite and infinite core representations respectively, is the required axial correction term.

The aximuthal distributions of neutron fluxes at the sample location are derived from a comparison of the results of the two dimensional 4 group (x,y) code PDQ3[5] and the one dimensional 4 group diffusion program AIM-5[6]. In the PDQ3 calculation the core, whose shape can be specified exactly, is surrounded by an all water medium. The radial and aximuthal source distributions in the core are both reasonable approximations to the averages expected during the core life. The radial source distribution in the AIM-5 calculation, in which the equivalent cylindrical core is surrounded by an all water medium, is identical to that chosen for P1MG.

The product of,

1) The spatial corrected P1MG results,

2) Axial correction term, and

3) Aximuthal correction term,

defines the three dimensional variation of neutron flux at the sample locations.

The technique described above overpredicts Saxton measurements by 30 per cent and the Yankee measured values by 14 per cent. In both reactors the measured results are averages for a set of specimens in a capsule located outside the thermal shield opposite a core corner. It has also given excellent agreement with measured data reported for the PM2A reactor. Based on the above evidence, it is concluded that the P1MG calculation, corrected as described, is conservative by approximately 20 per cent.

3. MEASUREMENT OF THE INITIAL NDTT OF THE REACTOR PRESSURE VESSEL BASE PLATE AND FORGINGS MATERIAL

The unirradiated or inital NDTT of pressure vessel base plate and forgings material is presently measured by two methods. These methods are the drop weight test per ASTM E208 and the Charpy V-notch impact test (Type A) per ASTM E23. The NDTT is defined in ASTM E208 as "the temperature at which a specimen is broken in a series of tests in which duplicate no break performance occurs at 10°F higher temperature". Using the Charpy V-notch test, the NDTT is defined as the temperature at which the energy required to break the specimen is a certain "fixed" value. For SA 533 Grade B Class I, and A508 Class 2 steel the ASME III TAble N-421 specifies an energy value of 30 ft-lb. This value is based on a correlation with the drop weight test and is referred to as the "30 ft-lb-fix". A curve of the temperature versus energy absorbed in breaking the specimen is plotted. To obtain this curve, 15 tests are performed which include three tests at five different temperatures. The intersection of the energy versus temperature curve with the 30 ft-lb ordinate is designated as the NDTT.

The available data indicates differences as great as 40 degrees between curves plotted through the minimum and average values respectively. The determination of NDTT from the average curve is considered representative of the material and is consistent with procedures as specified in ASTM E23. In assessing the NDTT shift due to irradiation, the translation of the average curve is used.

As part of the Westinghouse surveillance program referred to above, Charpy V-impact tests, tensile tests, and fracture mechanics specimens are taken from the core region plates, and core region weldments including heat-affected zone material. The test locations are similar to those used in the tests by the fabricator at the plate mill.

The uncertainties of measurement of the NDTT of base plate are:

1) Differences in Charpy V-notch foot pound values at a given temperature between specimens.

2) Variation of impact properties through plate thickness.

The fracture toughness technology for pressure vessels and correlation with service failures based on Charpy V-notch impact data are based on the averaging of data. The Charpy V-notch 30 ft-lb "fix" temperature is based on multiple tests by the material supplier, the fabricator, and by Westinghouse as part of the surveillance program. The average of sets of three specimens at each test temperature is used in determining each of five data points (total of 15 specimens). In the review of available data, differences of 0°F to approximately 40°F have been observed in comparing curves plotted through the minimum and average values respectively. The value of NDTT derived from the average curve is judged to be representative of the material because of the averaging of at least 15 data points, consistent with the specified procedures ot ASTM E23. In the case of the assessment of NDTT shift due to fast neutron flux, the displacement of transition curves is measured. The selection of maximum, minimum or average curves for this assessment is not significant since like curves would be used.

There are quantitative differences between the NDTT measurements at the surface, 1/4 thickness or the center of a plate. Differences in NDTT between 1/4T and the center in heavy plates have been observed to vary from improvement in the NDTT to increases up to 85°F. The NDTT at the surface has been measured to be as much as 85°F lower than at 1/4T.

The 1/4T location is considered conservative since the enhanced metallurgical properties of the surface are not used for the determination of NDTT. In additon, the limiting NDTT for the reactor vessel after operation will be based on the NDTT shift due to irradiation. Since the fast neturon dose is highest at the inner surface, usage of the 1/4T NDTT criterion is conservative.

Data is being accumulated on the variation of NDTT across heavy section steels at WAPD. Similarly, the Pressure Vessel Research Committee is sponsoring an evaluation of properties of pressure vessel steels in plates 6 to 12 inches thick. Preliminary data has shown NDTT difference between 1/4T and center of less than 20°F. The present criterion of using NDTT + 60°F at the 1/4T location without taking advantage of the enhanced properties at the surface of reactor vessel plates is conservative.

To assess any possible uncertainties in the consdierationof NDTT shift for welds heat affected zone, and base metal, test specimens of these three "material types" has been included in the reactor vessel surveillance program.

REFERENCES

1. "P1MG, A one-dimensional multigroup P1 Code for the IBM-704," Bohl, H., Jr. et al, WAPD-TM-135 (1959).

2. "Radiation damage exposure and embrittlement of reactor pressure vessels," Shure K. Nuclear Applications, Vol. 2 (April 1966).

3. "SPIC-I, An IBM-704 Code to calculate the uncollided flux outside a right circular cylinder," Gillis P. A. et al, WAPD-TM-176 (1959).

4. "PDQ4, A program for the soluton of the neutron diffusion equations in two dimensions on the Philco-2000," Cadwell, W. R., WAPD-TM-230 (1961).

5. "PDQ3, A program for the solution of the neutron diffusion equations in two dimensions on the IBM-704, Cadwell W. R., WAPD-TM-179 (May 1960).

6. "AlM-5, A multigroup one dimensional diffusion equation code," Flatt, H. P. and Baller, D. C. NAA-SR-4694 (March 1960).

QUESTION c

Please give details on method of prediction of NDT shift with fast neutron dose and quantitatively describe uncertainties therein, including considerations of weld regions and heat affected zones.

ANSWER

The predicted NDTT shift for an integrated fast neturon (E > 1 Mev) exposure of 2.6×10^{19} n/cm^2 is 246°F, the value obtained from the curve shown in Figure 3(c)-1 for 550°F irradiation.

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

In the surveillance programs, the evaluation of the radiation damage is based on pre- and post-irradiation testing of Charpy V-notch, tensile and wedge opening loading (WOL) fracture mechanics type test specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach, and are in accordance with ASTM E185, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors."

The reactor vessel surveillance programs use eight specimen capsules which are located about 3 inches from the vessel wall directly opposite the center portion of the core.

The capsules can be removed when the vessel head is removed. The capsules contain reactor vessel steel specimens from the shell plates located in the core region of the reactor and associated weld metal and heat affected zone metal. In addition, correlation monitors made from fully documented specimens of SA302 Grade B material obtained through Subcommittee II of ASTM

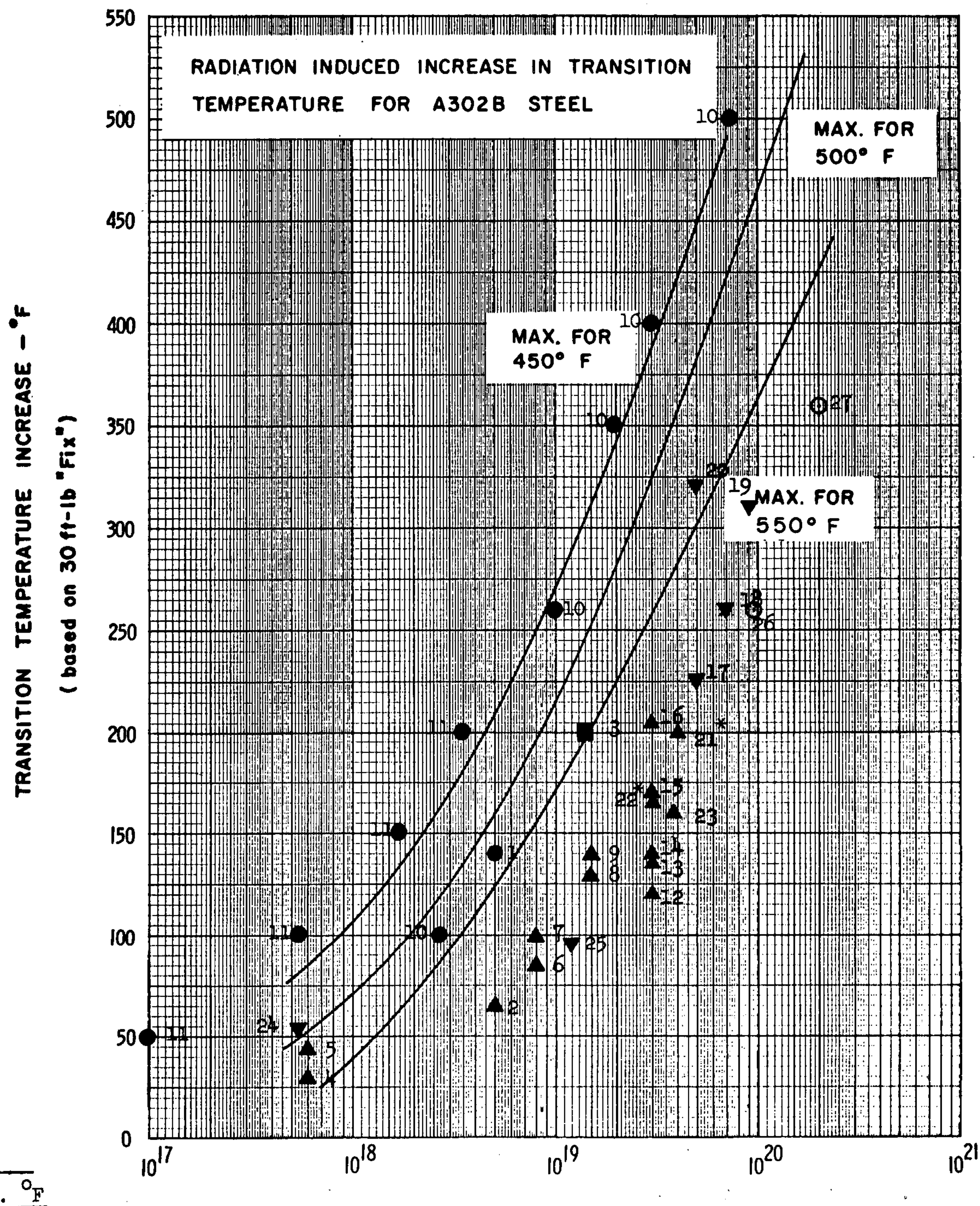
Committee E10 Radioisotopes and Radiation Effects are inserted in the capsules. The eight capsules will contain at least 27 tensile specimens, 256 Charpy V-notch specimens (which will include weld metal and heat affected zone material) and 42 WOL Specimens. Dosimenters including pure Ni, Al-Co (0.15% Co), Cd shielded Al-Co, CdO shielded Np-237 and CdO shielded U-238 are placed in impact specimens, tensile specimens or filler blocks drilled to contain the dosimeters. The dosimeters permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low melting alloys are included to monitor temperature of the specimens. The specimens are enclosed in a tight fitting stainless steel sheath to prevent corrosion.

The tentative schedule for removal of capsules is as follows:

Capsule	Estimated Exposure Time
1	Replacement of 1st region of core
2	Replacement of 2nd region of core
3	Replacement of 4th region of core
4	10 years
5	15 years
6	20 years
7 and 8	Extra capsules for complementary or duplicate testing or additional exposure.

Irradiation of the specimens will be higher than the irradiation of the vessel because the specimens are located in the vicinity of the core corners and are closer to the core than the vessel itself. Since these specimens will experience higher irradiation and are actual samples from the materials used in the vessel, the NDTT measurements will be representative of the vessel at a later time in life. Data from fracture toughness samples (WOL) are expected to provide additional information for use in determining allowable stresses for irradiated material.

The methods used to measure the initial NDTT of the reactor vessel base plate material are given in the attached Appendix to Question 3(a).



Code

Code	Temp. °F
●	450
■	490
▲	550
▼	475 to 540
⊙	600

Numbers 1 through 27 (see attached sheets)

FIGURE 3c-1

References for Figure 3c-1

RADIATION INDUCED INCREASE IN TRANSITION TEMPERATURE FOR A302B STEEL

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

**Transverse Specimens

Reference for Figure 3c-1 (Cont'd)

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

References for Figure 3c-1

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

QUESTION d

Please describe in detail the stress consideration to be allowable below NDT plus 60 and below NDT. State assumptions and give reasons, allowing for flaw size in particular.

ANSWER

The stress allowed in the vessel in relation to operation below NDTT and DTT (NDTT +60) to preclude the possibility of brittle failure are:

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

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

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

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

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

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

QUESTION e

Please give rationale for relationship between NDT and allowed stress emphasizing in particular the degree of conservatism which it is felt the circumstances require, and why.

ANSWER

The answer to this question is included in the answer to part (d).

QUESTION f

Define the flaw size and type in the pressure vessel which is accepted in the specifications. What flaws larger in size or special significance might not be detected, particularly in zones of irregular geometry?

ANSWER

Table 3(f)-1 summarizes the quality assurance program with regard to inspections performed on primary system components. In addition to the inspections shown in Table 3(f)-1, there are those which the equipment supplier performs to confirm the adequacy of material he receives, and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator are governed by ASME Code requirements. The inspection procedures and acceptance standards required on pipe materials and piping fabrication are governed by USAS B31.1 and Westinghouse requirements and are equivalent to those performed on ASME coded vessels.

Procedures for performing the examinations are consistent with those established in the ASME Code Section III and are reviewed by qualified Westinghouse engineers. These procedures have been developed to provide the highest assurance of quality material and fabrication. They consider not only the size of the flaws, but equally as important, how the material is fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the surfaces most subject to damage as a result of the heat treating, rolling, forging, forming and fabricating processes, receive a 100% surface inspection by Magnetic Particle or Liquid Penetrant Testing after all these operations are completed, although flaws in plates are inherently laminations in the center. All reactor coolant plate material is subject to shear as well as longitudinal ultrasonic testing to give maximum assurance of quality; all forgings receive the same inspection. In addition, 100% of the material volume is covered in these tests as an added assurance over the grid basis required in the code.

TABLE 3(f)-1

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

Component	RT*	UT*	PT*	MT*	ET*
1. Steam Generator					
1.1 Tube Sheet					
1.1.1 Forging		yes		yes	
1.1.2 Cladding		yes	yes		
1.2 Channel Head					
1.2.1 Casting	yes			yes	
1.2.2 Cladding			yes		
1.3 Secondary Shell & Head					
1.3.1 Plates		yes(a)			
1.4 Tubes		yes			yes
1.5 Nozzles (forgings)		yes		yes	
1.6 Weldments					
1.6.1 Shell, longitudinal	yes			yes	
1.6.2 Shell, circumferential	yes			yes	
1.6.3 Cladding			yes		
1.6.4 Nozzle to shell	yes			yes	
1.6.5 Support brackets				yes	
1.6.6 Tube-to-tube sheet			yes		
1.6.7 Instrument connections (primary and secondary)				yes	
1.6.8 Temporary attachments after removal				yes	
1.6.9 After hydrostatic test (all welds)				yes	
1.6.10 Nozzle safe ends (if forgings)	yes		yes		
1.6.11 Nozzle safe ends (if weld deposit)			yes		
2. Pressurizer					
2.1 Heads					
2.1.1 Casting	yes			yes	
2.1.2 Cladding			yes		
2.2 Shell					
2.2.1 Plates		yes(a)		yes	
2.2.2 Cladding			yes		
2.3 Heaters					
2.3.1 Tubing		yes	yes		
2.3.2 Centering of element	yes				
2.4 Nozzle		yes	yes		

TABLE 3(f)-1 (Continued)

Component	RT*	UT*	PT*	MT*	ET*
2.5 Weldments					
2.5.1 Shell, longitudinal	yes			yes	
2.5.2 Shell, circumferential	yes			yes	
2.5.3 Cladding			yes		
2.5.4 Nozzle Safe End (if forging)	yes		yes		
2.5.5 Nozzle Safe End (weld deposit)			yes		
2.5.6 Instrument Connections			yes		
2.5.7 Support Skirt				yes	
2.5.8 Temporary attachments after removal				yes	
2.5.9 All welds after hydrostatic test				yes	
3. Piping					
3.1 Fittings (Castings)	yes		yes		
3.2 Fittings (Forgings)		yes	yes		
3.3 Pipe		yes	yes		
3.4 Weldments					
3.4.1 Longitudinal	yes		yes		
3.4.2 Circumferential	yes		yes		
3.4.3 Nozzle to run pipe	yes		yes		
3.4.4 Instrument connections			yes		
4. Pumps					
4.1 Casting	yes		yes		
4.2 Forgings		yes	yes		
4.3 Weldments					
4.3.1 Circumferential	yes		yes		
4.3.2 Instrument connections			yes		
5. Reactor Vessel					
5.1 Forgings					
5.1.1 Flanges		yes		yes	
5.1.2 Studs		yes		yes	
5.1.3 Head Adapters		yes	yes		
5.1.4 Head Adapter Tube		yes	yes		
5.1.5 Instrumentation Tube		yes	yes		
5.1.6 Main Nozzles		yes		yes	
5.1.7 Nozzle Safe Ends (If forging is employed)		yes	yes		
5.2 Plates		yes(a)		yes	
5.3 Weldments					
5.3.1 Main Steam	yes			yes	
5.3.2 CRD Head Adapter Connection			yes		

TABLE 3(f)-1 (Continued)

Components	RT*	UT*	PT*	MT*	ET*
5.3.3 Instrumentation Tube Connection			yes		
5.3.4 Main Nozzles	yes			yes	
5.3.5 Cladding		yes (b)	yes		
5.3.6 Nozzle Safe Ends (if forging)	yes		yes		
5.3.7 Nozzle Safe Ends (if weld deposit)			yes		
5.3.8 Head adaptor forging to head adaptor tube	yes		yes		
5.3.9 All welds after hydrotest				yes	
6. Valves					
6.1 Castings	yes		yes		
6.2 Forgings		yes	yes		

*RT - Radiographic: UT - Ultrasonic: PT - Dye Penetrant:
MT - Magnetic Particle; ET - Eddy Current.

(a) 100% scanning for longitudinal wave technique and 100% shear wave technique
(b) UT of clad bond-to-base steel.

QUESTION g

What flaw size is accepted in the studs of the pressure vessel? What frequencies of stud inspection or replacement is planned? How many studs can fail without threatening the integrity of the closure?

ANSWER

The flaw size on the stud forgings will be determined by two ultrasonic inspections. A radial longitudinal beam inspection will be performed. The rejection standard will be 100% loss of back reflection or an indication in excess of 20% of the adjusted back reflection from a 1/2 inch diameter flat bottom hole. A radial inspection will be made using the angle beam technique. This inspection will carry the same rejection standards as for forgings.

The closure studs are inspected periodically using either magnetic particle tests and/or ultrasonic tests. The frequency of inspection will be specified the Final Safety Analysis Report.

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

QUESTION h

Please describe requirements concerning the support structure for the pressure vessel, including the degree of levelness over reactor life, which are needed to insure no problems due to local overstressing of the pressure vessel.

ANSWER

The reactor vessel support structure is discussed in the answer to Question 11 (Appendix 4A). It is designed to restrain vertical, lateral and rotational movement of the reactor vessel, while allowing for thermal growth by permitting radial shielding at each support on bearing plates. The degree of levelness provided for the pressure vessel is 1/2 a thousandth of a foot of diameter.

QUESTION i

Describe how small leaks in the pressure vessel would be detected and the action to be taken, should such occur. How is adequate response assured in the event of a previous existence of small leaks in other parts of the system?

ANSWER

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

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

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

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

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

The Technical Specifications will define maximum permissible leak rates which if exceeded will require the reactor to be shut down within a specified period.

The equipment described above provides indication of normal background which is indicative of a basic level of leakage from primary systems and components. Any increase in the observed parameters is an indication of change within the containment, and the equipment provided is capable of monitoring this change. The basic design criterion is the detection of deviations from normal containment environmental conditions.

QUESTION k

Are your considering procedures for detecting the propagation of cracks within the pressure vessel wall, i.e., acoustic emission?

ANSWER

Procedures such as acoustic emission are being investigated (probably 3 to 5 years before a method will be available). Accessibility to the pressure vessel is available.

QUESTION 1

State and justify the energy required to initiate failure of the primary system boundary. Can a control cluster ejection or any credible mechanism provide this amount of energy by reactivity insertion?

ANSWER

The reactor coolant boundary is capable of accommodating without further rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection. Details of the analysis will be provided in the Final Safety Analysis Report. Similar analyses for Unit No. 2 (to be presented in the Final Safety Analysis Report for Unit No. 2) verify this conclusion.

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

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

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

ITEM 2 (Letter 1 Question 4)

The PSAR contains pressure transient curves following the double-ended rupture of a primary coolant pipe for conditions wherein all engineered safeguards function, no engineered safeguards function, and the engineered safeguards function on emergency power only. In order that we may assess the margin of safety provided by these systems in the containment design and the relative effectiveness of each engineered safeguard, provide the following information:

a. Relate the available energy sources by showing the total energy that could be provided by (1) the primary coolant, (2) a 100% metal-water reaction, (3) the hydrogen-air reactio, and (4) the core decay heat at 10, 20, and 30 minutes. The relative energy sources should be provided on a percentage basis that totals 100% for each case. What is the total energy available from secondary sources (e.g., steam generators)?

b. Plot in graphical form up through one hour for your assumed model of post-accident conditions, (1) the ratio of decay heat energy in the containment atmosphere to primary coolant energy, (2) the ratio of metal-water energy in the containment atmosphere to primary coolant energy, (3) the ratio of H_2 recombination energy in the containment atmosphere to the primary coolant energy, (4) the ratio of total energy in the containment atmosphere to total available energy, and (5) the ratio of total energy in heat sinks to total available energy.

c. Indicate in graphical form the per cent of zirocium in the core available for reaction by providing a family of curves indicating the per cent of core clad at or above given temperatures and the zirconium assumed reacted, as a function of time with (1) no safety injection, and (2) full safety injection.

d. Provide the containment pressure transient curves following the MCA, assuming no further energy is added to the containment after the initial blowdown, for the cases wherein (1) all engineered safeguards function and (2) no engineered safeguards function and the containment and structures act as the only heat sink. Also show the per cent of total primary system energy lost as a function of time.

e. Provide the information in part (d) showing the increase in containment pressure resulting by (1) adding additional energy by the mechanism of steam generation equal to 50% and 100% of the original primary coolant energy, linearly with time in 1000 seconds and (2) adding additional energy stepwise equivalent to 20% of the Primary coolant energy at 500 and 1000 seconds by superheating the atmosphere. Also show the pound moles of air, steam, and hydrogen in the containment as a function of time.

f. Provide the containment pressure transient curves following the MCS for the cases in which the only engineered safeguards assumed to function are: (1) one high head and one low head safety injection pump, (2) one containment spray pump, and (3) four containment air recirculation coolers.

ANSWER

As stated in the answer to Item 16, Attachment E, Question 6.0, a Containment Integrity Evaluation is presented to provide additional information of the response of the containment following a loss-of-coolant accident. The type of analysis presented in the evaluations is intended to superseed the transients presented in the answers to Questions 4(a) through 4(f) of the February 28, 1966 letter which were included in Supplement 1 to the IPP #2 PSAR, Docket 50-247. It is felt that such evaluation provides a better conception of the containment capability.

ITEM 2 (Letter 1 Question 5)

The maximum specific power for the proposed fuel rods is higher than in any currently licensed reactor. In order to assess the conservatism of the proposed design, please provide the following information:

a. Summarize the peak heat flux factor (F_q), peak enthalpy rise factor ($F_{\Delta H}$), and the peak axial flux factor (F_z) for the following situations:

   (1) Nominal conditions for worst time in core life (using worst expected rod conditions).

   (2) Design conditions for worst time in core life (no engineering factors).

   (3) Hot channel conditions for worst time in life (with engineering factors).

b. Supply a distribution curve showing the fraction of the core operating above various power levels with their corresponding DNB ratios for condition (a-1).

c. For condition (a-2), provide the total number of fuel rods that are within 90% of the design peak power level and the corresponding DNB ratios (include the effects of instrument errors).

d. Repeat parts (b) and (c) for the overpower condition. If any channel has bulk boiling, or would require less than 5% additional power to cause boiling, tabulate these results indicating at what distance from the core top boiling ensues.

e. For a hypothetical 125% overpower condition, estimate whether any fuel rods approach design limits (e.g., DNB or center fuel melting).

f. For the hot channel positions, provide the DNB, exit quality, and center fuel temperature at 100%, 110% and 125% for the worst design conditions. In addition, arbitrarily raise the $F_{\Delta H}$ and F_z factors by 10% and tabulate as above for each condition.

ANSWER

(a) (1) Nominal conditions for worst time in core life (using worst expected rod conditions)

$$F_{\Delta H}^{T} = F_{\Delta H}^{N} F_{\Delta H}^{WFAB} = (1.38)\ (1.04) = 1.44$$

$$F_q^T = F_q^N = 2.13$$

$$F_z = 2.13/1.38 = 1.54$$

2. Design conditions for worst time in core life (no engineering factors)

$$F_{\Delta H}^T = F_{\Delta H}^N F_{\Delta H}^{WFAB} = (1.58)\ (1.04) = 1.64$$

$$F_q^T = F_q^N = 2.71$$

$$F_z = 2.71/1.58 = 1.72$$

3. Hot channel conditions for worst time in life (with engineering factors)

$$F_{\Delta H}^T = F_{\Delta H}^N F_{\Delta H}^E = (1.58)\ (1.075) = 1.70$$

$$F_q^T = F_q^N F_q^E = (2.71)\ (1.04) = 2.82$$

$$F_z = 2.71/1.58 = 1.72$$

<u>Definitions</u>

$F_{\Delta H}^T$ = Total enthalpy rise factor

$F_{\Delta H}^N$ = Nuclear radial peak to average factor

$F_{\Delta H}^{WFAB}$ = Enthalpy rise engineering factor without fabrication tolerances

F_q^T = Total heat flux factor

F_z = Nuclear axial peak to average factor

$F_q^N = (F_z)\ (F_{\Delta H}^{N})$

$F_{\Delta H}^{E}$ = Enthalpy rise engineering factor

F_q^E = Heat flux engineering factor

b. The distribution curve with the corresponding DNB ratios is illustrated in Figure 5.1. The following conditions wery employed:

Power = 100%
T_{in} = 549.7°F
P = 2250 psia
(a-1) conditions

c. As demonstrated in Figure 5.2, there are approximately 1770 fuel rods out of a total of 39,372 fuel rods that are within 90% of the peak power level. DNB ratios are shown in Figure 5.2. The following conditions were used:

Power = 102%
T_{in} = 553.7°F
P = 2220 psia
(a-2) conditions

d. Figure 5.3 specifies the distribution curve with the corresponding DNB ratios, for the following conditions:

Power = 106.5%
T_{in} = 549.7°F
P = 2250 psia
(a-1) conditions

DNBR AND POWER DISTRIBUTION

100% POWER, $F^N_{\Delta H} = 1.38$, $F^N_q = 2.13$

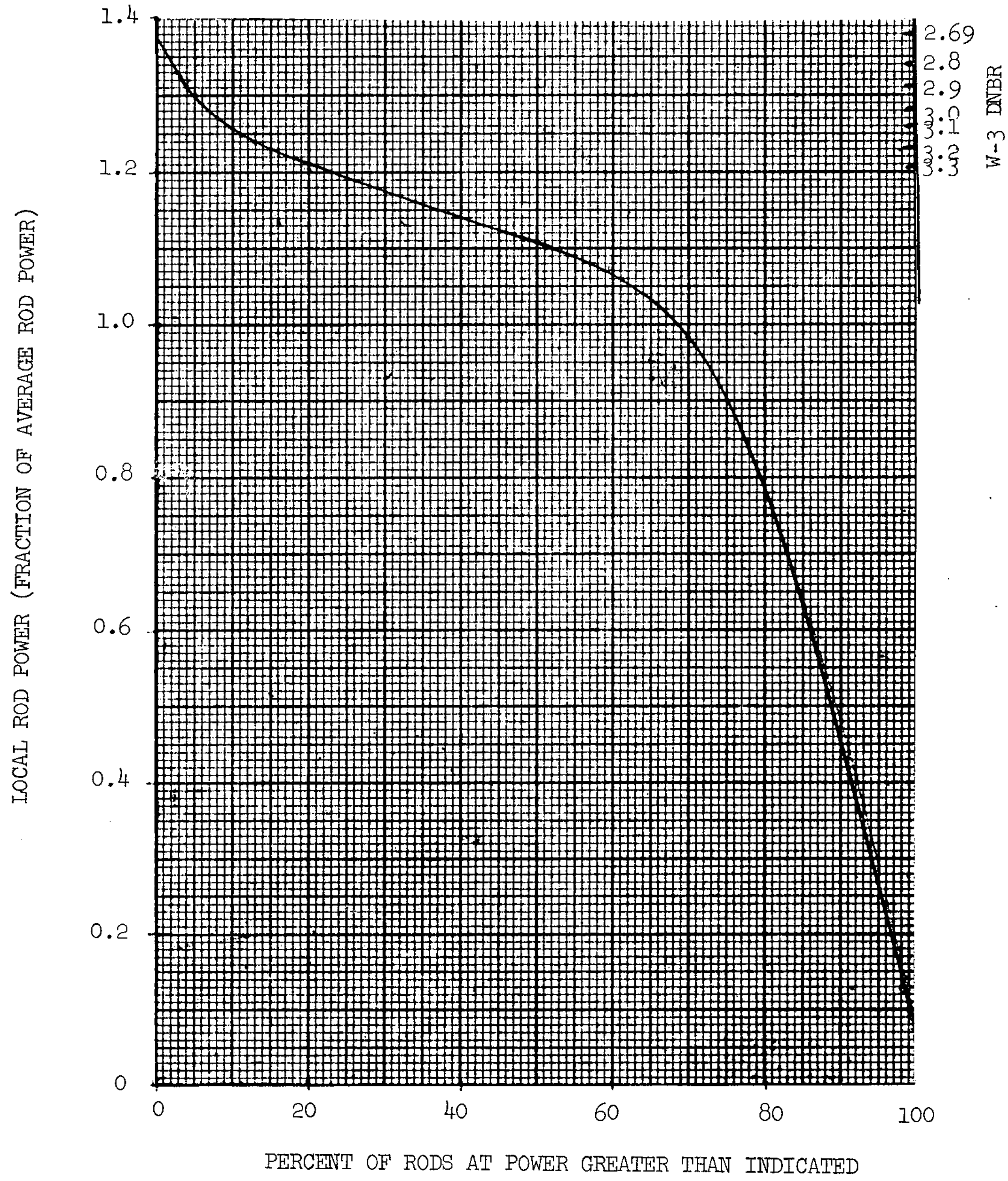


Figure 1
Supplement 1

DNBR AND POWER DISTRIBUTION

102% POWER, $F^N_{\Delta H} = 1.58$, $F^N_q = 2.71$

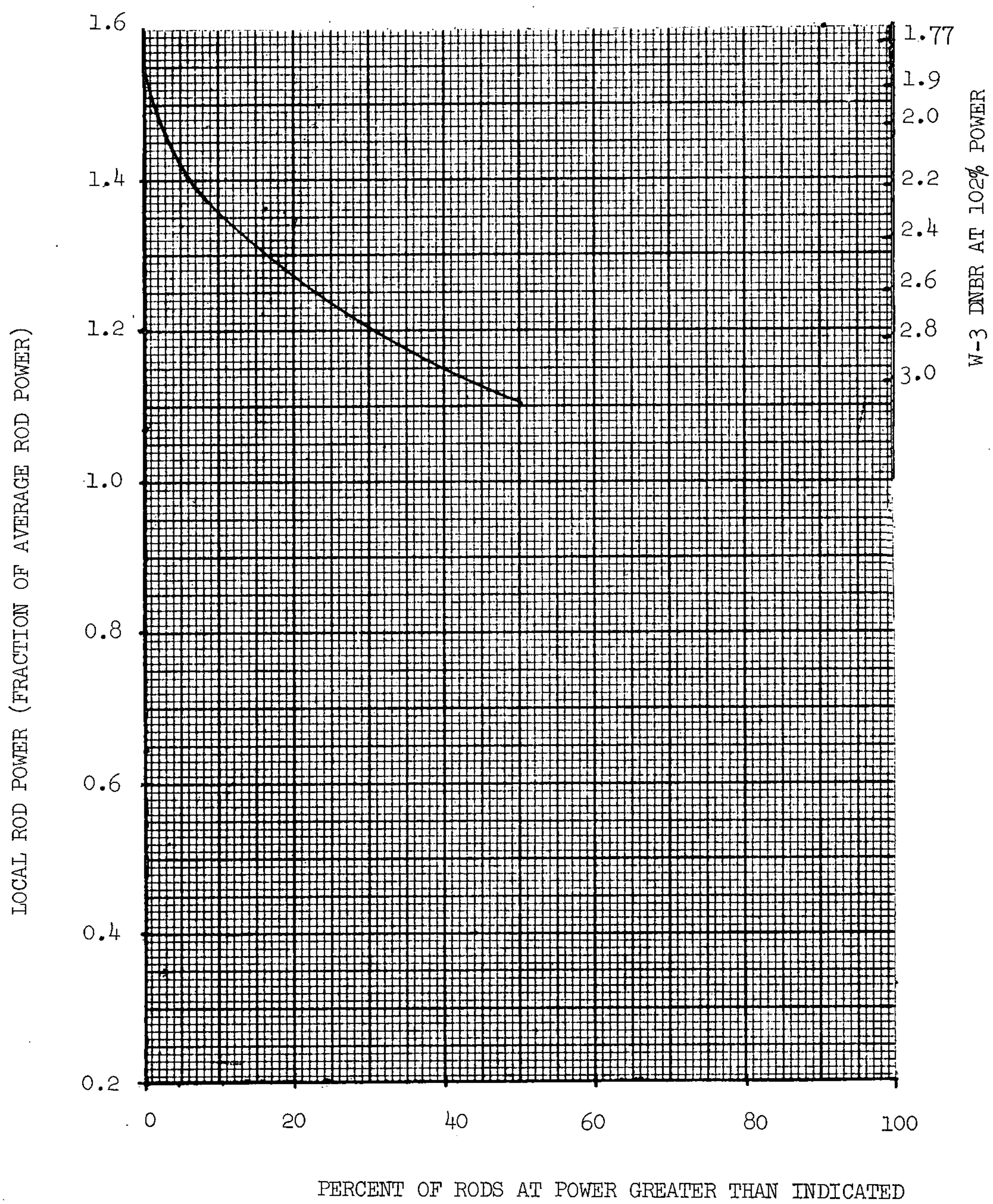


Figure 2
Supplement 1

DNBR AND POWER DISTRIBUTION
106.5% POWER, $F^N_{\Delta H} = 1.38$, $F^N_q = 2.13$

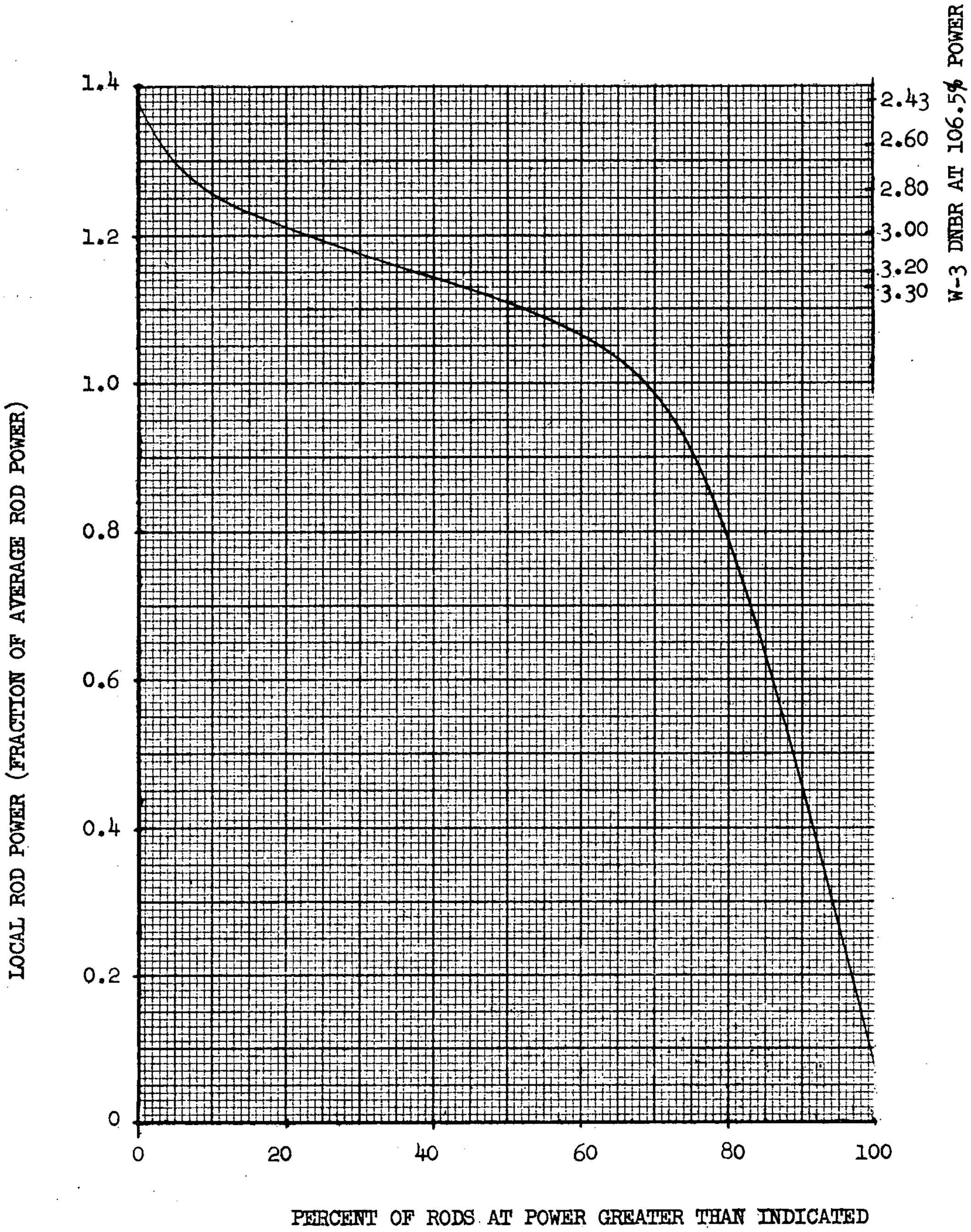


Figure 3
Supplement 1

Bulk boiling is discussed in the response to Item 10, Question A.2.

Figure 5.4 identifies the distribution curve with the corresponding DNB ratios for the following conditions:

Power = 112%

T_{in} = 549.7 °F

P = 2250 psia

(a-2) conditions

e. For a hypothetical 125% overpower condition, the center fuel temperature is 4580°F, and the minimum DNBR is 1.00. The following conditions were utilized:

T_{in} = 553.9°F

P = 2220 psia

F_H^N = 1.58

F_H^T = 1.70

F_q^N = 2.71

F_q^T = 2.82

It should be noted, however, that the reactor protection system limits the reactor overpower to 112%.

The analysis evaluating the number of fuel rods which may reach DNB for the 125% overpower condition is presented in Item 10, Question A.3.

DNBR AND POWER DISTRIBUTION

112% POWER, $F^N_{\Delta H} = 1.58$, $F^N_q = 2.71$

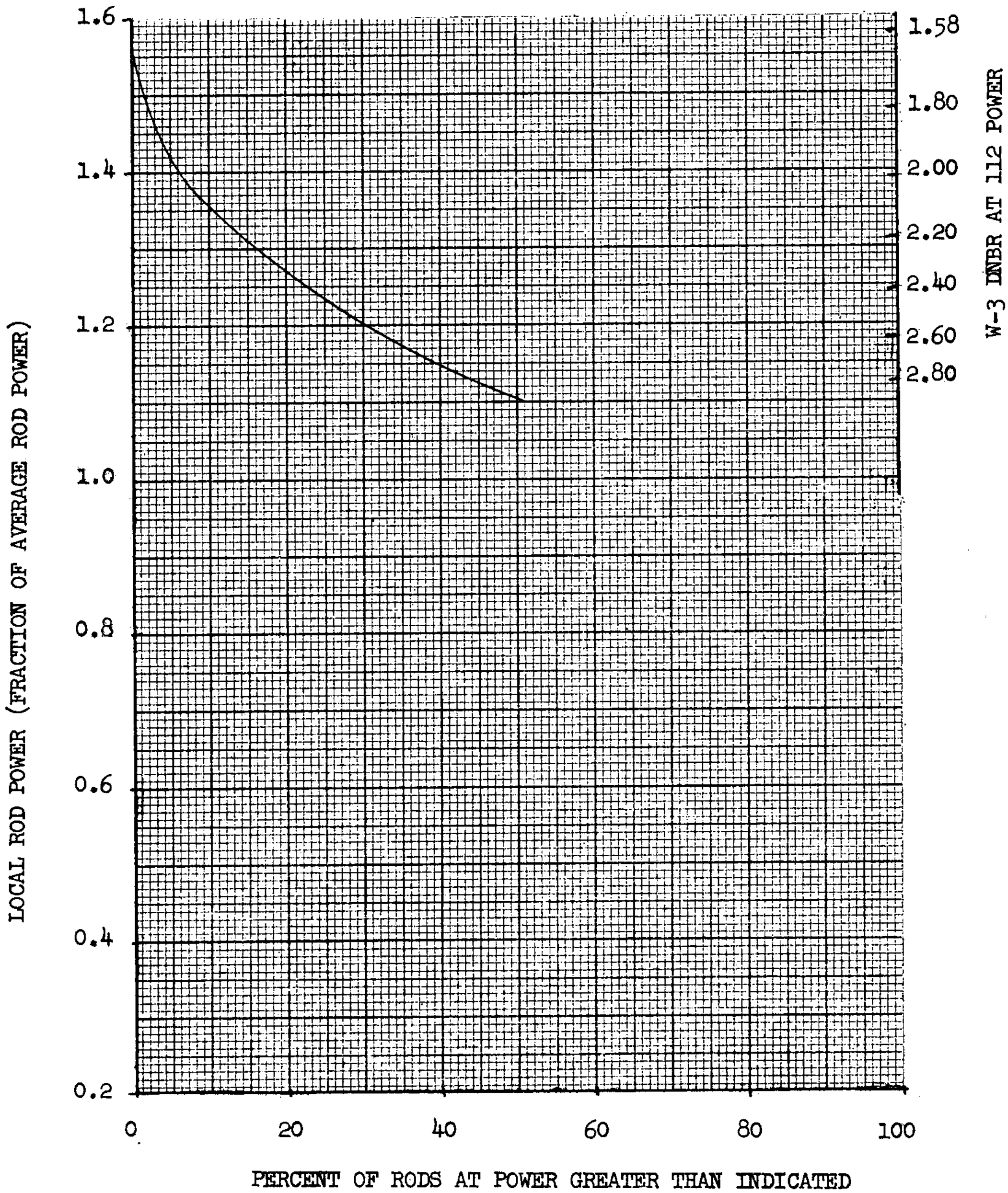


FIGURE 4

f.

Case	Power %	$F_{\Delta H}^{T}$	F_{q}^{T}	DNBR	Exit Quality %	Center Fuel Temperature °F
1	100	1.70	2.82	1.71	subcooled	4000
2	110	1.70	2.82	1.41	2.8	4220
3	125	1.70	2.82	100	8.4	4580
4	100	1.70	3.10	1.56	subcooled	4220
5	110	1.70	3.10	1.28	2.8	4480
6	125	1.70	3.10	0.91	8.4	4880
7	100	1.87	3.10	1.41	2.8	4220
8	110	1.87	3.10	1.10	6.9	4480
9	125	1.87	3.10	0.73	13.0	4880

The following conditions were employed for all cases:

T_{in} = 553.7°F
P = 2220 psia

It should be noted that the present reactor protection system limits the overpower to 112%. In addition, the design values of F_{q}^{T} and $F_{\Delta H}^{T}$ were raised arbitrarily 10% to answer this question. Exceeding these design values is not consist with the present design.

ITEM 2 (Letter 1 Question 6)

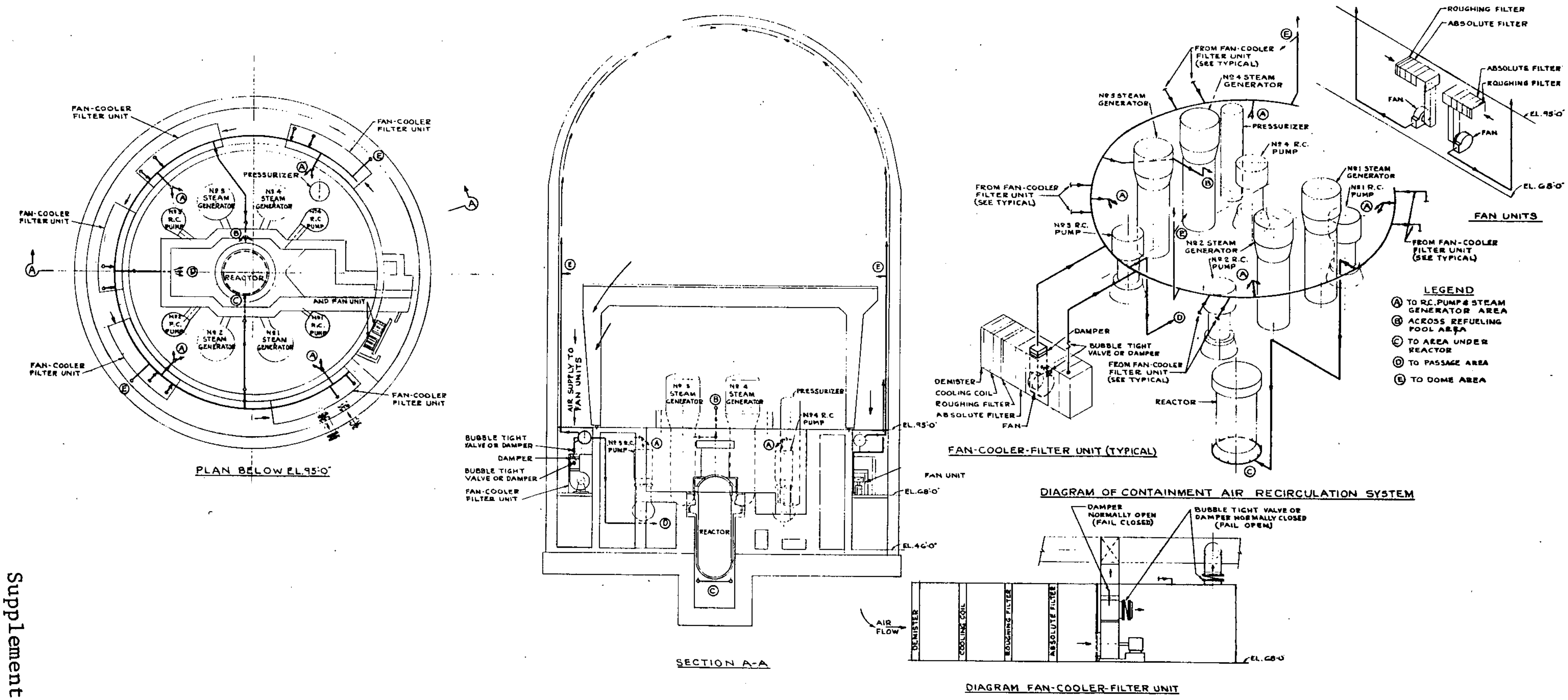
Provide a diagram of your conceptual layout of the internal air recirculation and iodine filtration systems showing the relative location of the input and exhaust ducts, fans, heating and cooling units, demisters, and charcoal filters. State and justify the estimated temperature and relative humidity of the containment atmosphere at each of the above locations for the anticipated conditions following the double-ended rupture of a primary coolant pipe. Describe the systems (including redundancy) provide to prevent ignition of the charcoal filters, and discuss the potential effects on containment pressure and off-site doses if total ignition of the filters is assumed. What experimental evidence can be given to justify the elemental and organic iodine removal efficiencies assumed in the PSAR? What is the basis for the selection of the fraction of organic iodine initially present, and for its growth rate throughout the accident? Also, what fraction of the total gaseous activity is assumed to be present in the fuel gaps?

ANSWER

The recirculation ventilating system consists of five (5) air handling units (four of the five units will be required to operate during an accident) each supplying air to a common distribution header. Branch ducts form this header proportion and direct air to areas below and above the operating floor as required. Figure 6.1 shows the conceptual layout of the system.

Design of electrical equipment located within the containment which is required to operate following a loss-of-coolant accident is discussed under Item 17, Attachment F, Question 5.

No charcoal filters are provided, however space is available for them should they later be deemed necessary.



Supplement 1

DIAGRAM OF CONTAINMENT RECIRCULATION SYSTEM
FIGURE 6-1

2 (1-6)
Page 2

ITEM 2 (Letter 1 Question 8)

The containment spray system is provided as an independent backup to the air recirculation and iodine filtration system. Discuss the experimental basis for the design of the containment spray system and indicate how the pressure reduction and iodine removal values were derived.

ANSWER

The heat transfer model used in predicting the effect of spray on containment pressure is in agreement with the experimental work of Hasson, et al[1] and of Brown[2], who investigated steam condensation on laminar water sheets and water drops generated by spray nozzles. These studies indicate that the bulk temperature of the liquid drop responds very quickly to changes in the surface temperature. In other words, conduction and convection of heat to the interior of the drop will not be a significant resistance to heat transfer.

When the gas film resistance is calculated, with allowance for the effect of a non-condensible component (air), it is shown that the surface temperature of the drop approaches the bulk containment atmospheric temperature during free fall. Since the interior of the drop also reaches this temperature, the heat removal by the drop is determined by a simple heat balance to be the sensible heat required to raise the average drop temperature from that of the spray inlet to that of the surroundings. This is the model employed to predict spray effect in the PSAR.

Information relating to Iodine removal is found in Appendix 6A of Item 14.

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1. D. Hasson, D. Luss, and U. Navon, "An Experimental Study of Steam Condensation on a Laminar Water Sheet," International Journal of Heat and Mass Transfer, Vol. 7, pp. 983-1001.

2. G. Brown, "Heat Transmission by Condensation of Steam on a Spray of Water Drops," Inst. of Mech. Eng. and ASME Proceedings of the General Discussion on Heat Transfer, Sept. 1951, p. 49.

ITEM 2 (Letter 1 Question 10)

Discuss the operation of the emergency diesel power supply system under accident conditions with no normal power sources available. Indicate how the proper equipment is selected for operation (assume failure of one bus or diesel) and how unnecessary loads are dropped to prevent overloading and possible tripping of the remaining diesels.

ANSWER

The answer to this Question is included in the response to Item 17, attachment F, Question 1.0.

ITEM 2 (Letter 1 Question 11)

Provide preliminary accident evaluations to support the results reported in the Preliminary Safety Analysis Report for the startup accident, steam line rupture, refueling, and control rod cluster ejection accident. For example, show such parameters as core reactivity, core temperature, and system pressures plotted against time for the worst condition during core lifetime. Consider the possible generation of curves that relate minimum reactor period to (a) integrated excusion energy and (b) average fuel temperature. These curves should consider cases of hypothetical reactivity insertions considerably greater than that resulting from the ejection of a single control cluster. For each accident, state the potential off-site doses.

ANSWER

Start-up Accident

The startup accident is assumed to occur as a result of an uncontrolled rod cluster withdrawal from a zero power condition. If the reactivity rate is excessive, an abnormally high rate of increase in flux will initiate an alarm in the control room.

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 rod stop - actuated when one of four power range channels indicate a power level above the preset value.

f. 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.

Steam Line Break

For analysis of steam line break see Item 16, Attachment E.

Rod Ejection Accident

In order to assume a rapid ejection of a control rod cluster from the core, a hypothetical rupture of the control rod mechanism housing must be postulated. For this accident, it will be demonstrated that the effects are definitely localized and that there is no resulting pressure surge that could further damage the reactor coolant system. The resultant 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. The amount of fuel damage that could result from such an accident is governed by the peak power attained in the transient which in turn depends on the worth of the ejected rod and the power distribution attained with the remaining control rod pattern.

The results of a rod ejection analysis are a function of the detailed physics design including the core enrichment pattern, the grouping of the control rods, their radial location and their axial location as a function of load. During the evaluation of this accident, all normal operating conditions are studied to determine the most severe condition including tolerances for instrumentation errors, reactivity coefficients etc. A key consideration in the detailed core physics design is the specification of the combination of parameters that will result in a calculation of ejected rod worth and power peaking due to the ejected rod that would limit the maximum fuel temperature to a value that would preclude any additional damage to the reactor coolant system.

The accident offers the potential for considerable core damage to the extent of possible gross fuel melting, dispersion of molten UO_2 in the coolant and a subsequent rapid pressure surge or shock. The reactor will be designed and operated so that this accident will not cause further rupture of the reactor coolant system. For purposes of this analysis fuel dispersion is assumed to occur when the average fuel temperature exceeds the UO_2 melting temperature. If there is any likelihood of this dispersal, a detailed analysis is made to demonstrate that the rapid dissipation of this energy into the water does not create a significant pressure surge to jeopardize either the integrity of the vessel or that of the adjacent fuel.

The core design for the Indian Point Unit #2 does not represent a significant departure from the San Onofre design for which detailed rod ejection Analyses have been made. The resultant ejected rod worth and peaking factors for the San Onofre design were such that in the worst case there was not even center rod melting. The maximum hot spot fuel center temperature only reached 4060°F. This corresponds to a maximum inserted reactivity of 0.8%δk and a heat flux hot channel factor of almost 10. On the basis of the San Onofre analysis it is expected that the criteria as stated for the Indian Point Unit #3 can be met with a high degree of confidence.

The effect of the additional heat transfer to the reactor coolant system during this transient is evaluated to determine the maximum system pressure including the effects of fuel dispersal in the coolant, if necessary. For the San Onofre Plant there was not even center melting and thus no danger of molten fuel being expelled. The maximum calculated pressure resulting from the rod ejection transient was 2316 psia.

Refueling Accident

During refueling, the primary coolant is borated sufficiently to yield k_{eff} = 0.9 for cold conditions with all rods in. This concentration is also sufficient to prevent criticality even with all control rods removed. Continuous mixing will be maintained through the reactor vessel by utilizing a residual heat removal loop. During this period, the neutron sources installed in the core and two separate B_{10} detectors with audible count rates provide direct monitoring of the core. Any appreciable increase in the neutron source multiplication including that caused by the maximum physical boron dilution rate (approximately 580 ppm per hour) is slow enough to give ample time to take corrective action i.e. turn off the makeup water pumps which are the only potential sources of clean unborated water. These pumps are tagged and locked out of service during the refueling operation so that dilution cannot occur.

Of the four accidents discussed above, only the rod ejection and the steam line rupture accidents provide any potential off-site exposures. The startup

and refueling accidents do not provide any potential for off-site exposure as they do not result in the release of radioactivity from the fuel. Criteria for the permissible activity in the secondary coolant and the potential off-site exposure from this accident are detailed in answer to Question 19d of the IPP #2 PSAR, Amendment 1, Docket No. 50-247.

The rod ejection accident is a loss-of-coolant accident with a small equivalent break area. The fission product release to the containment is dependent upon the detailed results of the analysis which will determine the extent of fuel dousage due to the ejected rod and associated reactivity transient. Quantative results for this accident are not available at this time as the evaluation cannot be performed until the final plant and core design are completed.

It can be stated, however, that the potential off-site exposures will be lower than those calculated for the hypothetical loss-of-coolant accident and presented in Chapter 12 of the PSAR. The missile shielding provided above the reactor vessel will stop the ejected control rod pressure housing and thereby maintain containment integrity. The rise in containment pressure due to reactor coolant blowdown will cause containment isolation and operation of the isolation valve seal water system as well as switching of the containment air recirculation system to the post-accident condition. Safety injection and reactor trip also would be initiated because of the low pressurizer level and pressure conditions.

Isolation of the containment in conjunction with the penetration and weld channel pressurization system will prevent release of fission products to the environment. Reactor trip and safety injection will terminate the accident and prevent further release of fission products to the containment and the air recirculation cooling System will reduce the containment pressure.

Rod Drop Protection

Should an RCCA fall into the core, the immediate effect is a reduction in reactor power and an increase in DNB margins. However, if power is restored to its initial value, either by negative moderator coefficient effect or by automatic control rod withdrawal, margin to DNB will be reduced because of the adverse power distribution. The rod drop protection system prevents return to a power level which might result in a DNB ratio of 1.30. The system designed in accordance with IEEE 279, consists of redundant means of detecting a dropped rod, redundant rod stops, and redundant means of reducing steam load.

Two independent systems are used to sense a rod drop:

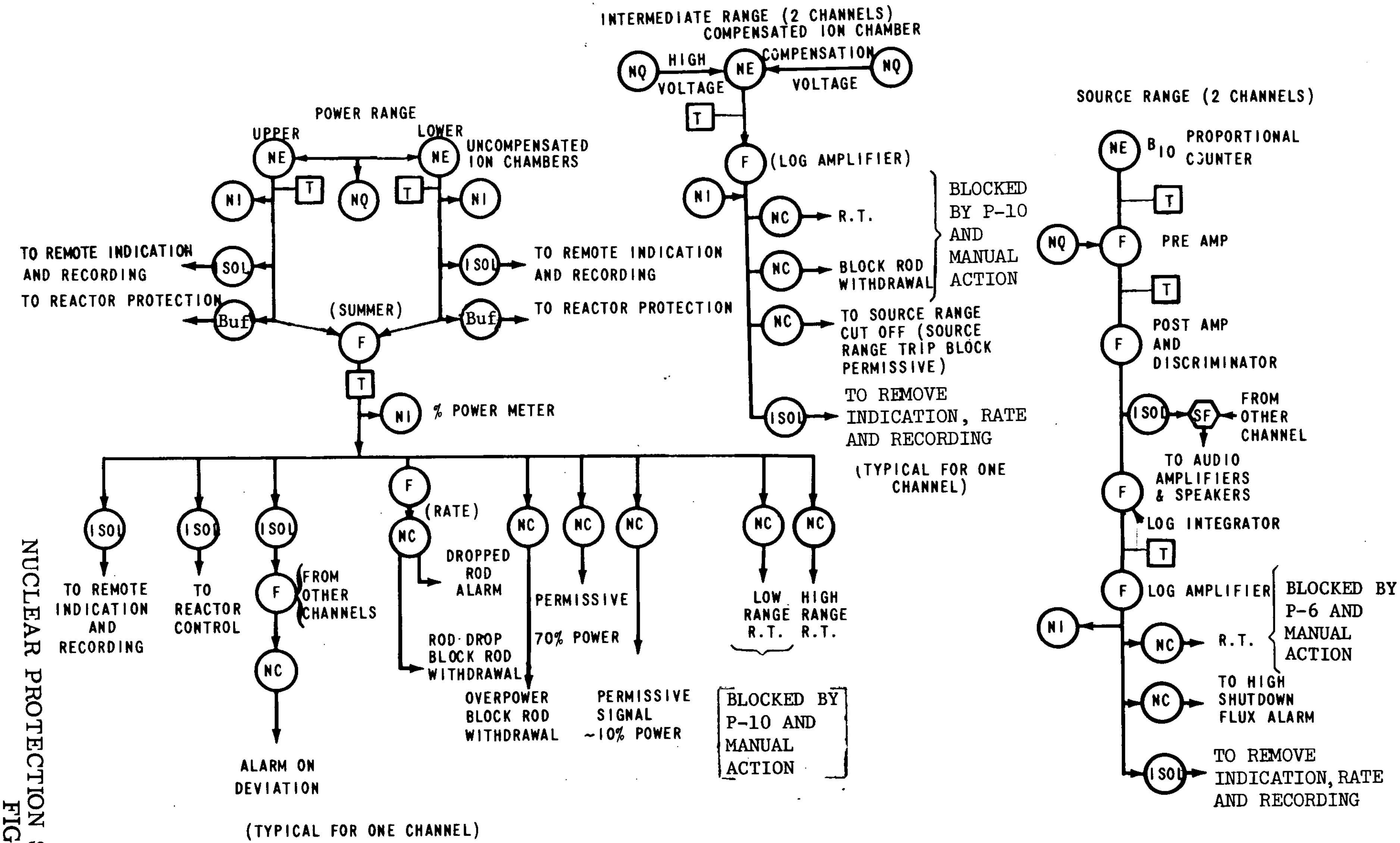
(1) Nuclear Instrumentation System - Each of the four power range nuclear channels contains a rod drop detection circuit, any one of which will initiate automatic action when a rapid decrease in ion chamber current occurs. This circuit consists basically of a comparison of each power range nuclear signal with the same signal taken through a first order log network. The sudden decrease in ion chamber current as a result of a rod drop 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. It will be demonstrated during initial startup that, when near full power, drop of any rod which causes a significant increase in hot channel factors will be detected on at least one nuclear channel.

(2) Rod Position Indication System - Any rod bottom signal from the rod position indication system will initiate protective action.

Figure 1 indicates schematically the dropped rod alarm and the Nuclear Protection System in general.

Both the rod stop and the turbine runback are redundant. Rod stop contacts are located in the rod control logic cabinet and in the rod speed control analog rack. The turbine runback is accomplished by reducing the turbine load limit to a preset value. This lowers the turbine control value position to that corresponding to the desired steam load. Two independent load limit reduction circuits, each including a load limit motor, are provided for redundancy.

The amount of the runback is to be determined by physics tests of dropped rod worths and hot channel factors during plant startup tests. The safety requirement of the runback is to preclude return to a power level that might result in core damage because of adverse hot channel factors. It is expected that the startup tests will show that dropped rod hot channel factors will not cause a DNBR less than 1.30 even at full power, and that the runback will be set for operational requirements. That is, the automatic load reduction would be large enough such that, with reasonable operator action, an orderly manual plant shutdown can be accomplished rather than a reactor trip on low pressurizer pressure.



NUCLEAR PROTECTION SYSTEM
FIGURE 1

ITEM 2 (Letter 1 Question 12)

The borated safety injection water may be diluted by the non-borated primary coolant or secondary system water following a major pipe failure. Analyze the consequences of adding diluted safety injection water to the reactor assuming several dilution factors, and provide the corresponding periods and energy releases if the control rods are and are not assumed to be inserted.

ANSWER

The borated safety injection water will be diluted to some extent by the spilled reactor coolant following a major pipe failure. The dilution of this water by spilled water from a simultaneous rupture of the secondary system is not considered credible; missile protection is provided for the steam generators and the support of these units is such as to accept reaction forces resulting from the complete circumferential severance of a reactor coolant pipe or steam line.

During safety injection, water is injected from the accumulators, the boric acid injection tank and the refueling water storage tank. Each of these sources contain boron at significantly higher concentrations than will exist in the primary coolant at any time during the core cycle. Primary fluid is released to the containment environment through a primary circuit failure in the form of both vapor and water. The spilled water runs to the containment floor and eventually to the containment sumps via the containment drain ducts. At the start of injection borated water is injected into the primary circuit where it mixes with the remaining primary water and subsequently spills to the containment floor via the original circuit breach. Also, during the injection phase borated water is sprayed into the containment where it condenses steam as it rains down onto the containment floor.

Recirculation will not start until a substantial quantity of the refueling water has been added; it is reasonable to assume that substantial mixing will take place before the water is returned to the core. Simple calculations

show that approximately 5% by weight of the injected water is required to mix with the reactor coolant spilled water in order to provide a 1% shutdown margin with the most reactive rod stuck out. Similarly 10% by weight of the injected water is required to mix with the reactor coolant water in order to provide a 1% shutdown margin with all the rods stuck out.

ITEM 2 (Letter 1, QUESTION 13)

The steam generators provide the primary mechanism for dissipation of primary system heat in the event of complete loss of power. Indicate the water sources and capacity available to the steam generators under these conditions. Discuss how this water can be delivered, and how long the reactor can be safely cooled by these sources.

ANSWER

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 heat 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 any steam generator. This system supplies 800 gpm of feedwater (200 gpm to each steam generator).

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 ring 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 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.

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.

ITEM 2 (Letter 1 Question 14)

Provide the following information regarding the proposed instrumentation system:

b. Discuss how the cluster control system has been designed so that rod insertion time is not delayed as a result of pressure gradients generated by potential blowdown forces.

c. Discuss how the position of critical isolation valves will be indicated in the control room.

d. A rupture of the tap feeding two of the three pressurizer-low-level channels can remove the intended automatic protection provided by this circuit. Please justify this proposed design.

e. Provide a list of all monitors that will be provided to indicate the reactivity status of the reactor, and the pressure, temperature and humidity conditions inside the containment after the MCA. Discuss the design lifetime of the critical components associated with this equipment when operated in the post-MCA containment environment.

f. Provide experimental evidence to indicate the sensitivity of the external ion chambers to changes in the axial and radial flux distribution. Relate this information to internal monitor readings, if possible.

ANSWER

b. The effect of hydraulic forces on rod cluster control insertion time during a loss-of-coolant accident is considered. Any delay in insertion time will be considered in establishing the time to achieve reactor shutdown, and the core fission energy generated prior to this shutdown will be considered in evaluating the energy releases to the containment under loss-of-coolant accident conditions.

c. All remote and automatically operated containment stop valves will have control switches and position indicating lights in the main control room.

Manual stop valves will be located to assure accessibility following the hypothetical accident and will be so designed to allow easy determination of valve position by local inspection. Manual valves which must be closed during power operation will be locked, closed and tagged. Opening of these valves during power operation may be performed only

under the administrative control of the licensed operator, and an operating staff member must remain in the area during the period that the valve is open.

d. The three pressurizer level transmitters will have three individual taps (3 sets).

e. The following instrumentation insures broad coverage of the effective operation of the safeguard system:

(1) Containment Pressure

Three channels, monitoring containment pressure, reflect the effectiveness of containment spray, and reactor cooling, in that high pressure indicates high temperatures and reduced pressure indicates reduced temperatures. The containment pressure transducers are outside the containment and are therefore not subject to the MCA environment.

(2) Containment Sump Level

Redundant containment sump level indicators will show that water has been delivered to the containment in the early stages of post MCA and, in the later stages, will show that the recirculation pumps are effective in recircualtion by maintaining the sump level. These transmitters will be designed to withstand MCA conditions.

(3) Refueling Water Storage Tank Level

These redundant (two) channels indicate that safety injection and spray have removed water from the storage tank. They are outside of the reactor containment.

(4) Safety Injection Pumps Dishcarge Pressure

This channel will clearly show that the safety injection pumps are operating. The transmitter is outside the containment.

(5) Pump Energization

All pumps will have indicator lights on the control board indicating closure of the motor feeded breakers or starters.

(6) Valve Position

All remote operated engineered safeguards valves will have limit switches and associated indicator lights on the control board to show proper positioning of the valves.

The valve position sensors on valves which must function in accident situations will be designed to withstand MCA conditions. The valve position sensors in combination with the valve and valve operator will be tested under pressure in a high temperature and humidity environment as experienced under MCA conditions.

(7) Residual Heat Exchangers

Combined exit flow is indicated to monitor operation of the residual heat exchangers. These transmitters will be designed to withstand MCA conditions.

(8) Air Coolers

The supply flow and exit temperature of each of the five coolers are alarmed in the control room if the flow is low or if the temperature is high. 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 locally by manually valving each one out in turn.

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

a. Residual heat removal pumps discharge pressure
b. Residual heat exchanger combined exit temperature
c. Containment spray test lines flow
d. Safety injection test line pressure and flow

e. Westinghouse had made a number of tests during the past four years to determine the extent to which out-of-core neutron detectors can depict conditions within the core. The conclusion from these studies was that a set of 4 long ionization chambers, each approximately equal in length to the core height, would provide the best means using external detectors both to measure average core power and to detect flux tilts. Each detector would be located opposite the "corner" of one quandrant and the internal construction of the chambers would be of two divided sections of equal length. The total current from the two sections would be used in the reactor protection system as giving the best measure of average core power, and the individual section currents would be used for detection of flux tilts.

The first of these studies was a tabulation of calibration corrections required during the operation of Core I at Yankee (Rowe). Where rod motions or other reasons caused the nuclear instrumentation detector readings to drift by more than 2 or 3%, recalibrations were made.

Plotting the corrections resulted in graphs which showed changes in the detector readings versus control rod group positions for all rod groups used during the life of Core I. Measurements of the flux distribution changes in the detector wells for detectors located opposite the bottom half and top half of the core were also compared with flux wire measurements made within the core, particularly in fuel assemblies nearest to the external detectors.

Tests have been performed on operating plants (SENA, San Onofre and Connecticut Yankee) to verify the ability of the out-of-core instrumentation to represent gross core power distribution. Results of these tests are reported in a topical report to be submitted to the AEC in September 1968. A one-dimensional (axial) power distribution procedure analysis has been compared to results obtained in an experiment performed at Haddam Neck. The results of the measured and calculated power distributions are given in the topical report to be submitted in September 1968.

In order to verify the ability of out of core nuclear detectors to represent gross core power distribution experimental data have been obtained from operating power plants such as SENA, San Onofre and Connecticut Yankee. Measurements were performed under various operating conditions such as boron dilution, load variations and single rod insertions to study the correlation between in-core power as measured by in-core instrumentation and out of core detectors in both axial and radial directions.

I. Axial Measurements

Measurements were performed at SENA, San Onofre and Connecticut Yankee to establish a quantitative relationship between in-core power distribution and out of core detector readings. In-core measurements at SENA and San Onofre were made with an aeroball flux mapping system and at Connecticut Yankee with a movable detector system. Out of core measurements at SENA and San Onofre were made with long ion chambers and at Connecticut Yankee with short ion chambers.

The parameter used to measure unbalance of power between upper and lower core halves is Symmetric Offset (SO). It is defined as the axial integral of power in the upper half of the core minus the integral in the lower half normalized to the total axial integral. In-core symmetric offset was derived from in-core power distribution as calculated by the in-core code. This code derives core power distribution based upon measured in-core activation rates. Out of core symmetric offset was calculated using ion chamber currents. Any sensitivity or configuration differences between upper and lower detectors were accounted for by normalizing the detector readings to in-core measurements under all rods out condition. Where long ion chambers were used each section of the chamber effectively integrated the flux over half core height.

Experimental results obtained from the SENA reactor for a typical long ion chamber shows that a linear correlation exists between axial unbalance in in-core power averaged over the core and out of core ion chamber readings. Out of core offset was calibrated to equal the in-core offset by multiplying the detector offset by a gain constant such that the offsets are equal.

Similar results were obtained from a long ion chamber at San Onofre. Again a linear relationship between out of core detector readings and in-core axial power distribution was demonstrated.

Data obtained from Connecticut Yankee during a xenon transient test are presented in the topical report to be released to the AEC in September 1968. Out of core detector offsets are compared with measured in-core offset as a function of time during a xenon transient test.

II. Radial Measurements

The ability of out of core measurements to detect radial power tilts in the core was verified at San Onofre during special rod insertion tests. Single control rods were inserted into the core to cause a tilt in core power distribution. Out of core detector signals were normalized to read the same power level as the core under normal operating conditions. With single rods inserted, measurements were made for in-core power distribution with the aeroball system and out of core detector readings.

The results obtained for tilts in the various quadrants (ratio of power in a quadrant to the average) are shown in the topical report to be released to the AEC in September 1968. Plant chambers respond nearly identical to core power distribution. Locations of various rods used to cause tilts are identified in the figure.

The experimental data were also used to study the relationship between radial hot channel factors to the ratio of maximum to average detector readings. With the exception of data for center rod where a diametral power tilt is not induced, the correlation between out of core response and fractional change in hot channel factor is linear.

ITEM 2 (Letter 1 Question 18)

Provide the anticipated pressure-flow characteristics for the safety injection and the charging pumps.

ANSWER

Safety Injection Pumps

The safety injection pumps are the horizontal-centrifugal type. The following pressure-flow characteristics have been specified:

Design Flow	400 gpm
Design Head	2500 ft
Shut-off Head	3500 ft
Maximum Flow	650 gpm
Number of Pumps	3

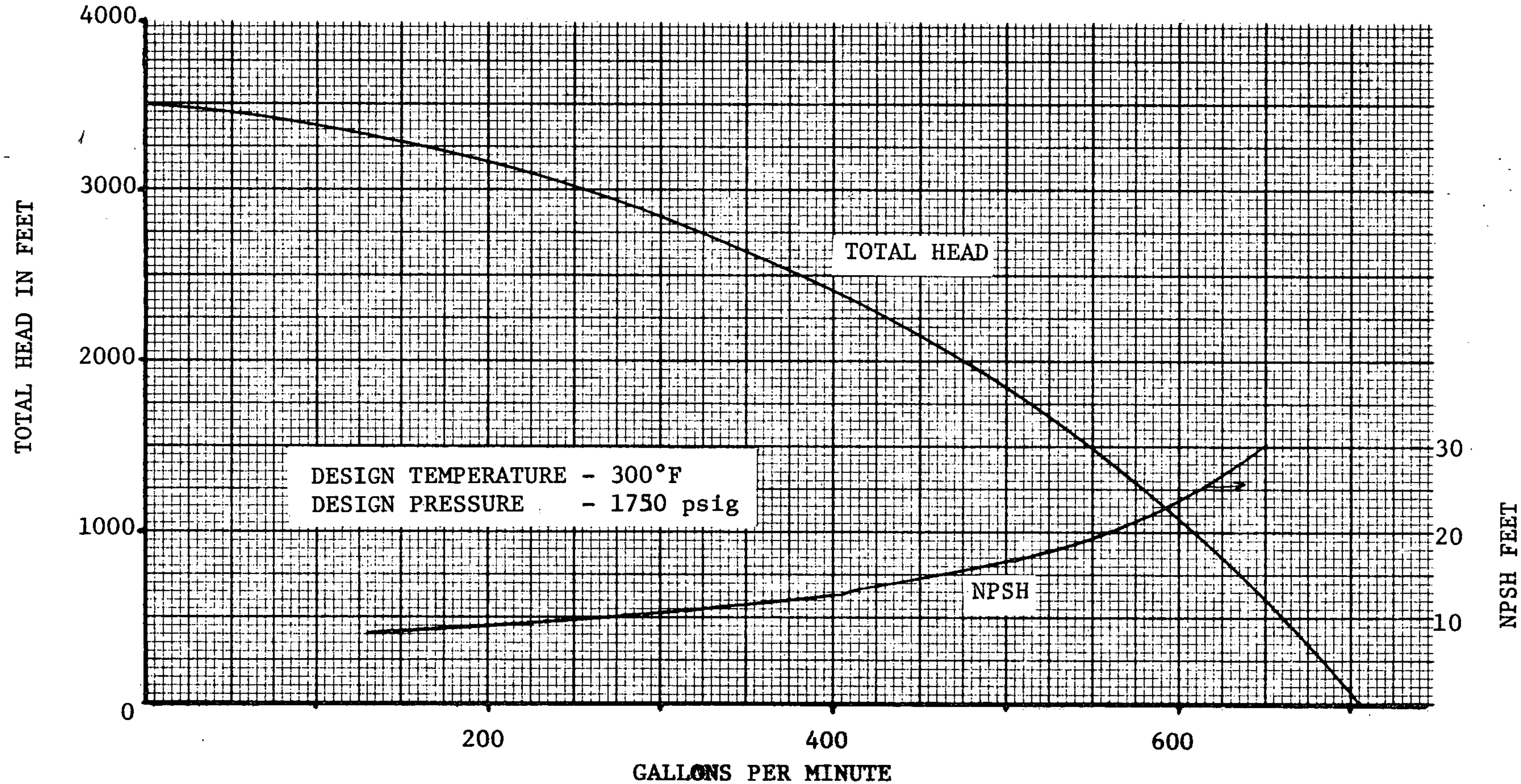
The anticipated head versus flow characteristic curve is attached.

Charging Pumps

The charging pumps are the postive displacement type with variable speed drive having the following preliminary characteristics:

Design Flow	100 gpm
Normal Head	2500 psi
Number of Pumps	3

SAFETY INJECTION PUMP PERFORMANCE



SAFETY INJECTION PUMP PERFORMANCE

Supplement 1

ITEM 2 (Letter 1 Question 19i)

State proposed design criteria, justification how these criteria will be fulfilled by the design proposed, and (where applicable) test methods:

i. To limit core drop if the upper support fails. Explain how the "in-core" instrumentation structure will be designed to limit the core drop. What are the consequences of the maximum potential reactivity insertion under these conditions?

ANSWER

Because of the care taken in the design and fabrication and the relatively low stresses imposed during normal operation, the failure of the core support flange and the subsequent drop of the core is considered to be incredible. If such an accident is postulated, the downward vertical displacement of the internals will be limited by the in-core instrumentation structure and energy absorbing devices attached to the instrumentation tie plate at locations concentric with the 4" diameter instrumentation guide tubes. In the event of this accident, the energy absorbers would contact the vessel bottom head. The load would transfer from the vessel to the energy device, through the instrumentation tie plate directly to the instrument guide tubes above the point of load application, and indirectly to other instrument guide tubes. The guide tubes, bolted to the instrumentation tie plate at their lower end, are supported from the underside of the core support forging at their upper end.

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

The free fall in the hot condition will be on the order of 1/2 inch, and there will be an additional strain displacement in the energy absorbing devices of approximately 3/4 inch. In the cold condition the free fall will be on the order of 1 inch and attain an additional strain displacement of approximately 3/4 inch.

In summary, the design criteria are:

1) To limit the downward vertical displacement to a safe distance.

2) To limit the imposed forces so that the reactor vessel is within ASME Section III Code Standards and that energy absorbing devices in the internals be used to absorb the dynamic energy of the internals drop.

In the event of this vertical displacement, core alignment will be maintained by the radial supports which are described in Chapter 3 of the Preliminary Facility Description and Safety Analysis Report. Control rods will insert under this condition.

The reactivity addition due to such a 1-1/4 inch drop with the rods in their most reactive position is of the order of 0.2 to 0.3% δk and is well within the capability of the reactor protection system to safely shut down the reactor.

Letter 2

Item 2 (Letter 2 Question 4)

A potential source of hydrogen following a primary piping failure could be radiolytic decomposition of the safety injection water initiated by the decay energy of the core. Please discuss the magnitude of the gases formed by this process and its potential effect on containment pressure and concentration of free hydrogen in the containment vessel.

ANSWER

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.[(1)] No thermal recombination occurs in solution; i.e. H_2 produced is quantitatively transported to the containment atmosphere.

[(1)] S. Gordon and E. H. Hart, Proc. Second Geneva Conf., 29 13 (1958); also R. G. Sowden, J. Nuc. Mat'ls 8 81-101 (1963).

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 yeild 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 1. The pressure effect of these gases is given in Figure 2, where the containment atmosphere is assumed to be cooled to its pre-accident temperature. Figure 3 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, a 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 pr to provide a recombiner system capable of sustaining a comcentration well below a flammable limit indefinitely.

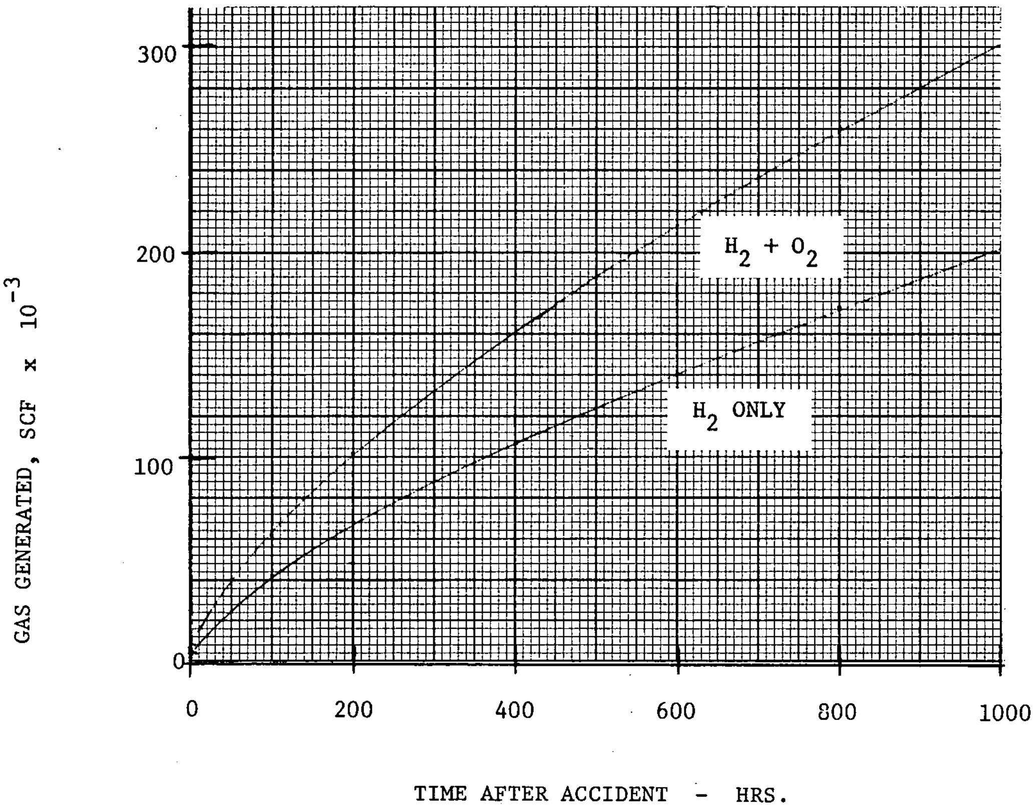


FIGURE 1

SUPPLEMENT 1

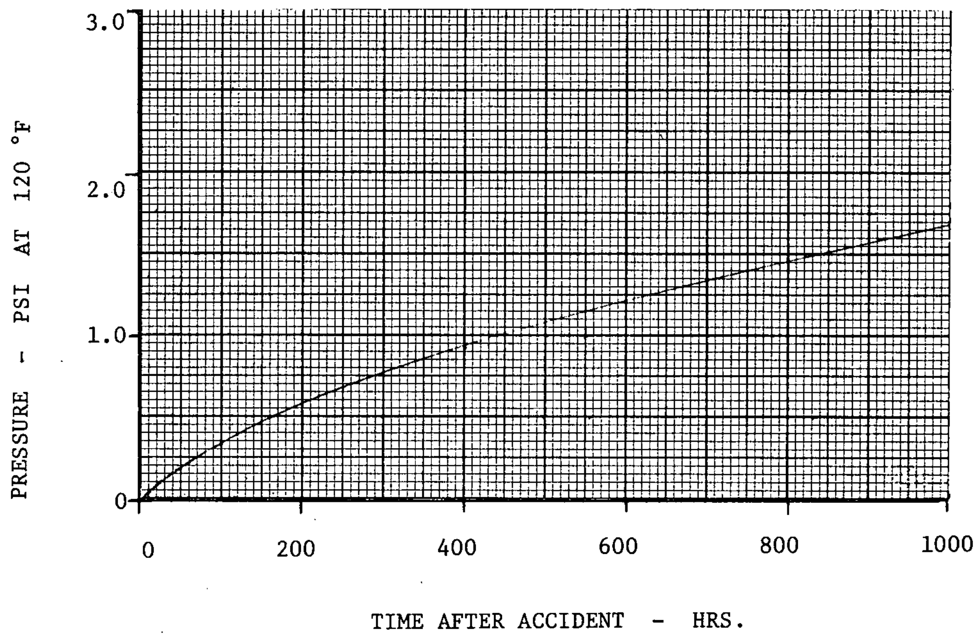


FIGURE 2

SUPPLEMENT 1

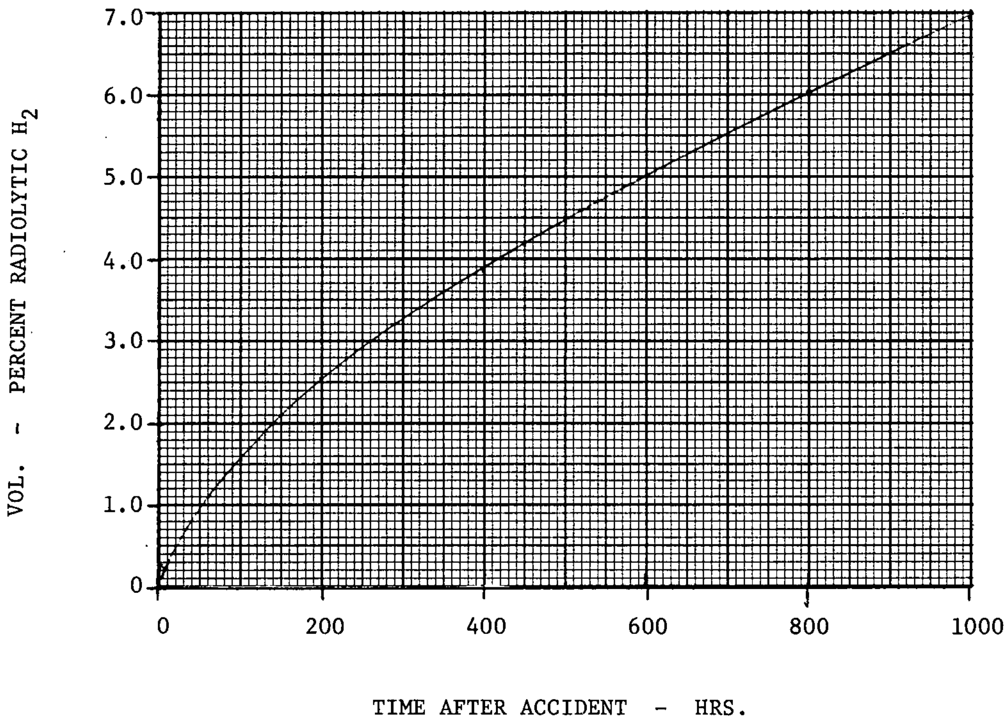


FIGURE 3

SUPPLEMENT 1

ITEM 2 (LETTER 2 QUESTION 5)

Provide an analysis of the primary system pressure, temperature, and power level transients that would result from a failure of a steam generator tube. State the amount of primary system radioactivity that could be leaked to the secondary system and to the atmosphere under these conditions. Discuss those design features of the secondary system which would limit the release of radioactivity to the atmosphere for this accident condition. How many tube failures can be tolerated before exceeding the pressure relief capacity of the secondary system safety valves?

ANSWER

In the event of a rupture of a single steam generator tube, the released reactor coolant radioactivity will be maintained essentially within the secondary and containment systems. Leakage to the environment will not exceed the dose limits of 10 CFR 20. The rupture of a single tube will not result in physical damage to equipment. The most adverse consequence of this incident would be the secondary system cleanup to remove the slightly radioactive, borated reactor coolant.

PROVISIONS TO HANDLE STEAM GENERATOR TUBE RUPTURE

Following a tube rupture, the gaseous activity transmitted to the main condensers through the steam line will cause the air ejector radiation monitor to alarm. The radiation monitor will divert the air ejector discharge into the containment. The design of the system will be such that the gaseous activity released to the environment prior to diverting the air ejector to the containment will not exceed the amount required to meet the 0.5 rem whole body dose criterion (about 13,500 curies of Xe-133). Flows to the atmosphere from the gland seal condenser amount to a small fraction of the total dose when integrated over the entire cooldown period of approximately four hours - assuming no isolation of the defective steam generator.

The steam dump which accompanies reactor trip is entirely to the turbine condensers. The steam dump valve capacity of approximately 40 per cent of full load flow is adequate to prevent popping of steam generator safety valves and dumping of radioactive material to the atmosphere.

Non-return valves in each main steam line permit isolation of individual steam generators once the primary pressure falls below the steam generator safety valve set pressure of 1100 psia. The increasing steam generator water level after trip, and the blowdown radiation monitor will indicate the faulty steam generator requiring isolation. With isolation of the faulty steam generator, the contamination of secondary plant components is limited.

In the event that the faulty steam generator is not isolated, the condensible fluids from the reactor coolant system will be stored in the main condensers. Emergency storage capacity within the condensers up to the lower air off-take connection amounts to 12,840 cubic feet, approximately double the maximum calculated discharge through both ends of a single broken tube over the entire cooldown period.

SHORT-TERM POWER, PRESSURE AND TEMPERATURE TRANSIENTS

Figure 5-1 shows the power level, temperature (coolant T_{avg}) and pressure transients associated with a single tube rupture. The analysis was performed using the following assumptions:

1. The faulty tube was assumed severed just above the tube sheet. Hot leg coolant discharge through a tube length equivalent to the tube sheet thickness. Cold leg fluid discharged through the entire length of a U-tube.

2. The calculation was performed on the basis of an uncontrolled incident. Only the automatic reactor trip points (high power, low pressure, safety injection) were considered, and no isolation of the faulty unit using the steam line non-return valves was assumed.

3. The effect of secondary side steam dump to the condenser following reactor trip was included.

4. The effect of charging flow was ignored.

This problem was analyzed on a detailed digital simulation model for a PWR plant. The computer model, computes time and space-dependent variables such as average channel fuel temperature, coolant temperatures in the primary loop and in the pressurizer, and shell side steam generator fluid conditions. Masses, energy, and state equations are satisfied for all coolant sections. Pressure, surge rates, and core reactivity are computed from coolant density changes. The standard six-delay group neutron kinetics equations are solved for a point core. Residual heat generation is included. Flow through the ruptured tube is assumed to be proportional to reactor coolant pressure for critical flow and proportional to the square root of the pressure differential for subcritical flow.

The reactor transient curves presented are for a slightly negative moderator coefficient. The power level remained essentially constant. Reactor trip is the result of low reactor coolant pressure and occurs at approximately seven minutes after the rupture. This case is representative of adverse conditions in which there is no operator action. For any value of the moderator coefficient, whether in its positive or negative range, the reactivity insertion rate is less than that incurred for a design load change. The reactor control system is therefore capable of maintaining essentially constant temperature and power until the trip is reached.

REACTOR COOLANT DISCHARGE TO SECONDARY PLANT

Figure 5-2 shows the transient pressure and integrated flow through a ruptured tube over the reactor cooldown period.

1. The initial flow through the break is at a maximum prior to reactor trip since the maximum pressure differential between primary and secondary sides of the steam generator exist at this time.

2. Pressure falls in the pressurizer as the charging pumps are initially unable to match the critical flow condition and relatively high reactor coolant through the break. The high flow condition and relatively high reactor coolant pressure are maintained during the period before trip as the pressurizer heaters attempt to maintain pressure in spite of a falling water level.

3. Reactor trip and turbine trip occur automatically on a low pressure signal about 436 seconds or seven minutes after the rupture. Automatic dump of secondary steam to the main condensers prevents popping of the main steam safety valves. Pressurizer level and reactor coolant pressure drop as the reactor coolant temperature is automatically reduced to the no-load average temperature by the automatic steam dump.

4. Safety injection is initiated just after reactor trip upon coincident pressurizer low pressure and low level signals. Operation of the high head safety injection pumps causes pressure to equilibriate at approximately 1200 psia and causes the water level to rise back into the pressurizer. Flow through the tube rupture stabilizes during this period as the steam dump system automatically maintains coolant temperature (and hence secondary steam pressure) at the nominal no-load value.

5. Cooldown of the plant is begun at 50°F/hr approximately twenty minutes after reactor trip. Heat is removed by controlled steam dump to the main condensers. During this period, a constant pressure margin of approximately 200 psi is maintained above saturation (and secondary side) pressure, to allow operation of the reactor coolant pumps during the cooldown period. Pressurizer water level is maintained by operation of a safety injection pump. The flow through the break remains essentially constant over this period. Figure 5-2 shows the transient assuming that the faulty steam generator has not been isolated.

6. It is expected that the unit will be isolated by a steam line non-return valve once the reactor coolant pressure is reduced below 1100 psia. In this event, the mass flow into the secondary system will terminate and no reactor coolant flow to the main condensers will occur after that time.

7. Should the isolation of the unit not take place, the entire secondary steam system will be isolated by closing off steam dump once the residual heat removal loop is placed in service about 4 1/2 hours after the rupture. At this time approximately 5760 ft^3 of reactor coolant will have been discharged to the secondary side.

RADIOACTIVITY RELEASE TO SECONDARY PLANT

The radioactivity release to the secondary plant has been evaluated consistent with the single tube rupture and failure to isolate the faulty steam generator and 1 per cent fuel defects. As shown in Figure 5-2, the total discharge from the reactor coolant to the secondary plant for these conditions is 5760 ft^3 or 163×10^6 cm^3. The following tabulation lists the fission product concentrations for one per cent defective fuel and the resultant discharge to the secondary plant under the most adverse condition.

PRIMARY COOLANT FISSION PRODUCT ACTIVITIES

Reactor Coolant Fission Product Concentration	Resultant Radioactivity Release to Secondary Plant
Xe-133 = 210. $\mu c/cm^3$	34,200 curies
Xe-135 = 8.8	1420
I-131 = 1.75	286
I-132 = 0.6	97.8
I-133 = 1.4	228
I-134 = 0.36	58.8
I-135 = 1.4	228
Cs-137 = .98	160
Cs-134 = 0.25	40.7
Kr-85 = 4.4	717

The activity in the turbine building from the above sources will not present tolerance problems for personnel during the accident and the subsequent plant cleanup.

RELIEF CAPACITY FOR A MULTIPLE TUBE FAILURE

Although the rupture of more than a single steam generator tube is considered improbable, an analysis has been made of the capacity of the main steam safety valves to pass the flow of a rupture of more than one steam generator tube.

The steam generator valves are sized to pass 3.215 x 10^6 lb/hr of steam for each steam generator. These valves can pass a water flow of approximately 4300 lb/second at 1100 psia saturation conditions. Considering a maximum flow severance of 80.5 lb/sec, the safety valves on any one steam line can handle the water flow from approximately 50 broken tubes.

SUMMARY

In summary,

1. The severance of a single steam generator tube will cause no significant power increase and will result in no core damage or fission product release.

2. The plant is designed to contain the effluent from the ruptured steam generator tube within the secondary and containment systems such that the release to surroundings with proper functioning of the equipment will not exceed the limits 10 CFR 20.

3. Even in the event of failure of equipment intended to contain the coolant activity release, the discharge of the entire inventory of coolant radioactivity will not cause off-site exposures in excess of 10 CFR 100 limits.

4. In the event of failure to isolate the faulty steam generator using a main steam non-return valve, the entire quantity of coolant lost through the ruptured tube is contained within the secondary system and no physical damage to equipment results. The only consideration is essentially one of cleaning up the secondary system.

5. The main steam safety valves in one main steam line have the capacity to pass water flow equivalent to the maximum through the double-ended severance of approximately 50 steam generator tubes. This condition is considered a highly improbable event and is not a basis for plant design.

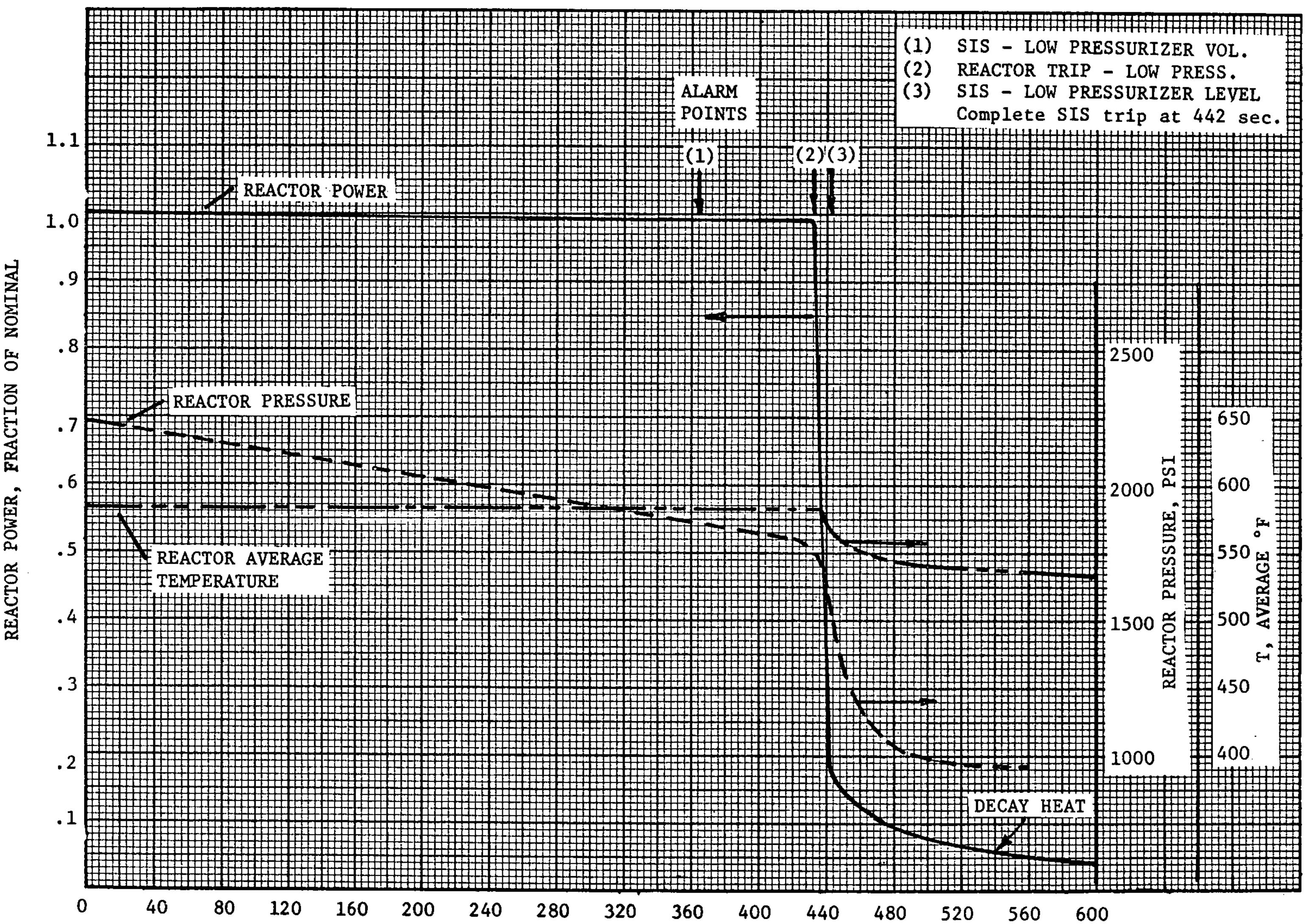

(1) SIS - LOW PRESSURIZER VOL.
(2) REACTOR TRIP - LOW PRESS.
(3) SIS - LOW PRESSURIZER LEVEL
Complete SIS trip at 442 sec.
ALARM POINTS
(1)
(2)
(3)
REACTOR POWER
REACTOR PRESSURE
REACTOR AVERAGE TEMPERATURE
DECAY HEAT
REACTOR POWER, FRACTION OF NOMINAL
1.1
1.0
.9
.8
.7
.6
.5
.4
.3
.2
.1
REACTOR PRESSURE, PSI
2500
2000
1500
1000
T, AVERAGE °F
650
600
550
500
450
400
0
40
80
120
160
200
240
280
320
360
400
440
480
520
560
600
TIME AFTER BREAK, SEC

FIGURE 5-1

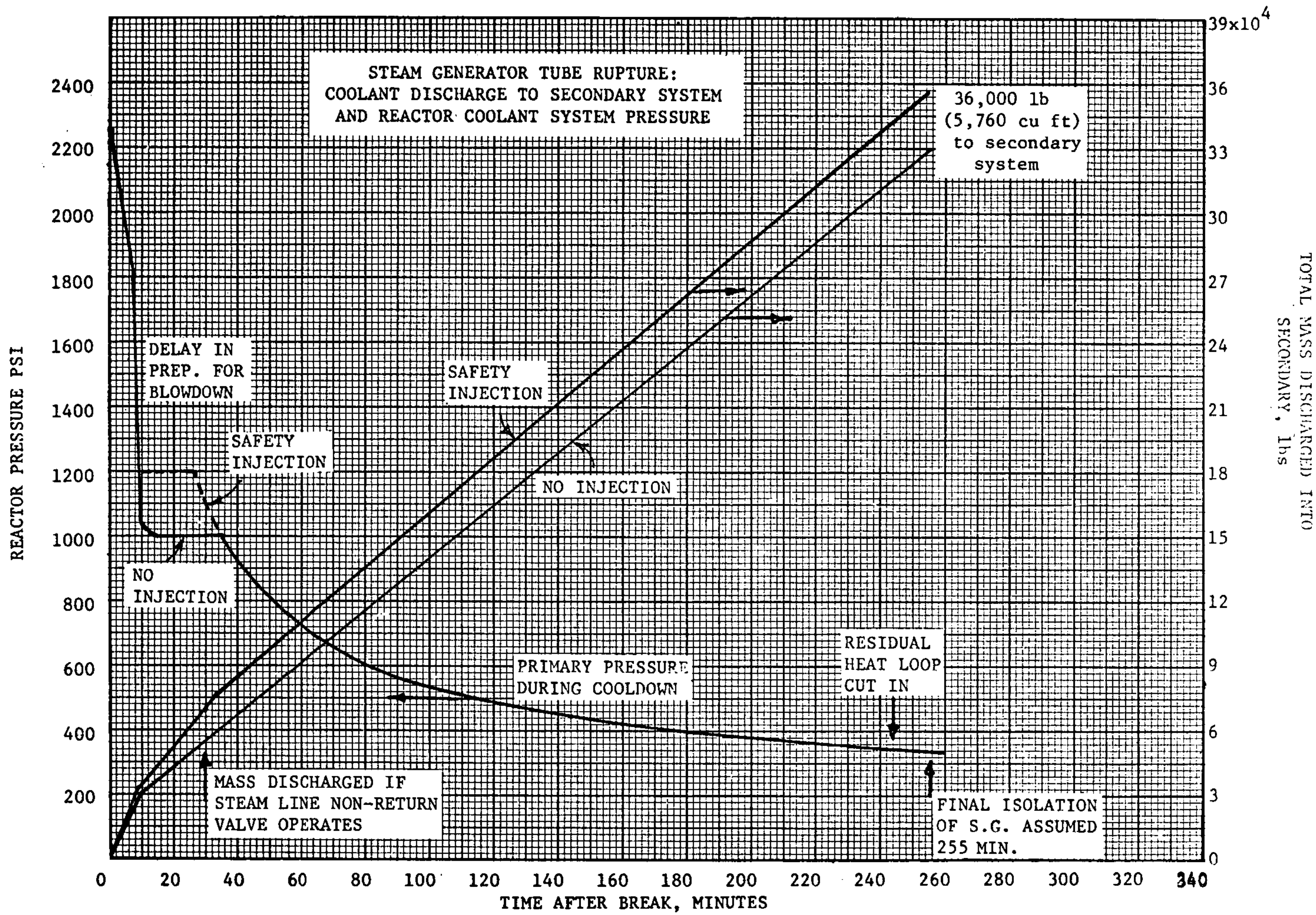

STEAM GENERATOR TUBE RUPTURE:
COOLANT DISCHARGE TO SECONDARY SYSTEM
AND REACTOR COOLANT SYSTEM PRESSURE
36,000 lb
(5,760 cu ft)
to secondary
system
DELAY IN
PREP. FOR
BLOWDOWN
SAFETY
INJECTION
SAFETY
INJECTION
NO INJECTION
NO
INJECTION
PRIMARY PRESSURE
DURING COOLDOWN
RESIDUAL
HEAT LOOP
CUT IN
MASS DISCHARGED IF
STEAM LINE NON-RETURN
VALVE OPERATES
FINAL ISOLATION
OF S.G. ASSUMED
255 MIN.
REACTOR PRESSURE PSI
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
20
40
60
80
100
120
140
160
180
200
220
240
260
280
300
320
340
TIME AFTER BREAK, MINUTES
39x10⁴
36
33
30
27
24
21
18
15
12
9
6
3
0
TOTAL MASS DISCHARGED INTO
SECONDARY, lbs

FIGURE 5-2

ITEM 2 (LETTER 2 QUESTION 7)

In order to evaluate the adequacy of the containment design for the consequences of a loss-of-coolant accident, it is desirable to consider the pressures that could result from energy added in a manner somewhat independent of a specific model. Please prepare a containment capability curve indicating the pressures resulting from various amounts of metal-water reaction occurring linearly with time in 500, 1000, and 2000 seconds. These curves should be drawn assuming (1) no engineering safeguards function, (2) only fan-coolers operate, and (3) the fan-coolers and spray pumps operate. Additional appropriate information such as containment atmosphere composition and temperatures should be presented.

ANSWER

Figure 7-1 presents the results of this study. The results illustrate the capability to accept energy release into the containment following a loss-of-coolant accident, in which the rate and extent of the metal-water portion of the total release are arbitrarily varied.

The study requested that various metal-water reaction energies be added to the containment linearly with time over several time periods. However, to be useful, representations for determining trends or sensitivity, other important energy source terms that are a consequence of the loss-of-coolant accident must be included in the results.

Therefore, the following ground rules were established for this sensitivity study:

1. In all cases, the initial pressure peak is caused by the blowdown of the reactor coolant system.

2. Starting at 10 seconds, reactor core residual heat generation is brought into the containment as produced in the form of steam. For conservatism, this energy term is based on infinite core operating time at full power prior to the accident. The core residual heat contributions are 45.4×10^6 and 112×10^6 and 220×10^6 BTU at 300, 1000, and 2000 seconds, respectively.

3. The initial core stored energy totaling 28.5×10^6 BTU is brought into the containment as steam at a uniform rate over 2000 seconds.

4. The reactor coolant system hot metal above the level of the break (including the approximately 160,000 ft^2 of steam generator tube heat transfer area) is assumed to be in direct contact with the containment steam-air mixture. The total heat release to the containment steam-air phase is 6×10^6, 20×10^6, and 40×10^6 BTU at 300, 1000, and 2000 seconds, respectively. A transfer of 40×10^6 Btu amounts to nearly all the stored energy in one steam generator. Since these hot metal areas are within stagnant flow regions of the reactor coolant system, the actual rate of heat release would be much slower than assumed. This conservative assumption was also used in all containment pressure transients previously presented in the PSAR.

5. The zirconium-water reaction energy is brought into the containment in the form of steam at a uniform rate until the specified time at which the reaction is completed. The corresponding hydrogen-oxygen recombination energy is added directly to the containment steam-air mixture at a uniform rate over the same time period. It is assumed that the hot hydrogen burns as it discharges from the break into the containment atmosphere.

The main parameters studied were extent of zirconium water reaction, time to complete the reaction, and containment cooling engineered safeguards systems in operation.

Various metal water reactions equaling 33.3 per cent, 66.6 per cent, and 100 per cent of the total core Zr mass were arbitrarily assumed. In energy terms, a 100 per cent reaction is equal to 113×10^6 BTU. The corresponding hydrogen-oxygen recombination energy is 89×10^6 BTU.

Arbitrary completion times for the reaction were also assumed. The three reaction fractions were each studied at assumed completion times of 300, 1000, and 2000 seconds.

Each combination of reaction fraction and completion time was then analyzed for three levels of engineered safeguards in operation. The first level includes five fan coolers and two containment spray pumps and structural heat sinks. The second level is based on running either five fan coolers or two spray pumps and structural heat sinks. The third level assumes only containment structural heat sinks are available.

To more clearly illustrate the containment sensitivity to various amounts of zirconium-water reaction, the results are presented in the form of allowable metal-water reaction per cent versus time to complete the reaction. The allowable reaction is defined as the metal-water reaction that causes containment pressure to reach a defined pressure.

As shown by the juncture of the solid line curves on Figure 7-1, reaction of 75 per cent occurring immediatedly after blowdown will not cause containment pressure to exceed the containment load capacity of 1.5 P or 70.5 psig. With five fan coolers and two spray pumps operating, a 100 per cent reaction could be tolerated if the time of completion were greater than 120 seconds. With either five fan coolers or two spray pumps running, the minimum allowable completion time for 100 per cent reaction is increased to 370 seconds. These times indicate a high degree of effectiveness in the containment engineered safeguards. Referring to the specific core thermal transient model discussed in the PSAR, a conservative steam limited reaction of 45 per cent was predicted to accur in 2300 seconds. If containment structure is the only heat sink, the allowable reaction percentage decreases with increasing completion time because of continued energy addition to the containment from core residual heat generation and heat transfer from the steam generators. This illustrates the relatively greater importance of integrated core residual heat than zirconium-water reaction over long time

periods. Considering the reliability and redundancy of the containment cooling and recirculation system, such a situation is highly improbable.

As shown by the dashed line curves on Figure 7-1, the allowable reaction shifts downward when the design working limit of 47 psig is used. With safeguards operating, the net result is to extend the minimum time of completion for the reaction.

ALLOWABLE METAL-WATER REACTION VS TIME

TO COMPLETE REACTION

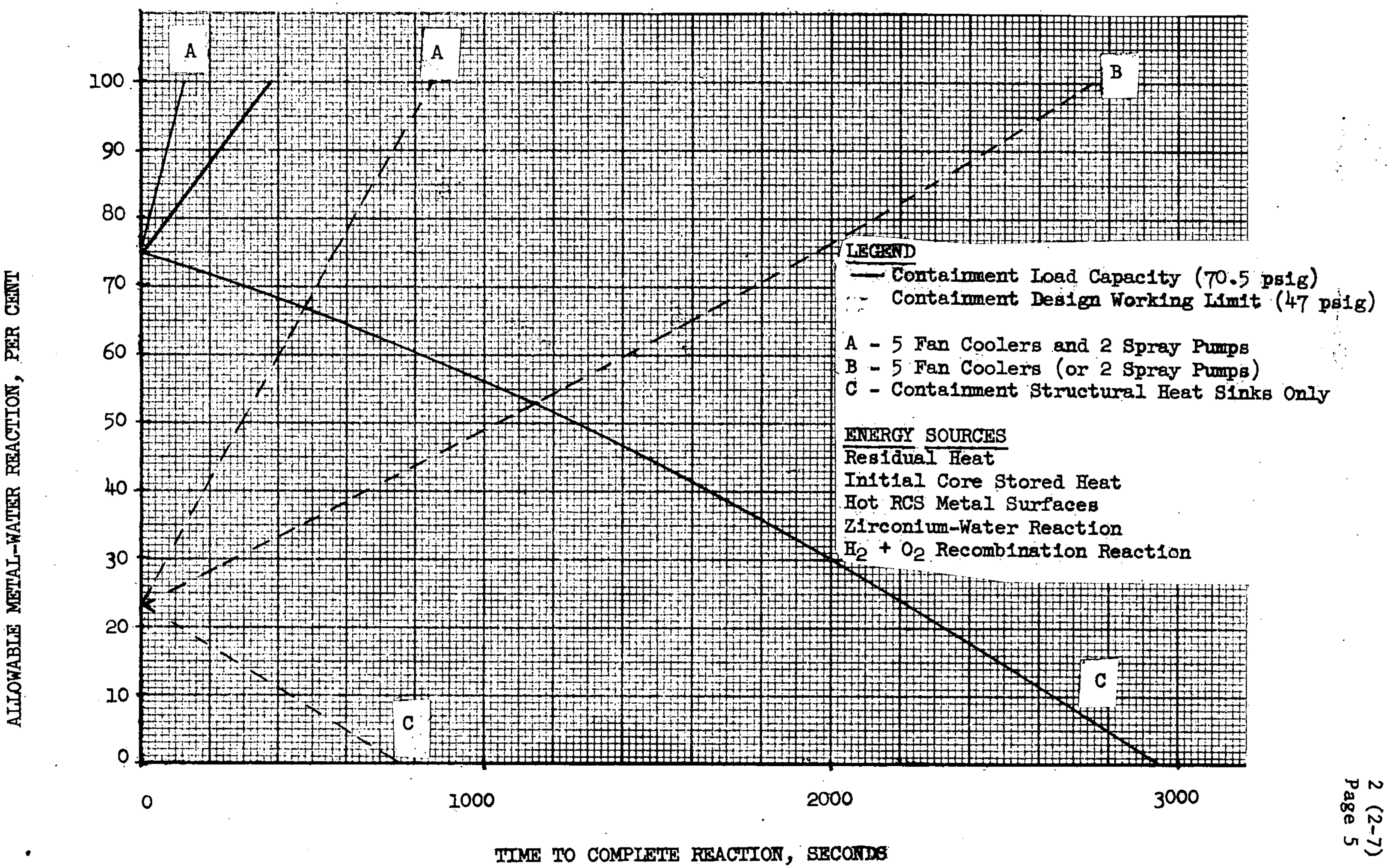


FIGURE 7-1

Supplement 1

ITEM 2 (LETTER 2 QUESTION 8)

We note that locked-open valves are to be installed in some sections of the safety injection system within the containment vessel. In view of the importance of this system to protect the reactor core and to maintain containment vessel integrity, we believe the position of these valves should be determined by other than procedural control when the reactor is in operation and the containment is closed. Please discuss contemplated means for assuring that these valves are open.

ANSWER

The system contains one valve inside the containment which is locked-open to ensure a flow path for safety injection. The valve in question is the maintenance isolation valve for one of the residual heat exchangers and will be closed very infrequently. Maintenance work on the residual heat exchangers can be performed only with the plant shut down and such a condition would require that the residual heat removal loop be in operation to cool the reactor coolant system so that both heat exchangers cannot be isolated simultaneously.

The administrative controls which pertain to these valves as well as to other valves and equipment throughout the plant, are being used successfully in other power stations, both nuclear and conventional, and in the Consolidated Edison System.

According to Consolidated Edison Company practice, such a valve would be locked in the fully-open position according to the instructions of the licensed plant operator. The construction of the valve is such that its position is plainly visible.

Upon closing the locked-open valve, the valve is tagged and the issuance of the tag number, the responsibile party, and the reason for issuing the tag are made a part of the plant operating record. Removal of the tag is similarly noted in the operating record along with the name of the responsible party.

Prior to startup, a checklist of safety items must be completed. The position of all such valves in the safety systems must be visually examined and indicated on this list which must then be signed by the operator performing the check and reviewed by the general watch foreman.

With such administrative procedures, a physical check is made of each device which is important to plant safety. These procedures provide the necessary assurance that the proper valves in the safety injection system will be open. No additional features are planned.

ITEM 2a

The response to Item 2a is given in the Second Supplement to the PSAR under the heading of "Responses to AEC Letter of February 19, 1968 - Containment Design Information."

Item 3

The Advisory Committee on Reactor Safeguards in its August 16, 1966 report to the Chairman on Indian Point Unit No. 2 and in its December 20,1967 report to the Chairman on Diablo Canyon Nuclear Power Plant Mode specific comments on certain aspects of the plant design which warrented careful attention. An analysis and evaluation relative to all applicable points raised by the Committee should be included.

ANSWER

ACRS August 16, 1966 letter

a) "The emergency core cooling systems are of particular importance, and the ACRS believes that an increase in the flow capacity of these systems is needed; improvements of other characteristics such as pump discharge pressure may be appropriate."

The present ECCS design as proposed for IPP Unit No. 3, and presented in the PSAR, represents a substantial improvement over the design for IPP Unit 2 at the time of the ACRS letter dated August 16, 1960.

In this respect, the design of the ECCS system now incorporates four accumulators in addition to the three high head injection pumps and the two low head residual heat removal pumps. Accumulator operation requires no initiating signal.

b) "The forces imposed on various structural members within the pressure vessel during blowdown in a loss-of-coolant accident should be reviewed to assure adequate deign conservatism."

A study of blowdown forces is in progress and is discussed in Item 4 of this supplement.

c) "Design and fabrication techniques for the entire primary system should be reviewed thoroughly to assure adequate conservatism throughout and to make full use of practical, existing inspection techniques which can provide still greater assurance of highest quality."

The design and fabrication inspection techniques for the primary system are discussed in Item 2, Question 3 of this supplement.

d) "Great attention should be placed in design on in-service inspection possibilities and the detection of incipient trouble in the entire primary system during reactor operation. Methods of leak detection should be employed which provide a maximum of protection against serious incidents."

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

For further discussion of leak detection, refer to Item 2 (Letter I Question 3i).

e) "Attention should also be given to quality control aspects, as well as stress analysis evaluation, of the containment and its liner. The Committee recommends that these items be resolved between the AEC Regulatory Staff and the applicant as adequate information is developed."

The quality control aspects of the containment and its liner are found in response to Item 5 of this supplement. A stress analysis

evaluation will be performed for the containment and its liner and will be presented in the AEC prior to the FSAR.

f) "Westinghouse representatives reported that the magnitude of such reactivity transients could be reduced by installation of solid burnable poisons in the core to permit reduction of the soluble boron content of the moderator, thereby reducing the positive moderator coefficient. The Committee agrees with the applicant's plans to be prepared to install the burnable poison if necessary."

Burnable poison rods are being incorporated in the IPP Unit No. 3 core to maintain a negative moderator coefficient at normal operating conditions. Additional information concerning the installation of burnable poisons is presented in answer to Item 4 and Item 9 of this supplement.

ACRS Letter of December 20, 1967

a) "The Regulatory Staff should review analyses of possible effects upon pressure vessel integrity, arising from thermal shock induced by ECCS operation."

The response to analyses relative to reactor vessel thermal shock induced by ECCS operation is discussed in answers to Item 4 and Item 12 of this supplement.

b) "Further evidence should be obtained to show that fuel-rod failures in loss-of-coolant accidents will not significantly affect the ability of the ECCS to prevent clad melting."

Further development work is in progress concerning fuel rod behavior during loss-of-coolant accident conditions. Refer to the Rod Burst Program discussed in Item 4 of this supplement.

c) The applicant has proposed using signals from protection instruments for control purposes. The Committee believes that control and protection instrumentation should be separated to the fullest extent

practicable. The Committee believes that the present design is unsatisfactory in this respect but that a satisfactory protection system can be designed during the construction of this reactor. The Committee wishes to review an improved design prior to installation of the protection system."

The proposed control and protection system is currently being reviewed with the AEC as noted in Item 18 of this Supplement.

d) "Consideration should also be given to the development and utilization of instrumentation for prompt detection of gross failure of a fuel element."

A development program is currently in progress as discussed in the Failed Fuel Monitor Program of Item 4 in this Supplement.

ITEM 4

On page 2-59 of the PSAR, some features of the proposed Indian Point Unit No. 3 have been identified with Research and Development (R&D) programs which would resolve safety questions in the design of the proposed facility. These programs are presently being carried out or are planned in the near future. The following information should be discussed regarding each of these items:

1. A delineation of the responsible organization for each R&D item.

2. A description of the program for each R&D item and an analysis of the adequacy of the program to solve the problem.

3. A proposed schedule of completion for each R&D item as related to the proposed facility construction schedule.

4. A discussion of alternatives in the event the program results do not corroborate their objectives.

ANSWER

The design of the plant will be based upon proven concepts which have been developed and successfully applied to the design of pressurized water reactor systems. Results of work already completed under the Nuclear Safety Research and Development Program being conducted by the Atomic Energy Commission will be incorporated in the design and evaluation of applicable portions of the engineered safety features.

The term "research and development" as used in this section is the same as that used by the Commission in Section 5.2 of its regulations as follows:

> "(n) 'Research and development' means (1) theoretical analysis, exploration or experimentation; or (2) the extension of investigative findings and theories of a scientific nature into practical application for experimental and demonstration purposes including the experimental production and testing of models, devices, equipment, materials and processes."

The research and development to be done for this plant is to confirm the engineering and design values normally used to complete equipment and system designs. It does not involve the creation of new concepts or ideas.

The technical information generated will be used either to demonstrate the safety of the design and more sharply define margins of conservatism, or will lead to design improvements.

The schedules for developing this technical information is compatible with the plant schedule (commercial operation in 1972) such that definite results will be available before the plant design is complete, and in time to consider alternatives in development programs, changes in design or in plant operating conditions in the event that the program results do not corroborate their objectives. However, considerable information is on hand to indicate that the anticipated program results will be obtained.

The specific areas in which additional information will be developed and which are required for plant operation are as follows:

1) Core Stability Evaluation
2) Rod Burst Program
3) Containment Spray Program

Other research and development programs are being carried out primarily to provide technical information which can be applied for component or system organization in future plants. While these programs will give added confirmation of the conservatism of the proposed design for the IPP #3 plant, their completion is not essential for the resolution of outstanding safety questions. These programs include the following:

1) Burnable Poison Program
2) Saxton Loose Lattice Irradiation Program
3) Zorita Irradiation Program
4) In-Core Detector Program
5) ESADA DNB Program
6) Failed Fuel Monitor Program
7) Loss of Coolant Analysis Program
8) FLECHT (Full Length Emergency Cooling Heat Transfer Test) Program
9) Flashing Heat Transfer Program
10) Blowdown Forces Program
11) Reactor Vessel Thermal Shock Analysis Program

PROGRAMS REQUIRED FOR PLANT OPERATION

1. Core Stability Evaluation

The purpose of this program is to establish means for the detection and control of potential xenon oscillations and for the shaping of the axial power distribution for improved core performance.

In the transition to Zircaloy 12-foot cores, a potential for axial xenon oscillations was created. Part length control rods have been incorporated in the design to provide a means of controlling xenon oscillations should they occur and to permit a more optimum control of core power distribution.

A neutron flux monitoring system has been developed consisting of long ion chambers, each split in two sections, located outside the reactor vessel. A similar system is in operation at the SENA reactor. Such a system will be employed in all Westinghouse reactors now under construction, and will provide the reactor operator with a simple means for continuously monitoring axial power distribution and thus to position the part length rods. Identical systems will be installed and operational in 11 reactors well before plant operation, including the Ginna, Indian Point 2 and Beznau plants in 1969.

The mechanical, nuclear and thermal-hydraulic and safety aspects of the part length rod design were submitted to the AEC in a report, WCAP-7072, in mid-1967 in conjunction with the Diablo Canyon application. Further analytical work is continuing to investigate the behavior of the reactor under various operational conditions to determine the effects of part length rod operation as well as malpositioning of these rods. These analytical studies will be verified in operating reactors well before plant operation.

In addition, a carefully planned plant startup test program will be carried out, which will include verification of the effects of part length rods over the full range of travel. Similar tests will be conducted in subsequent start-ups after refueling to verify the effects of these rods with the core of fresh fuel combined with burned fuel.

On power distribution monitoring, the development program consists of correlations between out-of-core measurements and detailed in-core maps in operating reactors, e.g., SENA and San Onofre. Tests performed to date include forcing various axial power shapes with full length rods and comparing the measured out-of-core readings with detailed in-core measurements. Excellent correlation has been obtained in the form of a linear function relating to the power generation in the upper and lower half of the core as observed out-of-core and in-core.

This gross power measurement can be readily related to allowable core limits as demonstrated by analysis. The part length rods can, therefore, be positioned to maintain the difference between top and bottom detector readings within an allowable range. These relationships will be developed for the plant in the course of detailed design.

Further tests at operating reactors will be performed at various operating loads, core depletion, conditions of xenon distribution, operating temperatures, and boron concentrations. Further substantiation of these results will be obtained in the reactors going on the line prior to plant operation. Also, a detailed start-up program will be conducted prior to full power operation to fully demonstrate the required capability of the monitoring system.

The referenced SENA and San Onofre data is in progress of being documented in a general topical report to be submitted to the Commission shortly.

2. Rod Burst Program

This program will determine clad deformation characteristics and extent of flow blockage under simulated loss-of-coolant accident conditions. This is to substantiate the present assumption that rod bursting during a loss-of-coolant accident will not cause gross core geometry distortion and that the core will remain in place and essentially intact to such an extent that effective core cooling is not impaired. In addition, experimental data will be obtained on the behavior of the fuel rod during the core reflooding stage of the loss-of-coolant accident in order to establish a realistic upper bound for the peak fuel clad temperature criteria for use in design evaluations. This upper bound is expected to be well above the present peak temperature predictions for the accident conditions.

During a loss-of-coolant accident, the core must be preserved in its original geometry to such an extent that effective core cooling is not impaired. In the blowdown phase of the accident, core geometry distortion may be due to clad bursting. The clad temperature may get sufficiently high (1200-2000°F) so that bursting of the clad would occur by virtue of the internal gas pressure and the significant reduction of clad strength. Clad bursting is of concern because of the possibility of blocking the flow channel so that core cooling would be insufficient to prevent fuel rod melting.

The deformation and burst characteristics of the clad are dependent upon clad strength and ductility as functions of temperature, prior irradiation and material properties (including hydride levels in Zircaloy). Experimental tests to evaluate all of these inter-related parameters is being conducted by Westinghouse.

Recent Argonne[1] experimental data on fuel rod quenching caused by the cold safety injection water hitting the fuel rods and causing rod shattering has initiated concern about keeping the fuel rod geometry intact. It should be noted that these experiments were for much more severe conditions than those experienced by PWR cladding during a loss-of-coolant accident in that significantly greater metal/water reactions had occurred.) This program will demonstrate that peak fuel clad temperatures are below that where the clad could shatter under quench conditions.

The overall program consists of the following tests:

A. <u>Rod Burst Tests - Unirradiated Clad</u>

Eleven transient heating burst tests simulating loss-of-coolant heating rates on unirradiated reference cladding with very small void volume.

B. <u>Rod Burst Tests - Unirradiated Hydrided Clad</u>

Transient heating burst tests in unirradiated, hydrided clad to determine:

1) Effect of void volume - Five tests will be performed on nearly empty tubes, to determine the maximum effect of stored energy on fracture characteristics.

2) Parametric experiment - Sixteen burst tests will be performed to investigate the effect of pellet-to-clad gap, pellet cracking, heating rate, and initial gas pressure at levels typical for a PWR loss-of-coolant accident.

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(1) ANL 7438, Argonne National Laboratory Reactor Development Progress Report, March, 1968.

C. Quench Tests - Unirradiated Hydrided Clad

Twenty-four specimens will be tested to investigate the effect of two heating rates and maximum temperatures, varying amounts of Zircaloy-water reaction prior to quench, and two quench methods. The qeunch water temperature will correspond to that of the emergency core cooling following a loss-of-coolant accident.

D. Rod Burst Tests - Irradiated Clad

Samples of Zircaloy cladding irradiated in Yankee will be subjected to loss-of-coolant simulated burst tests similar to those currently being performed on unirradiated cladding.

The tests are being conducted at the Materials Testing Laboratory of the Westinghouse Atomic Power Divisions in Pittsburgh. The expected completion dates for the tests noted above are as follows:

A.	Rod Burst Tests - Unirradiated Clad	Completed (Results have been presented to the AEC in the Zion Station PSAR Vol. III, Dockets 50-295 and 50-305, Responses to ACRS August 31, 1967 letter to PG&E Diablo Canyon Site)
B.	Rod Burst Tests - Unirradiated, Hydried Clad	November 1968
C.	Complete Quench Tests	December 1968
D.	Rod Burst Tests - Irradiated Clad	July 1969

Data from the rod burst tests on unirradiated cladding 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 of approximately 1/32 to 3/16 inch. This would result in a flow area blockage of 10-15% for a single rod. These tests also indicate that the burst pressure versus clad temperature data is as much as 50 to 200% higher than the design curve used in present rod burst evaluations.

The experimental data obtained from this program will be used to substantiate the present assumption that rod bursting during a loss-of-coolant accident will not cause gross core geometry distortion and that the core will remain inplace and substantially intact to such an extent that effective core cooling is not impaired. Analytical studies with flow blockages of the amount seen in the preliminary data obtained from these tests have already been completed and show that the fuel rod failure due to rod bursting during a loss-of-coolant accident has a negligible effect on the ability of the Emergency Core Cooling System to effectively cool the core.

Upon completion of this program, experimental data will have been obtained to determine a realistic clad temperature criteria for use in design evaluations.

3. Containment Spray Program

The purpose of this program is the development of design details for a containment spray system utilizing reactive chemical additives to promote fission product iodine absorption.

Following a loss-of-coolant accident, sprays are actuated in the containment to reduce the pressure of steam. Boric acid H_3BO_3, is a required constituent of these sprays to prevent dilution of the boron concentration in water collected by the recirculation sump and subsequently used for core cooling. Due to the low solubility of elemental iodine (I_2) in boric acid compared with that in alkaline solutions it has been determined to blend sodium hydroxide, NaOH, with the boric acid solution in the event of a major accident making the solution an aggressive absorber for iodine when contacted with the containment atmosphere.

Ideally, the containment sprays can remove iodine at a rate sufficient to reduce the integrated two-hour leakage of elemental iodine by a factor of 20-60, depending on physical dimensions and spray rates required for containment cooling. Design of the spray system is being evaluated and modified to obtain maximum benefit from chemical additives in the spray to reduce iodine leakage from the containment.

Work completed to date by Westinghouse has developed spray nozzle performance characteristics as well as engineering data leading to the selection of NaOH as the preferred spray additive. The second leading candidate, which was a combination of NaOH and sodium thiosulfate, $Na_2S_2O_3$, was not selected due to higher water radiolysis rates, $Na_2S_2O_3$ decomposition with potential sulfur deposition, and higher corrosion rates of certain materials. Further investigations with NaOH - H_3BO_3 solutions developed data on iodine absorptive capacity, material compatibility and radiolytic stability for use in final system design.

Parallel research at Oak Ridge National Laboratory (ORNL) has explored the basic mass transport theory underlying the absorption of I_2 by reactive sprays. This work, conducted in small and intermediate scale containment simulations and in wind-tunnel experiments, have shown good agreement with the empirical model used by Westinghouse to predict spray performance in full-sized containment. Larger scale simulations, approximately one-half of the full height of PWR plant sprays, have been initiated at Battelle Northwest Laboratory (BNWL). Westinghouse concurrently evaluates the results of these programs to confirm the validity of the analytical model, but holds no contractual relations with these groups relevant to the spray program.

Parallel work at ORNL is also directed to the search for chemical additives which promote trapping of organic iodide vapors. These forms constitute a relatively minor fraction of the total inventory of iodine, but have proven almost insensitive to sprays which can be used in containment systems. Currently, as there is no prospective successful spray additive for organic iodine trapping, Westinghouse has no plans to extend its development activities in this direction.

A subsidiary program which is the study of radiolysis in emergency core cooling water, was initiated as a part of the spray additive evaluation. It is being continued for the purpose of studying mass transport effects on the net yield of hydrogen from radiation absorption in the post-accident containment water phase. Results will provide a basis for assessing potential hydrogen buildup when the containment is held isolated for extended periods of time. Westinghouse sponsors this program, which parallels a similar effort at ORNL covering a broader range of solution chemical compositions.

The portions of this program still in progress, i.e. refinement of the analytical model and radiolysis testing, will continue through calendar 1968. If further data become available from large scale simulations after this date, correlation with the Westinghouse model will be checked and requirements made if warranted.

It is expected that design details of a typical PWR containment spray system employing the information from this program will be complete in mid-1969. The uncertainty margins evolved from comparison of analytical model predictions with test results and from further consideration of industrial gas absorption experience will be applied to the determination of a reasonable factor of reduction of iodine leakage for use in safety analysis.

OTHER AREAS OF RESEARCH & DEVELOPMENT

Other areas of research and development, as outlined below, are those which will give added confirmation that the proposed designs are conservative.

These programs are in progress and are being carried out basically to provide technical information which can be applied to component or system improvements in future plants.

1. Burnable Poison Program

This program will verify the calculated reactivity worth and effect of borosilicate glass on power distribution, and will demonstrate satisfactory mechanical performance in a reactor environment.

Burnable poisons are to be added to the initial core to eliminate the positive moderator coefficient previously expected at operating temperatures early in the first fuel cycle. The design basis and critical experiments were described in a report, WCAP-7113, "Use of Burnable Poison Rods in Westinghouse Pressurized Water Reactors," dated October, 1967.

This report, submitted to the AEC, Division of Reactor Licensing, also outlined the program for future development. The program for the burnable poison rods includes performing critical experiments to verify calculated reactivity effects, calculation of critical loading using two dimensional diffusion theory, and corroboration of mechanical and nuclear design by evaluation of rod performance in the Saxton reactor.

Critical experiments conducted at the Westinghouse Reactor Evaluation Center have demonstrated that standard analytic methods can be used to evaluate the reactivity worth of these burnable poison rods. These experiments were performed using 2.72 per cent enriched UO_2 clad in Zircaloy. The reactivity worth was measured for solid glass rod and two thicknesses of glass tubing for several ratios of glass to fuel.

The Boron content of the glass was 12.8 weight per cent. These parameters are representative of the plant design. Application of these standard methods to reactor design indicates that the moderator temperature coefficient of reactivity will be negative at operating conditions.

In-core testing of these rods is presently underway in the Saxton reactor. Basic information now available showing satisfactory performance is to be further confirmed in this reactor environment, with testing continuing through mid-1969.

The first application of these rods in a commercial power reactor will be in the GINNA plant, scheduled to start up in 1969. Complete results of the program will therefore be available several years prior to startup of Indian Point Unit 3.

2. Saxton Loose Lattice Irradiation Program

There is a need for additional data on PWR fuel performance in bundle form at high linear power rating combined with high burnup, although data has been obtained for the variables individually. The excellent operating experience achieved with the Zircaloy-4 clad mixed oxide Saxton Plutonium Project fuel in Core II together with the ability to extend the burnup of this fuel to significantly higher levels by positioning the individual fuel rods at a higher water-to-fuel ratio in new mechanical assemblies has led to the decision to undertake a "loose lattice" operation in Saxton Core III. The objectives of this mode of operation are to:

1) validate fuel element performance predictions, including determination of linear power/burnup failure limits,

2) demonstrate the performance capability of Zircaloy clad oxide fuel elements over a broad spectrum of burnup and power levels,

3) obtain depletion characteristics and transuranic isotope generation data for high burnup mixed oxide fuel.

The more open lattice of the reconstituted Core III assemblies also permits satisfying current thermal design criteria while operating individual fuel rods at very high linear power levels. The maximum power/burnup fuel operating conditions which are achieved in the loose lattice assemblies are directly responsive to specific ACRS concern for further experimental confirmation of fuel performance capability at worst anticipated transient heat generation rates and maximum burnups.

Existing Saxton Core II Zircaloy-4 clad PuO_2-UO_2 fuel assemblies will be modified to effectively increase their lattice pitch by reinstalling individual fuel rods in new assemblies for further irradiation during Core III operation. This modification utilizes the significant increase in the reactivity capability of Zircaloy clad plutonium fuel which can be obtained by increasing the water-to-fuel ratio. Saxton Core III will consist of seven plutonium loose lattice assemblies reconstituted from the plutonium assemblies now in Core II, two new PWR reference UO_2 assemblies, and twelve UO_2 partially depleted fuel assemblies from Core I and Core II operation.

It is planned to operate the lead rods in Saxton Core III loose lattice assemblies at a peak design linear power rating of 24.0 kw/ft. End-of-life peak burnups of 45,000-55,000 MWD/MTM will be achieved. Because the Saxton Core II mixed oxide fuel rods were designed for relatively low peak burnups (20,000 MWD/MTM) and operation at design peak power densities $\leq$ 16 kw/ft, operation of the higher burnup Core II rods at the higher linear power levels planned for this test program will produce fuel failures. Prior experience at Westinghouse and other sites has shown that these failures will be limited to localized cracks in the cladding which do not result in loss of fuel. The rods in which failures are to intentionally be induced will be located in the center removable 3 x 3 subassembly to permit easy access when removal of these rods becomes necessary. Linear power/burnup levels for the balance of the loose lattice rods have been more conservatively specified, but will demonstrate fuel performance at conditions significantly exceeding current PWR design limits. This mode of test operation will permit power/burnup limits to be established while operating safely and in full compliance with the reactor license Technical Specifications.

The new and reconstituted Core III fuel assembly designs permit removal and replacement of individual fuel rods. This permits on-site visual and nondestructive examination of rods at varying levels of burnup during a planned shutdown and vessel head removal at the midpoint of Core III operation. Selected rods will also be transferred to the Westinghouse Post-Irradiation Facility hot cells for destructive examination and further evaluation during this shutdown. A similar program of non-destructive and destructive examinations is planned at the end of Core III life for rods operated in a range of high linear power and burnup levels.

Design and fabrication of fuel assembly components and specification of detailed test conditions is being carried out by the Westinghouse Nuclear Fuel and PWR Plant Divisions. Conduct of the test will be a joint Westinghouse-SNEC effort. Westinghouse Nuclear Fuel Division will conduct the post-irradiation examination at the Waltz Mill facility. Evaluation of the test results will also be performed by the Nuclear Fuel Division.

Planning of this program was begun in early 1967 and is now essentially complete. Fuel assembly components are now being fabricated at Westinghouse Nuclear Fuel Division, Cheswick facilities. The reactor licensing procedures will begin in the early fall of 1968. Completion of Saxton Core II operation is scheduled for October 1968; reconstitution operations will be conducted at the Waltz Mill site from November 1968 to January 1969. Initial Core III high power operation is scheduled in March 1969. Intermediate on-site fuel examination will be performed in March 1970, and the first destructive post-irradiation results are expected in June 1970. End-of-life post-irradiation examinations and evaluations will be conducted in the last half of 1971.

The results of this extensive fuel test program will both define linear power/burnup failure limits and provide additional confirmation of the adequacy of current fuel performance margins by more quantitatively validating design methods.

Specific results from this program can be applied in plants beginning operation in late 1971.

3. Zorita Irradiation Program

The Westinghouse Nuclear Fuel Division and the Union Electra Madrilena (UEM), S. A. have agreed to the inclusion of four advanced assemblies in the Zorita reactor to demonstrate the reliability of contemporary fuel rod designs for high power, high burnup application. Since Zorita is the first PWR to operate at 2250 psi, the temperature, pressure, and chemical environment are similar to reactors presently being designed or under construction. Consequently, these assemblies provide the first opportunity to evaluate the performance of Zircaloy-4 clad fuel rods that have operated at peak design linear power levels of up to 20 kw/ft and peak pellet burnups of up to 55,000 MWD/MTU in a commercial PWR.

The design of the Zorita advanced assemblies is such that 51 of the rods in each assembly can be removed for examination or replacement during normal refueling operations. Sixteen of the rods in each assembly are designed for operation at peak linear power levels of 18-20 kw/ft. Post-irradiation visual and dimensional examinations will be performed on individual fuel rods during scheduled head removals. The two Region 2 assemblies will be removed after two cycles of operation; the two Region 3 assemblies will be removed after three cycles. Fuel rods will be examined after each cycle to evaluate the effects of increasing exposure. The lead burnup rods will be removed from Region 3 assemblies at the end of the second cycle to limit the peak exposure after three cycles to less than 55,000 MWD/MTU.

Present program plans specify a total of 56 fuel rods from the four advanced assemblies to be visually and dimensionally examined in the Zorita fuel storage pool. Eighteen of these rods will also be subjected to extensive destructive examinations. The objectives of this effort are to:

1. Demonstrate the overall reliability of contemporary PWR fuel element designs for use in Indian Point Unit 2, Diablo Canyon Unit 1 and other future PWR cores.

2. Evaluate the effects of long-term exposure under actual service conditions on the reference design Zircaloy-4 fuel element cladding.

3. Evaluate the effects of various combinations of linear power level and burnup on overall fuel performance.

Agreement has been reached with the Spanish AEC (JEN) to perform the destructive post-irradiation examination in the JEN hot cells with Westinghouse providing program direction and engineering assistance. The on-site examinations will be performed cooperatively by Westinghouse, UEM and JEN technical personnel.

The Zorita Irradiation Program was initiated in 1965. Design, fabrication and insertion of the four advanced assemblies have been completed. Plant startup is in progress and full power operation will begin in August 1968. Initial non-destructive fuel performance data will be obtained upon completion of the first cycle of plant operation in February 1970, at which time on-site visual and dimensional examinations will be conducted. Several fuel rods will then be shipped to JEN hot cells in Madrid for post-irradiation evaluation.

Data from the post-irradiation examination of Cycle I fuel rods are expected by December 1970. Data from Cycle II and III examinations are expected by February 1972 and April 1973, respectively.

The results of the examination of selected special fuel rods will extend the present knowledge of long-term exposure effects on cladding and fuel operating at design linear power levels in excess of the maximum levels planned for currently committed plants. Successful operation of the test rods will permit more quantitative determination of the fuel performance margin provided by current design practice, thus the results of this program will apply primarily to future plants and may permit later up rating of cores for plants currently committed. Sustained successful operation of special Zorita fuel rods at peak design power levels in excess of those planned for presently committed plants will also increase assurance that the fuel for current plants has adequate performance margins to accommodate transient overpower operation.

4. In-Core Detector Program

The purpose of this program is to develop fixed in-core detectors with a lifetime to make them suitable for continuous monitoring of the power distribution in a PWR core.

Experimental measurements in SENA, San Onofre and Connecticut Yankee have shown that gross power distribution monitoring can be satisfactorily performed with out-of-core detectors. The out-of-core detectors, particularly long ion chambers, have been found effective for monitoring both axial and radial gross power distribution. However, as a possible backup to out-of-core instrumentation, a program is underway to develop and evaluate fixed in-core detectors for continuous core power monitoring. Detectors are already available which have the required performance characteristics, but these have not yet been qualified for long term operation.

In-core detectors presently available from various manufacturers will be evaluated to determine their lifetime and sensitivity characteristics.

In-core detectors will be obtained from various vendors and evaluated at Saxton, Yankee, San Onofre and other reactors for linearity, response time, sensitivity and lifetime characteristics. Four experimental detectors have already been installed in the Yankee plant.

A test program will be carried out in the early plants to determine the need and number of incore detectors if required for power distribution monitoring and control applications.

The evaluation of detectors underway at Saxton, San Onofre, Yankee, Western New York Nuclear Research Center and Other facilities will be completed in December, 1970. The lifetime evaluation program will be completed by the end of 1971.

ESADA DNB Program

This program will provide experimental rod bundle DNB data with non-uniform rod axial flux distributions. These DNB data will be directly applicable to present Westinghouse pressurized water designs.

DNB studies have been conducted at Westinghouse and other facilities with uniform axial heat flux rod bundles and with uniform and non-uniform heat flux single rod test sections. The Westinghouse rod bundle data indicated the 2σ lower limit of the DNB ratio at DNB was 1.08. Thus, the design criteria for 1.30 has substantial margin. These data are the basis for existing DNB correlations used in reactor design. DNB data for rod bundle geometries and non-uniform rod axial flux distributions will be supplied under this program.

The reference design test assembly will be a 4 x 4 square array of 0.422 inch O.D. rods with an 8 foot heated length. Each test assembly will have an axially non-uniform heat flux distribution. The effects of radial variation in power and mixing promotion devices will also be investigated.

The first phase of this program consists of the construction of a high pressure and temperature water loop for conducting the heat transfer and hydraulic tests. Upon completion of the loop, 12 test assemblies will be tested to provide DNB data with axial non-uniform heat flux distributions.

This program will be conducted at Columbia University under the direction of Westinghouse Atomic Power Divisions, Pittsburgh, Pennsylvania.

The loop construction was completed in May, 1968. Testing will be conducted during the period June, 1968 through September, 1969.

The experimental rod bundle DNB data with non-uniform rod axial flux distributions will be directly applicable to all present Westinghouse pressurized water reactor designs, and will indicate their margin to DNB. These data will also be used to optimize future designs.

6. Failed Fuel Monitor Program

The purpose of this program is the instrumentation for prompt detection of fuel failure.

An early indication of an increase in fission product concentration in the Reactor Coolant System, symptomatic of a fuel failure, is obtained from a change in gross gamma activity in the area of the coolant letdown system. Precise information of the coolant activities is obtained by laboratory analysis of a coolant sample. Although failed fuel detection is presently performed by the letdown monitor a program is underway to develop a rugged and dependable system with a short response time and high sensitivity to fission product release. Various prototype systems will be installed and tested at Saxton to determine the best system.

The following types of failed fuel detectors will be evaluated at the Saxton reactor:

1) Delayed neutron monitor
2) Coolant gamma activity monitor
3) Gross gamma monitor
4) Letdown monitor.

Main coolant water at operating pressure will be continuously circulated in the set-ups for delayed neutron and coolant gamma monitors to establish the background at normal operating conditions and changes in the signal levels will be monitored for indication of failed fuel. Sufficient lag times (20 to 60 seconds) will be provided for decay of the short lived products of water irradiation. Evaluation will be made of the response time, reliability and sensitivity of both the systems. Monitoring for gross gamma activity along the main coolant pipe will also be performed. This system is expected to have fast response but relatively insensitive to fuel failures because of high background level. Methods for increasing the fuel failure sensitivity of this scheme will be investigated. Performance of the iodine monitor in the letdown system will also be evaluated in conjunction with the other failed fuel detection systems.

Installation of the systems at Saxton was completed in August, 1968, and are currently being tested. A detailed evaluation of the performance of the systems will be completed by late 1969.

Relative evaluation of the schemes at Saxton under actual operating conditions of a PWR should yield information on the sensitivity, speed of response and reliability of various schemes.

7. Loss-Of-Coolant Analysis Program

This program will integrate the more realistic heat transfer models obtained from experimental and analytical development programs into the design codes used to evaluate loss of coolant accident.

Design and safety assessment of the PWR Emergency Core Cooling System requires an understanding of the complex heat transfer mechanisms characteristic of the loss-of-coolant accident sufficient to predict the influence of system performance of objectives and to show the margins of safety afforded by the final design.

Westinghouse accomplished this by using simplifying conservative assumptions where rigorous analysis was not practicable and test data were lacking. The designer has employed pessimistic reasoning in an effort to account adequately for uncertainties.

A more precise and realistic basis for sizing the Emergency Core Cooling System permits the designer to optimize the systems and equipment design with a clearly defined performance margin.

The design codes will incorporate heat transfer models developed from the Flashing Heat Transfer Program.

This work is being performed by Westinghouse Atomic Power Divisions in Pittsburgh, Pa.

The program is currently in progress and it is expected to have the heat transfer models incorporated into the design codes by October, 1968.

The incorporation into the design codes of tese new heat transfer models will provide the designer with a more realistic design tool to optimize the Emergency Core Cooling System and clearly define the performance margin. This program, upon completion, will provide a quantative evaluation of the conservative assumptions presently used to predict the core thermal transients.

8. FLECHT (Full Length Emergency Cooling Heat Transfer Test) Program

The purpose of this program is to investigate experimentally the thermal behavior of a simulated pressurized water reactor core during the core recovery period which follows a loss-of-coolant accident.

Considerable effort has been expended in experimental verification of spray cooling and core flooding as satisfactory methods for emergency core cooling. The present design assumptions are conservative. The main purpose of this test program is to continue the general evaluation and to provide data at higher temperatures and degraded conditions.

The subsidiary items of investigation will enhance the understanding of phenomena which have required a theoretical approach in formulating design criteria to date. These include, but are not limited to, the behavior of fuel rods in the approach to melting or metal-water reaction temperatures; the flooding level required to terminate fuel rod temperature excursion; the effects of thermal shock on fuel rod integrity; and the sensitivity of core cooling to variations in pressure, temperature, quality and quantity of flow. The specific information to be obtained is:

1. Time-temperature behavior of the cladding.

2. Conditions which result in clad melting, including the effects of coolant injection delay times.

3. Effectiveness of core coolant systems to prevent core damage.

4. Effect of pressure on emergency core cooling capability.

5. Effect of a soluble poison on emergency core cooling capability.

The program will assess the effects of all pertinent independent variables on fuel element cladding temperature during the core recovery phase of loss-of-coolant accidents, over a range which will include conditions postulated for all current plants. In addition, the minimum flooding levels to maintain steady state conditions for a range of power densities will be determined.

The effects of fuel distortion will be studied by introducing additional flow resistance in the form of collars which slip over the test elements. The effect of space grids with mixing vanes will also be tested on a test assembly.

This work will be done by Westinghouse at its Atomic Power Laboratories in Forest Hills, Pa. under Subcontract No. C-619 from the Phillips Petroleum Company who in turn are under contract to the Atomic Energy Commission, AT (10-1) - 205. Work was initiated on these tests in April, 1968 and will continue through February, 1970.

This program is expected to supply experimental data on the core behavior during the reflooding phase of a loss-of-coolant accident. It will assist in developing new analytical models to describe the core recovery phenomena. This information will be used to evaluate the margins and optimise the systems used for emergency core cooling.

9. Flashing Heat Transfer Program

This program is to obtain experimental data to evaluate the heat transfer behavior in the core during various phases of the loss-of-coolant accident.

The present analytical techniques used to evaluate the effectiveness of the Emergency Core Cooling System use conservative heat transfer coefficients based on current data. The ultimate objective of the program will be a reduction in the peak clad temperatures now predicted for loss-of-coolant accidents.

The events following a loss-of-coolant accident can be divided into the following categories:

1. Depressurization of the Reactor Coolant System.

   Present analytical techniques assume stable film boiling exists on the fuel rod after DNB. Although it is well known that transition boiling occurs and that this is expected to produce significantly higher heat transfer coefficients, no credit is taken for transition boiling.

2. The period when the core is uncovered.

   Cooling of the core is due to laminar of forced convection of steam flow and to radiation between fuel rod and the steam. Presently, a limited amount of data is available for steam at elevated pressures (greater than one atmosphere), such as those which exist at the end of blowdown, to determine the emissivity of steam and the radiative heat transfer mechanism. The data for other gases at these pressures indicates significant improvement in radiation heat transfer. Full benefit of this pressure effect is not considered in the present analysis.

3. Core reflooding phase.

   The rising clad temperature transient is turned around after the lower portion of the core has been reflooded. The heat transfer during this period is associated with the two-phase flow which is present due to entrainment. The present analytical model used to describe the phenomena occuring during core reflooding has been compared with existing experimental data and indicates entrainment occurs earlier than predicted.

This program is divided in three parts:

1. Blowdown Heat Transfer Test - Depressurization phase of the Reactor Coolant System during a loss-of-coolant accident.

   The objective of the blowdown heat transfer test is to determine transient heat transfer coefficients under blowdown conditions.

   The range of variables investigated is representative of that in a PWR in the blowdown phase. The work is being done at Westinghouse Atomic Division Test Engineering Laboratories, Forest Hills, Pennsylvania.

2. Low Pressure Heat Transfer Test - Period when core is uncovered.

   The objective of this program is to measure local heat transfer coefficients for the total simultaneous convective and radiant heat transfer from uniformly and non-uniformly heated tube surfaces at elevated temperatures to low pressure saturated and superheated steam in fully developed laminar and turbulent forced convection.

   The desired operating conditions, as specified, are characterized by high surface temperature, large film temperature differences and relatively low pressures and flow rates. Existing experimental data do not cover completely such conditions. The effects of radiant heat transfer precludes at present accurate predictions of local heat transfer coefficients by extrapolation of existing data. These texts will indicate the benefits which may be obtained during this phase of the accident.

In addition, an experimental program is being performed to measure the radiation properties of superheated steam. The emmissivity of water vapor will be measured over a temperature and pressure range of 300-1500°F and 15-75 psia, respectively.

This work is being done at the University of Michigan, Department of Mechanical Engineering.

3. Cold Water Injection Test - Core reflooding phase.

The objective of the cold water injection test is to investigate conditions corresponding to the core recovery phase of a loss-of-coolant accident. The range of variables investigated is representative of that in a PWR during the core reflooding phase. The work is being done at Westinghouse Atomic Division Test Engineering Laboratories, Forest Hills, Pa.

This program was initiated in early 1967. Testing has been compled for the blowdown heat transfer test and the cold water injection test. Data reduction, evaluation and analysis are currently in progress. Completion is expected in October 1968. Testing is still in progress in the low pressure heat transfer test with some data already being taken. Fabrication of the steam emissivity test facility is nearing completion. Final results from this program is expected by October 1968.

The experimental data obtained from this program will be used to develop new analytical models or make refinements to present models in order to more realistically evaluate the core behavior during the loss-of-coolant accident. It is expected that peak cladding temperatures, when analyzed using the data from this program, will be reduced over 200°F from those predicted with present techniques.

Approval of plant operation is not dependent on completion of this program since present core conditions evaluated using conservative assumptions are will within the specified design criteria associated with the loss-of-coolant accident. The program is expected to aid in showing that added margins to present design criteria do exist.

10. Blowdown Forces Program

The objective of this program is to develop digital computer programs for the calculation of pressure, velocity, and force transients in the Reactor Coolant System during a loss-of-coolant accident, and to utilize these codes in the calculation of blowdown forces on the fuel assemblies and reactor internals to assure that the stress and deflection criteria used in the design of these components are met.

In general the existing codes such as WHAM, BURST, and SATAN were used to simulate the entire blowdown phase of a loss-of-coolant accident. It was found that the most severe loadings are generated during the sub-cooled portion of blowdown which occurs only for a short time at the beginning of blowdown. Westinghouse is now developing a special prpose program to deal specifically with the pressure, velocity and force transients during this period.

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.

The basic analytical model used in BLODWN-1 is the same as that which is in the WHAM computer program developed by Kaiser Engineers for the LOFT program and has been compared with the LOFT data with good results. Westinghouse will continue comparisons as new data becomes available.

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

(1) J. Parmakiar, "Water-Hammer Analysis," Prentiss Hall, (1955).

(2) V. L. Streeter and E. B. Wylie, "Hydraulic Transients, McGraw-Hill, Page 19, (1967).

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 on metastable flow of hot 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 embodied in the methods of analysis wherein the time dependent differential equations, which represent conservation laws, are solved simultaneously by finite difference approximations.

Owing to its capability to handle branching, BLODWN-1 can be used for representation of a 3-dimensional flow, in spite of the fact that the flow model itself is one-dimensional.

This work is being conducted by Westinghouse Atomic Power Divisions in Pittsburgh, Pa. The program to describe the subcooled portion of blowdown has been completed. Extension of the digital computer program to consider two-phase flow is now underway and is expected to be completed by January 1969.

Analyses using BLODWN-1 to evaluate the effects of blowdown forces in existing plants are now in progress. Preliminary results indicate that previous SATAN analyses of these effects were conservative, and so indicate a greater margin of safety of the core and internals design for present plants.

11) Reactor Vessel Thermal Shock Analysis Program

This program considers the effect of the safety injection water on the reactor vessel following a postulated loss-of-coolant accident. As such it is of concern in the highly unlikely event of a large reactor coolant break where the accumulators rapidly discharge into a hot reactor vessel. It must be shown that the injection of this cooling water does not adversely affect the core cooling function. That is, it does not breech the integrity of the reactor vessel such that it is no longer capable of holding water and maintaining the core covered.

The addition of the accumulator system for rapid core flooding following a large loss-of-coolant accident focused more attention on the possible adverse effects of thermally shocking the reactor vessel, particularly the irradiated region. Analyses were initially performed using three failure modes: the ductile yielding mode, the fatigue yielding mode and the brittle fracture mode.

The ductile and fatigue modes involve straight forward analysis indicating that vessel integrity is maintained. The analysis of the brittle fracture mode, however, involves the relatively new method of applying fracture mechanic techniques. Preliminary calculations were performed, taking into consideration information obtained in discussions with the AEC staff and the ACRS, using a simplified analysis and very conservative assumptions. The analysis considered fracture toughness for a highly irradiated vessel corresponding to forty years of operation and instantaneous application of 70°F water on the hot vessel surface. A continuous, circumferential crack was assumed to exist, and the analysis considered the effect of the thermal stress development on this crack. The assumption, considering the continuous crack, was found to be a worse case than an assumed small local flaw and localized crack propagation. The results, while indicating that no vessel failure would occur, were not complete and more rigorous analyses were required to incorporate all the pertinent physical effects.

The results of these analyses are highly dependent on the fracture toughness of the material properties through the vessel wall, which are a function of irradiation and temperature.

A rigorous analysis of the problem was performed using elastic fracture mechanics, and the same initial circumferential crack condition. This analysis used an axisymmetrical finite element digital computer program developed by Westinghouse Atomic Power in conjunction with the Westinghouse Research Laboratory. This program considers the effects of changes in the stress field as the crack propagates into the vessel wall.

Under the most adverse assumptions, the analysis indicates that the crack may penetrate a significant distance into the vessel wall. However, the results are very sensitive to both the fracture mechanics properties of irradiated heavy section steel, and heat transfer coefficients and cooling water temperature. Also, for the crack propagation distance calculated, the linear elastic analysis is considered too conservative since plastic effects must be considered. Another similarly conservative analysis for a shorter irradiation period, where the elastic model is considered applicable, indicate that the vessel integrity will be maintained.

Efforts are being made to obtain the effects of temperature and irradiation on the fracture toughness of the material through the wall of the vessel.

The heavy section steel technology program at Oak Ridge National Laboratory, due for completion by 1973, will also give information on material properties. Westinghouse is also participating in Euratom-funded fracture mechanics program to obtain irradiated fracture toughness properties.

Further investigation is also being performed to determine the sensitivity of various heat transfer coefficients and water temperatures on the results of the analysis.

Preliminary discussions have been held with representatives of the AEC and ACRS. The detailed analysis considering the elastic fracture mechanism method along with the various sensitivity studies is expected to be completed during September, 1968.

The results of the reactor vessel thermal shock program will be used to evaluate the effects of the safety injection water, following postulated loss-of-coolant accidents, on the integrity of the reactor vessel.

Should these analyses show that complete penetration is possible, systems will be added to the plant to cool and cover the core when reactor vessel integrity is lost.

ITEM 5

A comprehensive description of the quality control procedures to be followed during the construction of Unit No. 3 was not provided in the PSAR. The description should include, as a minimum, an outline of methods, procedures, and frequency of inspection, the organization responsible for inspection, the authority of this organization, and the documentation of results.

ANSWER

GENERAL

Herein is described the applicant's quality assurance program for the IPP #3 project. This program is comprehensive and covers all phases of construction both off and on site. Included are all areas of activity which have influences on plant integrity, including design (drawings and specifications), manufacture and field erection and installation and all related activities such as cleanliness control, shipment and storage.

The program places special emphasis on the reactor coolant and safety systems, the containment and the other components necessary for the safety of the nuclear portion of the plant. The description which follows delineates the quality assurance organization and procedures but does not repeat the design and specification requirements set forth in the PSAR.

There are three principal organizations active in the area of Quality Assurance to insure the safety and integrity of the completed plant: Consolidated Edison, the applicant-owner, Westinghouse, the prime contractor and United Engineers and Constructors, the Architect-Engineer-Constructor and a sub-contractor to Westinghouse. The direct activities of each organization are in turn supplemented by the quality control activities of other organizations, either in the form of specific subcontracts for this purpose or as a part of their responsibility as a supplier

of material or equipment, conforming to specification. Where one of these organizations does not have the first line quality control responsibility, a surveillance (auditing and monitoring) function is performed by the contractor in any contractor-subcontractor relationship.

Westinghouse has the prime responsibility to provide all material and equipment for all construction. This responsibility is in turn dispatched through various channels. Westinghouse is the direct supplier of some equipment while in other cases it is supplied by way of purchase from other manufacturers. All of the construction materials and the remainder of the equipment are purchased by United Engineers. The relationship among these organizations is shown on Figure 1.

The scope and nature of the quality assurance functions of each of these three participants and their relationship to one another are discussed as follows.

CONSOLIDATED EDISON

While much of quality assurance activity will be carried out by the Prime Contractor and the Architect-Engineer, Consolidated Edison recognizes that it has ultimate responsibility in this area.

Con Edison's quality assurance function includes monitoring the Westinghouse and United Engineers efforts in critical areas through an independent detailed vendor surveillance by the United States Testing Company and company engineers during selected phases of in-shop manufacturing and on-site surveillance during the plant construction phase. In addition, there is continuous on-site surveillance by Con Edison personnel experienced in the construction field.

USTC keeps informed of the manufacturing status of critical items and makes visits from time to time to the various manufacturing facilities on behalf of Con Edison. Physical and chemical certifications are reviewed

for compliance with applicable specifications. Test procedures involving radiograph, ultrasonics, etc. are reviewed against accepted standards and fabrication techniques are reviewed for general shop practice. Selected tests and inspections are witnessed and test results including radiographic films are reviewed. Where procedural non-conformances or questionable test results of a minor nature are found, USTC requests an on-the-spot procedural revision or retest. Where procedural nonconformances or questionable test results of a significant nature are found, USTC contacts the quality control engineering representative at Con Edison who assesses the situation. Westinghouse is contacted directly by Con Edison for necessary corrective action when required. Written reports of all USTC surveillance visits are forwarded to Con Edison and Westinghouse.

Con Edison maintains on-site a permanently assigned Superintendent of Construction and his full-time staff whose prime function is to ensure that the on-site work is accomplished in accordance with the contractual requirements, and to ensure that the on-site quality control programs are being properly implemented.

The superintendent has the authority to stop work in any area that he considers can affect the technical adequacy or safety of the plant.

Day to day progress is closely monitored by the superintendent. When major work is in progress USTC personnel supplement the Con Edison Personnel. Concrete pours in the containment structure are witnessed and concrete samples are tested for physical, chemical and mechanical properties. Rebar placement, cadwelding, liner welds and installation of critical systems components and piping are witnessed and the necessary inspection reports are reviewed. Written reports of all USTC surveillance visits and concrete sample tests are forwarded to Con Edison and Westinghouse.

WESTINGHOUSE

In the capacity of Prime Contractor, Westinghouse is responsible for the provisions of all material and equipment and for all construction. In discharging this responsibility, Westinghouse recognizes the importance of Quality Assurance throughout all stages of design, fabrication and construction, and accordingly maintains a comprehensive overall Quality Control Program. This program assures that design, engineering, materials and workmanship employed in the fabrication and construction of the facility meet safety, operability, maintainability and reliability objectives previously established. Particular emphasis is placed on the Nuclear Steam Supply System and other critical features.

Complete records and documentation pertinent to the required quality of materials, workmanship and testing are maintained by the Quality Control and Reliability Section.

Organization

The Organization Chart for Westinghouse relating to Quality Control is shown in Figure 2.

Component Design and Quality Control Groups have the responsibility for insuring off-site Quality Control, i.e., up to and including component fabrication and dispatching for shipment.

The Manager of Quality Control and Reliability is responsible for establishing and implementing the overall quality control program through three specialized Quality Control units each headed by its own manager. There are two mechanical Quality Control groups, one is responsible for vessels and tanks and the other is responsible for the balance of the mechanical equipment. There is also a Quality Control group for all electrical instrumentation and control equipment. There are senior Quality Control Engineers in each unit who have expertise in specific categories of equipment and who carry out the Quality Control Engineering

planning. These engineers, supplemented by the efforts of Quality Control representatives, also do surveillance of the suppliers.

The Nuclear Power Service Group has the responsibility for insuring on-site Quality Control, i.e., from receiving all equipment and materials on-site, through erection, to plant start up.

In addition, the Reliability Group performs an independent audit function over all fabrication Quality Control functions as well as over all of the on-site Quality Assurance Functions.

The Reliability Unit is staffed by experienced specialists in electrical and mechanical equipment design, testing, and installation. These personnel serve to audit the design, quality control and site installation activities of the entire PWR division.

All of the Groups are staffed by capable and experienced engineers who collectively provide the experience and effort to implement the overall Quality Control Plan to assure the quality of the finished plant.

Components Supplied By Westinghouse

All major components are supplied by Westinghouse, either directly through Westinghouse equipment manufacturing division or by way of purchase from other manufacturers.

There are four stages in the quality control program during component fabrication to ensure the required degree of quality of the finished product. They are:

A. Supplier Evaluation
B. Equipment Specifications
C. Purchase Order Review
D. Supplier surveillance during fabrication

A. Supplier Evaluation

An evaluation of prospective suppliers is conducted prior to award of a contract for important components. This evaluation establishes that the supplier has acceptable design, manufacturing and quality control capability. Elements considered in conducting the evaluation include:

1. Previous experience with the supplier
2. Physical plant facilities
3. Quality control program
4. Personnel qualifications
5. Material control and inspection
6. In-process inspection
7. Assembly and test capability
8. Tool and gage control
9. Special processes required
10. Non-destructive testing
11. Inspection and test equipment

Responsibility for this evaluation is a function of the Quality Control Groups.

B. Equipment Specification

Individual equipment specifications cover all aspects of equipment design, manufacture, inspection and testing. For Class I components, such as those in the reactor coolant system, a specification which defines the supplier's quality control requirements is made a part of each purchase order. Specific requirements may include:

1. Drawings and change procedures.
2. Procedures and revisions covering welding, heat treating and other process control documents.
3. Inspection plans covering contracted and sub-contracted work.

4. Identification and disposal of non-conforming material or components.
5. Quality control records.
6. Material control.

These requirements are similar to those of Appendix IX to Section III of the ASME Boiler and Pressure Vessel Code.

Responsibility for assuring conformity with these requirements is a function of the Quality Control Group.

C. Purchase Order Review

Purchase orders, including the applicable drawings, welding specifications, non-destructive test procedures and other process control documents required to manufacture, inspect, and test the equipment are reviewed by Westinghouse Engineering, Material, and Quality Control personnel to make sure they include all contract requirements, meet applicable codes and quality control requirements and are compatible with the supplier's capabilities. In cases where existing inspection techniques are found to be inadequate to assure requisite quality, discussions are held with the engineers concerned, and the necessary adjustments are made. This procedure is applied to all types of material and components and provides the means for maintaining surveillance over the quality of procurements.

D. Supplier Surveillance

The Design Engineer/Quality Control Engineer team develops specific Quality Control Plans which detail the inspections, surveillance, record verification, and surveillance which Westinghouse Quality Control personnel perform in suppliers' plants. They cover the entire cycle from purchase order placement through manufacture, inspection and test. These plans are developed as follows.

The many requirements of the equipment specification and its referenced specifications that establish the quality of equipment are listed in a Quality Requirements Summary for each category of equipment. This serves as a working tool from which the Westinghouse Quality Control Plan is developed. This Quality Control Plan is the action plan which details the inspection and surveillance done by Westinghouse Quality Control Engineers and field representatives in the suppliers' shops. It covers the auditing of the supplier's quality control organization, system, and procedures; surveillance of key shop operations such as welding, non-destructive testing and production testing, specific inspections such as radiographic, ultrasonic, material test reports, key dimensions, and other important requirements.

The design engineer also visits the supplier's plant periodically and monitors overall compliance with specification requirements. Particular emphasis is placed on inspections of the first units of fabricated new designs and from new suppliers.

Through the Westinghouse visit to the supplier's plant, the supplier's Quality Control system, in-process work, testing and records are evaluated to assure that he meets contract requirements. The extent of surveillance performed in a supplier's facility depends on the complexity of the component, the supplier's past performance, the observed effectiveness of the supplier's own quality control procedures, the relative importance with respect to safety of the component being fabricated, and whether the component is of an established or new design. The Quality Control Plan emphasizes the basic responsibility of the supplier to have systems and procedures to meet the requirements of the specifications and drawings and to control the quality of his product. The supplier is required to inspect and test his products and to furnish objective evidence that control requirements have been met. Direct evidence of performance is obtained by actual examinations by Westinghouse Quality Control representatives. For example, close attention is given to welder qualification

to ASME Code, non-destructive testing records of both supplier and subvendor, and records of heat treating. The Westinghouse in-process control is aimed at minimizing or preventing defects or errors during fabrication.

In the cases of the reactor vessel, Westinghouse has a full time resident inspector at the manufacturing facility.

Each supplier is required to inspect and test his products and to furnish evidence that all requirements of the contract have been met. This evidence includes proof of (1) conformance to specifications for raw materials and those manufacturing steps through which they have passed, (2) the acceptability of manufactured and purchased items, and (3) the accuracy of all test equipment used in evaluating product quality. Westinghouse inspection engineers review the supplier's quality control procedures and record keeping for conformance to Westinghouse requirements. Critical examination is made of the supplier's quality control records, reports, and inspection certificates gathered during production. Direct evidence of conformance is obtained by product inspection.

In this manner, Westinghouse determines whether or not a supplier has met the specified requirements through the use of objective evidence which is obtained by or from the supplier, and then verified and evaluated by Westinghouse inspection engineers.

Shipment of Components

The detailed requirements for preparation of equipment for shipment are included in the "Equipment Specifications." These include sealing of all openings, protection of nozzle preparations, the use of desicants if required, etc. Where required, the suppliers submit detailed plans for review and approval.

The reactor vessel supplier will provide a cover and seal system to protect all internal surfaces and external stainless steel and machined surfaces from exposure to ambient environments during shipment, storage at the site, and installation. The protective means will be either: 1. Pressurized inert gas with covers or 2. Multiple series of removable surface film and barriers with covers.

For the reactor internals, the lower assembly will be shipped on an up-ending skid, shock-mounted to limit loads transmitted to the assembly during shipment. Prior to installation onto the skid, the lower internals will be wrapped in a plastic film and sealed. Internal bracing will be used inside the assembly. The upper internal assembly will be shipped in a shock mounted dual purpose shipping assembly stand in the vertical position. This package will also be wrapped and sealed in a plastic film. Both the skid and the stand will have a protective metal covering to provide weather protection and long-term storage protection at the site.

All other components will have protection, as required, against mechanical or environmental damage during shipment and/or site storage.

Inspection and Installation of Equipment in the Field

For components and equipment supplied by Westinghouse or its subcontractors specifications are prepared not only for design manufacturing, cleanliness requirements and shipment, but also specifications and procedures are provided for on-site storage, erection, quality control and testing.

During component installation, Westinghouse Nuclear Power Services provides a capable and experienced group of specialists to monitor all construction related activities on the Nuclear Steam Supply System, Engineered Safeguards and Critical Structures. This group is staffed to provide coverage in all phases of construction such as welding, mechanical, electrical,

systems, instrumentation and control, and startup. The primary responsibility of this staff is to insure proper erection of the Nuclear Steam Supply System, Engineered Safeguards and Critical Structures as outlined by Westinghouse specifications and procedures. This surveillance includes visits to selected shops of suppliers to the Architect Engineer-Constructor to insure that established procedures of inspection and documentation are properly followed. Secondary functions of this staff will be to provide technical direction and assistance to the constructor during critical operations and to insure that adequate documentation is being maintained.

The Nuclear Power Service Staff is augmented by the assignment of a resident Quality Assurance Engineer. This engineer is responsible for quality and documentation of all construction activities on the Nuclear Steam Supply System. He will provide additional surveillance of critical operations, follow problems or deficiencies until disposition, aid staff specialists in performance of their duties when necessary, and monitor construction records for completeness.

In addition to the independent control and surveillance activities performed by the engineers at the site, the Reliability Engineering unit of the Engineering Department is responsible to audit the construction site to assure that proper procedures, documentation and controls are being followed.

Non-Conforming Components or Material

All non-conforming components or material, whether discovered at a supplier's facility or at the construction site, are documented, reviewed and disposed of as follows: All details pertinent to the non-conformity are shown on applicable forms. In all cases, the non-conforming component or material is positively identified and separated from acceptable items or items awaiting inspection. All cases of non-conforming components

or material are reviewed by Westinghouse design and Quality Control engineers for resolution. Westinghouse's management is kept informed of all cases of major importance with recommendations for proper disposition.

Quality Control Records

A complete set of quality control records is maintained for each component by its manufacturer and/or purchaser and preserved for Westinghouse. These records include certified test reports, letters of compliance, product inspection reports, non-destructive test reports, and reports of non-conforming material. Records of personnel, procedures and equipment qualifications are also maintained as required by Westinghouse's specifications. Completed records for the components are transmitted periodically to Westinghouse so that all required data will be available for the completed plant.

Audits

All phases of the design review activity, manufacturing quality control, and construction quality assurance in the PWR Plant Division are routinely audited by the Reliability Engineering Function. Further, the overall PWR Plant Division quality assurance program, its systems, procedures and controls, as described above, are audited periodically by the corporate Manufacturing Controls Section. Their findings are reported formally to division management.

UNITED ENGINEERS AND CONSTRUCTORS

In the capacity of Architect-Engineer, United Engineers and Constructors has the responsibility for the design of all systems and structures which are not designed by Westinghouse as a part of the Nuclear Steam Supply and associated Engineered Safeguard Systems. In addition, United Engineers specifies and purchases all equipment within their scope of design responsibility. Furthermore, they prepare all construction drawings and specifications and manage all construction work.

In its construction management capacity, UE&C will carry out much of the "first line" on-site quality control, including receipt inspection, identification, on-site storage, and initial inspection and testing during erection. Receipt inspection will be carried out to determine whether the particular item is ready for installation including checking for damage, sealing, completeness, cleanliness, and the presence of all documentation required of the supplier. Equipment will be labeled or segregated, where appropriate, to assure that proper identification is maintained.

If erection cannot proceed immediately, the small items are placed in a permanent warehouse, and the very large items are stored outdoors, off the ground, and covered. Openings remain sealed until erection except when further inspection or pre-erection work may be required; afterwards, they are resealed until installed.

The requirements for the highest grade commercial cleanliness which can be obtained practically are observed during construction. Cleanliness specifications have been prepared with full awareness of the constraints imposed by the field conditions. The necessity of removing foreign material which could cause difficulties during operation is stressed. Gross dirt and debris are removed continually from the building area during erection. Equipment is protected as required and kept reasonably clean, but final cleaning is reserved until last at which time systems that will contain main coolant or are connected to the main coolant system will be cleaned and rinsed with demineralized water.

The equipment and materials are installed in accordance with prescribed erection procedures. These procedures include such items as sequence of installation when required and specifications for welding, paying particular attention to methods that are not standard to the construction industry. Included in the welding specifications are non-destructive tests, such as dye penetrant and radiography.

The work is done by craftsmen skilled in their respective trades. Welders are given the necessary qualification tests as required by the applicable codes.

UE&C maintains an on-site quality control group which is independent of construction management and which monitors the construction activity at the site. This is described in more detail hereinafter.

Organization

The Organization Chart for UE&C relating to Quality Control is shown in Figure 3. The Quality Control Section is staffed by capable and experienced personnel and is divided into two principal operations: Home Office and Field.

The Quality Control Operation is headed by the Manager of Reliability and Quality Assurance in the Home Office who reports directly to the Vice President, Administration.

A Home Office Quality Control Engineer is assigned to the projects reporting to the Supervising Engineer in matters relating to project operating control and reporting directly to the Manager of Reliability and Quality Assurance in matters relating to policy and technical control. His responsibility is to establish the quality assurance plan for United Engineers scope of project responsibility, assure adequacy of specifications, audit suppliers facilities and procedures and perform liaison duties between Home Office and Field Quality Control groups.

The Field Quality Control group is supervised by the Field Supervisor - Quality Control who reports to the Construction Manager in matters relating to project operating control and is responsible to the Manager of Reliability and Quality Assurance in matters relating to policy and technical control.

Quality Control Engineers assigned primarily to one basic project phase, i.e., (1) structural, (2) mechanical, (3) electrical and instrumentation and controls, and (4) piping, will report to the Field Supervisor - Quality Control.

Quality Control inspectors are assigned to various phases of the project as required and are supervised by the appropriate Quality Control Engineer.

Records

Records consist of items such as:

1. receipt inspection and storage inspection reports
2. installation procedures
3. non-destructive testing reports and radiographs
4. deficiency reports

The Field Quality Control group is responsible for maintaining records pertinent to and necessary for the quality control of materials and workmanship for UE&C activities on the project. These records are maintained throughout construction and final testing. At the completion of the project these records will be turned over to the Owner. Throughout the course of the project, these records are available for review by the Owner, regulatory agencies and/or their authorized representatives.

The Home Office Quality Control Engineer and the Field Supervisor -Quality Control develops the "filing system."

The filing system is maintained in a numerical order. Numbers are the same as the applicable specification which is also the purchase order number for the equipment or service applicable. As the total system is completed, all records pertinent thereto will be assembled in a "system package" and so maintained for submittal to the Owner.

Piping welding records are by system and a system package contains, but is not necessarily limited to material certifications, shop inspection and test documentation, performance test records, field welding data sheets, non-destructive testing reports, radiographs and isometrics. Again these records are maintained as the work progresses and data assembled into the final package when the system is completed. Where an item is used "across the project" (such as reinforcing steel), separate records are maintained in a "General" category.

Details of Quality Control and Inspection Work

A. UE&C Purchased Items - Vendor Surveillance

Where vendor plant surveillance is necessary, a program is organized and implemented according to the following plan:

1. Review purchase order to determine that quality control requirements are contained or referenced therein.

2. Review vendor's Quality Control Procedures for the subject item(s). Approve these or request changes.

3. Make a Manufacturer's Facilities Survey to assure vendor's capability and to determine his understanding regarding Quality Control documentation and test requirements. Complete the "Manufacturer's Facilities Survey" report.

4. Determine extent of shop inspection necessary to assure compliance with terms and intent of the order. Complete "Work Sheet for Testing Requirements and Reports".

5. During vendor inspections, in addition to the specific purpose of the inspection (e.g., witnessing shop tests or reviewing radiographs), the Quality Control representative shall:

a. Audit applicable mill test reports and certifications for full compliance to specification requirements.

b. Audit welding procedures and special techniques.

c. Review individual welder qualifications for conformance with applicable ASME, AWS or UE&C codes.

d. Review techniques and procedures for required inspection methods and tests.

e. Determine that vendor quality control procedures are being carried out as approved.

6. Prior to any vendor surveillance, the Field Supervisor Quality Control or the Home Office Quality Control Engineer prepares an "Inspection Check List" for the Quality Control representative making the visit. This list contains specific details to be checked during the visit, including dimensional tolerances, test procedures, test pressures, applicable codes, etc.

The Quality Control Engineer ensures that the Quality Control representative reviews the necessary codes, specifications, and drawings pertaining to the inspection.

B. Scheduling

Scheduling of quality control work is done with complete coordination with the construction, scheduling, engineering and purchasing groups connected with the project. Use of the CPM print-outs is made in this regard for long range scheduling, using separate sortings where helpful.

Copies of shipping schedule distribution and piping progress sheets for items are reviewed thoroughly as updated so that quality control schedules are kept on a current and correct basis.

C. Site Quality Control - Structural

1. Concrete Materials Approvals and Mix Design - Materials entering into production of structural concrete for the project are sampled and tested in full accordance with project specification requirements and applicable ASTM and ACI standards prior to any placement or concrete in the permanent structures. Using approved materials, mix proportions are determined in accordance with project requirements and ACI 613.

2. Mill test certificates are required and reviewed for cement used on the project. Separate storage facilities are maintained at the batch plant so that no mixing of cement types or brands takes place.

3. Concrete aggregates are periodically sampled and tested. The batch plant is inspected to ascertain that only previously approved aggregates are used.

4. The concrete batch plant is inspected for conformance to ASTM C-94. During concreting operations, the batch plant is checked to determine that approved mixes are being batched and all batching is being done in accordance with project specifications.

5. All concreting operations are inspected in the field including forms, reinforcing, placing, curing and stripping. Slump tests, air content checks, and concrete cylinders are made as required.

6. All concrete materials testing, design mixes, batch plant inspection, site inspection, and cylinder testing are performed by an independent testing laboratory hired by the contractor. The Field Quality Control Organization shall oversee these areas to assure compliance with specifications.

7. Reinforcing Steel - Mill test reports are required and reviewed covering each heat of material supplied. Material is inspected upon receipt at the site for identification marks and condition. Fabricated high strength bars are checked for cracking at bends when received at the site.

8. Cadweld Splices - Mill test reports covering sleeves and cartridges are required. These items in the storage areas are checked to assure that no deterioration has taken place. All operator qualification records are maintained. All production splices are visually inspected and samples randomly selected for testing to destruction. A plot plan of all splices is maintained showing splice location and number, operator number, position, site, date made and date inspected. A log of all test results is maintained and results are charted. The Structural Engineer is informed of all test results and results without the specification are "red flagged."

9. Structural Steel - Mill test reports are required and reviewed. Material is inspected when received at the site for proper identification and condition.

   Where shop fabrication is required, a vendor surveillance program is set up as described previously.

   Erection of structural steel is subject to inspection as required. Connections are particularly checked for soundness and tightness.

10. Field Welding of Structural Steel - Welding procedure test and operator certifications are checked for conformance to applicable ASME, AWS or UE&C specifications. Only certified welders are allowed to perform structural welding on the permanent portion of the structure. Previous qualification of welders may be accepted as certification for the project if allowed by the governing code or specification.

    Field welds are inspected and where required, non-destructive tests are performed, witnessed or reviewed by a representative of the Quality Control group.

    Radiographs are reviewed and a report of the tests is maintained in the Quality Control file.

11. Containment Liner - Particular attention is placed on the fabrication of the vapor containment liner. Mill test reports for liner plates are checked and become a part of the permanent quality control file.

    Welding procedures and welder qualifications are checked and certifications become a part of the permanent quality control file.

    Weldings and non-destructive testing are audited and non-destructive test reports are reviewed and become a part of the permanent quality control file. These tests include the leak testing of the pressure channels.

    Procedures for the final containment pressurization and test are reviewed and this test is witnessed. Records pertinent to the test become a part of the permanent quality control file.

D. Site Quality Control - Mechanical Equipment

1. Receiving Inspection - Items are checked when received at the site for shipping damage, sealing and generally to determine if the item is ready for installation. All records and documentation required from the supplier must be on site and acceptable before the item is accepted for installation. (This includes shop inspection reports noted in Vendor Surveillance section above.)

2. The Field Supervisor - Quality Control prepares a check list of inspection requirements, limiting factors, and acceptance criteria. These inspections must be completed and results satisfactory prior to acceptance of an item into the structure.

3. Any non-destructive tests (including pressure and leak rate) required by the specifications are performed, witnessed, or reviewed by a member of the Quality Control group. Radiographs are reviewed and a report of the tests is maintained in the Quality Control file.

4. Field welding on mechanical equipment is subject to the same criteria as noted under "Field Welding of Structural Steel." Testing requirements of these welds is considerably more extensive for the mechanical portion than for the structural portion.

E. Site Quality Control - Mechanical - Piping

1. Receiving Inspection - Piping is checked when received at the site for damage, sealing, and cleanliness (where required). All records and documentation required from the supplier must be on site and acceptable before any piping item is accepted for installation.

2. Welders and welding procedures are certified in accordance with ASME Code requirements. For those welders or procedures requiring field certifications, tests for certification are witnessed by a representative of the Quality Control Group. Qualification records become a part of the Quality Control file.

3. Field welds in the critical systems and others where required are numbered and these numbers will be the permanent identification of the joint. Numbering of these joints is the responsibility of the Welding Supervisor and he provides the Quality Control Engineer with a list of these numbers and the location of the joints to which they apply.

4. Non-destructive tests (including pressure and leak tests) are performed, witnessed, or reviewed by a member of the Quality Control group. Radiographs are reviewed and the "Piping System Inspection Record" completed. This record and all radiographs become a part of the Quality Control file when the system is completed.

5. The cleanliness of piping systems is carefully checked in full accordance with project specifications.

6. During the course of field welding on the critical piping systems, a "Record and Schedule of Field Welds" is maintained and available for checking progress. It is the responsibility of the Welding Supervisor to maintain this record.

F. Site Quality Control - Electrical, Instrumentation and Controls

1. Receiving Inspection - Items are inspected when received at the site for damage to the crating or equipment and for proper identification. All records and documentation required from the supplier must be on site and acceptable before the item is accepted for installation. This includes all shop inspection (usually limited to witnessing of acceptance tests) reports noted in Vendor Surveillance section above.

2. The inspection of installation is primarily a check upon the workmanship in mounting and wiring the equipment to code and specification requirements.

3. Pre-operational checks - After installation, components are tested for their proper electrical and mechanical functions without energizing the equipment. These tests are conducted to applicable USAS, IEEE, and NEMA standards and the "Start-up Check List for Electrical Equipment" as prepared by the Electrical Engineer and approved. On critical or other major items, the tests are witnessed by a representative of the Quality Control group.

4. Operational Checks - After successful pre-operational tests, equipment is further checked for proper mechanical and/or electrical functions with the equipment energized. Tests are conducted in accordance with applicable USAS, IEEE, and NEMA standards and the "Start-up Check List for Electrical Equipment" as prepared by the Electrical Engineer and approved. On critical or other major items, the tests are witnessed by a representative of the Quality Control Group.

INDIAN POINT NO. 3

PARTICIPATING ORGANIZATIONS IN QUALITY ASSURANCE

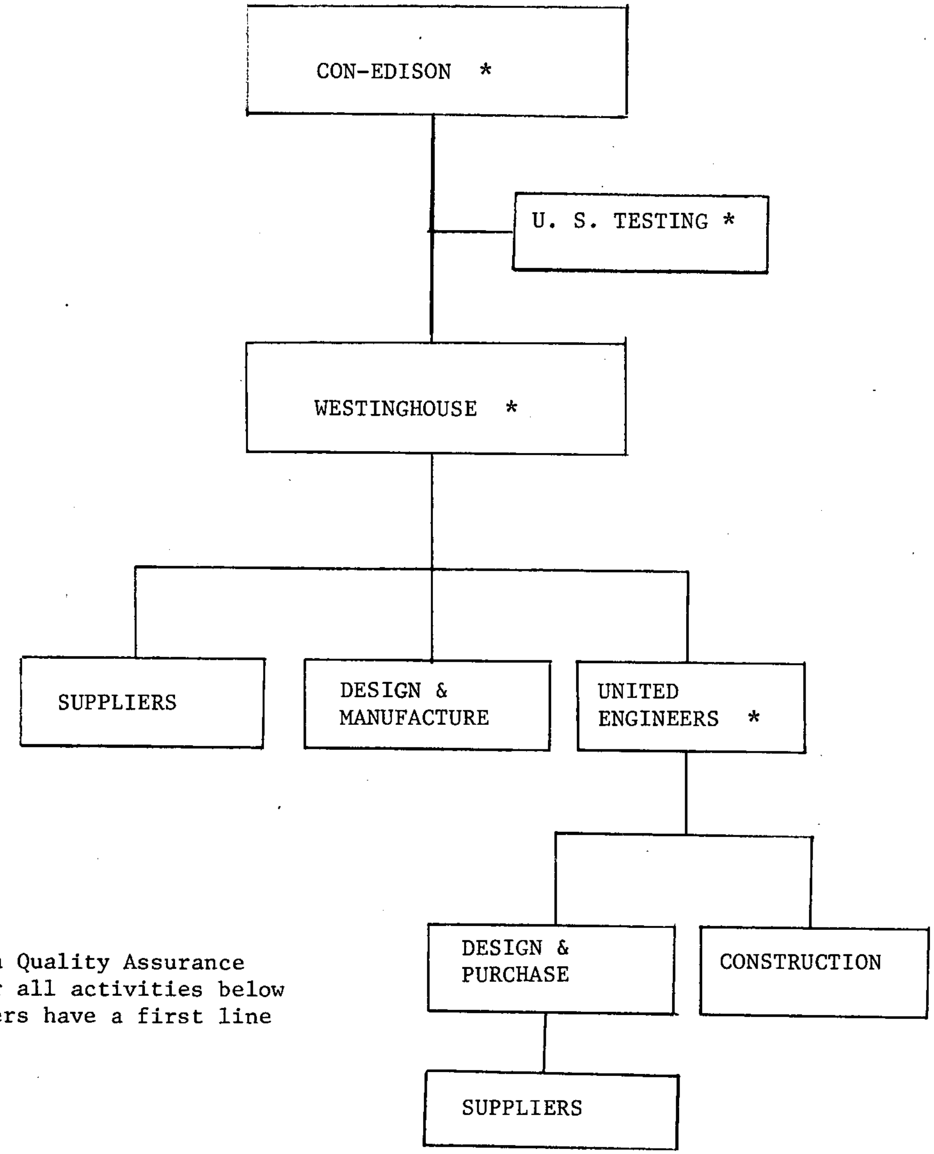

FIGURE 1
Supplement 1

* Indicates performance of a Quality Assurance surveillance function over all activities below this level of chart. Others have a first line responsibility.

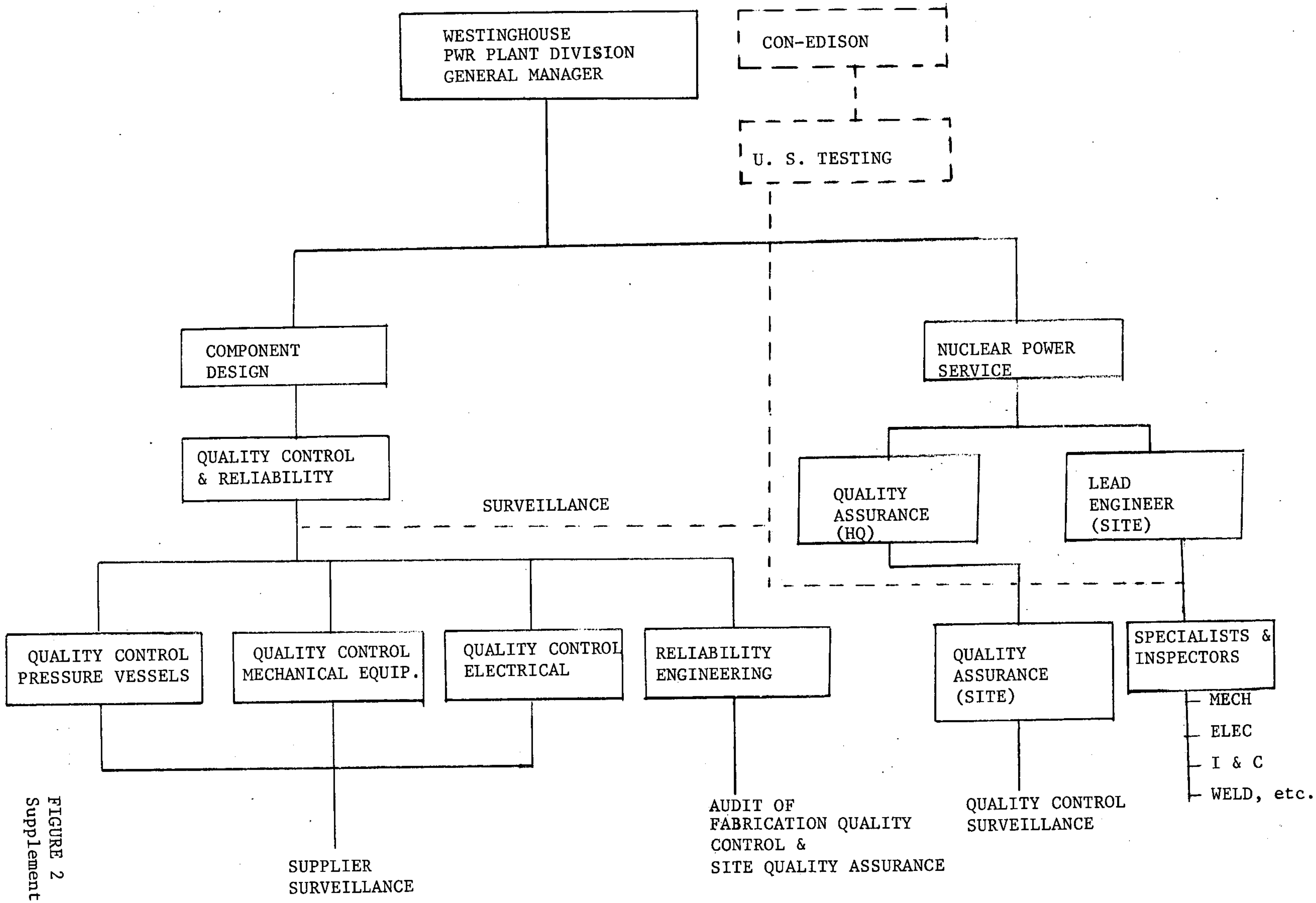

FIGURE 2
Supplement 1

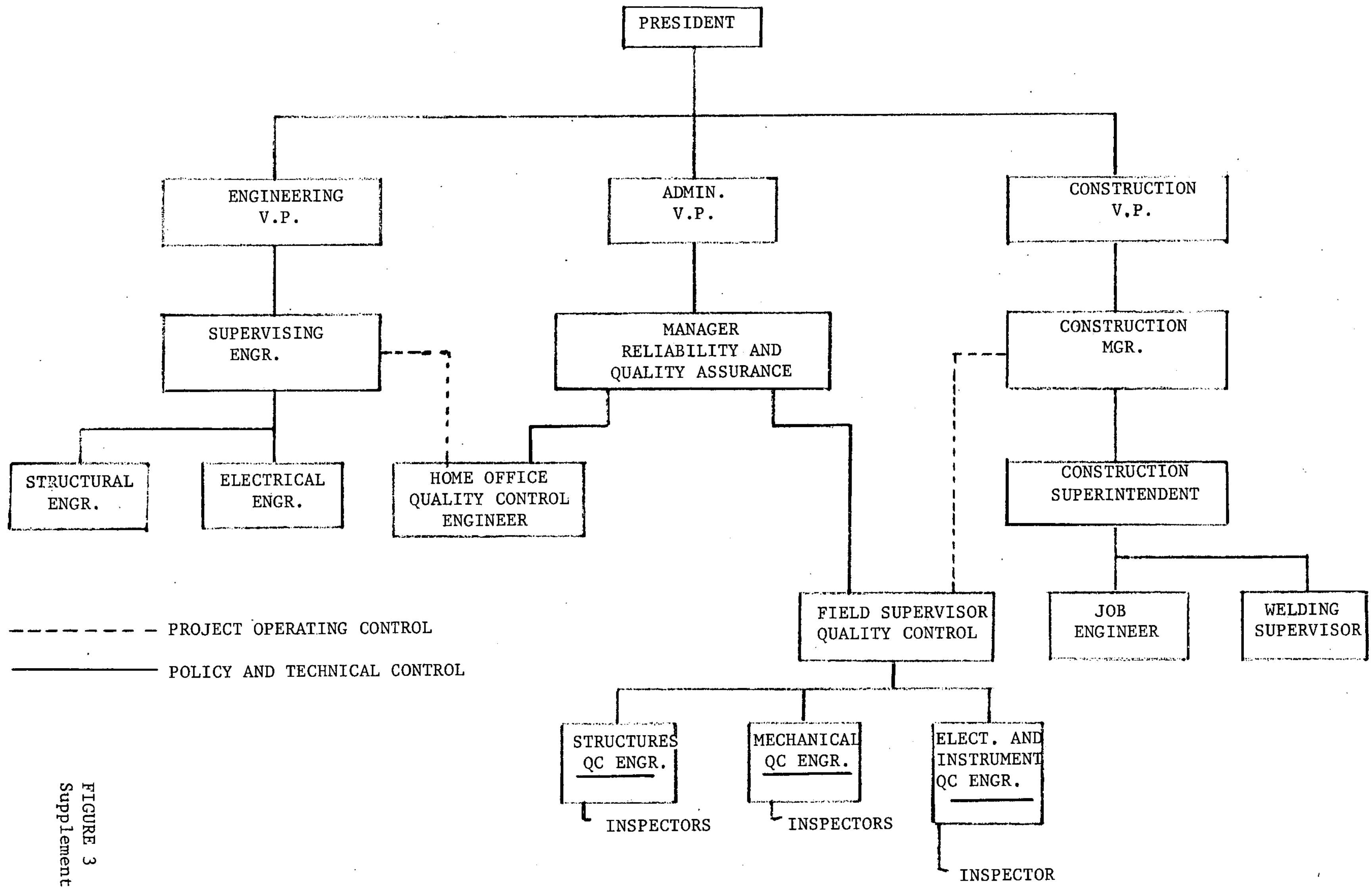

FIGURE 3
Supplement 1

UNITED ENGINEERS AND CONSTRUCTORS
PROJECT ORGANIZATION
QUALITY CONTROL RELATIONSHIPS

ITEM 6

Design criteria provided are inadequate regarding the ability to accomplish a safe and orderly plant shutdown in the event of a tornado. The design criteria for the tornado should (1) indicate the design wind loading and pressure drop considered, and the basis for their selection; (2) identify the equipment which will be designed to withstand these loadings; and (3) discuss the ability of the plant components and systems to withstand tornado-originated missiles. In this respect, our criteria are that structures important to safety should be designed for tornado winds corresponding to 300 mph tangential velocity, traverse velocity of 60 mph, and a differential pressure drop of 3 psi in 3 seconds, with stresses limited to 90% of yield stress in steel and 85% of ultimate stress in concrete. Missiles associated with tornado winds should also be considered.

Answer

The plant is safeguarded from tornadoes by the combined use of buildings and structures designed to withstand tornadoes and of redundancy of components. The following buildings and structures will be designed to withstand tornado winds corresponding to 300 mph tangential velocities, traverse velocities of 60 mph and a differential pressure drop of 3 psi in 3 seconds with no loss of function:

- Control building
- Primary auxiliary building
- Containment
- All connecting ducting for essential cabling and piping

Tornado wind loads are converted to equivalent static structural loadings in accordance with the applicable portions of the wind design methods described in ASCE Paper No. 3269 "Wind Forces on Structures." The provisions for gust factors and variation of wind velocity with height do not apply. The following factored load equation is used for those structures designed to resist tornado wind effects.

$$C = (1 \pm 0.05)\ D + 1.0\ W' \qquad (1)$$

where:

C = Required load capacity of section.

D = Dead load of the structure plus any normal operating live loads.

W' = Tornado wind load to include the pressure drop effect where applicable

The stress criteria used for this load criterion will be no gross yield of the primary structure with the yield stress levels revised by the capacity reduction factors as defined in Section 5.1.2.4 of the PSAR. Redistribution of stress in redundant structures due to plastic hinge formation is allowed.

The above buildings and structures will also be designed to withstand various postulated tornado-generated missiles, including the following:

a) 4" x 12" x 12' plank at 300 mph

b) 4000 lb passenger car at 50 mph not exceeding 25 ft. above the ground.

All components and equipment essential to safety will be housed within the tornado-proof structures described above, with the following exceptions. For these components and systems, adequate tornado protection is provided by redundancy.

1) Redundancy is provided for the vital 480 volt system by three independent systems. Onsite there are three emergency diesel generators; offsite there is a 138 kv above-ground system and a 6.9 kv underground system.

2) The emergency feed requirements of the steam generators are assured by tornado protected pumps and redundant water supplies.

3) The water requirements of the primary system are assured by the availability of primary water storage tank, the refueling water tank and the boric acid tanks.

4) Service water supply is assured by redundancy of two supply lines, four screens and six pumps of which only two pumps, one screen and one supply line are required for prolonged shutdown. The intake structure itself is tornado proof.

The effect of tornados on the spent fuel pit is currently being evaluated; unless it can be demonstrated that the effects are insignificant, tornado protection will be provided for the spent fuel pit.

Three general criteria have been adopted for the design of Indian Point No. 3 plant in tornado conditions.

1) A tornado will not cause a loss-of-coolant accident.
2) A tornado will not impair the ability to safely shut the plant down.
3) A tornado following a loss-of-coolant accident will not impair the long term safety of the plant.

<u>Criterion I</u>

The reactor coolant system is contained entirely within the confines of the containment vessel. For the Tornado to cause a loss of coolant accident the Tornado or Tornado produced missiles must penetrate the containment vessel. The design is such that penetration of the containment vessel is not considered to be credible.

<u>Criterion II</u>

There are two phases of reactor shutdown that must be considered; a shutdown to hot shutdown condition and a shutdown to cold condition.

<u>Shutdown to hot shutdown condition</u> - the reactor requires a number of basic services when held for an extended period in the hot standby condition:

x

(a) Residual Heat Removal

(b) Reactivity Control, i.e., as fission poisons decay

(c) Pressurizer Pressure and Level Control

(d) Auxiliary Building and Control Room Ventilation

(e) Electrical Systems.

These services require that a number of systems and equipment will continue to operate following a tornado.

(a) Residual Heat Removal

Following a normal plant shutdown an automatic steam dump control system 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 may be controlled both locally and remotely from the control room. In the event of loss of compressed air, local operation would be adopted.

For continued operation of the electric feed pumps the 480 volt system must be assured. This is discussed under item (e).

In addition, the diesel generator requires the continued supply of fuel oil and service water; adequate redundancy and protection exist for this purpose.

Vital instruments and controls are provided both locally and in the control room.

(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; control may be local or from the control room. 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 water 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.

The vital items of this equipment are housed within the containment and the reinforced concrete auxiliary building, the service water system is protected by means of redundancy. In order to guarantee the operation of the system the 480 volt system must again be assured.

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 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 although the essential systems are considered to be adequately protected within the auxiliary building and containment building.

(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 safely protected within the P.A.B.; it is only necessary to supply water for makeup. Water may readily be obtained from separated sources; that in the volume control tank, boric acid tanks, monitor tanks, primary storage tank, and refueling water storage tank.

Similarily to the two previous service requirements the 480 volt system must be assured with the additional electrical load of the pressurizer heaters. Vital instruments and controls are provided both locally and in the central control room.

(d) Ventilation

The most essential ventilation requirements apply to the containment since in order to guarantee the satisfactory operation of the instrumentation and central systems the containment air temperature must be controlled to a tolerable level. This system again requires the satisfactory operation of the service water and electrical systems.

(e) Electrical Systems

Protection is provided for the 480 volt switch gear and supply redundancy is provided by the diesel generators, gas turbine generator, the two above-ground incoming lines and the one below ground incoming line. The 6.9 kv system is fed by either the 13 kv gas turbine generator or by an undergound 13 kv feeder from the Buchanan substation. The Buchanan substation consists of four buses.

Shutdown to cold condition - Plant cooldown is not an immediate requirement following major damage due to a tornado. For a cooldown the basic services required are:

(a) Residual Heat Removal
(b) Reactivity Control
(c) Pressurizer Pressure and Level Control
(d) Ventilation
(e) Electrical Systems

A cooldown would not be attempted until full equipment facilities had been guaranteed.

Criterion III

Following a loss of coolant accident the residual heat is removed through internal recirculaton conditions with the facility for external recirculation if required. This duty implies the continued operation of the auxiliary cooling water system together with the associated electrical and service water supplies. The recirculation systems are protected by the tornado proof containment and auxiliary buildings. The electrical and service water systems are assured by redundancy as previously discussed.

ITEM 7

A safety analysis and the results thereof should be provided relative to the ability of the facility to accommodate the consequences of an explosion or fire in the gas pipeline which passes through the site. This discussion should indicate the location and proximity to the site of any gas pumping station near the site.

ANSWER

The gas transmission system passing though the Indian Point site consists of two parallel mains installed in separate trenches 20 feet apart. Indian Point is situated approximately 4 miles from the gas pumping station located in Stoney Point, New York.

A review of the design and construction of the mains, as well as the operational and maintenance procedures employed by the Algonquin Gas Transmission Company, will support the contention that a failure of either pipe at Indian Point would be a highly unlikely accident.

DESIGN AND CONSTRUCTION

The design and construction of both mains followed the guidelines of the American Standard Code for Pressure Piping, Gas Transmission and Distribution Systems (ASA 31.8), and the New York State Safety Code which in many respects, has more stringent requirements than the ASA Code.

It should be noted that the Federal Power Commission's Report on the Safety of Interstate Natural Gas Pipelines, dated April 1966, revealed that less than 42% of the nation's transmission line mileage was subject to any state safety regulation when installed. Of the 1,058 failures reported in the nation's 204,730 miles of pipelines, only 13 (1.2%) occurred in states with ASA Code and State Codes more stringent than ASA Code requirements, of which New York is one. These states have 9% of the nation's pipeline mileage.

Out of the 1,058 failures, 74 (7%) were reported to have resulted in fires. Of these 74 only 1 (.09%) occurred in states with the more stringent safety regulations than the ASA Code. This failure was caused accidentally by a bulldozer. Due to the stringent inspection procedure of the Algonquin Transmission Company, it has only experienced one failure of its pipelines since 1950, which occurred in New Jersey, and was caused by a ship's dragging anchor.

The first pipeline was installed in Indian Point in 1952; the second line in 1965. Both pipes are made of 52,000 psi minimum yield strength steel, conforming to the American Petroleum Institute Specification 5LX52.

The 1952 pipe has an outside diameter and wall thickness of 26" and 0.281" respectively. Hoop stress calculations on a pipe of these dimensions and material show that the pipe is capable of withstanding internal pressures of 1125 psi before yield point stresses develop.

The outside diameter and wall thickness of the 1965 pipe are 30" and 0.438" respectively. This pipe is calculated to withstand internal pressures of 1520 psi before yield point stresses develop.

Both mains have welded joints. As a quality control measure, a random number of these joints were X-rayed to establish the soundness of the welds. However, the most convincing proof of the weld quality and the adequacy of the pipeline design, was the ability of each main to sustain the pressure generated during the hydrostatic tests.

The 1952 line was tested to its yield strength and was able to withstand the 1125 psi internal pressure. The 1965 line was tested to 92% of yield, 1390 psi, and also withstood the internal pressure.

Considering the results of these tests, it is safe to assume that in a new state, the pipes could sustain internal pressures at least the equivalent of the hydrostatic test pressure.

It then becomes relevant to examine those factors that could cause an overstress in the pipe and subsequently lead to a pipeline failure at Indian Point. The following factors must be considered:

a) An internal pressure increase beyond yield.

b) Stresses induced in the pipe due to settlement.

c) Weakening of the pipe walls due to corrosion.

To eliminate the danger of a pressure buildup, a network of relief valves, set to operate at 750 psi, have been installed in both pipelines. Since these relief valves will prevent a pressure increase beyond 750 psi, the pressure in the 1952 line will never esceed 67% of the hydrostatic test pressure and the 1965 one will never exceed 54% of the hydrostatic test pressure.

Since both mains at Indian Point have been installed in trenches cut through rock, stresses due to settlement cannot occur.

Two protective measures have been taken to prevent a weakening of the pipe due to corrosion. To prevent corrosion, the pipes were given two coats of coal tar enamel, followed by a wrapping of fiberglass fabric and a third coat of coal tar enamel. Before backfilling the trenches the coatings on both pipes were tested to insure against weak spots in the coating or possible areas overlooked during the coating process. In addition a network of cathodic protection has been provided for both mains.

As described hereinafter the pipelines are operated and maintained with the same regard for safety shown in the design.

OPERATION AND MAINTENANCE

Although the 1952 line was designed to operate at a pressure of 750 psi, the New York State Public Service Commision, in August 1953, directed the Algonquin Gas Company to limit the operation of the main to a maximum pressure of 650 psi. Algonquin complied with this directive and the main now operates at only 58% of yield. The 1965 line was designed and is now operating at 750 psi, which is only 48% of yield.

It is well to note that much of the nation's transmission line milage operates at 72% of yield, considerable higher than the operational levels at Indian Point.

In the field of maintenance, Algonquin has employed a more comprehensive program than the average industry wide practice. To check for leaks, Algonquin conducts a monthly foot patrol and a bi-weekly airplane patrol of the mains. Under this surveillance, any leak that might develop would be detected before a hazardous conditon could arise. In addition to the patrols, Algonquin performs monthly tests on all of the relief valves and automatic shut-off valves in the system to make certain taht the valves function properly. A monthly check is also made of the cathodic protection system.

Under the heading of maintenance, it should be mentioned that measures have been taken to avoid the most common cause of pipeline failure which is an accidental puncturing of a main by construction or farm equipment.

Although the mains are over 300 feet from the closest point of Unit #3 construction site, as an additional safety measure, Consolidated Edison has staked out the exact location of the mains and signs have been installed warning heavy equipment to stay clear. However, once Unit #3 is in operation, construction at the site will be completed and the possibility of construction damage to the mains will no longer exist.

In the light of the foregoing discussion, it can be concluded that conditions which might lead to a pipeline failure have either been provided for in the design of the pipes, or do not exist at the Indian Point site.

However, postulating a pipeline failure at Indian Point, two possibilities must be considered. The first possibility would be a rupture or explosion of the main, but with no fire occurring. This has been the most common situation according to the Federal Power Commission's Safety Report. In the event of an explosion, protection must be provided against concussion damage and missile damage in the form of flying pipe fragments. The distance of the plant from the mains will provide adequate protection for both cases.

As a matter of reference, the following table sets forth the shortest distances between the pipeline and the nearest corner of the proposed buildings:

	Feet
Turbine Hall Heater Bay	630
Service Building	540
Diesel Generator Building	470
Primary Auxiliary Building	480
Fuel Storage Building	570
Containment	660
Refueling Water Stroage Tank	400
Primary Water Storage Tank	380
Hold-up Tank Pit	410

Regardless of distances however, all critical structures of Unit #3 are being designed to resist earthquake damage and tornado generated missile damage, and will therefore be resistant to concussion damage and damage due to flying pipe fragments resulting from a gas main explosion.

The second case to consider would be a fire resulting from the explosion. The fire would most likely be in two stages; a primary stage consisting of the burning gas, and a secondary stage consisting of fires caused by the burning gas.

The primary fire would be of short duration since automatic shut off valves would isolate the ruptured section of the main within 4 minutes. Those valves are located at both banks of the Hudson River and at Gomer Street in Yorktown, 10.4 miles from the plant. The secondary fire would be set in the trees surrounding the gas mains. It should be noted that even if the rupture occurred at the closest point to the plant and the wind

blew the flames toward the plant, it is extremely unlikely that the flames would reach the plant 400 feet away.

The most important fact to note when considering the separation distances is that there is over 100 feet of clear space between the plant and the closest row of trees. This clearance serves as an ideal firebreak to prevent a fire from spreading to the plant.

Even if a fire could reach the plant, there is no flammable material in the building construction that could burn. The Containment Building, Primary Auxiliary Building, the Control Room, and Waste Hold-up Tank Pit are all of massive concrete construction and could survive a fire of almost any duration. The Refueling and Primary Storage Water Tanks are normally filled with water and are therefore fire resistant. The remaining buildings are of steel frame construction covered with metal wall siding. Even if exposed to the direct flames, records indicate that the insulation in the siding will afford 15 to 20 minutes of fire protection for the structural steel. This is longer than the primary fire could continue to burn 500 feet from its source. The only building in which the collapse of the structural steel could create a hazard is the Fuel Handling Building. However, it is situated behind the Primary Auxiliary Building, and the Refueling and Primary Storage Water Tanks.

CONCLUSION

On the basis that the failure of either gas main at Indian Point would be a highly unlikely accident, that the Unit #3 structures are constructed of non-flammable materials, and the arrangement of the plant and the distance of the plant from the mains provide substantial protection from fires originating at the mains, it must be concluded that the presence of the gas transmission lines at Indian Point does not endanger the safe operation of Unit #3.

ITEM 8

A discussion was not included relative to the probability of significantly reducing cooling water flow from the river by blockage of the intake by debris during a flood, or as a result of collapse of the intake structure or its foundations during an earthquake.

ANSWER

The arrangement of the main circulating water pumps and service water pumps are similar to IPP#2. The service water pump suction chamber is located between two main circulating water pump chambers.

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 pump 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.

It is concluded therefore, that blockage of the intake by debris during a flood does not pose a problem.

As stated in Appendix A of the PSAR, the intake structure is designed as Seismic Class I, and is therefore not subject to collapse under earthquake loading.

Item 9

We understand from oral discussions with your representatives that burnable poison and part-length control rods are planned which would (1) limit the moderator temperature coefficient of reactivity to a negative value and (2) improve the spatial stability of the core to xenon oscillations. Since use of such modifications has not, as yet, been completely documented, your application should be amended to describe and analyze these modifications in terms of their effect on plant safety. This analysis should include a discussion of the effect of these modifications on the threshold for hydrodynamic instability and your consideration of the ability to determine local heat flux. In this respect, we believe that more extensive in-core instrumentation may be required and require your analysis of this necessity.

ANSWER

Refer to topical reports WCAP 7113 (previously submitted as Appendix B to Indian Point Unit No. 2 PSAR, Supplement 7, Docket 50 - 247) and WCAP 7072 (previously submitted as Appendix C to Diablo Canyon Site, Pacific Gas and Electric Company PSAR, Supplement 1, Docket 50 - 275).

Item 10 (Attachment A Question 1.0)

ATTACHMENT A

1. Discuss the "error band" or accuracy to which the following can be determined:

   a. Core inlet temperature
   b. Mass flow rate
   c. Inlet pressure
   d. Engineering hot channel factors
   e. Local heat flux
   f. Total core thermal power
   g. Relate each of the above to an error band on the DNB ratio and combine the effects of these variables to give an overall "error band" on the minimum DNB ratio.

ANSWER

a. Core inlet temperature can be determined to within $\pm$ 2°F from the T_{avg} and ΔT instrumentation, including readout errors. There is an additional 2°F allowance for control errors and deadband, so that inlet temperature is maintained within $\pm$ 4°F in steady-state operation.

b. Reactor coolant flow is not a variable and is not directly measured to high absolute accuracy. Conservative design and startup test verification insure that the flow is not less than design.

c. Pressurizer pressure is measured to an accuracy of $\pm$ 10 psi. The core inlet pressure is not directly measured, but is about 50 to 75 psi higher than in the pressurizer because of the static head and dynamic pressure drops in the core and reactor vessel outlet. For conservatism, core pressure is generally assumed equal to pressurizer pressure. In steady-state operation, the pressure control system maintains pressure within $\pm$ 30 psi of the set point including the $\pm$ 10 psi measurement error.

d. The design total hot channel factors which are used in core thermal analysis already include engineering hot channel factors to account for variations in fabrication tolerances and flow conditions. The engineering heat flux hot channel factor (See INT PSAR Section 3.2.2.2) can be evaluated statistically from the measurements taken on actual fuel assemblies. A probability limit of three standard deviations is used to evaluate this factor for design at a 99.9% confidence level. Discussion of the enthalpy rise engineering hot channel factor is referred to the PSAR Section 3.2.2.2.

e. Local heat flux is the product of the core power and the hot channel factors. The hot channel factors at power are maintained at less than their design value as verified by periodic incore maps.

f. In steady-state, the total core thermal power can be determined to within ± 2% error by steam cycle calorimetric measurements. During a transient, changes from steady-state can be determined from the nuclear flux and ΔT indications.

g. The effect on the DNB ratio of using the lower envelope of the above-mentioned "error band" is presented in the following table.

	% Decrease in DNB ratio from nominal design value
a) Core Inlet Temperature ($T_{nominal}$ + 4°F)	4.2
b) Mass Flow Rate,	0
c) Inlet pressure ($P_{nominal}$ - 30 psi)	1.3
d) Engineering Hot Channel Factors ($F_q^E = 1.04$, $F_{\Delta H}^E = 1.075$)	0.
e) Local Heat Flux	0.
f) Total Core Thermal Power (102%)	3.

A conservative overall analysis assuming all of the above errors act at once results in a minimum DNB ratio of 1.67 versus the minimum DNB ratio at nominal conditions of 1.82.

Item 10 (Attachment A Question 2.0)

Discuss the effect a small increase in power would have on the number of channels experiencing bulk boiling. Present in tabular or graphic form indication of number of channels experiencing bulk boiling vs percent of full power for power levels up to 125% of full power.

ANSWER

Under nominal system conditions the core does not experience bulk boiling until 115% of full power occurs. At 120% of nominal power 200 flow channels experience bulk boiling, approximately 0.5% of the core. The increase in the number of flow channels experiencing bulk boiling is essentially linear in the range from 115 to 125% of nominal power. The results are summarized in the following table.

Power (% of 3025 MWt)	Number of Flow Channels Experiencing Bulk Boiling
100	0
110	0
115	0
120	200
125	375

Item 10 (Attachment A Question 3.0)

Indicate the statistical number of fuel rods experiencing DNB assuming heat flux and enthalpy hot channel factors 110% of design value and power levels of 100, 105, 110, 112, and 125% of design value.

ANSWER

The statistical number of rods which could experience DNB assuming heat flux and enthalpy hot channel factors 100% of design value is tabulated below for a range of power levels. Nominal system pressure, inlet temperature and flow were used in the analysis. Also presented are results using the design hot channel factors ($F_{\Delta H}$ = 1.70, F_q = 2.82) which are themselves approximately 10% in excess of the best estimated values.

Power Level, % of Nominal	Statistical Number of Rods That May Experience DNB*	
	Hot Channel Factors 10% in Excess of Design Values	Design Hot Channel Factors (Approximately 10% in Excess of Best Estimate Value)
100	1.5	0.3
105	5.4	0.9
110	20.0	2.8
112	32.0	5.5
125	503.0	88.0

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

Item 10 (Attachment A Question 4.0)

State the power levels at which (1) a minimum DNB ratio of 1.30 is reached, assuming design hot channel factors, and (2) fuel center melting is expected to occur.

ANSWER

For nominal system pressure, inlet temperature, and flow a power level of 119% of nominal may be reached before the minimum DNB ratio of 1.30 will be reached. Fuel center melting (5000°F) is expected to occur at 142% of nominal power.

Item 10 (Attachment A Question 5.0)

Indicate the peak fuel burnup anticipated. Relate this exposure to fuel clad integrity considering fuel expansion, fuel fission gas release, and clad corrosion. Consider the effect that flaws or defects sufficiently small to escape inspection during fabrication would have on clad integrity. Cite applicable experience which justifies any claims made.

ANSWER

The peak fuel burnup anticipated in IPP-III is between 45,000 and 50,000 megawatt days/metric ton Uranium. The fuel mechanical analysis studies are related to these peak burnup conditions. The effects of fuel expansion, fission gas release, clad corrosion, fuel swelling are considered in these studies. Even under these peak conditions, the fuel element design criteria are met and no problems with fuel clad integrity are anticipated. Because of the stringent manufacturing and inspection techniques, no flaws or defects are anticipated during fabrication. The helium leak check performed is usually adequate to insure that small defects are not present in the fuel elements. However, it is possible because of operator error or strange circumstances that a small defect could escape inspection during fabrication.

To evaluate the effect of a small defect, experimental studies have been performed in the Saxton reactor. Zircaloy clad fuel rods were purposely defected with a 15 mil diametral hole through the cladding at the peak heat flux region. Two rods were tested, one which was unirradiated at the beginning of the test and another which had been previously irradiated to 10,000 megawatt days/metric ton Uranium. The rods operated for approximately 60 days at a peak power rating of approximately 13 kw/ft. Examination of the clad in the defect region and in sections above and below the defect indicated that the defect had not propagated. These results indicate that even if Zircaloy rods containing small defects are inserted into a reactor, it is anticipated that no catastrophic failure would result.

Item 10 (Attachment A Question 6.0)

Provide the fuel pellet temperature profile and the fuel clad pressure time history used in calculating fission gas release from the fuel at the end-of-life.

ANSWER

The average pellet temperature and the internal fission gas pressure generated as the function of core life are illustrated in Figures 1 and 2, respectively. The discontinuties indicate the transfer of elements during refueling. These results are calculated using the best estimated power distribution history data.

AVERAGE FUEL TEMPERATURE vs. TIME

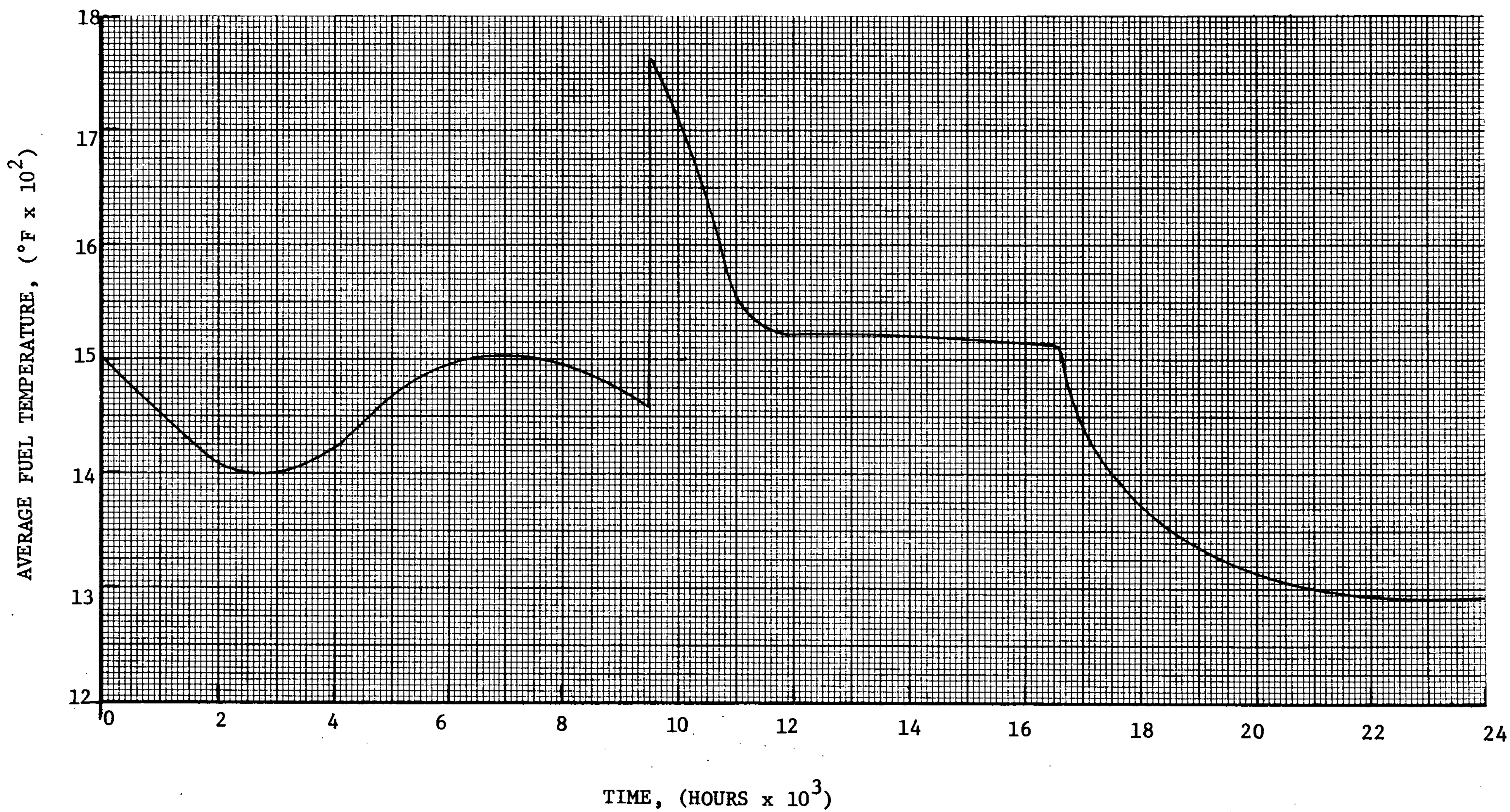

FIGURE 1

INTERNAL PRESSURE vs. TIME

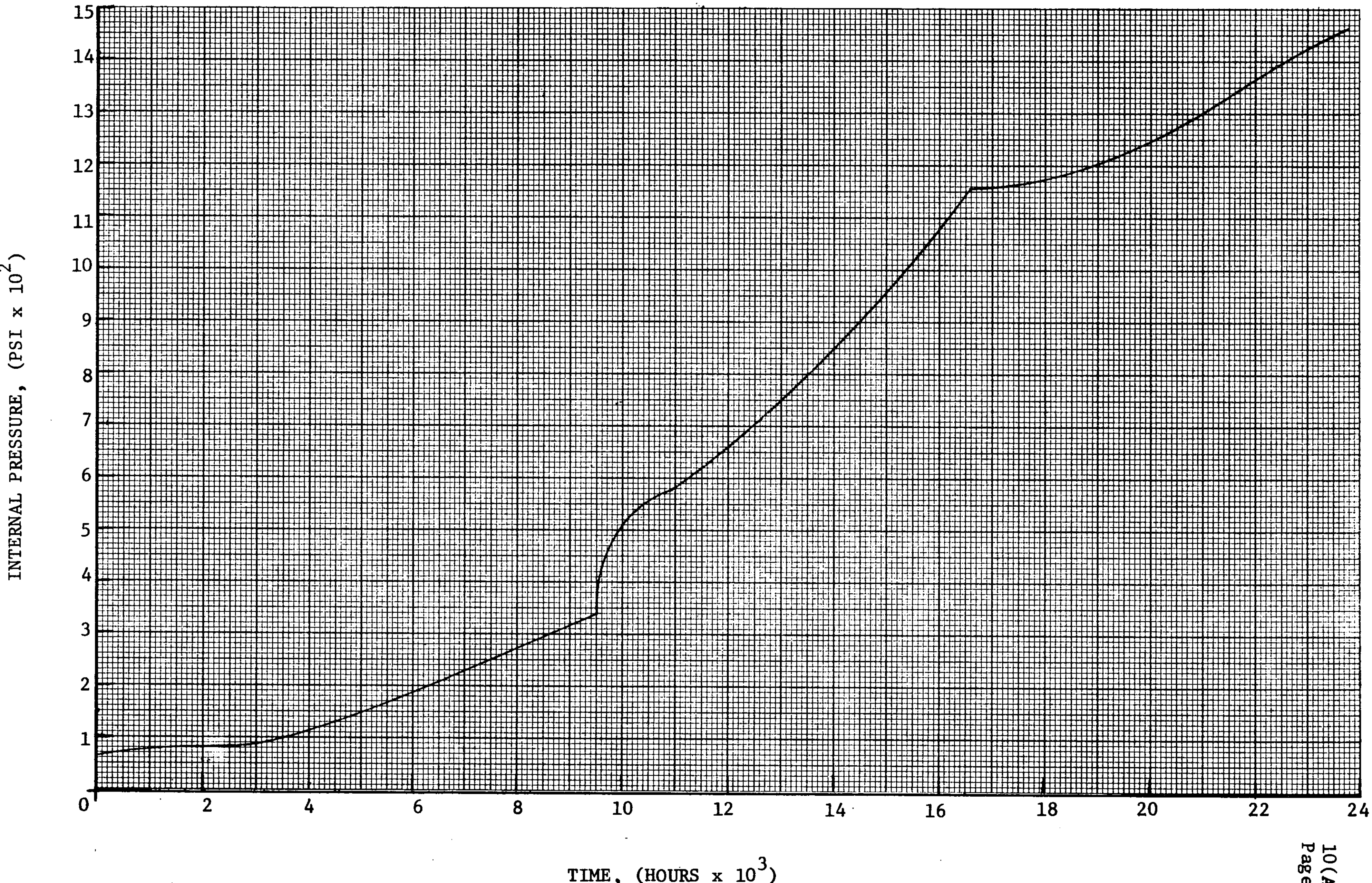

FIGURE 2

Item 10 (Attachment A Question 7.0)

Present and discuss the margins available in clad stress and strain and in peak fuel temperature between the limiting values and the values of these parameters when the core is operating at 112% of full power at the end-of-life.

ANSWER

The core is not operated at 112% under normal operating conditions and this condition would only be reached under accident conditions. The clad stresses and strains and the peak fuel temperature at 112% of full core power at the end-of-life are summarized as follows:

	Maximum at 112% Power at End-of-Life	Design Limit Value	Margin
a. Clad Stress	40,500 psi	46000 psi	5500 psi
b. Clad Strain	0.2%	1.0%	0.8%
c. Peak Fuel Temperature	2400°F	4740 °F	2340°F

The above values do not necessarily occur simultaneously in the same rod. The margins for each of the above parameters are deemed adequate.

ITEM 11

There was a deficiency of information in the PSAR concerning the construction quality assurance and inspection programs of nuclear vessels and piping. We regard such programs as an important step in assuring safe function of nuclear plants. There was no discussion of such items as structural support design, construction, installation and alignment, and alignment tolerance checks to assure that the reactor vessel, turbines, and steam generators will be properly installed and aligned and remain so following earthquake or other postulated accident. In developing this area, you should note the Commission's Tentative Regulatory Supplementary Criteria for ASME Code-Constructed Nuclear Pressure Vessels dated August 23, 1967, and provide and discuss the following.

a. Provide a tabulation of all the nuclear pressure vessels in the Class I (seismic design) systems in the facility. The tabulation should include a notation of whether the vessel design is complete, the stage of fabrication of the vessel, and the extent to which each of the vessels will comply with each of the 34 supplementary criteria.

b. For each vessel, provide a discussion that presents the reasons why total compliance is not feasible for each criterion not met in its entirety.

ANSWER

Appendix 4A includes pertinent information on support structure for Reactor Coolant System components and Primary System component interactions. Section 6.2.1.6 of the PSAR (page 6-20) contains information concerning inspection and installation of equipment in the field. Appendix A referred to in Appendix 4A is at this time included as the answer to Item 15. Construction and installation of reactor vessels, turbines, steam generators, etc. are done by written procedure. Design tolerances are written into this procedure and records are kept by a quality assurance group at the plant site. Item 5 is a comprehensive description of the quality control program for the Indian Point Unit No. 3 plant.

The tabulation requested in parts (a) and (b) is given in Table 1.

APPENDIX 4A

REACTOR COOLANT

Support Structures for System Components and Primary System Component Interactions.

The Reactor Coolant System Components are designed as Class I seismic components as discussed in Appendix A. The design of their supports consider seismic loadings.

Reactor Vessel

The reactor vessel support structure consists of a circular box section ring girder, fabricated of carbon steel plates. The bottom flange of the girder is in continuous contact (except for openings for neutron detectors) with a non-yielding concrete foundation.

The reactor vessel has four supports located at alternate nozzles. Each support bears on a support shoe, which is fastened to the support structure. The support shoe is a structural member that transmits the support loads to the supporting structure. The support shoe is designed to restrain vertical, lateral and rotational movement of the reactor vessel, but allows for thermal growth by permitting radial sliding at each support on bearing plates.

Steam Generators

The steam generators are supported within a caged structural system consisting of four connected columns all welded together, fabricated of carbon steel members, with provisions for limited movement of the

structure in a horizontal direction to accommodate piping expansion with a system of "Lubrite" plates, hydraulic snubbers, guides and stops. The "Lubrite" plates, hydraulic snubbers, guides and stops are designed as rigid support to resist the action of seismic and pipe break loads.

Reactor Coolant Pump

Each reactor coolant pump is supported on a three-legged structural system consisting of three connected columns fabricated of carbon steel members, structural sections and pipe. Provisions for limited movement of the structure in any horizontal direction to accommodate piping expansion is accomplished with a sliding "Lubrite" base plate arrangement and a system of tie rods and anchor bolts which restrain the structure from movement beyond the calculated limits.

Pressurizer

Pressurizer is supported on a free-standing structural system, consisting of six connected columns fabricated of carbon steel members, all welded together and secured at the base by anchor bolts.

Component Interactions

The criteria for movement of the reactor vessel, under the worst combination of loads, i.e., normal plus 0.15g plus reactor coolant pipe rupture loads, assures that the radial movement of the reactor vessel will not exceed two inches. Preliminary evaluation of the supports and support structures of the major components of the Reactor Coolant System, i.e., reactor vessel and steam generators, shows that movement of these components is only fractions of an inch. More specifically, the maximum reactor vessel displacement is expected to be essentially equal to the deflection of the reinforced concrete structure that supports the reactor vessel. Preliminary evaluation of the proposed design indicates that the deflection of the concrete vessel support structure will be limited to 1/4 inch.

The relative motions between Reactor Coolant System components will be controlled by the structures which are used to support the reactor vessel, the steam generators, the pressurizer and the reactor coolant pumps.

The supports will be designed to limit the stresses in the pipe to the stress limit given in Table 1 of Appendix A.

The design of steam generator stops and other restraining devices will be based on loadings resulting from a dynamic response analysis for ground acceleration.

Stops or travel restraining devices will be used as necessary to limit stresses in the piping connecting the steam generator to the reactor vessel to acceptable values.

Evaluation of the loads on the shell will be performed in accordance with the ASME Boiler and Pressure Vessel Code. The results of Pressure Vessel Research Committee work reported in Welding Research Council Bulletin No. 107 will be used for evaluating the effects of thrust and moment loads on shell connections.

Preliminary evaluation shows that for the relatively rigid components attached to the steam generator shell (such as the supported tubes or tube sheet), the seismic response accelerations contribute negligible loading in comparison with the loadings resulting from the steam generator operating pressure.

TABLE 1

Tentative Regulatory Supplementary Criteria for ASME Code Constructed Nuclear Pressure Vessels

VESSELS IN CLASS I SEISMIC CLASSIFICATION

Component	Code	IPP-3 Status	Compliance With Supplementary Criteria
a. Reactor Vessel	ASME III, Class A	In Fabrication	See Note 1
b. Rod Drive Mechanism Housing	ASME III, Class A	In Design	See Note 1
c. Steam Generators			
Primary Side	ASME III, Class A	In Fabrication	See Note 1
Secondary Side	ASME III, Class C*	In Fabrication	See Note 1
d. Pressurizer	ASME III, Class A	In Design Major material ordered	See Note 1
e. Regenerative Heat Exchanger	ASME III, Class C	In Design	NA
f. Volume Control Tank	ASME III, Class C	In Design	NA
g. Boric Acid Tank	No Code	In Design	NA
h. Seal Water Heat Exchanger			
Tube Side	ASME III, Class C	In Design	NA
Shell Side	ASME VIII	In Design	NA

* 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.

Component	Code	IPP-3 Status	Compliance With Supplementary Criteria
i. Non-Regenerative Heat Exchanger			
Tube Side	ASME III, Class C	In Design	NA
Shell Side	ASME VIII	In Design	NA
j. Mixed Bed Demineralizers	ASME III, Class C	In Design	NA
k. Cation Bed Demineralizers	ASME III, Class C	In Design	NA
l. Holdup Tanks	ASME III, Class C	In Design	NA
m. Gas Stripper Package	ASME III, Class C	In Design	NA
n. Boric Acid Evaporator Packabe	ASME III, Class C	In Design	NA
o. Base Removal Ion	ASME III, Class C	In Design	NA
p. Primary Makeup Water Storage Tank	No Code	In Design	NA
q. WDS Sump Tank	No Code	In Design	NA
r. Spent Resin Storage Tank	ASME III, Class C	In Design	NA
s. Gas Decay Tank	ASME III, Class C	In Design	NA
t. Waste Holdup Tank	ASME III, Class C	In Design	NA
u. Waste Evaporator	ASME III, Class C	In Design	NA
v. Residual Heat Exchanger			
Tube Side	ASME III, Class C	In Design	NA
Shell Side	ASME VIII	In Design	NA

Component	Code	IPP-3 Status	Compliance With Supplementary Criteria
w. Component Cooling Heat Exchanger	ASME VIII	In Design	NA
x. Component Cooling Surge Tank	ASME VIII	In Design	NA
y. Spent Fuel Pit Heat Exchanger			
Tube Side	ASME III, Class C	In Design	NA
Shell Side	ASME VIII	In Design	NA
z. Spent Fuel Pit Demineralizer	ASME III, Class C	In Design	NA
aa. Accumulator Tanks	ASME III, Class C	In Design	NA
bb. Refueling Water Storage Tank	No Code	In Design	NA
cc. Chemical Additive Tanks	ASME III, Class C	In Design	NA
dd. Excess Letdown Heat Exchanger			
Tube Side	ASME III, Class C	In Design	See Note 1
Shell Side	ASME VIII		

NA - Not Applicable

Note 1 - These vessels comply with the supplementary criteria with the modifications proposed by the letter of March 14, 1968, by Mr. J. C. Rengel, Vice President and General Manager, Westinghouse Atomic Power Divisions, which comments on the AEC Supplementary Criteria.

ITEM 12

The PSAR was deficient in discussing or presenting analysis or results of analysis, indicating whether or not the reactor vessel can accommodate, without failure, thermal shock incident upon or induced by operation of the emergency core cooling system at end of design life. Such analysis should have considered both ductile yielding and brittle fracture modes of analysis and the conditions or assumptions, criteria, and bases used in the analysis should have been provided.

ANSWER

Preliminary analyses have been completed and submitted to the AEC for their review and comments. A technical meeting was held with representatives of the AEC technology group and the ACRS and their consultant. The assumptions, methods of analysis and results were discussed and it was concluded that further analyses are required to complete the final evaluation.

Further evaluation was performed to obtain the effects of temperature and irradiation on the fracture toughness, through the wall, of the vessel material, based on available data. Also, further investigation done relative to the effects of various heat transfer coefficients on the results of the analysis.

The results of the above analyses were reviewed with the AEC. A report summarizing these evaluations is currently in preparation.

Section 12.3 will be updated upon completion of the above work. Further discussion of this question is found in the answer to Item 4 on the Research and Development Program.

ITEM 13

Your PSAR does not adequately discuss the effects that organic forms of iodine resulting from fuel meltdown may have on the capability of the proposed iodine removal system. Our preliminary evaluation indicates that off-site radiological consequences of accidents are sensitive to assumptions regarding organic or other forms of radio-iodine in the containment following an accident. Further information is needed to substantiate the fraction of iodine existing in the organic form assumed in your calculations. Recently published literature (BNL-11329) indicates that up to 16% of the total radio-iodine released from UO_2 fuel heated to temperatures of 1000°C to 1300°C in a steam-hydrogen atmosphere may be in an organic form. Means which show promise for removing organic iodine from containment atmospheres are in the early stage of development and will require continued research effort. Considering these facts in conjunction with the relatively high population density about your Indian Point site, our calculations indicate that you will be unable to meet the 10 CFR 100 guidelines for both the two-hour site boundary exposure, and the thirty-day low population zone boundary exposure for a TID-14844 fission product release, unless you can demonstrate that a significant amount of the organic iodine species can be removed from the containment atmosphere by containment sprays or other means. We anticipate that this will mean a research and development program which has progressed to the point where preliminary data is encouraging and which is outlined in sufficient detail so that we can estimate the likelihood of successful completion prior to the issuance of a construction permit.

ANSWER

The report BNL-11329 indicates that up to 16 per cent of the total radio-iodine released from UO_2 fuel heated to temperatures of 1000°C to 1300°C in a steam-hydrogen atmosphere may be in an organic form. The application of the results of this experiment to PWR loss of coolant environment conditions is unrealistic. In recently published literature (BMI-1829, Feb. 22, 1968) it is stated that the mechanism for formation of methyl iodine is largely unknown, and without knowledge of the formation mechanisms of methyl iodide it is almost impossible to predict what its concentration will be in an accident environment. The experiment described in BNL-11329 does not simulate the physical and chemical environment of the containment following a loss of coolant accident. To extrapolate the experiment results to a PWR accident condition is impossible considering the differences. The

major difference in the two environments is the presence of a high radiation field which can change significantly the formation mechanisms and rates for methyl iodide as described in BNL-12219 (January, 1968) and BMI-1829. Furthermore, the concentration of iodine affects the formation rate of organic iodine, the temperature and pressure vary significantly through the beginning of a loss of coolant accident, and the physical size of the reactor containment system is inconsistent with the experimental conditions of the BNL-11329 report. Therefore, the 16 per cent organic iodine release seems inconsistent with the low release expected in a reactor accident situation as indicated in the previous experimental work.

Because of the extremely low probability of an accident involving major fission product release (presupposing lack of timely injection of core cooling water) and subsequent continuous leakage (presupposing failure of the isolation valve seal water or channel pressurization systems), Westinghouse has deemed unwarranted the further conservatism introduced by postulating higher organic iodide yields than appear reasonable based on the most applicable experimental evidence. In order to ensure that this matter is adequately cared for, however, the following lines of action will be pursued.

First, an in-depth review will be made of the thermodynamic and chemical kinetic factors governing CH_3I formation. This study will take particular note of the conditions of radiation and temperature, and of the relative abundance of hydrogen and organic molecules present in the reactor accident environment. The objective of this study will be to formulate a model which explains the variability of observed concentrations of CH_3I found in some experiments and sets a conservative upper limit for such concentrations in the reactor accident environment. When and if this study produces a body of information which in the judgment of Westinghouse and the applicant forms a convincing basis for ruling out the credibility of significant yields of CH_3I in the containment, a presentation to that effect will be submitted for the review of the Regulatory Staff.

As a second and parallel line of action, the design of the Indian Point Unit 3 facility will proceed on the basis that capability may be required to tolerate the formation of CH_3I by 5 per cent of the maximum fission product iodine inventory of the core (i.e., 10 per cent of that assumed transported to the containment atmosphere in the TID-14844 accident model). This capability would take the form of layout space, structural provisions, air handling capability, and all relevant service capacities to incorporate an adsorbing or exchange-trapping bed of the requisite efficiency for CH_3I removal. The reference design of this bed would be that which offers the greatest assurance of performance attainment by present technology; that is, a bed of activated charcoal granules, impregnated with one or more non-radioactive iodine compounds. Such units, which are commercially available, have been proven to have substantial and dependable efficiency for CH_3I trapping at air relative humidities of 90% or less, and in large-scale demonstrations have shown promise at relative humidities up to 100% . The shortcomings of such units are recognized:

a) The introduction of a large quantity of combustible material, exposed to a potentially heavy burden of radioactive contaminants raises the possibility of hot-spots and charcoal ignition should forced air circulation be lost.

b) Waterlogging of the charcoal can cause substantial loss of efficiency.

Both of these shortcomings can be corrected by proper engineering. To minimize the hazards of ignition, the reference design would employ high ignition point charcoal; delayed actuation to permit the inorganic iodine to be essentially completely scavenged by the sprays rather than deposited on the charcoal beds; and reduced charcoal quantities commensurate with the duty of the system to effect only the needed reduction in potential organic iodine leakage. In addition, of course, appropriate devices would be incorporated in the design to detect and extinguish combustion within a bank of charcoal units should it occur.

With respect to waterlogging, appropriate moisture eliminating equipment would be installed to dry and pre-condition the air entering the charcoal units so that the possibility of waterlogging is precluded. Under the proposed design basis, it would be feasible to turn off the containment sprays prior to filter operation, further minimizing the chance of wetting the charcoal.

At a point in the detailed design and procurement program when sufficient flexibility remains for these decisions, a determination will be made of the potential iodine leakage source with only the chemical spray system and no filter. If additional organic iodine removal capacity is needed, based on the organic yield which is justifiable, the adsorber units will be incorporated. At that time consideration will also be given to alternate adsorbent media which may have advantages over charcoal and which may have been developed to a status of proven efficiency during the interim. It is believed, however, that the system as proposed offers an adequate backup with reasonable assurance that all necessary engineering data can be developed on a timely basis.

ITEM 14 (ATTACHMENT C QUESTIONS 1.0-1.4)

In order that we may assess the ability of the fan-cooler units to function as proposed, the following information is requested:

14 (C-1.1)

A preliminary design of the heat exchanger surface and fan assembly including configuration and dimensions.

14 (C-1.2)

Heat transfer performance for the unit for the spectrum of accident conditions including flow rates, temperatures, pressures, and compositions for both steam-air flow and cooling water.

14 (C-1.3)

An outline of the analytical procedures used in designing the heat transfer surface and in determining its performance and the basis for these analytical procedures.

14 (C-1.4)

A discussion of experimental verification of the heat exchanger design and further experimental verification which may be required.

ANSWER

14 (C-1.1)

Each fan cooler unit comprises a demister, cooling coils, roughing and absolute filters, motor and fan. The fan is of the direct driven centrifugal type and the cooling coils are plate fin-tube type comprising copper plate fins vertically orientated on copper tubes. The coils are provided with drain pans and piping to prevent flooding during accident conditions. The condensate is drained to the containment sump.

The cooling coil section will be approximately 12 feet high, 9 feet wide, and 2 feet in the direction of air flow. Drain pans are placed at three foot vertical intervals. The coils will be 116 tubes high and 10 tubes in the direction of air flow. The fins are spaced 8 1/2 per inch on the tubes and the tube thickness is 5/8 inches OD, and are in sections 2 feet by 3 feet.

During normal operation air is drawn thru the roughing filter and cooling coils by the fan. The moisture separator and HEPA filter constitute a separate flow path which is connected to the normal flow path during accident operation.

Each fan is designed to supply 65,000 cfm at containment design condition; 271°F 47 psig and 0.175 lb/ft^3. There is 2000 gpm of service cooling water supplied to each unit during accident conditions. The maximum river water inlet temperature of 85°F results in a maximum outlet temperature of 161°F. The service water system pressure at locations inside the containment is 15-20 psig. Each air handling unit is capable of removing 76.4×10^6 Btu/hr from the containment atmosphere under accident conditions.

Figures 14 (C-1.2)-1 thru 14 (C-1.2)-4 indicated the estimated variation in air recirculation fan-cooler heat removal with changes in significant parameters. The figures are:

Figure 14 (C-1.2)-1 - Heat removal rate as a function of cooling water flow rate.

Figure 14 (C-1.2)-2 - Heat removal rate as a function of inlet cooling water temperature.

Figure 14 (C-1.2)-3 - Heat removal rate as a function of containment atmospheric saturation pressure and temperature.

Figure 14 (C-1.2)-4 - Heat removal rate as a function of steam-air mixture volumetric flow rate.

FAN-COOLER HEAT REMOVAL RATE AS A FUNCTION OF COOLING WATER, FLOW RATE

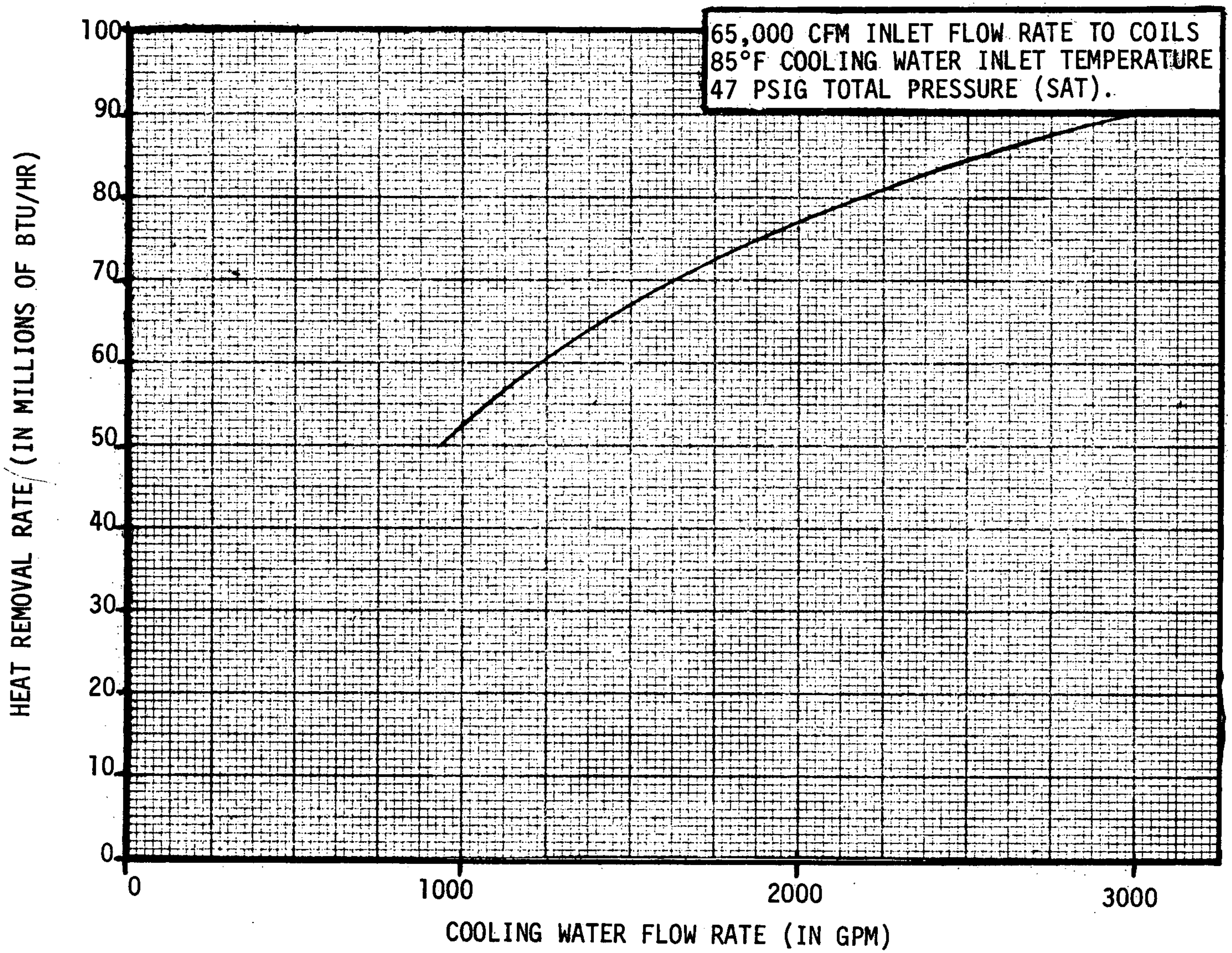

FIGURE 14(C-1.2)-1

SUPPLEMENT 1

FAN-COOLER HEAT REMOVAL RATE AS A FUNCTION OF
INLET COOLING WATER TEMPERATURE

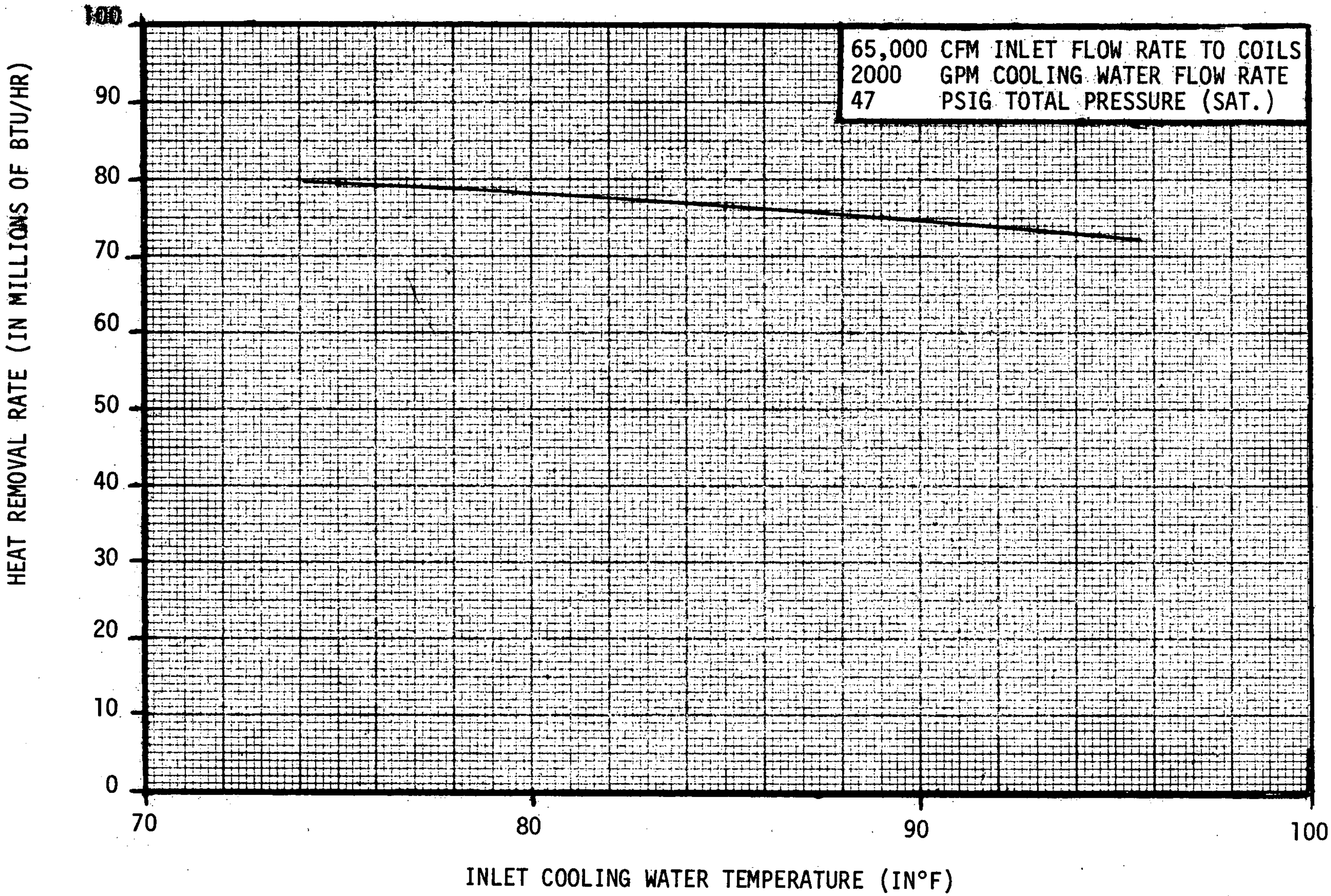

FIGURE 14(C-1.2)-2

SUPPLEMENT 1

FAN-COOLER HEAT REMOVAL RATE AS A FUNCTION
OF CONTAINMENT ATMOSPHERIC SATURATION PRESSURE
AND TEMPERATURE

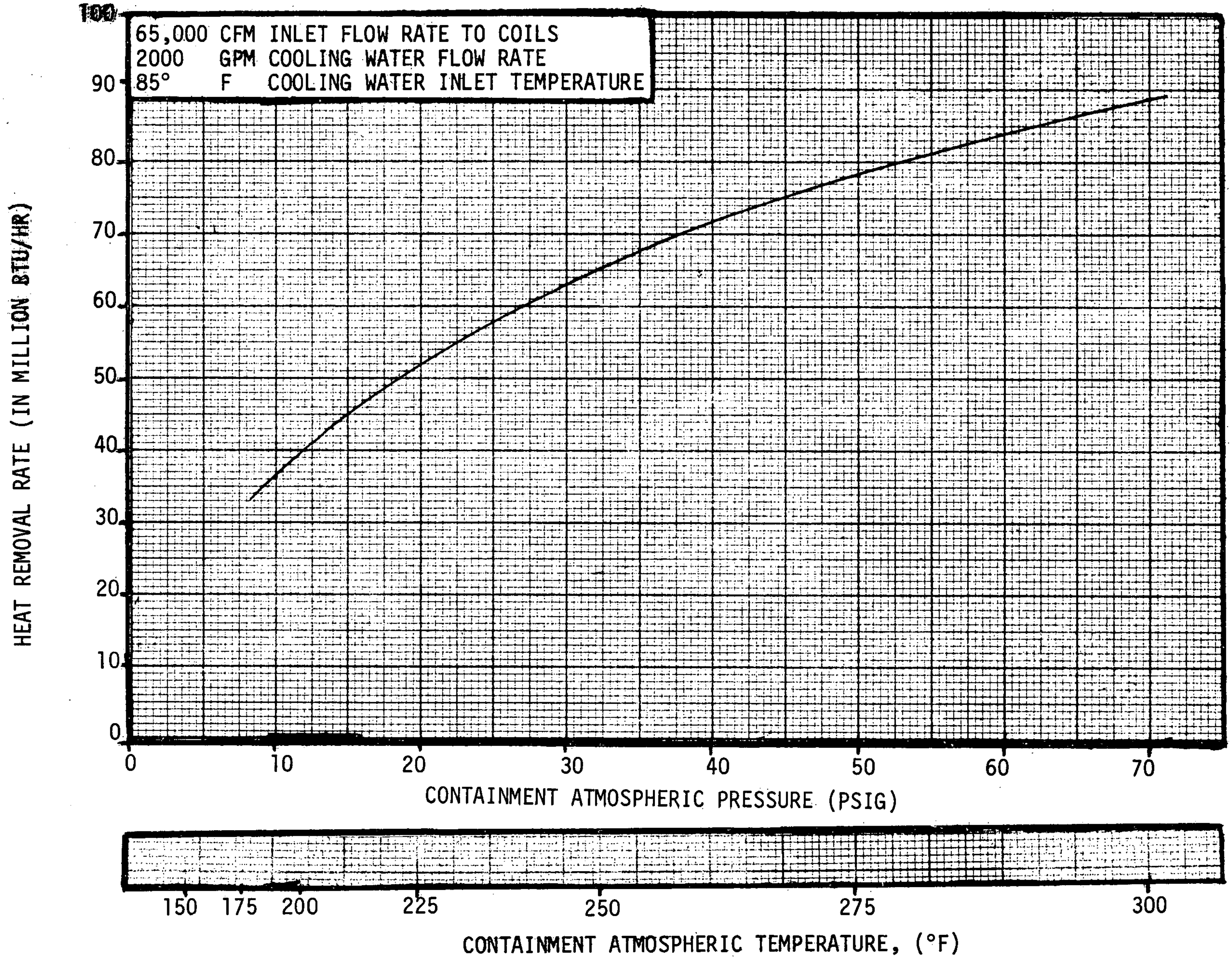

FIGURE 14(C-1.2)-3
SUPPLEMENT 1

FAN-COOLER HEAT REMOVAL RATE AS A FUNCTION OF
STEAM-AIR MIXTURE VOLUMETRIC FLOW RATE

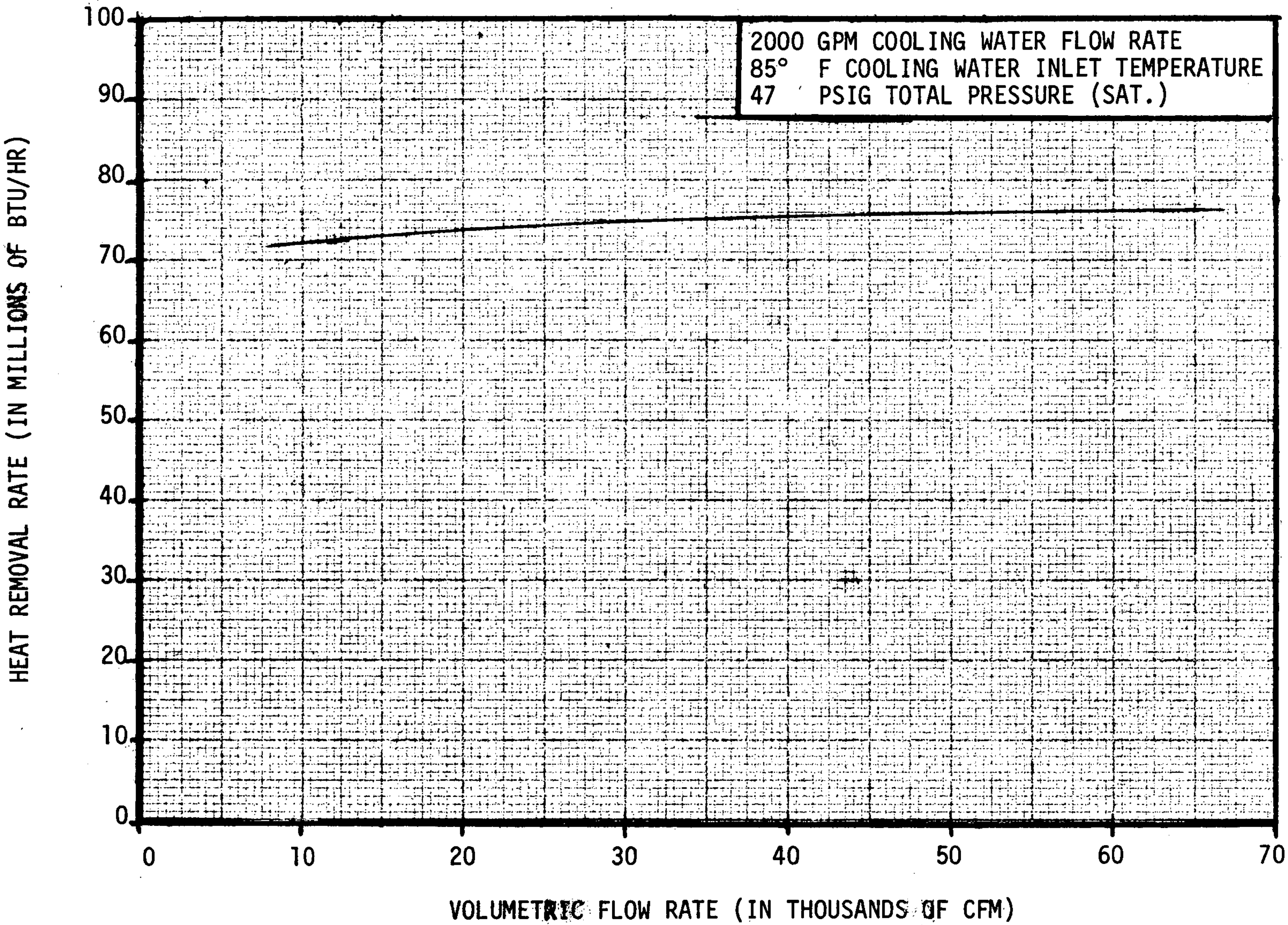

FIGURE 14(C-1.2)-4

SUPPLEMENT 1

14 (C-1.3)

The following is a description of the analytical procedure used to determine plate-fin heat exchanger performance in an environment of high steam partial pressure (saturated) such as would exist in the containment structure following a loss-of-coolant accident.

It is initially assumed that a heat exchanger of given size has been chosen for other purposes and it is desired to determine the performance of this particular unit after a loss-of-coolant accident. 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}. 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 fluid stream, which involves the effect of convective heat transfer and mass transfer. With this value 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 fluid stream. In the process, from an energy balance, one determines the temperature of the interface between the bulk fluid and the condensate; this is necessary for determining the mass transfer rate from the stream. 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/film. If the assumed and computed values are not the same, a new estimate is made and calculations are 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 fluid 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 air-steam fluid composition and thermodynamic state. The updated fluid state is used as inlet conditions for the next row of fin surface. The process is now repeated for the second, third etc. rows until the air-steam mixture leaves the heat exchanger.

The analytical procedure described above is based on standard heat and mass transfer calculations. Nomenclature is given on page 6-97.

The mass transfer coefficients are derived from analyses and reports of experimental data contained in references 1, 2 & 3 given on page 6-98. From reference 1 the mass flow rate of condensate is defined by

(1) $$\dot{m} = \bar{h}_D \left(\rho_{sg} - \rho_{sw}\right)$$

From ref. (1), pp. 471-473, experimental data for mass and heat transfer correlate well by the expression

$$\frac{\bar{h}_D}{u_s} (Sc)^{-2/3} = St\ (Pr)^{-2/3}$$

as shown in Figure 16-10 of ref. (1). Thus

(2) $$\bar{h}_D = u_s \cdot St \left(\frac{Sc}{Pr}\right)^{2/3}$$

$$\bar{h}_D = \frac{u_s \cdot h}{\rho_C \, u_s} \left(\frac{Sc}{Pr}\right)^{2/3}$$

As reference (1) points out, for large partial pressures of the condensing components, equation (2) must be corrected by a factor P_t/P_{am}. Thus $\bar{h}_D$ is defined by

(3) $$\bar{h}_D = \frac{h}{\rho C}\left(\frac{P_t}{P_{am}}\right)\left(\frac{Sc}{Pr}\right)^{2/3}$$

This is essentially the same result as reported by ref. (2), pg. 343 and reference (4).

Reference (1) 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

$$\dot{m} = \frac{h}{\rho C}\left(\frac{P_t}{P_{am}}\right)\left(\frac{Sc}{Pr}\right)^{2/3} (\rho_{sg} - \rho_{sw})$$

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

$$\dot{m} = \frac{h}{\rho C}\left(\frac{P_t}{P_{am}}\right)(\rho_{sg} - \rho_{sw})$$

The heat transfer rate due to condensation is computed from

$$q_1 = \frac{\dot{m}\lambda h}{\rho C} \left(\frac{P_t}{P_{am}}\right) (\rho_{sg} - \rho_{sw})$$

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.

ρ and C are evaluated at the local bulk gas temperature

The heat transfer coefficient, h, was determined from experiments on plate-fin coils which had 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 mehtods used are accurate.

The pressure of non-condensable gases is taken into consideration by virtue of the fact that the theory behind the analyses assumes that the condensable vapor must diffuse through a non-condensable gas.

Nomenclature

$\dot{m}$ 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 temperature, lbm/ft^3

u_s free stream 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 vapor-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 pressure, $\dfrac{P_{aw} - P_{ag}}{\ln \dfrac{P_{aw}}{P_{ag}}}$, 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

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14 (C-1.4)

A testing program is being developed to confirm the performance of heat exchangers and the design procedures under post accident conditions. Details of the testing program will be presented to the USAEC by separate correspondence early in 1969.

Item 14 (Attachment C Questions 2.0-4.3)

The following information is requested relative to the heat removal and iodine scavanging functions of the containment spray:

14 (C-2.1)

Since nozzle performance with respect to particle size is more critical with respect to iodine removal than with respect to steam condensation, provide nozzle performance curves of drop size vs. pressure drop and relate to optimum drop size for iodine removal. Indicate your criterion regarding size distribution for any given pressure drop.

14 (C-2.2)

Discuss the considerations that are required in header (or nozzle) design to assure proper drop size distribution from each nozzle, and the effect debris from the sump might have on spray performance.

14 (C-2.3)

In view of the importance of demonstrated function of engineered safety systems simulating, as closely as reasonable, accident conditions, discuss why you consider that a functional test of the completed Indian Point spray system is not required. Describe your plans for an engineering scale test of a typical system which could qualify the system design as having been proof tested, and evaluate the applicability of the test to the installed system.

14 (C-3.0)

Describe the screens provided in the sump, indicating arrangement and mesh sizes. Can the largest credible particle passing lengthwise through the screens result in a clogged nozzle or in pump damage?

14 (C-4.0)

Design of the sodium thiosulfate iodine removal spray system is based on the work of Griffiths. This paper presents an analytical approach for determining the potential performance of sprays in removal of iodine from the containment atmosphere following a loss-of-coolant accident. These are some questions regarding the application of the work of Griffiths to the system as proposed for the Indian Point plant.

14 (C-4.1)

The removal coefficient for iodine is directly proportional to the diffusivity of iodine through air. This diffusivity has been shown to vary inversely with pressure and directly with temperature. It appears that the diffusivities used by Griffiths are for a pressure of one atmosphere. Discuss the corrections made to the Griffiths' model to account for the pressure and temperature conditions in the containment following blowdown.

14 (C-4.2)

The containment spray system is planned to be used for both iodine removal and steam condensation following loss-of-coolant accidents. The condensation process will involve the deposition of a significant film thickness of water (condensate) on each drop for the drop sizes proposed in this system. Discuss the effect of this liquid undoped film on iodine removal considering iodine to be in elemental, ionic, and organic forms. Perform a sensitivity study illustrating the variation of iodine removal coefficient with containment atmospheric relative humidity. Describe the model and absorption mechanism assumed for each of the three iodine forms noted above.

14 (C-4.3)

Outline any laboratory scale or engineering scale tests that are being planned to support the claimed performance of the thiosulfate spray iodine removal system. Include a discussion of the chemical stability of the thiosulfate solution when exposed to the radiation fields and temperatures which will exist following a loss-of-coolant accident. Consider both hydrogen produced by radiolysis of the water as well as thiosulfate destruction and subsequent release of absorbed iodine. Indicate the sensitivity of the stability of the system to pH.

<u>ANSWER</u>

14 (C - 2.1)

Reference is made to Appendix 6A which is submitted as an amendment to the PSAR. It is noted that sodium hydroxide rather than sodium thiosulphate is currently being proposed as the spray additive.

14 (C - 2.2)

As above.

14 (C - 2.3)

A comprehensive program of plant testing is formulated for all equipment systems and system control vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop, integrated tests of the system as a whole, and periodic tests of the actuation circuitry and mechanical components to assure reliable performance, upon demand, throughout the plant lifetime.

The initial tests of individual components and the integrated test of the system as a whole complement each other to assure performance of the system as designed and to prove proper operation of the actuation circuitry.

Permanent test lines for all the containment spray loops are located so that all components up to the isolation valves at the containment may be tested. These isolation valves are checked separately.

The air test lines, for checking that spray nozzles are unobstructed connected downstream of the isolation valves. Air flow through the nozzles is monitored by the use of a smoke generator.

Periodic sampling confirms that proper sodium hydroxide concentration exists in the tank.

14 (C - 3.0)

Provisions are made in the design of the containment sump so that debris in the containment will not block the sump suction lines, damage recirculation system equipment or plug the containment spray nozzles.

The low flow velocity approaching the sump will permit the bulk of the debris denser than water to settle to the floor of the containment rather than enter the sump. Baffles at the sump will protect the sump screens from damage and prevent floating debris from entering the sump. Two screens of graduated mesh size will prevent entrance of other suspended matter of a size that could jeopardize the recirculation system. The fine screen mesh opening will be 1/4 inch.

The spray nozzles are of the ramp bottom design with a 3/8 diameter orifice. The nozzles are not subject to clogging by particles 1/4 in. in maximum dimension.

14 (C-4.0) - Refer to Appendix 6A

14 (C-4.1) - Refer to Appendix 6A

14 (C-4.2) - Refer to Appendix 6A and WCAP-7153.

14 (C-4.3) - Refer to WCAP-7153.

APPENDIX 6A

IODINE REMOVAL EFFECTIVENESS EVALUATION OF THE CONTAINMENT SPRAY SYSTEM

1.0 PURPOSE OF CHEMICAL MODIFICATION

The containment spray system in this pressurized water reactor facility is one of the engineered safety features systems employed following a loss-of-coolant inside the containment to reduce the pressure and temperature of the containment atmosphere. The flow rate and inlet subcooling of the spray are sufficient to provide thermal capacity for condensing steam produced by dissipation of heat in the reactor and its associated systems. Minimum operability of this system with on-site power and under a single-component failure contingency will prevent pressurization of containment above the design pressure with a substantial capacity margin.

The spray system, by virtue of the large surface area provided between the liquid droplets and the containment atmosphere, affords an excellent means of absorbing the soluble components from the gas phase. If the solubility of the component is sufficiently high, the rate of absorption is limited only by the mass transfer rate of the absorbing species through the gas film. In the case of I_2 vapor, elimination of all but the gas film resistance would permit the absorption by sprays to proceed with a removal half-life of less than two minutes, as will be shown later. However the solubility of I_2 in the refueling water used as spray is limited, as indicated by the partition coefficient[1]

$$K_c = 0.0125 \frac{\text{mol/liter gas}}{\text{mol/liter liquid}}$$

in acidic solution. While this coefficient corresponds to an equilibrium favoring solution of 60-80% of the iodine by the liquid (considering the gas/liquid volume ratios of conventional PWR containments) it is expected that the liquid phase mass transfer resistence would severely limit the

removal rate. Assuming a liquid film coefficient of 0.01 cm/sec and a gas film coefficient (to be derived later) of the order of 10 cm/sec, the overall mass transfer coefficient, V_T, is obtained as follows:

$$\frac{1}{V_T} = \frac{1}{v_G} + \frac{Kc}{V_L} = \frac{1}{10} + \frac{0.0125}{0.01} = 0.1 + 1.25 \qquad (1)$$

$$V_T = 0.74 \text{ cm/sec}$$

If the I_2 were infinitely soluble ($K_c \sim 0$), the value of V_T would approach 10 cm/sec in this example.

To obtain the advantages of an order-of-magnitude improvement of absorption rate and nearly complete removal of I_2 at equilibrium, the chemistry of the spray solution is modified by adding NaOH, raising the pH to 9.5. According to the known behavior of elemental iodine in highly dilute solutions the hydrolysis reaction

$$I_2 + OH^- \longrightarrow HIO + I^-$$

proceeds nearly to completion[2] at pH >8. The iodide form is highly soluble, and HIO readily oxidixes to IO_3^- in the oxygenated medium, this form being likewise soluble:

$$HIO + O_2 \rightarrow IO_3^- + H^+ \qquad (2)$$

Griffith[1] suggested that the use of chemical additives which undergo ionic reactions with aqueous I_2 would improve the absorption rate to the point where the gas film mass transfer resistance became limiting, implying that $K_C/V_L << 10^{-1}$. His paper called attention to sodium thiosulfate ($Na_2S_2O_3$) as a likely reagent for this purpose and mentioned NaOH as another candidate. Subsequent experiments in a spray medium have shown that both additives bring about absorption rates indicative of gas film control, verifying the desired rate capability.

The selection of NaOH instead of $Na_2S_2O_3$ for this application followed an evaluation program which revealed certain disadvantages for $Na_2S_2O_3$. The results of this evaluation program are detailed in the Proprietary Westinghouse report, WCAP-7153.

By contrast, the same testing program revealed no instability of the solution formed by adding NaOH alone to the borated spray. Corrosion rates of copper and copper-alloy heat exchanger tubing were reduced by more than an order of magnitude compared with high pH $Na_2S_2O_3$ and were acceptably low (< 0.01 nils/month at 200°F) for the application. These tests showed that pitting or local corrosion did not occur.

For engineering reasons, therefore, further testing was centered on the use of NaOH as the spray additive leading to the development of a technical basis for its inclusion in the plant engineered safety features as a means of "Fixing" absorbed iodine, enhancing the natural rate of deposition of I_2, and thus lowering the calculated off-site thyroid dose resulting from a postulated release of fission products to the containment atmosphere. In summary, this work supports the following conclusions comprising the technical basis for spray absorption process for iodine removal:

1) The conversion of absorbed I_2 to I^- and IO_2^- in pH 9.5 borate solution is quite rapid, such that the absorption process is gas film diffusion controlled.

2) Mass transfer follows the Ranz-Marshal rate equation[(3)] for soluble gases, as demonstrated by containment simulation tests performed with a nozzle design atmospheric conditions, iodine concentration and spray chemical composition in close approximation to the design basis accident.

3) Under a range of conditions bracketing the possible accident modes, the spray experiments and calculations show irreversible and effective iodine removal, (i.e., K_c is not reduced with time if pH is maintained) compatability with the vital materials and processes of the containment system and high mechanical reliability.

2.0 TECHNICAL BASIS FOR IODINE REMOVAL FACTOR

2.1 ANALYTICAL MODEL AND ASSUMPTIONS

The removal of a soluble component by a reactive spray under conditions of a constant mass transfer rate coefficient is exponential:

$$C = C_o e^{-\lambda_s t} \qquad (3)$$

The removal constant λ_s can be expressed as the product of a mass transfer coefficient, v_G, and the effective absorbing surface area, A, per unit volume.

In addition to the basic assumption that the absorption is gas-film resistance controlled, the following idealizations are made to simplify the physical model:

1) All droplets behave as spheres of diameter equal to that of the surface-mean diameter droplet, d.

2) Droplets fan at their terminal velocity, u_t, from the spray nozzle to the operating deck, a distance h.

3) Iodine concentration in the gas is uniform.

The effective absorbing area is then

$$A = \frac{6 F h}{u_t V_c d} \qquad (4)$$

where A = absorbing area per unit volume

F = volumetric spray flow rate

V_c = containment free volume

For a given droplet size, the terminal velocity and the mass transfer coefficient are temperature and pressure dependent. In the expression for λ_s, then, these variables can be treated as a dimensionless ratio:

$$\lambda_s = v_G A = \left(\frac{v_G}{u_t}\right) \frac{6Fh}{V_c d} \qquad (5)$$

When the remaining parameters are expressed in engineering units, λ_s in reciprocal hours is given by

$$\lambda_s = 1470 \left(\frac{v_G}{u_t}\right) \frac{F\,h}{V_c d} \qquad (6)$$

where F = spray flow gal/min

h = fall height, ft

V_c = volume, cu. ft.

d = droplet diameter, cm

For the various classes of Westinghouse PWR containments, the following values of the physical parameters are conservatively approximated as follows:

	F	h	V_c	d	$\frac{F\,h}{V_c d}$
Two Loop	1250	70	970,000	0.1	0.90
Three Loop	880	77	2,100,000	0.1	0.32
Four Loop	2340	104	2,600,000	0.1	0.94

The value of $v_G/\bar{u}_t$ for 0.1 cm droplets in a saturated air-steam atmosphere of pressure P* is plotted in Figure 1. These data are obtained from ORNL-TM-1911.[4] It is apparent that as pressure decreases during the

*Defined as the mixture of air and steam produced by adding steam to dry air at an initial temperature of 30°C and 1 atmosphere pressure, at constant volume.

x

post-accident period, the value of $v_G/\bar{u}_t$ and hence the removal coefficient λ_s will increase. The removal rate is underestimated, therefore, by assuming for purposes of analysis the value of this ratio at the design condition of the containment. The results, calculated from equation (6), are then:

	$v_G/\bar{u}_t$	λ_s
Two loop (60 psig, 286°F)	0.0224	29.6 hr^{-1}
Three loop (42 psig, 264°F)	0.0236	11.1
Four loop (47 psig, 270°F)	0.0231	32.0

The "half-life" for removal of elemental iodine is obtained from the following expression for exponential decay

$$T_H = \frac{0.693}{\lambda_s} \times 3600 \quad \text{(in seconds)} \qquad (7)$$

The "dose reduction factor" applicable for elemental iodine is the ratio of the average two-hour inventory of I_2 without removal to the average with removal. It is given by the following expression

$$DRF_2 = \frac{2\lambda_s}{1-e^{-2\lambda_s}} \qquad (8)$$

The calculated values of λ_s for the three plant types yield the following values of I_2 half-life and I_2 dose reduction factor:

	T_H	DRF_2
Two loop	84. sec	59.
Three loop	225.	22.
Four loop	78.	64.

Concerning other airborne forms of iodine, the removal mechanisms can be characterized in the following way:

HI - Hydrogen iodide may constitute an important fraction of the liberated iodine if oxygen is excluded from the reactor during the melt. The higher diffusivity of HI, compared with I_2, and the fact that favorable partition between vapor and liquid does not require that the absorbed HI molecule undergo chemical reaction, would lead to removal of HI by sprays no less rapid than I_2.

CH_3I - A small fraction of the available iodine will exist as organic iodides of which methyl iodide is the most important. There is preliminary evidence that absorption and chemical decomposition of CH_3I occurs in the reference spray solution. The rate of absorption, which is expected to be liquid-film diffusion of liquid-phase reaction rate controlled, is so slow that the reduction of the two-hour average inventory of CH_3I vapor is less than the probable error in predicting that inventory. No credit for removal is taken in calculating the two-hour dose due to organic iodide leakage.

Particles - Spray may have an important effect on particle removal by increasing the rate of steam condensation. When the bulk flow of steam to the condensing surface is great enough to mask the diffusive motion of particles (as would be the case when cold droplets contact the containment atmosphere during the high-steam period), sub-micron particles are efficiently captured by the spray.(5) Larger particles would be removed by high efficiency particulate air (HEPA) filters or would agglomerate and settle out by gravity, reducing their importance as a potential leakage path at all. In evaluating the potential benefit of sprays in reducing post-accident iodine leakage, no quantitative consideration is given here to particle removal by condensation, because the phenomenon is independent of the chemical modification of the spray solution.

2.2 EXPERIMENTAL VERIFICATION

The droplet size assumed in the spray calculations summarized above was 0.10 cm or 1000 μ. The spray pattern produced by a 3/8 in. aperture ramp bottom nozzle of the type used in these facilities was measured photographically at various operating nozzle pressures. A ramp bottom nozzle with a 3/8-inch orifice diameter was selected because it is capable of producing the smallest mean droplet size without employing impingement baffles, swirl vanes, or other features which might trap particles of debris. Note that a 1/4-inch mesh screen is used to trap debris which might otherwise enter the recirculation pump suction from the containment sump.

Should some of the spray nozzles become plugged, considerable performance margin is available. For example, at the time recirculation from the sump would be employed, the condensing capacity of one spray above exceeds the residual heat rate of 25%. Similarly, at the same point in time, there is a greatly reduced dependence on sprays for continued iodine removal because most of the absorbable iodine has been removed prior to recirculation. Theory predicts that over 15 half-lives for removal of elemental iodine have elapsed during the period when clean spray water is being delivered from the storage tanks. One may conclude, therefore, that plugging of about one fifth of the nozzles in one spray system, complete outage of the other spray system, and disability of all five fan coolers could be tolerated at the time of recirculation without losing the ability to transfer residual heat from the containment atmosphere. A statistical analysis of the droplet images produced the following results:

Nozzle Pressure psi	Flow Rate gpm	Number Avg. diameter, μ	Surface Avg. diameter, μ
20	10.9	960	1340
30	12.9	830	1126
40	15.2	735	1012
50	16.9	685	961

In this table, the "number average diameter" is defined as

$$D_{NA} = D_G \exp\left(\frac{\ln^2 S_G}{2}\right)$$

where D_G = 50% undersize diameter

S_G = standard deviation, or ratio of 84% to 50% undersize diameter.

The "surface average diameter" is the diameter at which a uniformly sized array of drops would present the same surface to volume ratio as the observed array of drops. It is this parameter which is significant to the absorption rate model.

The spray system is designed to deliver rated flow with a minimum available nozzle pressure of 40 psi above the containment design pressure. Generally, the nozzle pressure will be more than 40 psi above the actual containment pressure, making the assumption 1000 μ a realistic one.

A more meaningful demonstration of effective droplet size and verification of the overall mass transfer model is obtained from the spray test program at Nuclear Safety Pilot Plant (NSPP). Data from these tests are published through the regular ORNL reporting channels.[6,7] The data treatment in this program uses the same basic analytical model as has been presented here, and the results are entirely consistent with the premise that I_2 absorption by Na-OH - H_3BO_3 spray is gas-film controlled.

Applying the removal expression (Eq. 8) to the NSPP system for a typical test condition of 44 psig, 266°F the values corresponding to our plant parameters are

	F	H	V_c	d	$\frac{F\,h}{V_c\,d}$
NSPP	15	17	1330	0.100	1.92

The ratio $v_G/\bar{u}_t$ for 0.100 cm droplets in a 44 psig steam-air atmosphere, (Figure 1) is 0.0234, giving $\lambda_s = 66\ hr^{-1}$ and a half-life of 38 sec.

As reported by the NSPP investigators[7,8] a run made at these conditions (run No. 30) with $NaOH-H_3BO_3$ spray resulted in an iodine removal half life of 24 to 44 seconds depending on the method of sampling. Generally speaking, NSPP results at a variety of conditions have shown that analytical models based on the gas-film controlled drop-wise absorption theory (Ranz-Marshall equation) tend to underpredict the experimental absorption rate.

Further spray testing is being conducted at the Containment Systems Experiment (CSE). Preliminary results of tests at room temperature but with fall heights (35-50 feet) more comparable to full scale than those in NSPP have been published.[9] These tests have shown that substantial decontamination of the containment atmosphere with respect to I_2 occurs with a half-life in good agreement with those calculated by the Ranz-Marshall equation.

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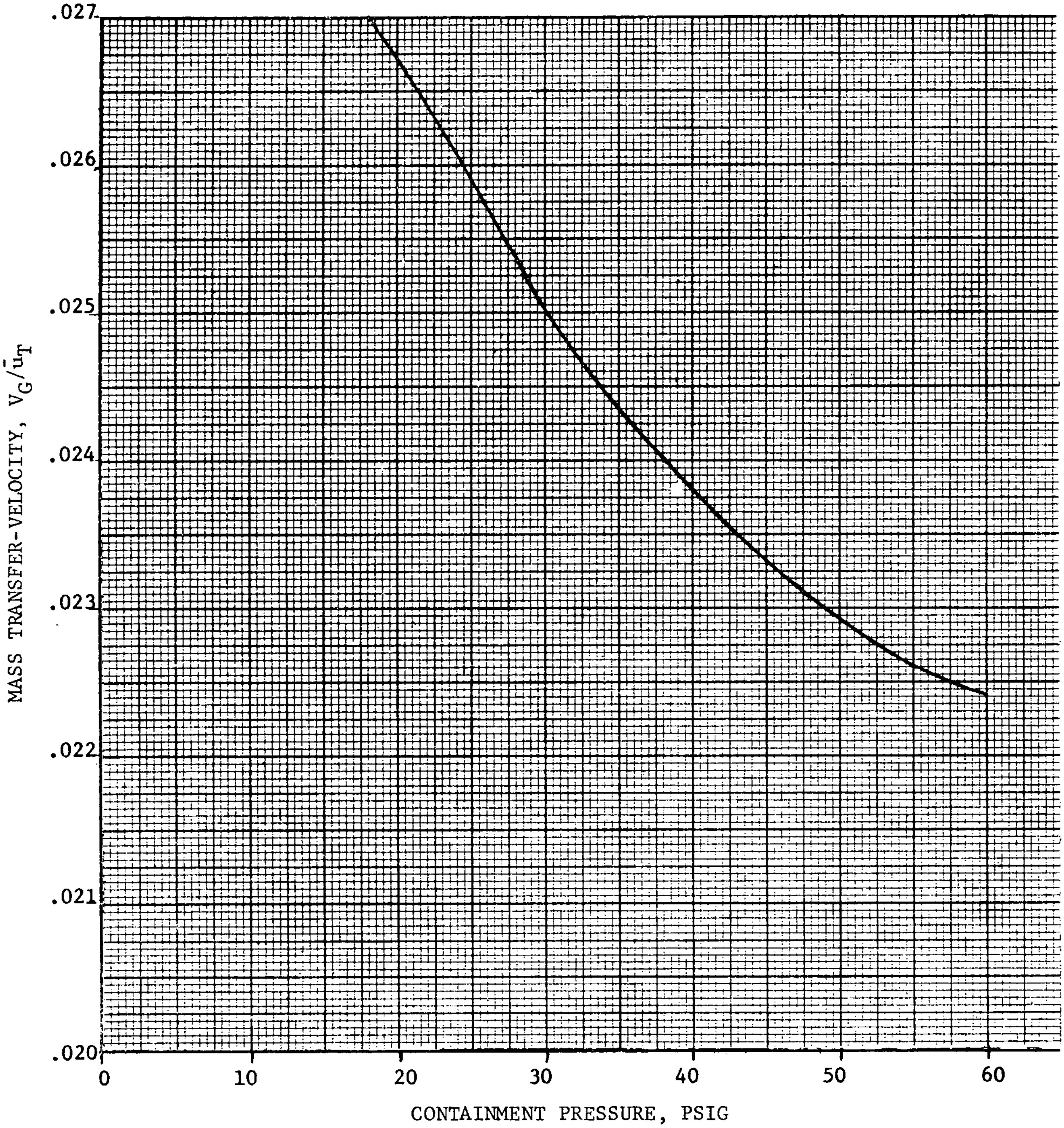
8. L. F. Parsly, Jr., and J. K. Cranzreb, "Removal of Iodine Vapor from Air and Steam-Air Atmospheres in the Nuclear Safety Pilot Plant by Use of Sprays," ORNL-4253, June 1968.

9. J. D. McCormack, R. K. Hilliard, and L. F. Coleman "Large-Scale Spray Tests in the Containment Systems Experiment (CSE)" ANS Transactions, 11 1, June 1968.



MASS TRANSFER-VELOCITY RATIO OF SPRAY DROPLETS AS A FUNCTION OF CONTAINMENT PRESSURE
FIGURE 1

ITEM 14 (Attachment C, Questions 5.0 - 7.0)

14 (C-5.0)

A system used to inject sodium thiosulfate into the containment spray water is presented in the PSAR and the systems' operation is briefly described. A complete evaluation of the adequacy of the system as proposed requires additional information as follows:

14 (C-5.1)

Present a more comprehensive description of this system and its operation, indicating the advantage of the system proposed over other concepts, such as aspiration from the thiosulfate or the application of controlled gas pressure to the tank.

14 (C-5.2)

Provide the preliminary design of the sodium hydroxide storage tank with regard to the inlet system and diffuser which will enable the solution to be forced out of the tank with little or no mixing.

14 (C-5.3)

Describe the method of system calibration used to achieve the desired injection rate.

14 (C-5.4)

Describe the engineering tests that have been performed on this system or similar systems to assure proper function of this concept.

14 (C-5.5)

State the shelf life of the sodium thiosulfate as stored in this tank. What periodic testing is contemplated to check chemical stability?

14 (C-6.0)

Submicron particulate fission products (<0.1 micron) including iodine absorbed on submicron particles, may not be efficiently removed by the spray system or the absolute filters in the fan-cooler system. State and justify by referencing suitable experimental information the proportion of fission products that are assumed to be present in the containment atmosphere in submicron particulate form following a loss-of-coolant accident.

14 (C-7.0)

Discuss the chemical stability of the chemical additive following the design-basis accident. Consider the effects of temperature, radiation, and possible chemical reactions with materials inside containment. Identify the products of thiosulfate reaction or degradation and estimate the concentration of these products.

ANSWER

14 (C-5.1)

The principal components of the Containment Spray System which provides containment cooling and iodine removal following a loss-of-coolant accident consists of spray pumps, one sodium hydroxide tank, spray ring headers and nozzles, and the necessary piping and valves. The containment spray pumps and the sodium hydroxide injection tank are located in the auxiliary building and take suction directly from the refueling water storage tank.

The spray water is injected into the containment through spray nozzles connected to four 360° ring headers located in the containment dome area. Each of the spray pumps supplies two of the ring headers.

The spray system will be actuated by he coincidence of two sets of two out of three high containment pressure signals. This starting signal will start the pumps and open the discharge valves to the spray header. The valves associated with the spray additive tank will be opened by the operator upon indication that a significant release of fission products has occurred or is imminent. Emergency procedures will set forth guidelines for this action, based on one or more of the following phenomena: high containment pressure in combination with loss of reactor coolant system pressure; occurrence of high radiation levels in combination with elevated containment pressure; liquid level signals indicative of accumulator discharge into the Reactor Coolant System. If required, the operator can manually actuate the entire system from the control room, and periodically, the operator will actuate system components to demonstrate operability.

During the period of time that the spray pumps draw from the refueling water storage tank approximately 10% of the spray flow is diverted from the spray pump discharge line through the sodium hydroxide tank. The liquid from the tank then mixes with the liquid entering the suction of the pump. The result is a solution suitable for the removal of iodine from the containment atmosphere.

14 (C-5.2)

Sodium Hydroxide Tank

The capacity of the stainless steel tank is sufficient to contain enough sodium hydroxide solution which, upon mixing with the refueling water from the refueling water storage tank, the boric acid from the boron injection tank, and the borated water contained within the accumulators and primary coolant, will bring the concentration of sodium hydroxide in the final mixture in the containment sump to approximately 0.6 weight per cent solution of caustic and approximately 1.7 weight per cent boric acid. This maintains pH of at least 9.3 and assures the continued iodine removal effectiveness of the containment spray during the recirculation phase of operation after the supply of borated water in the refueling water storage tank has been exhausted. The design pressure of the tank is the sum of the refueling water storage tank head and the total developed head of the containment spray pumps at shutoff. A deflector plate is provided so that incoming refueling water thoroughly sweeps the contained solution towards the outlet of the tank. The contents of the tank are flushed into the spray water in approximately 15 minutes, hence all of the sodium hydroxide will be added prior to depletion of the refueling water.

A level indicating alarm is provided in the control room if, at any time, the solution tank contains less than the required amount of sodium hydroxide solution. Periodic sampling confirms that proper sodium hydroxide concentration exists in the tank.

The design parameters are presented in Table 1.[1]

16.4.3 (AEC 5.3)

The only criterion is to displace the contents of the sodium hydroxide tank before the refueling water storage tank is emptied.

14 (C-5.4)

Reference is made to Appendix 6A.

14 (C-5.5)

Reference is made to Appendix 6A and WCAP 7153.

14 (C-6.0)

Spray may have an important effect on particle removal by increasing the rate of steam condensation. When the bulk flow of steam to the condensing surface is great enough to mask the diffusive motion of particles (as would be the case when cold droplets contact the containment atmosphere during the high-steam period), submicron particles are efficiently captured by the spray.[1] Larger particles would be removed by high efficiency particulate air (HEPA) filters or would agglomerate and settle out by gravity, reducing their importance as a potential leakage source, if indeed they could penetrate the leakage path at all. In evaluating the potential activity release from the containment at the present time too little information is available to state the amount of fission product as submicron particulate matter that would be present in the containment atmosphere following a loss-of-coolant accident. However a partition factor of 10^{-8} between particulate matter has been measured and described in Reference 2.

14 (C-7.0)

Reference is made to Appendix 6A and WCAP 7153.

(1) P. Goldsmith and R. A. Stindhcombe, "The Cleanup of Submicron Particles by Condensing Steam", AERE-M-1214 (1963).

(2) Ashworth, C. P., Barton, D. B. and Robbins, C. H., Nuclear Eng., 7, pg. 313, 1962.

TABLE 1

SODIUM HYDROXIDE TANK DESIGN PARAMETERS

Number	1
Total Volume (empty), gal.	5100
Minimum volume at operating conditions (solution), gal	5000
NaOH concentration, w/o	30
Design temperature, °F	300
Design pressure, psig	300
Operating temperature, °F	building ambient
Operating pressure, psig	200
Material	Carbon steel, ss clad
Code	ASME III, Class C

Item 15 (Attachment D Questions 1.0-4.0)

1. For reactor vessel internals, and for each type of component of the reactor coolant system and the engineered safety features, provide the following:

   a. The proposed stress or deformation limits for primary tensile or membrane loads.

   b. The proposed stress or deformation limits for combined primary loads (tension and bending).

   c. The margin of safety between the limits in b above and the expected collapse or failure condition.

   Consider the following loadings:

   a. Load combinations, including normal design loads and the design earthquake loads.

   b. Load combinations, including normal operating loads and the maximum earthquake loads.

   c. Load combinations, including normal operating loads, maximum earthquake loads, and applicable design-basis accident loads.

2. Identify specific reactor internals which must maintain their functional performance capabilities to assure safe shutdown of the reactor. Provide calculated (or estimated) maximum limits of deformation or stress, at which inability to function occurs, for each component identified. Also, supply the calculated (or estimated) maximum design limit value, and the expected deformation or stress. In all cases, identify the applicable loading combination and state the proposed margin of safety.

3. In cases where limit load analysis is to be employed, describe the method in detail. If strain hardening effects will be considered in the analysis, supply the actual stress-strain curves for the principal materials of the Class I components involved. Provide also a realistic estimate of the maximum allowable strain based on appropriate material properties at the applicable temperature in order than an estimate of the margin of safety can be made.

   Provide also information that will permit evaluation of the effect of irradiation on the material properties and of the effect of irradiation, welds, and material imperfections on the deformation limits proposed.

4. Supply criteria or specific information on the interaction forces, deformation and stresses connected with the relative motions between the reactor vessel, steam generators, or other large components. Indicate how these relative motions will be controlled by snubbers or other means, and what reaction forces (and corresponding stresses) will be transmitted to the pipes.

DESIGN CRITERIA FOR STRUCTURES AND EQUIPMENT

1.0 DEFINITION OF SEISMIC DESIGN CLASSIFICATIONS

All equipment and structures are classified as Class I, and Class II, or Class III as recommended in:

a) TID-7024, "Nuclear Reactors and Earthquakes" August, 1963 and,

b) G. W. Housner, "Design of Nuclear Power Reactors Against Earthquakes", Proceedings of the Second World Conference on Earthquake Engineering, Vol. I, Japan 1960, Pg. 133, 134 and 137.

Class I

Those structures and components including instruments and controls whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of excessive amounts of radioactivity. Also, those structures and components vital to safe shutdown and isolation of the reactor.

Class II

Those structures and components which are important to reactor operation but not essential to safe shutdown and isolation of the reactor and whose failure could not result in the release of substantial amounts of radioactivity.

Class III

Those structures and components which are not directly related to reactor operation or containment.

All components, systems and structures classified as Class I are designed in accordance with the following criteria:

1. Primary steady state stresses, when combined with the seismic stress resulting from the response to a ground acceleration of 0.05g acting in the vertical and 0.1g acting in the horizontal planes simultaneously, are maintained within the allowable stress limits accepted as good practice and, where applicable, set forth in the appropriate design standards, e.g., ASME Boiler and Pressure Vessel Code, USAS B31.1 Code for Pressure Piping, ACI 318 Building Code Requirements for Reinforced Concrete, and AISC Specifications for the Design and Erection of Structural Steel for Buildings.

2. Primary steady stress when combined with the seismic stress resulting from the response to a ground acceleration of 0.10g acting in the vertical and 0.15g acting in the horizontal planes simultaneously and the stresses from a Reactor Coolant System pipe rupture, are limited so that the function of the component, system or structure shall not be impaired as to prevent a safe and orderly shutdown of the plant.

All Class II structures and components are designed on the basis of a static analysis for a ground acceleration of 0.05g acting in the vertical and 0.1g acting in the horizontal directions simultaneously.

The structural design of all Class II structures meets the requirements of the applicable building code which is the "State Building Construction Code," State of New York, 1961. This code does not reference the Uniform Building Code.

Table 15.-1 gives the damping factors used in the design of components and structures.

The design of Class I structures and components utilizes the "response spectrum" approach in the analysis of the dynamic loads imparted by the earthquake. The analysis is based upon the response spectra shown on Figures A-1 and A-2 of Appendix A of the PSAR.

The following method of analysis is applied to Class I structures and components, including instrumentation:

1. The natural period of vibration of the structure or component is determined.

2. The response acceleration of the component to the seismic motion is taken from the response spectrum curve at the appropriate period.

3. Stresses and deflections resulting from the combined influence of normal loads and the seismic load due to the design earthquake (0.05g acting in the vertical and 0.1g acting in the horizontal planes simultaneously) are calculated and checked against the limits imposed by the design standard.

4. Stresses and deflections resulting from the combined influence of normal loads and the seismic loads due to the maximum potential earthquake (0.1g acting in the vertical and 0.15g acting in the horizontal planes simultaneously) and Reactor Coolant System pipe rupture are calculated and checked to verify that deflections do not cause loss of function (excluding the broken pipe) and that stresses do not produce rupture.

Where the vibrator system is of a highly complex geometric shape, such as piping systems, the maximum response from the response curve with the appropriate damping factor is selected. By using this conservative value and demonstrating that the stresses are satisfactory, it becomes unnecessary to perform any further analysis to determine the natural periods of the system.

2.0 CLASSIFICATION OF PARTICULAR STRUCTURES AND EQUIPMENT

Examples of particular structure and equipment classifications are given below. These classifications are not intended to be all-inclusive.

Item	Class
Buildings and Structures	
Containment (including all penetrations and air locks, the concrete shield, the liner and the interior structures)	I
Spent fuel pit	I
Service water screenwell	I
Control Room	I
Diesel generator room	I
Auxiliary building	I
Intake structure (to the extent that water is always available to the service water pumps)	I
Turbine structure	III
Buildings containing conventional facilities	III
Equipment, Piping and Supports*	
Reactor Control and Protection System	I
Radiation Monitoring System	I

* Class I components (equipment, piping, instrumentation, etc.) located in or supported on Class II structure will be protected from earthquake damage or will be backed up by other Class I components located in or supported by a Class I structure.

Nuclear Process Instrumentation and Controls	I
Reactor	I
Vessel and its supports	
Vessel internals	
Fuel assemblies	
RCC assemblies and drive mechanisms	
Supporting and positioning members	
In-core instrumentation structure	
Reactor Coolant System	I
Piping and valves (including safety & relief valves)	
Steam Generators	
Pressurizer	
Pressurizer relief tank	
Reactor Coolant pumps	
Supporting and positioning members	
Engineered Safety Features	I
Safety Injection System (including safety injection and residual heat removal pumps, refueling water storage tank, accumulator tanks, boron injection tank, residual heat exchangers and connecting piping and valving)	
Containment Spray System (including spray pumps, spray headers, spray additive tank and connecting piping and valving)	
Containment Air Recirculation Cooling and Filtration System (including fans, coolers, ducts, valves, absolute filters and demisters)	

Auxiliary Building Ventilation System	I
Condensate storage tanks	I
Pressurizer relief tank	II
Residual heat removal loop	I
Containment Penetration and Weld Channel Pressurization System	I
Component cooling loop	I
Isolation Valve Seal Water System	I
Sampling System	II
Spent fuel pit cooling loop	II
Fuel transfer tube	I
Emergency Power Supply System	I
Diesel generators and fuel oil storage tank	
D-C power supply system	
Power distribution lines to equipment required for transformers and switchgear supplying the engineered safety features	
Control panel boards	
Motor control centers	
Control Equipment, facilities and lines necessary for the above Class I items	I

Waste Disposal System	I
Chemical drain tank	
Waste holdup tanks	
Sump tank	
Gas decay tanks	
Spent resin storage tank	
Reactor coolant drain tank	
Compressors	
Waste evaporator	
Waste evaporator feed pump	
Waste holdup tank pumps	
Sump tank pumps	
Interconnecting waste gas piping	
Waste Disposal System	III
All elements not listed as Class I	
Containment crane	I
Manipulator and other cranes	III
Conventional equipment, tanks and piping, other than I and II Classes	III
Emergency Boiler Feed, Service Water and Fire Protection Systems' pumps and piping	I

The Chemical and Volume Control System is considered Class I except for those items listed below,

Batching tank	II
Monitor tanks	II
Monitor tank pumps	II
Chemical mixing tank	II
Resin fill tank	III

3.0 CLASS I DESIGN CRITERIA FOR VESSELS AND PIPING

Reference is made in this section of the Appendix to WCAP-5890, Rev. 1, "Ultimate Strength Criteria to Ensure No Loss of Function of Piping and Vessels under Earthquake Loading." by R. A. Wiesemann, R. E. Tome and R. Salvatori. Following discussions with the AEC DRL Staff, the criteria presented in WCAP-5890, Rev. 1, for the combination of normal load plus maximum potential earthquake loads plus pipe rupture loads associated with a major loss-of-coolant accident will be modified. Details of the manner in which this modification will be developed are given at the end of this section in Note 1.

The loading combinations which are employed in the design of Class I piping, vessels, supports, reactor vessel internals and other applicable components are shown in Table 15.2 with reference to WCAP-5890, Rev. 1. This Table also indicates the stress limits which are used in the design of the listed equipment for the various loading combinations.

To be able to perform their function, i.e., allow core shutdown and cooling the reactor vessel internals must satisfy deformation limits which are more restrictive than the stress limits shown in Table 15.2. For this reason the reactor vessel internals are treated separately.

Piping, Vessels and Supports

The reasoning for selection of the above mentioned loading combinations and stress limits is as follows:

1) For the design earthquake, the nuclear steam supply system is designed to be capable of continued safe operation. Equipment and supports needed for this purpose are required to operate within normal design limits as shown in Line 2 of Table 15.2.

2) In the case of the maximum potential earthquake, it is necessary to ensure that components required to shut the plant down and maintain it in safe shutdown condition do not lose their capability to perform their safety function. This capability is ensured by maintaining the stress limits as shown in line 3 of Table 15.2. No rupture of a Class 1 pipe can be caused by the occurrence of the maximum potential earthquake.

3) For the assumed case of reactor coolant pipe rupture, limit stresses in the unbroken reactor coolant system legs and other Class I vessels and pipes will be as noted in Line 4 of Table 15.2.

4) For the extremely unlikely event of simultaneous occurrence of maximum potential earthquake and reactor coolant system pipe rupture the design of Class I piping and components, excluding the broken pipe, is checked for no loss of function, i.e., the capability to contain fluid and allow fluid flow. This is assured by limiting the various stress combinations within the limits shown in Line 5 of Table 15.2.

The minimum margin of safety between the design stress limit and the expected collapse condition is that for the case of pure tension and is defined as

$$\frac{S_{ultimate} - S_{design}}{S_{design}}$$

In the more practical cases of design, piping and vessels will always experience some combination of tension and bending. For these combinations of loads the margin of safety is larger than that for pure tension, as shown by the limit curves contained in WCAP-5890, Rev. 1. Therefore, it is conservative to base the margin of safety on pure tension. Table 15.3, illustrates the margin to safety between the stress limits for various load conditions and the expected failure or collapse condition for typical materials.

Reactor Vessel Internals

Design Criteria for Normal Operation

The internals and core are designed for normal operation conditions and subjected to loads of mechanical hydraulic, and thermal origin. The response of the structure under the design earthquake is included in this category.

The stress criteria established in Section III of the ASME Boiler and Pressure Vessel Code, Article 4, has been adopted as a guide for the design of the internals and core with exception of those fabrication techniques and materials which are not covered by the Code, such as the fuel rod cladding. Seismic stresses are combined in the most conservative way and are considered primary stresses.

The members are designed under the basis principles of: (1) maintaining distortions within acceptable limits, (2) keeping the stress levels within acceptable limits, and (3) prevention of fatigue failures.

Design Criteria for Abnormal Operation

The abnormal design condition assumes blowdown effects due to a pipe break combined in the most unfavorable manner with the effects associated with the maximum potential earthquake.

For this condition the criteria for acceptability are that the reactor be capable of safe shutdown and that the engineered safety features are able to operate as designed. Consequently, the limitations established on the internals for these types of loads are concerned principally with the maximum allowable deflections. The deflection criteria for critical structures under normal operation, plus the maximum potential earthquake and blowdown excitation are presented in Table 15.4.

The allowable limit deflection values given in Table 15.4 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 much more restrictive than those given by the failure stress criteria.

Movement of Reactor Coolant System Components

The criterion for movement of the reactor vessel, under the worst combination of loads, i.e., normal plus the maximum potential earthquake plus reactor coolant pipe rupture loads, is that the radial movement of the reactor vessel will not exceed the clearance between a reactor coolant pipe and the surrounding concrete to prevent excessive shear load on the RCS pipe, should this limit be more restrictive than those listed in Table 15.2.

The relative motions between Reactor Coolant System components will be controlled by the structures which are used to support the reactor vessel, the steam generators, the pressurizer and the reactor coolant pumps in such a way that the stresses in the various components and pipes do not exceed the limits as established in Table 15.2.

Tests to Demonstrate the Conservatism of the Limit Curves

A series of tests has been performed at Westinghouse Material Testing Laboratory in Pittsburgh to demonstrate the conservatism of the limit curves. Carbon steel and stainless steel pipes have been tested under axial and transverse loads to measure combinations of tensile/compressive and bending stresses required to cause failure. Specimens, about 1.5 foot long have been cut from out of stock 1.5 inch nominal diameter Schedule 160 pipes. The materials employed were SA 106B carbon steel and Type 304 stainless steel. These specimens were kept internally pressurized to 3000 psia for the entire duration of the tests. The experimental data is still under evaluation. Tables 15.5 and 15.6 summarize the tests that have undergone preliminary evaluation and the results of this evaluation.

Standard ASTM tensile specimens have been modelled from pieces of test pipes and stress-strain curves have been measured. These curves have been conservatively approximated with trapezoidal stress-strain curves as indicated in WCAP-5890, Rev. 1. The limit curves for both SA 106B carbon steel and Type 304 stainless steel for the test conditions, have been calculated and are reported in Figures 15.1 and 15.2. The experimental points, i.e., stress intensities versus axial stresses as listed in Table 15.5 and 15.6 are shown in Figure 15.3 and 15.4. Also shown in these figures are the limit curves as calculated by the use of the trapezoidal stress-strain curves up to the ultimate stress. Comparisons between the experimental points and the design limit curves show the conservatism of the latter.

Whenever a structure is subject to combinations of axial and transverse loads, the stress distribution across each cross section is similar to the actual stress-strain curve as measured by standard ASTM Tensile tests. Therefore, if the materials in use show strain hardening, the strain hardening will be reflected in the stress distribution across each section of the structure being analyzed. In WCAP-5890, Revision 1, the actual strain hardining behavior has been conservatively approximated by the use of the simplified trapezoidal stress-strain curve. The applicability of the deviations in WCAP-5890, Rev. 1, to the strain hardening domain is further discussed in Reference 6 of WCAP-5890, Rev. 1.

The employment of qualified welding procedures and qualified welders and thorough inspections assures that welds on Class I components and piping will have little, if any, effect on the tensile properties of basic materials.

A series of tests recently performed at Westinghouse APD laboratories revealed no difference in tensile properties between welded and non-welded pipes. The experimental data are still under evaluation but preliminary results are shown in Table 15.5 and 15.6 and Figures 15.1 and 15.2.

Accidental imperfections of the order of magnitude of those that pass inspection are also expected to be of no significance. Because of chemistry control of the employed coolants and periodic inspections, corrosion is not anticipated to be a problem.

The only component affected by irradiation is the reactor vessel. Irradiation of the reactor vessel is significant only in the area adjacent to the core. High stress areas, i.e., nozzle to shell junctures, are only slightly affected by irradiation. The neutron exposure of these areas will be calculated and its effect on the stress-strain curve evaluated. The corrected stress-strain curve will then be used in the development of the limit curves using the method presented in WCAP-5890, Rev. 1.

<u>Note 1</u>

This second revision of WCAP-5890 will be issued in the near future. The revision will affect only the stress limits for the combination of normal loads plus maximum potential earthquake loads plus pipe rupture loads associated with a major loss of reactor coolant accident. The change will reflect recent agreement with the AEC. Details of the manner in which this revision will be developed are as follows:

1) Use material data to develop stress-strain curves.

Typical stress-strain curves of Type 304 stainless steel (Fig. 15.3), Inconel 600 (Fig. 15.4) and SA 302B low alloy steel (Fig. 15-5) at 600°F have been generated from tests using graphs of applied load versus cross-head displacment as automatically plotted by the recorder of the tensile test apparatus. The scale and sensitivity of the test apparatus recorder assure accurate measurement of the uniform strain.

For materials other than these three, stress-strain curves will be developed by conservative use of pertinent available material data (i.e., lowest values of uniform strain and initial strain hardening). Should the available data not be sufficient to develop a reliable stress-strain curve, three standard ASTM tensile tests of the material in question will be performed at design temperature. These data will be conservatively applied in developing a stress-strain curve as described above.

2) Normalize the ordinate (stress) of the stress-strain curves to the measured yield strength.

3) Use 20% of uniform strain as defined on the curve developed under Item 1 as the allowed membrane strain.

4) Establish the normalized stress ratio at 20% of uniform strain on the normalized stress ratio-strain curves developed under Item 2.

5) Establish the value of the membrane stress limit.

Multiply the normalized stress ratio in Item 4 by the applicable code yield strength at the design temperature to get the membrane stress limit. As an alternate, the actual physical properties as determined from standard ASTM tensile tests on specimens from the same heats may be used to determine the membrane stress limit. If such an approach is adopted, sufficient documentation will be provided to support the actual material properties used.

6) Develop limit curves for the combination of local membrane and bending stresses.

The limit curves will be developed by using the analytical approach presented in WCAP-5890, Rev. 1 and the stress-strain curve up to the membrane stress limit as developed under Item 5. It is anticipated that these limit curves will be within the limit curves discussed with the AEC DRL Staff during the meetings of November 30 and December 1, 1967 for the same materials.

TABLE 15.1

DAMPING FACTORS

Component	Per Cent of Critical Damping
Containment Structure	
(a) Hypothetical earthquake (larger)	5.0
(b) Design earthquake (smaller)	2.0
Concrete Support Structure of Reactor Vessel	2.0
Steel Assemblies:	
(a) Bolted or Riveted	2.5
(b) Welded	1.0
Vital Piping Systems	0.5
Concrete Structures above Ground:	
(a) Shear Wall	5.0
(b) Rigid Frame	5.0

TABLE 15.2 LOADING COMBINATIONS AND STRESS LIMITS

LOADING COMBINATIONS	VESSELS	PIPING	SUPPORTS
1. Normal Loads	$P_m \leq S_m$ $P_L + P_B \leq 1.5\ S_m$	$P_m \leq S$ $P_L + P_B \leq S$	Working Stresses or Applicable Factored Load Design Values
2. Normal + Design Earthquake Loads	$P_m \leq S_m$ $P_L + P_B \leq 1.5\ S_m$	$P_m \leq 1.2\ S$ $P_L + P_B \leq 1.2\ S$	1-1/3 Working Stresses or Applicable Factored Load Design Values
3. Normal + Maximum Potential Earthquake Loads	$P_m \leq 1.2\ S_m$ $P_L + P_B \leq 1.2\ (1.5\ S_m)$	$P_m \leq 1.2\ S$ $P_L + P_B \leq 1.2\ (1.5\ S)$	Deflections and Stresses of Supports Limited to Maintain Supported Equipment Within Their Stress Limits
4. Normal + Pipe Rupture Loads	$P_m \leq 1.2\ S_m$ $P_L + P_B \leq 1.2\ (1.5\ S_m)$	$P_m \leq 1.2\ S$ $P_L + P_B \leq 1.2\ (1.5\ S)$	Deflections and Stresses of Supports Limited to Maintain Supported Equipment Within Their Stress Limits
5. Normal + Maximum Potential Earthquake + Pipe Rupture Loads	Limit Curves of WCAP-5890, Rev. 1 as Modified by Note 1 of This Answer	Limit Curves of WCAP-5890, Rev. 1 as Modified by Note 1 of This Answer	Permanent Deflections of Supports Limited to Maintain Supported Equipment Within Limit Curves of WCAP-5890, Rev. 1 as Modified by Note 1 of This Answer

Where P_m = primary general membrane stress; or stress intensity

P_L = primary local membrane stress; or stress intensity

P_B = primary bending stress; or stress intensity

S_m = stress intensity value from ASME B & PV Code, Section III

S = allowable stress from USAS B31.1 Code for Pressure Piping

TABLE 15.3 - MINIMUM MARGINS OF SAFETY

LOADING CONDITIONS

Material	Normal plus Design Earth-quake Loads	Normal plus Maximum Potential Earthquake	Normal plus pipe Rupture Loads	Normal plus* Maximum Potential Earthquake plus Rupture Loads
SA302Gr.B	200%	150%	150%	27%
Inconel 600	228%	172%	172%	43%
316 SST	222%	169%	169%	60%
A212Gr.B	346%	272%	272%	55%

* Based upon limit curves computed using Note 1 of this attachment

TABLE 15.4 INTERNALS DEFLECTIONS UNDER ABNORMAL OPERATIONS

(Inches)

	Calculated Deflection (Preliminary)	Allowable Limit	No Loss-of-Function Limit
Upper Barrel, expansions/compression (to assure sufficient inlet flow area/and to prevent the barrel from touching any guide tube to avoid disturbing the RCC guide structure).	0.072	3	6
Upper Package, axial deflection (to maintain the control rod guide structure geometry).	0.005	1	2
RCC Guide Tube, cross section distortion (to avoid interference between the RCC elements and the guides).	0	0.035	0.072
RCC Guide Tube, deflection as a beam (to be consistent with conditions under which ability to trip has been tested).	0.2	1.0	1.5
Fuel Assembly Thimbles, cross section distortion (to avoid interference between the control rods and the guides).	0	0.035	0.072

TABLE 15.5 - TESTS AND PRELIMINARY TEST RESULTS ON SA 106B CARBON STEEL PIPE SPECIMENS (internal pressurization = 300 psia)

	Pseudo-elastic Axial Stress Normalized to the Yield Stress	Pseudo-elastic Bending Stress Normalized to the Yield Stress	Pseudo-elastic Stress Intensity Normalized to the Yield Stress	Strain % (gage length)
Pure Tension (no weld)	1.736 1.840	0.0 0.0	1.770 1.874	22.61% (12") 22.32% (12")
Pure Bending*	0.10	>2.348	>2.382	>35.4% (1")
Tension + Bending* (no weld)	>1.375 >1.585 >1.845	>1.030 >0.565 >0.266	>2.440 >2.180 >2.145	>7.75% (6") -- --
Compression + Bending* (no weld)	>-1.130	1.08	2.410	--
Pure Tension (circumf. weld)	1.852	0.0	1.886	20.05% (12")
Pure Bending* (circumf. weld)	0.10	>2.580	>2.614	>30.19% (1.5")
Pure Bending ** (rejected circum. weld)	0.10	2.460	2.494	14.51 (2")

* The limit capability of the test apparatus has been reached before failure of these specimens was approached.

** This test was performed on a welded pipe specimen which has been rejected by the inspector prior to the test for gross lack of penetration in weld. This was the only test in which the weld failed. The specimen exhibited substantial ductility prior to failure.

TABLE 15.6 - TESTS AND PRELIMINARY TEST RESULTS ON 304 STAINLESS STEEL SPECIMENS (internal pressurization = 3000 psia)

	Pseudo-elastic Axial Stress Normalized to the Yield Stress	Pseudo-elastic Bending Stress Normalized to the Yield Stress	Pseudo-elastic Stress-Intensity Normalized to the Yield Stress	Strain
Pure Tension (no weld)	2.495	0.0	2.53	52.1% (12" gage length)
Pure Bending* (no weld)	+0.10	>±2.91	>2.945	>30.0% (1" gage length)
Tension + Bending* (no weld)	>+2.09	>±0.42	>2.55	--
Pure Bending (with circumf. weld)	+0.10	>±3.27	>3.30	>25.0% (1.5" gage length)
Pure Tension (with circumf. weld)	+2.46	0.0	2.49	44.6% (12" gage length)

* The limit capability of the test apparatus has been reached before failure of these specimens was approached.

DESIGN LIMITS COMPARED TO EXPERIMENTAL POINTS

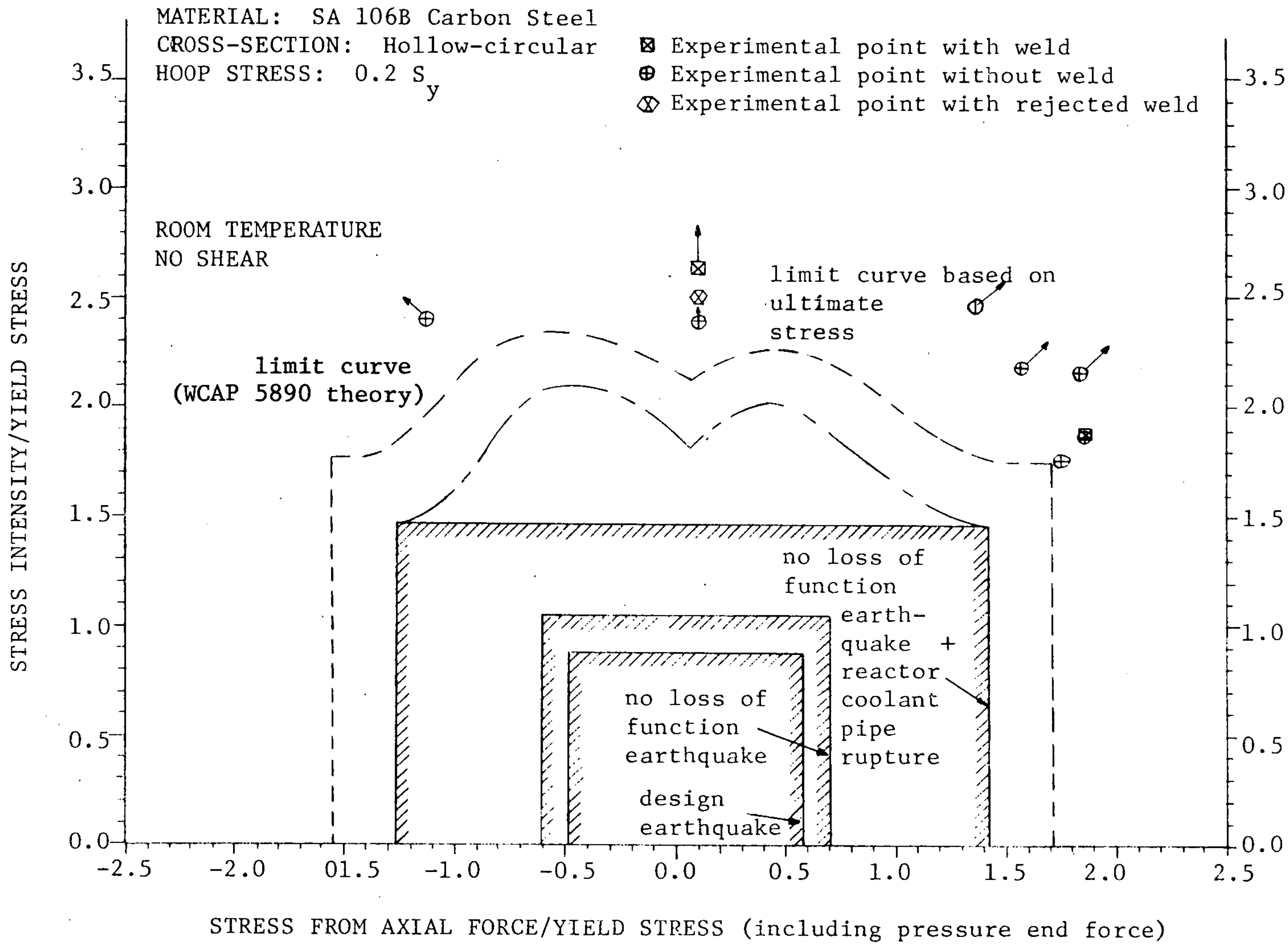


FIGURE 15.1
Supplement 1

DESIGN LIMITS COMPARED TO EXPERIMENTAL POINTS

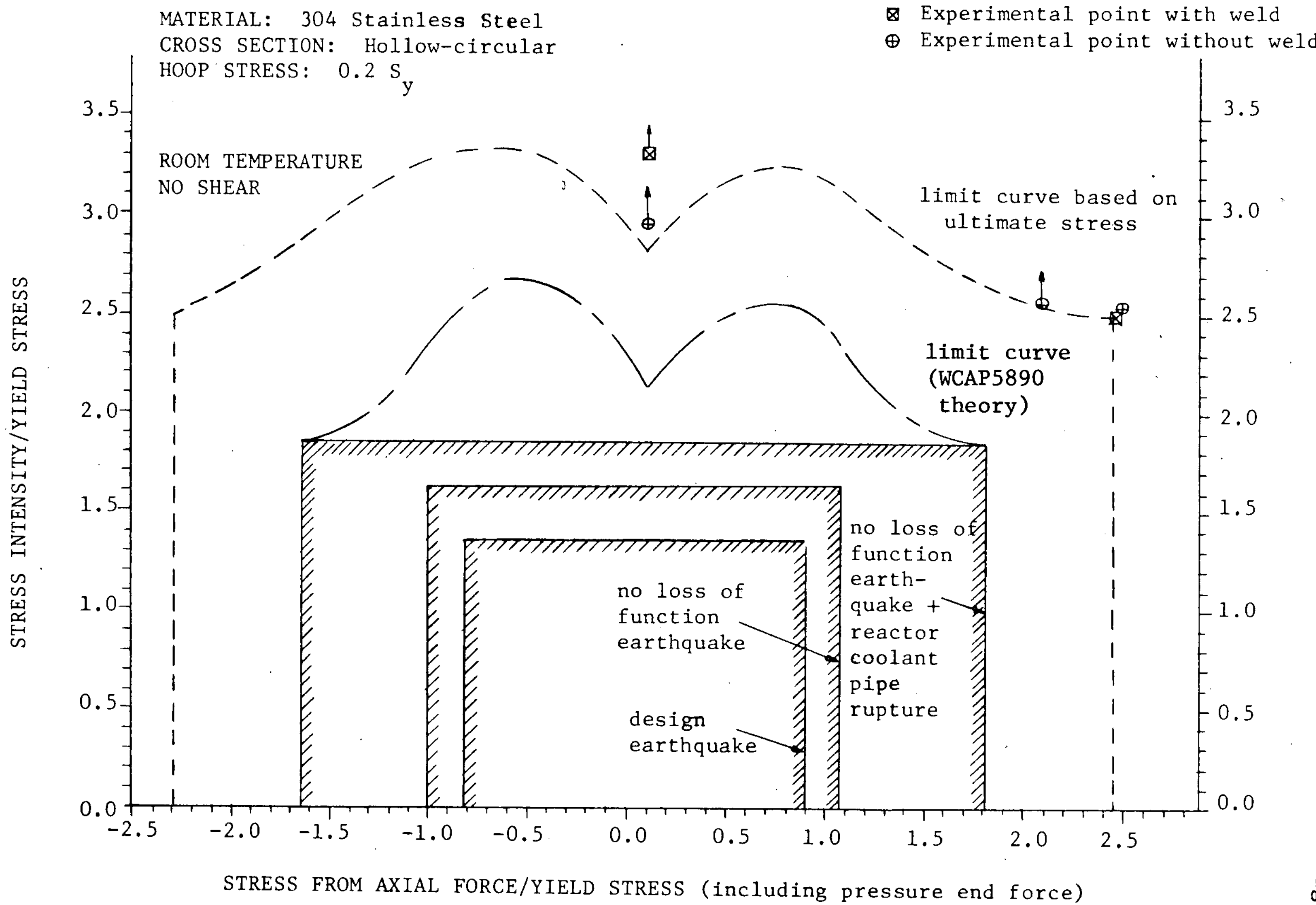


FIGURE 15.2
SUPPLEMENT 1

TYPICAL STRESS STRAIN CURVE
STANDARD ASTM TENSILE TEST
MATERIAL: 304 STAINLESS STEEL
TEMPERATURE: 600°F

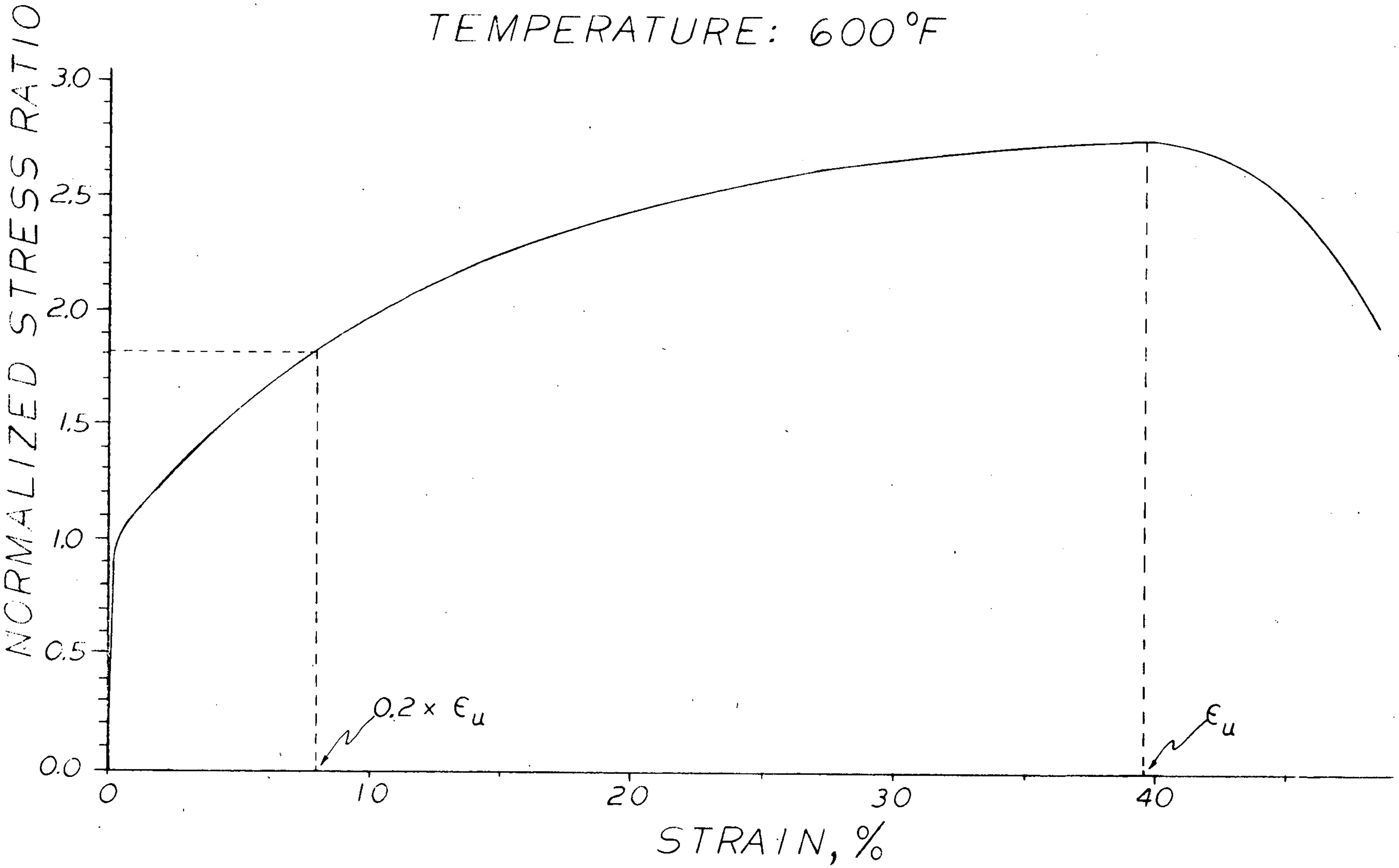


FIGURE 15.3
SUPPLEMENT 1

TYPICAL STRESS STRAIN CURVE
STANDARD ASTM TENSILE TEST
MATERIAL: INCONEL 600
TEMPERATURE: 600 °F

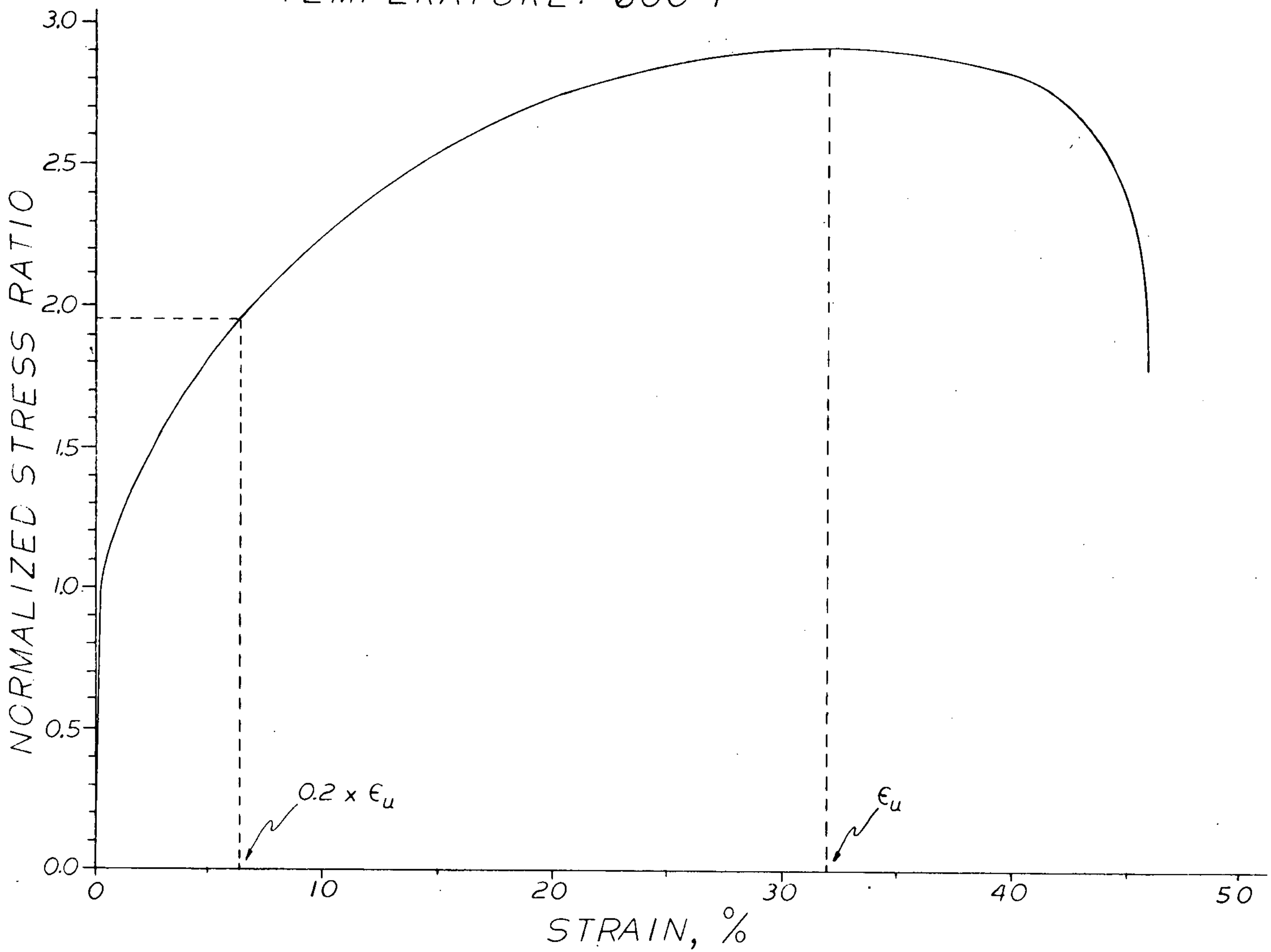


FIGURE 15.4
SUPPLEMENT 1

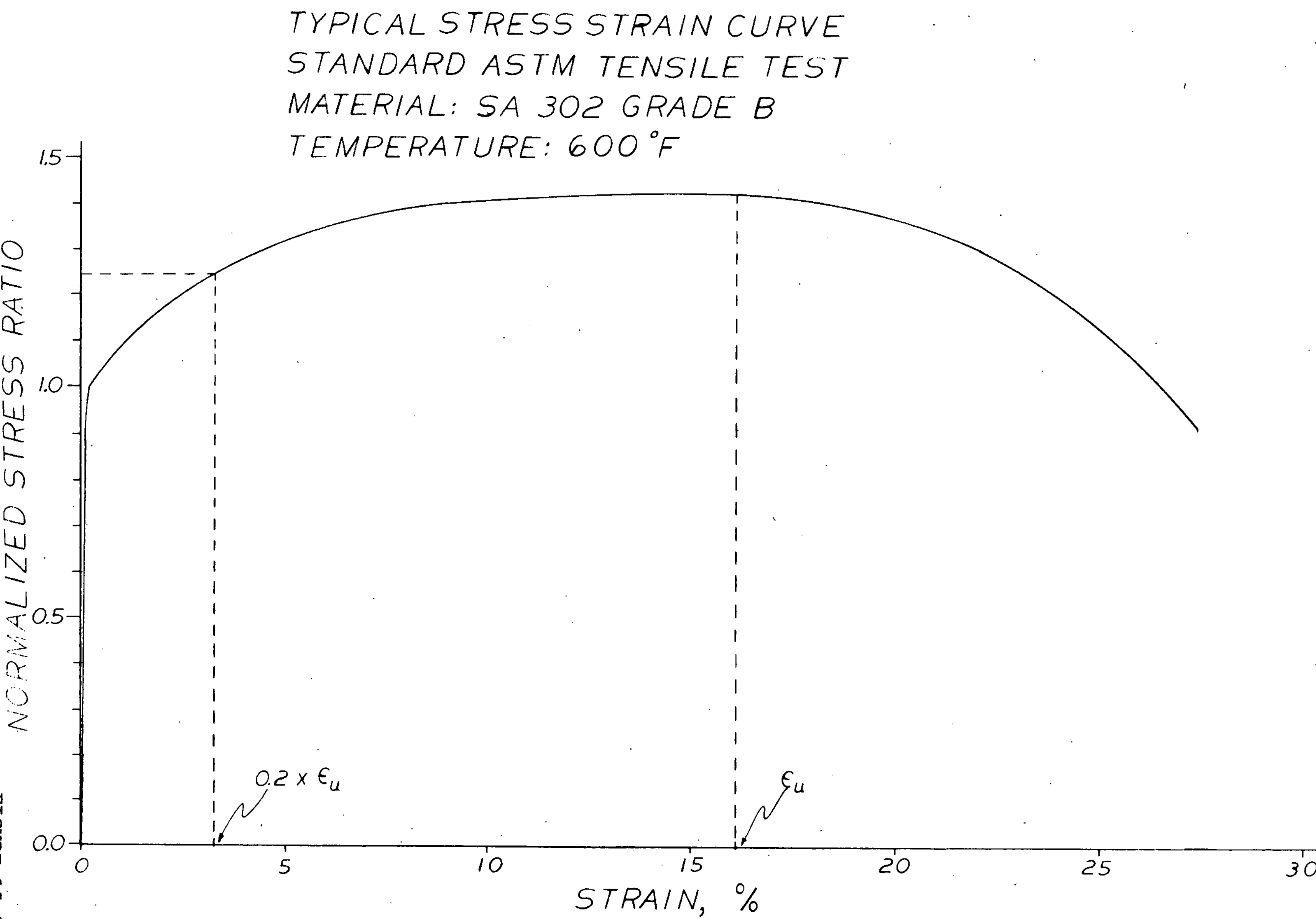


FIGURE 15.5
SUPPLEMENT 1

ITEM 16. (Attachment E Question 1.1)

Provide an analysis of the following "start-up accident" for a representative spectrum of initial power levels: Simultaneous withdrawal of all rods not already in the full-out position, assuming the excursion is terminated only by action of the "High Nuclear Flux" channels and the inherent negative feedback characteristics of the reactor itself.

ANSWER

The RCC assembly mechanisms will be powered by two power supply units. Each unit is capable of driving only one RCC assembly bank at a time. By means of a switch, the reactor operator may select either manual or automatic control for the movement of the RCC assemblies. In manual control he may select any bank of RCC's but the single switch is used for the selection so that not more than one bank can be moved at a time unless in an overlap condition. In automatic control, the banks are moved in their normal, programmed sequence with one bank moving at a time. In both automatic and manual control positions there is some overlap as one control bank near its full out position and the next control bank begins to move from its full in position. Thus, the design of the RCC mechanism power supply system prevents the simultaneous withdrawal of more than the equivalent of two RCC banks (either control or shutdown banks).

Since the sources of power available to drive the banks is limited by design as stated above, the banks must be withdrawn sequentially by transferring the power from one bank to the next. Any failure in the control or transfer system which could connect two or more banks to one power unit would not permit withdrawal of the banks because of insufficient power and would probably result in the dropping of the RCC banks into the core for the same reason.

Relay interlocks, whose function is to prevent withdrawal of two or more banks concurrently, are also provided and when combined with the single selector switch and the limited power capability of the mechanism power units, the protection provided for prevention of simultaneous withdrawal of all RCC assemblies meets the single failure criterion.

Following are results for a typical pressurized water reactor. Figure 1 shows the effect of initial power level on peak heat flux 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 this 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 2 through 4 show the transient behavior for a reactivity insertion rate of 8×10^{-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 2 shows the nuclear power increase. The power is seen to increase to the trip point in 9.3 seconds.

The nuclear power overshoots to approximately 720 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 3. The beneficial effect of the inherent thermal lag of the fuel is evidenced by a peak heat flux of only 36.6 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 4 shows the response of the average fuel, cladding and coolant temperatures. The fuel temperature increases to 992°F. The average coolant temperature increases only to 556°F.

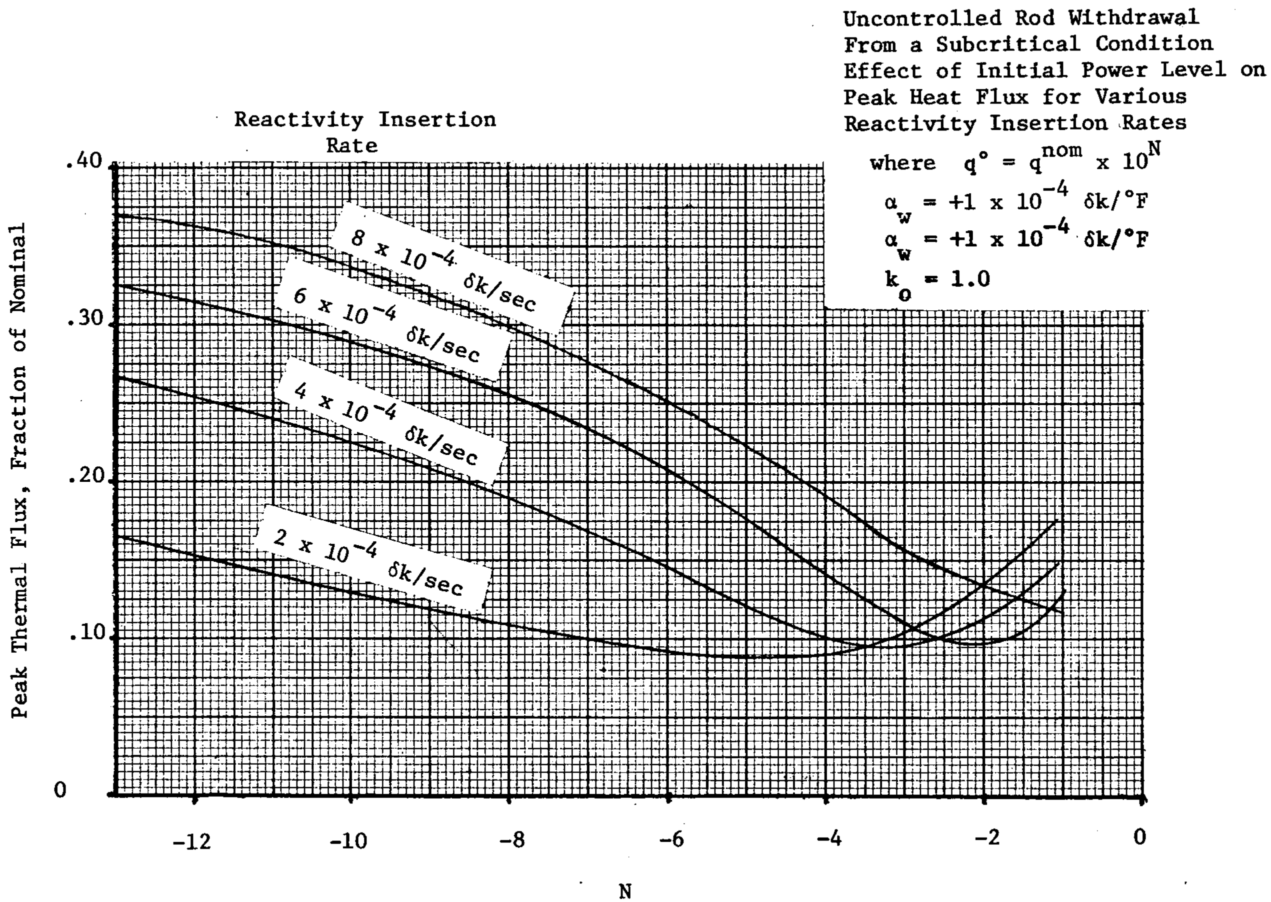


FIGURE 1
Supplement 1

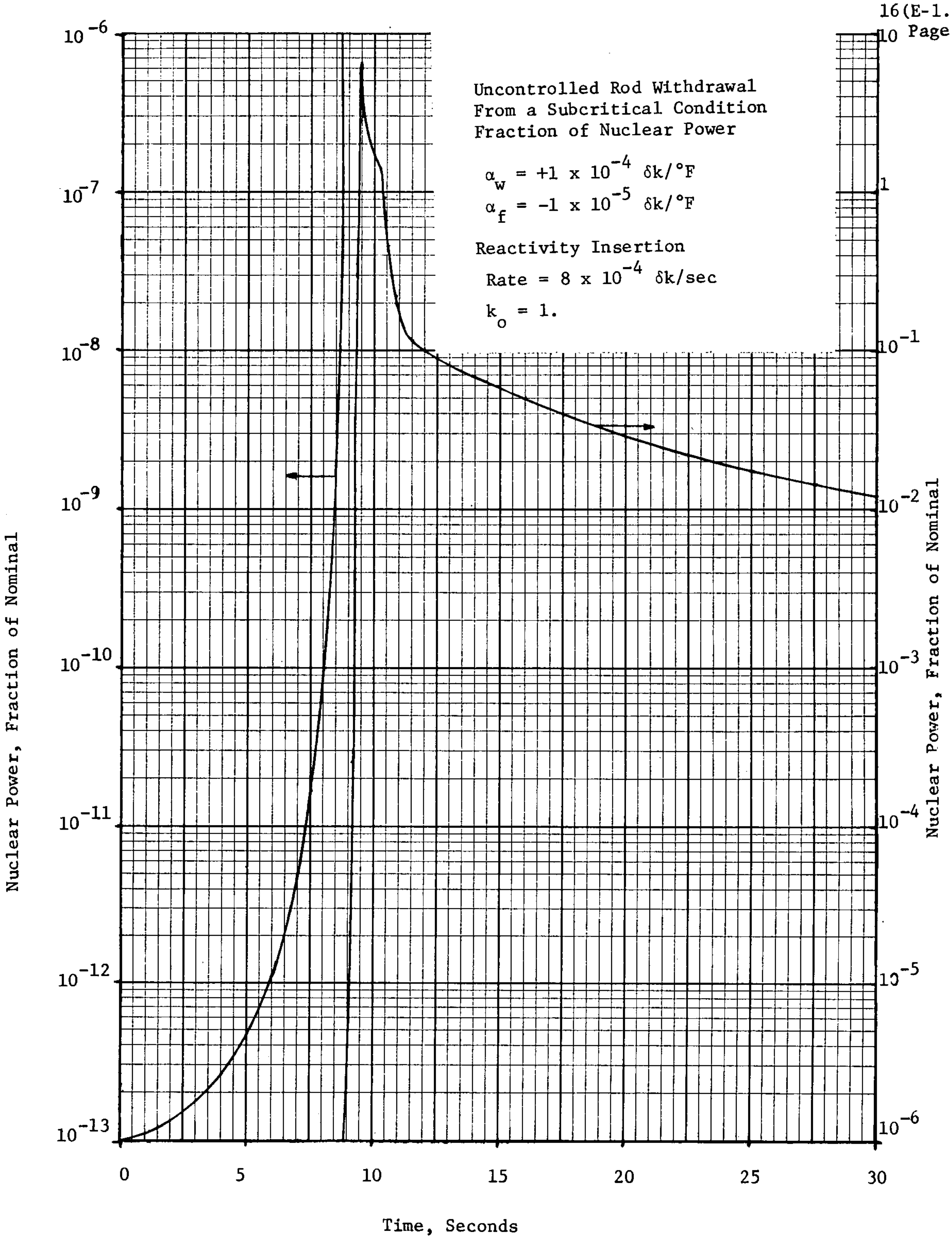

Uncontrolled Rod Withdrawal
From a Subcritical Condition
Fraction of Nuclear Power
$\alpha_w = +1 \times 10^{-4}$ δk/°F
$\alpha_f = -1 \times 10^{-5}$ δk/°F
Reactivity Insertion
Rate = 8×10^{-4} δk/sec
$k_o = 1.$
10^{-6}
10^{-7}
10^{-8}
10^{-9}
10^{-10}
10^{-11}
10^{-12}
10^{-13}
Nuclear Power, Fraction of Nominal
10
1
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-6}
Nuclear Power, Fraction of Nominal
0
5
10
15
20
25
30
Time, Seconds

FIGURE 2

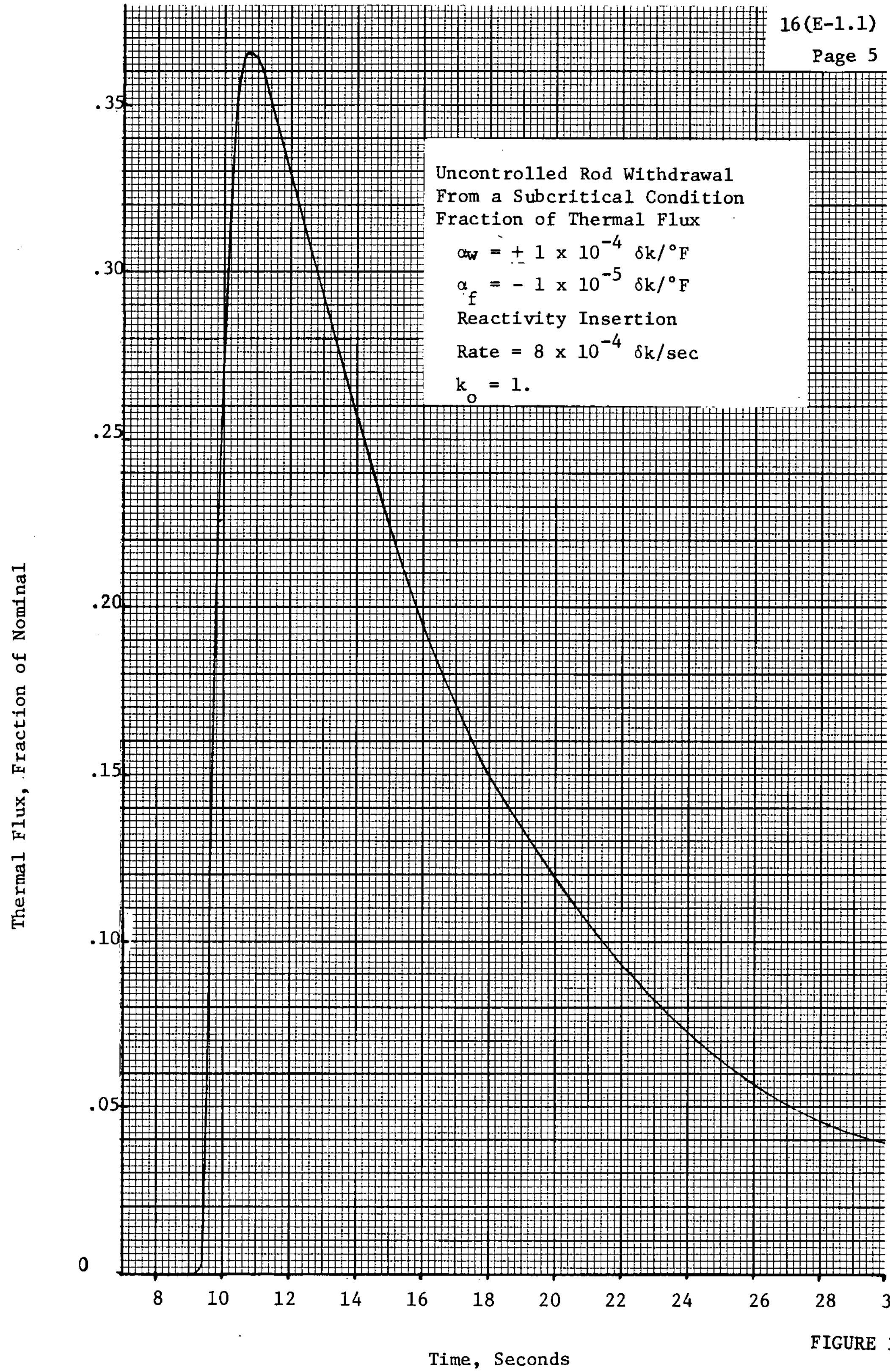

Uncontrolled Rod Withdrawal
From a Subcritical Condition
Fraction of Thermal Flux
$\alpha_w = \pm 1 \times 10^{-4}\ \delta k/°F$
$\alpha_f = - 1 \times 10^{-5}\ \delta k/°F$
Reactivity Insertion
Rate = $8 \times 10^{-4}\ \delta k/sec$
$k_o = 1.$
Thermal Flux, Fraction of Nominal
.35
.30
.25
.20
.15
.10
.05
0
8
10
12
14
16
18
20
22
24
26
28
30
Time, Seconds

FIGURE 3

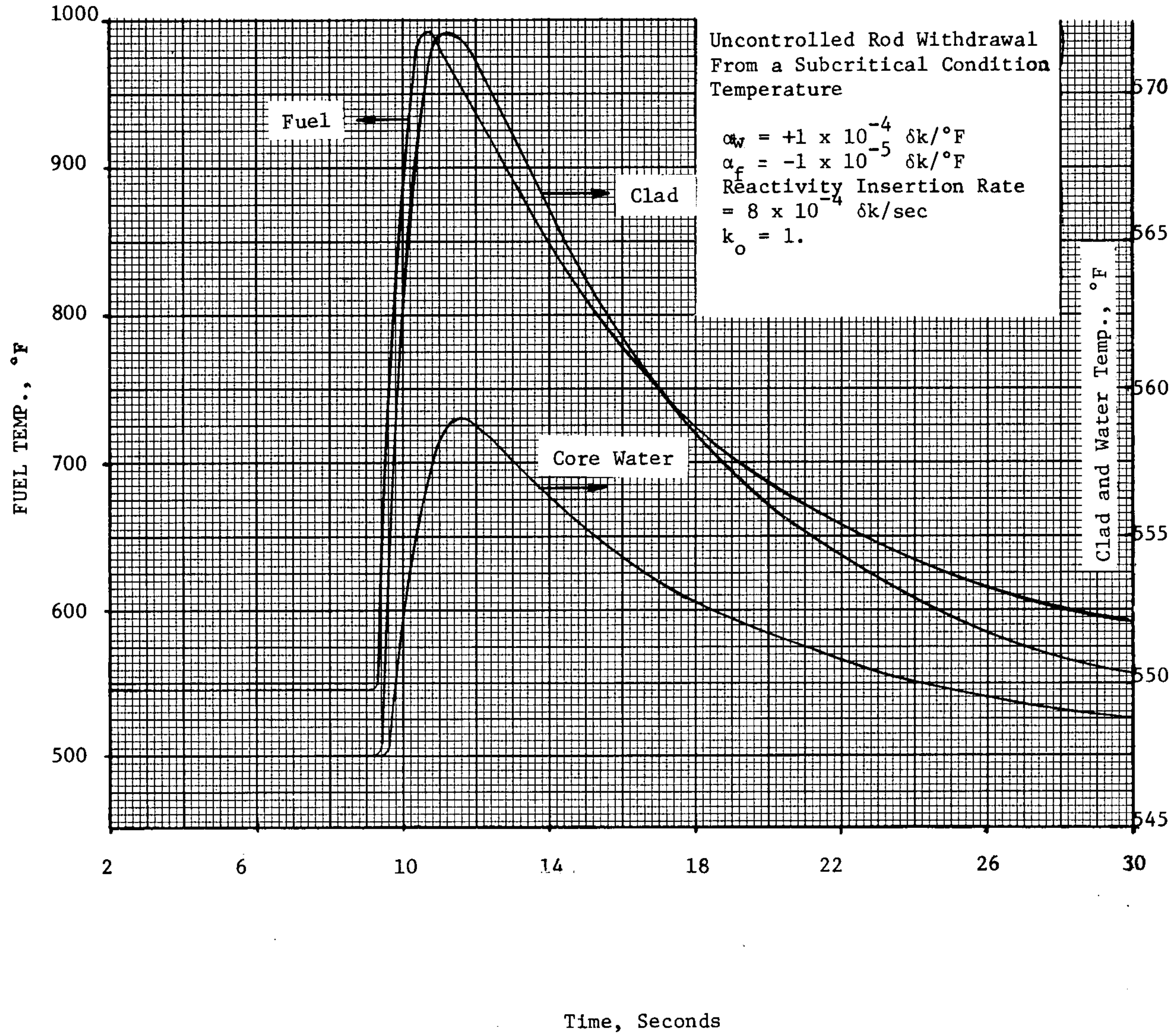


FIGURE 4

SUPPLEMENT 1

ITEM 16 (Attachment E Question 1.2)

State your criterion regarding the minimum acceptable DNB ratio for a transient resulting from an uncontrolled rod withdrawal.

ANSWER

The minimum allowable DNB ratio for a transient resulting from an uncontrolled rod withdrawal is 1.30.

ITEM 16 (ATTACHMENT E QUESTION 2.1)

Indicate the basis for the assumed pressurizer water level prior to the initiation of a turbine trip. What margin will be provided to assure that a "solid" hydraulic primary system will not occur? Evaluate this accident assuming one pressurizer safety valve does not open.

ANSWER

A reactor trip is derived from high water level in the pressurizer with two out of three trip logic. This trip will preclude operation with a solid system. The combination of the fixed high pressure reactor trip and the high water level trip will ensure a trip in the event of a loss of load accident without a turbine trip. Sufficient margin is provided to prevent water relief from the pressurizer including overshoot following the trip actuation independent of the initial water level below the high level setpoint. The closer the initial level is to the high level setpoint, the xmaller the steam volume and the high pressure trip is acutated sooner. Note that a turbine trip automatically actuates a reactor trip directly and would prevent any insurge into the pressurizer. There are also relief valves attached to the pressurizer which are actuated by a high pressure signal.

Primary side safety valve actuation is prevented by acutation of either the high pressurizer pressure trip or the high pressurizer water level trip. Therefore, there is no effect if one pressurizer safety valve is assumed not to open.

ITEM 16 (Attachment E Question 3.1)

Tabulate by isotope the primary system fission product inventory assumed to be present with 1% failed fuel. Describe the model used in deriving this inventory including fuel temperature, diffusion coefficients, size of perforation, and the mode of operation of the demineralizers, and rate of boron dilution.

ANSWER

The fission product activity in the reactor coolant during operation with small cladding defects (approximately 5 mils) in 1% of the fuel rods is computed using the following differential equations:

For parent nuclides in the core, (subscript i)

$$\frac{dN_{C_i}}{dt} = FPY_i - (\lambda_i + \phi\sigma_i + \eta_i)\, N_{C_i}$$

For daughter nuclides in the core, (subscript j)

$$\frac{dN_{C_j}}{dt} = \lambda_i N_{C_i} + FPY_j - (\lambda_j + \phi\sigma_j + \eta_j)\, N_{C_j}$$

For parent nuclides in the coolant,

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

for daughter nuclides in the coolant,

$$\frac{dN_{w_j}}{dt} = D\nu_j\, N_{C_j} - \left(\lambda_j + R\eta_j + \frac{B'}{B_o - tB'}\right) N_{w_j} + \lambda_i N_{w_i}$$

where:

N = population of nuclide

D = fraction of fuel rods having defective cladding

R = purification flow, coolant system volumes per second

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

ϕ = thermal neutron flux neutrons/cm^2-sec

σ = microscopic cross-section, cm^2

F = fissioning rate fissions/Mw-sec

P = power generated in that portion of fuel which has cladding defects, Mw

Y = independent isotope yield, atoms/fission

Subscript C refers to core

Subscript w refers to coolant

The parameters used in the calculation of the reactor coolant fission product inventory, including pertinent information concerning the expected coolant cleanup flow rate, boron dilution rate, and demineralizer effectiveness, are presented in Table 1. The results of the calculations are presented in Table 2. In these calculations the defective fuel rods are assumed to be uniformly distributed throughout the core and the fission product escape rate coefficients are therefore based upon an average fuel temperature. The fuel temperatures assumed in the fuel are given in Table 3.

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 4.

TABLE 1

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.	Pruficiation flow rate (normal), gpm	75
6.	Effective cation demineralizer flow, gpm	7
7.	Volume control tank volumes	
	a. Vapor, cu.ft.	130
	b. Liquid, cu.ft.	270
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	2.3×10^{-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 2

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
Br-84	2.71×10^{-2}	I-133	1.36
Rb-88	2.59	T-134	1.95×10^{-2}
Rb-89	5.99×10^{-2}	I-134	3.56×10^{-1}
Sr-89	2.02×10^{-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^{-4}	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^{-3}
I-132	5.99×10^{-1}	Pr-144	2.31×10^{-3}

Gaseous Fission Products	μc/cc (573°F)
Kr-85	4.42
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 3
FUEL TEMPERATURES

Local Temperature °F	Fraction of Core Fuel Volume Above Given Temperature
4100	0.
3700	0.002
3300	0.018
2900	0.070
2500	0.155

TABLE 4

TRITIUM PRODUCTION IN THE REACTOR COOLANT

Basic Assumptions:

Plant Parameters:

1.	Core thermal power	3216 MWt
2.	Coolant water volume	12,600 ft^3
3.	Core volume	1,152.5 ft^3
4.	Core volume fraction	
	a. UO_2	0.3023
	b. Zr + SS	0.1035
	c. H_2O	0.5942
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.	Boron dilution rate (base load operation)	1.75 ppm/day
8.	Burnable poison boron content (total-all rods)	18.1 kg
9.	Fraction of tritium in core (ternary fission + burnable boron) diffusing thru clad	0.3**
10.	Ternary fission yield	8 x 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 because of the formation of zirconium tritide. Shippingport data indicates that less than 1% of ternary tritium produced is released to the coolant. Although this data cannot be used directly, it does indicate that Zircaloy will reduce tritium diffusion.

TABLE 4 (Continued)

11. 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 > Mev = 75	
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

12. Cooling water flow: 7.5×10^5 gpm 15×10^{14} cc/yr

CALCULATIONS (per year)

		Initial Cycle	Equilibrium Cycle
A.	Tritium from Core (Curies)		
	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
	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

ITEM 16 (Attachment E, Question 3.2)

Indicate the minimum number of simultaneous steam generator tube ruptures which would (1) result in operation of the steam system safety valves, (2) cause fuel cladding damage, and (3) uncover the core hot-spot. Relate these to core water level. Plot minimum DNB ratios, reactor water level, equilibrium primary system pressure and peak secondary system pressure vs. number of tubes ruptured to indicate the sensitivity of the number of tubes rupturing to core conditions. Assume double-ended tube rupture at the peak of the U-bend.

ANSWER

For reasons to be discussed later in this answer, the multiple spontaneous occurrence of gross tube failures in a single incident is not considered credible. In order to perform a rigorous analysis of the flow dynamics of blowdown through multiple tube ruptures one must understand and define mathematically the physical configuration of the ruptures. Because no reasonable mechanism exists for the multiple ruptures, it is instead just as meaningful to analyze the consequences of a pipe rupture, equivalent in terms of discharge rate to various multiples of the single tube rupture discharge rate.

Such an analysis reveals that the core cooling system will prevent clad damage for break discharge rates equal to or smaller than that resulting from a broken pipe between 4 inches and 6 inches in diameter. The discharge rates which bracket the onset of clad damage correspond to 18 and 40 times the discharge from a single severed steam generator tube. Actually the ratio would be much larger owing to the fact that the discharge from a tube failure will be limited by the back pressure in the steam generator. Ultimately the tube discharge would terminate when the reactor coolant system and the steam generator reached pressure equilibrium. The operator can initiate cooldown through the unaffected steam generators.

The discharge rate required to lift a secondary safety valve is about 15 times the rate from a single severed tube.

These conclusions are based on single-failure mode performance of the core cooling system. Clad damage is prevented in those cases where the top of the core does not become uncovered.

The discharge rate required to cause the top of the core to become uncovered is 18 to 40 times the rate from a single severed tube.

The incredibility of multiple simultaneous tube failures is supported by the following reasoning:

1. At the maximum operating internal pressure the tube wall sees only about 1530 psi, compared with a calculated bursting pressure in excess of 11,100 psi based on ultimate strength at design temperature (factor of 7.3); and compared with a prefabrication test pressure of 7,000 psi (factor of 4.5).

2. The above margin applies to the longitudinal failure mode, induced by hoop stress. This failure mode is the least likely to cause propagation of failure tube-to-tube. An additional factor of two applies to ultimate pressure strength in the axial direction tending to resist double-ended failure (total factor of 14.6).

3. Failures induced by fretting, corrosion, erosion or fatigue, in addition to being rendered extremely improbable by design, are of such a nature as to produce tell-tale leakage in substantial quantity while ample metal remains to prevent severance of the tube (a small fraction of the original tube wall section, as indicated by the margin derived in 2). Thus it is virtually certain that any incipient failures that would develop to the point of severe leakage requiring a shutdown for repair would happen long before the large safety margin in pressure strength is lost.

x

ITEM 16 (Attachment E Question 3.3)

Justify the iodine separation factor assumed in your analysis of steam generator tube rupture. If it is based on experimental data, relate experimental conditions to those present during a tube rupture.

ANSWER

In the analysis of the steam generator tube rupture, release of activity through the air ejector from the secondary water leaving the condenser at low pressure is based upon an iodine partition factor of 10^{-4} or less from experiments performed in Canada[1] and in Russia.[2] This release is from a system under highly hydrolized conditions, a pH less than 8, and with a highly dilute solution having an iodine concentration less than 10^{-4} mole/liter.

The partition factor of 10^{-4} has been observed with pressure of the order of 20 psia, pH=7.5 - 8.0. Although at the present time no data are available for much lower pressures comparable with those inside the condenser it is shown in Reference 2, page 738, Fig. 4 that the factor decreases with decreasing pressure for values of pH above 7. Hence a lower value might be used. In addition no credit has been taken for the transient of the secondary concentration in the steam generator and in the condenser after tube rupture, which could further decrease any possible release of activity.

In the hypothesis of loss of power, bypass to the condenser is not possible and steam is released to the atmosphere through the main relief valves. This condition is approximately represented by the upper curve in Fig. 5 page 738 of Reference 2. For a steam generator pressure of 1050 psia and a pH value between 8 and 9, the average partition factor is seen to be 4×10^{-3}. Since the moisture carry-over doesn't exceed 0.25%, the total partition factor is 6.5×10^{-3}.

[1] L. C. Watson, A. R. Bancroft and C. W. Hoelke, "Iodine Containment by Dousing in NPD-11", AECL-1130 Atomic Energy of Canada Limited, Chalk River, Ontario, October 27, 1960.

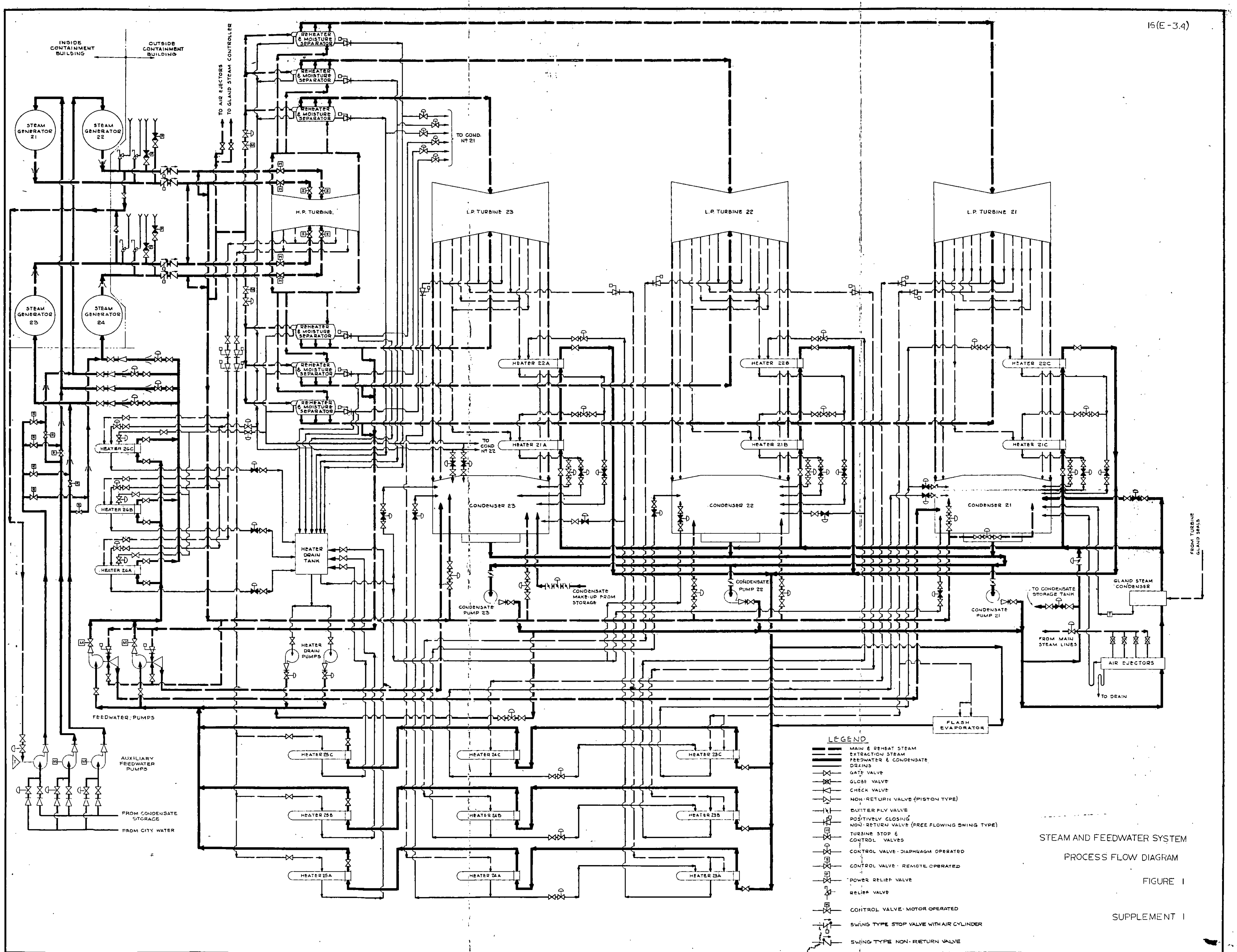
[2] M. A. Styrikovich, O. I. Martynova, K. Ya. Katkovskaya, I. Ya. Dubrovskii, and I. N. Smirnova, "Transer of Iodine from Aqueous Solutions to Saturated Vapor," Atomriaya Energiya, Vol. 17, No. 1, pp. 45-49, July 1964.

ITEM 16 (Attachment E Question 3.4)

It appears that rupture of the steam bypass line upstream of the stream bypass valve could result in simultaneous blowdown of all four steam generators. Analyze the consequences of this rupture indicating maximum fuel clad temperature and extent of fuel damage.

ANSWER

Reference is made to the revised Steam and Feedwater System process flow diagram shown in Figure 1. 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 four main steam lines are interconnected local to the turbine. The isolation valves contain free swinging discs which are normally held up out of the main steam flow by an air piston. These valves are closed upon a signal of either high steam flow in 2/4 main steam lines or 2/3 high containment pressure. These valves are designed to close in less than 5 seconds. Thus a rupture of the steam bypass line upstream of the steam bypass valve can not result in a simultaneous blowdown of all four steam generators.



STEAM AND FEEDWATER SYSTEM
PROCESS FLOW DIAGRAM
FIGURE 1
SUPPLEMENT 1

ITEM 16 (Attachemnt E Question 3.5)

In the event of a steam-line rupture coincident with the failure of an RCC assembly to scram, return to criticality and core damage may occur. State your criterion regarding the extent of core damage which is considered acceptable and plot power level and effective multiplication factor vs. time. Assuming a primary system fission product inventory equivalent to that resulting from this limiting core damage plus an additional 1% failed fuel, calculate the dose to the environs considering the maximum primary-to-secondary steam generator tube leakage at which the loop will be allowed to operate. Peform a similar analysis considering the consequences of rupture of the steam bypass line.

ANSWER

Core protection is provided in the same manner as for reactivity insertion accidents by the combination of overpower and over-temperature trips (i.e., when the resulting reactor power increase exceeds the setpoint of either the overpower or overtemperature protection systems).

The design bases for the hypothetical steam break accident are as follows:

1) With a stuck rod and minimum engineered safety features*, the core remains in place and eventually intact so as not to impair effective cooling of the core.

2) With no stuck rod and all equipment operating at design capacity, insignificant (or no) cladding rupture occurs.

The analysis presented here is only of a preliminary nature since final design details are not completed. More detailed analyses will be performed during final design and it will be shown that criteria 1 and 2 above are met.

The steam line break constitutes an uncontrolled heat removal from the Reactor Coolant System which is limited by the steam line non-return and trip isolation valves. These valves cannot preclude the blowdown of one steam generator if the break occurs upstream of the isolation valve.

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*Refer to Chapter 6 of the PSAR for listing of minimum safety features.

In this case, there is a rapid cooldown of the Reactor Coolant System, which results in a reduction in shutdown reactivity margin after trip. The resultant coolant contraction has the characteristics of the beginning of a loss-of-coolant accident and results in the initiation of engineered safety features as the pressurizer is emptied. By boron addition, these systems compensate for the temperature effect on reactivity in all anticipated cases. If the flux distribution resulting from a single stuck RCCA is sufficiently distorted, DNB may occur at relatively low power local to the stuck assembly, causing clad deformation and possible clad rupture is limited core region. Only in the extreme cases, where the combined effects of a stuck RCC assembly, the largest steam rupture and the most adverse reactivity coefficients are imposed, does the possibility exist for sufficient power generation to cause core damage. The extent of possible damage is determined during detailed design analysis. The duration and magnitude of the peak power generation is limited so that conditions are significantly below those which can lead to fuel clad melting. Public and plant personnel safety is assured by the fact that the transient is self-limiting as rising coolant temperatures, blowoff of available coolant in the secondary steam systems and the continued boron addition act to terminate the transient, and by the fact that any fission products released are confined to the Reactor Coolant System. There is no primary system overpressure transient following a steam-line rupture.

In the analyses no fuel damage is assumed to occur unless DNB is reached. Although DNB may not be sufficient to cause fuel damage, it has been assumed that DNB leads to possible clad perforation when determining the number of damaged fuel rods. At the primary system pressures occurring during the transient a clad temperature of approximately 1100°F is required to cause fuel failure. This clad temperature will not be reached unless DNB occurs.

Analyses on transients caused by a steam line break have been performed for a higher power rating than that of Indian Power Unit 3 power level and are presented here. Detailed analyses for Indian Point Unit 3 are currently being prepared.

Figures 1, 2, and 3 show the typical effect upon the core and primary system of a 4.6 sq. ft. break at the exit of a steam generator (complete severance of a main steam line). It can be seen in Figure 1 that the reactor returns to criticality about twenty-five seconds after the accident. Heat flux rises in one minute to a peak value of 40% of the average full power value. As shown typically in Figure 2, the first reversal in the reactivity transient occurs at about thirty-five seconds as a result of the return to power and the resultant increase in fuel temperature combined with the arrest of the Reactor Coolant System cooldown. The rapid decrease in reactivity at about eighty-seven seconds is a result of safety injection water containing 20,000 ppm boron reaching the core from the boron injection tank. This effectively terminates the power transient. The small reactivity increase at ninety seconds is a consequence of the effect of reactor coolant loop transit time and the emptying of the break affected steam generator.

Detailed physics calculations of stuck rod worth and the power distribution associated with the stuck rod are not yet completed. However, assuming the design value of 1% shutdown with a stuck rod a steam line break inside the containment could lend to DNB and possible clad perforation (no melting or zirconium-water reaction). If any damage were to occur it would be limited to the vicinity of the stuck control rod and would not damage in excess of 10% of the fuel rods.

Figures 4 and 2 show the typical effects of break outside of the containment and downstream of the flow nozzle. The effective size of this break is 1.5 sq. ft. Criticality occurs at about forty seconds with the peak heat flux rising to 25% of the average full power value in about eight seconds. The power transient is terminated when 20,000 ppm boron safety injection water from the boron injection tank reaches the core at 110 seconds. The rapid decrease in core reactivity that is seen for the larger 4.6 sq. ft. break following injection does not occur in this case because the steam generator is not yet emptied. The steam generator empties in this accident in somewhat less than three minutes. For this case no DNB is expected even with a stuck rod.

Figure 5 shows the typical transient for a steam line break equivalent to the opening of one steam generator safety valve. In this case water containing 2500 ppm boron enters the Reactor Coolant System at about 285 seconds from the Safety Injection System lines and is sufficient to prevent any power generation. At 380 seconds, the Safety Injection lines are cleared of 2500 ppm borated water and 20,000 ppm boron injection tank enters the Reactor Coolant System and provides sufficient negative reactivity to keep the reactor well below criticality while the steam generator empties and causes further cooldown.

Figure 3 shows the reactor coolant pressure and pressure in the break-affected steam generator as a function of time. The secondary pressure, as expected, reduces to ambient when the steam generator is emptied.

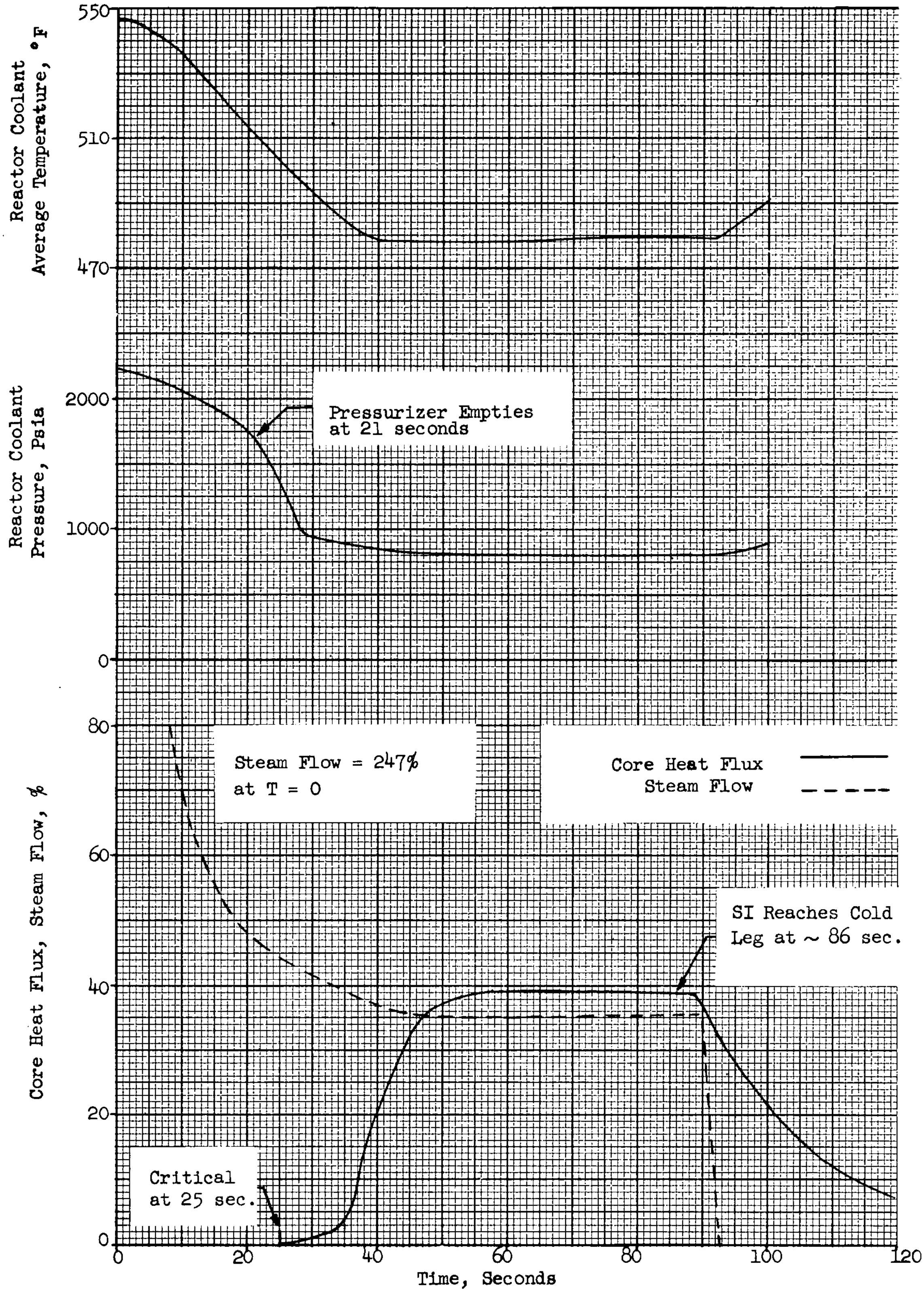
In all of the above anlyses, reactor coolant pumps are assumed to be operating at full flow since this creates the most rapid primary system cooldown. Minimum engineered safety features are also assumed.

Analyses have been performed for a steam line break accident for which it is assumed that a 10 gpm tube leak exists during normal operation prior to the accident.

The calculated leakage rate during and after depressurization of the affected steam generator secondary side is shown in Figure 6. In the calculations it has been conservatively assumed that the secondary side pressure drops almost instantaneously to atmospheric pressure and the primary system pressure decreases linearly to 350 psia from 2250 psia in four hours. This corresponds to a cooldown rate on the order of 50°F/hr. At this time the residual heat removal system is placed in operation to continue cooling the primary system. The primary system pressure during this period is assumed to be maintained constant at 350 psia. The residual heat removal system is capable of cooling the primary system to less than 200°F in four hours. Eight hours following the accident the reactor coolant pumps can be shut off and the primary system pressure reduced to atmospheric. At this time, the primary-to-secondary leakage drops essentially to zero. The leakage has been assumed proportional to the square root of the primary-to-secondary differential pressure.

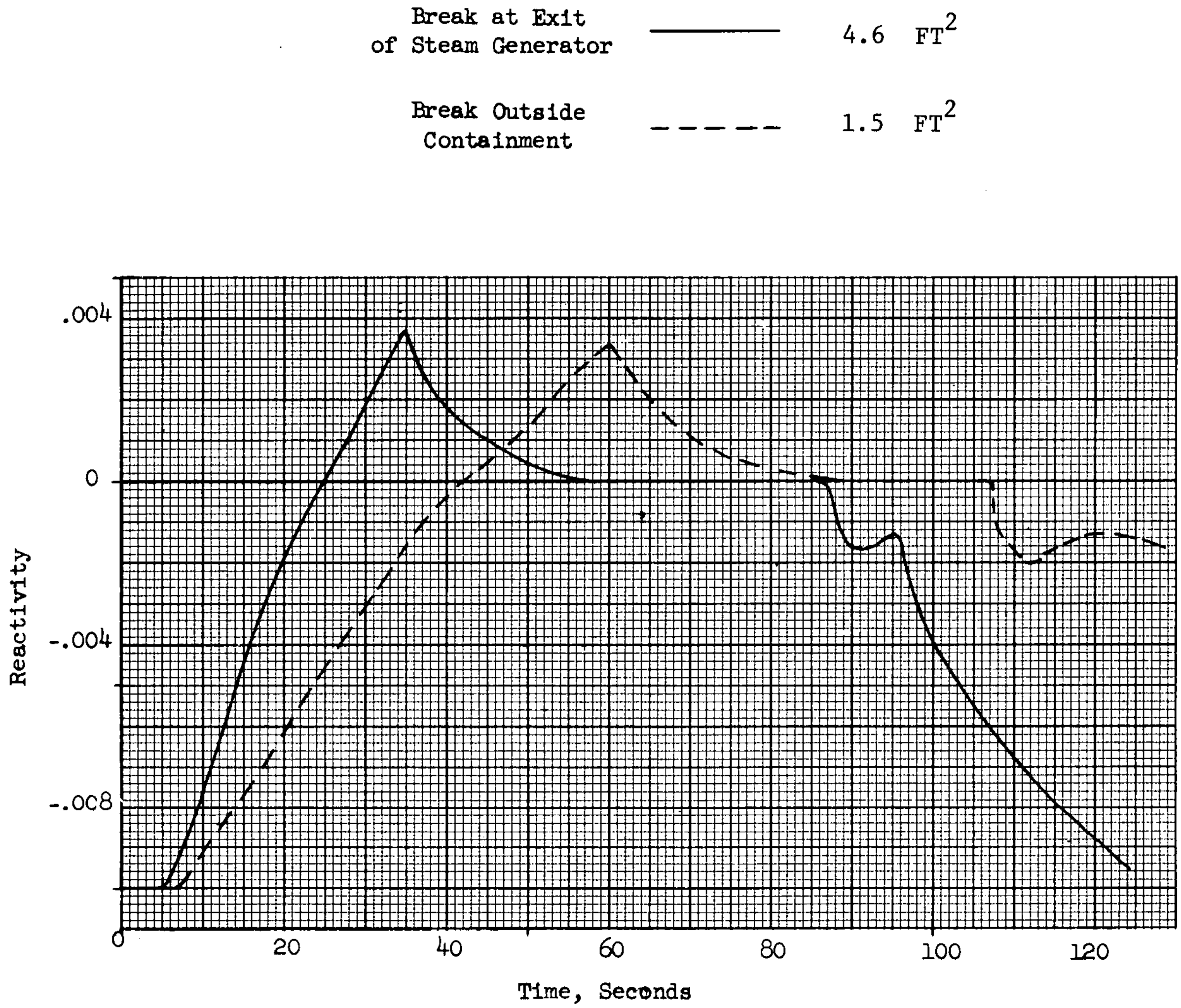
The integrated leakage of reactor coolant to the steam generator secondary side over the eight hour period is approximately 442 cu. ft. This amounts to only 3.6% of the total reactor coolant volume. It is therefore assumed that 3.6% of the equilibrium reactor coolant activity corresponding to operation of 3216 MWT with small cladding defects in 1% of the fuel rods would be available for release through the steam line break during this eight hour cooldown period. The resultant dose will be well below the suggested limits of 10 CFR 100.

Following rupture of a steam bypass line, the isolation valves in the main steam lines will close and prevent steam flow through the break. If it is assumed that an isolation valve in one of the four main steam lines fails to close, then steam flow in the associated main steam line will be limited by the steam flow nozzle as in the case described above.



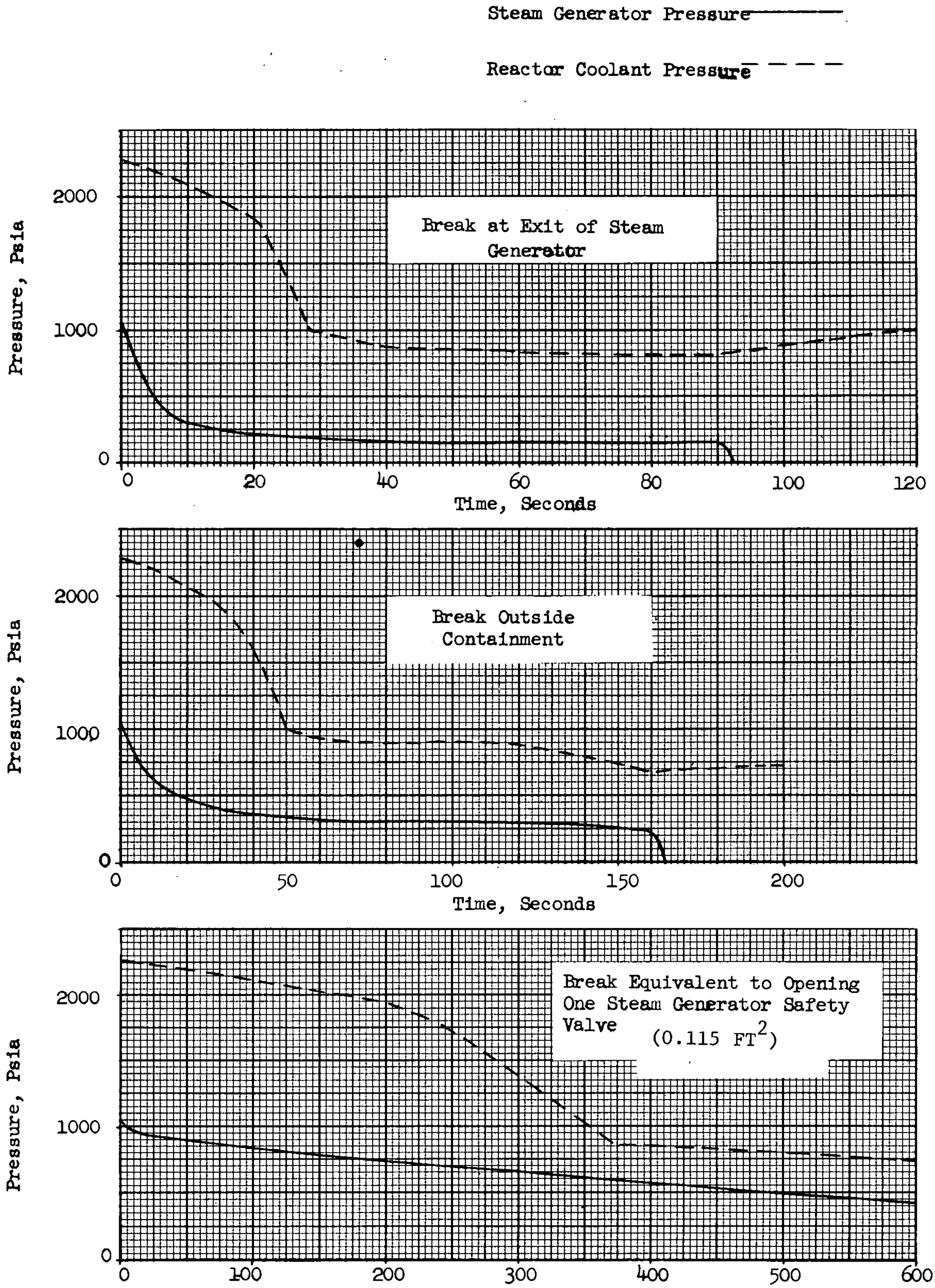
STEAM LINE BREAK AT EXIT
OF STEAM GENERATOR
FIGURE 1

SUPPLEMENT 1



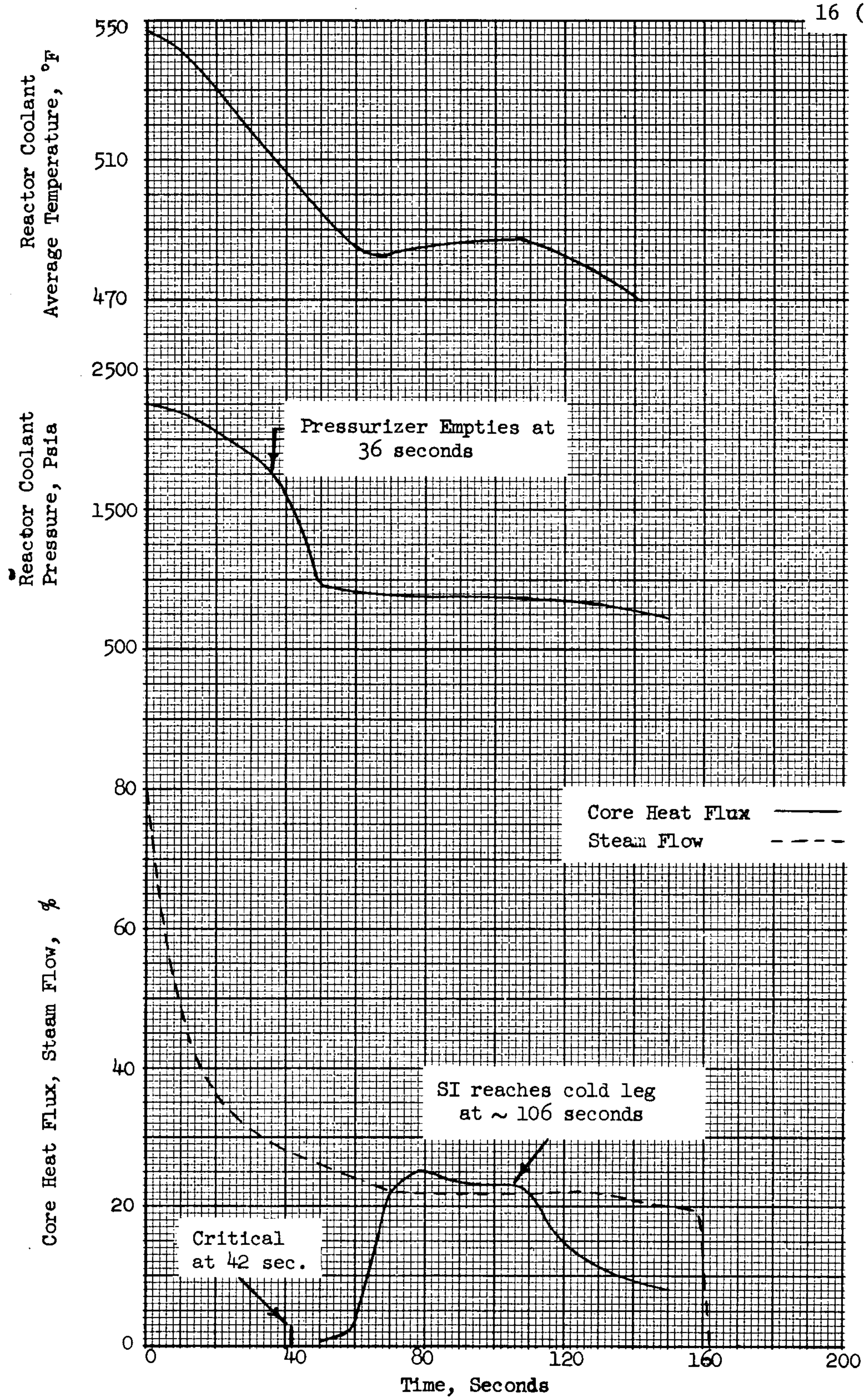
REACTIVITY TRANSIENTS FOR STEAM LINE BREAK

FIGURE 2



STEAM GENERATOR AND REACTOR COOLANT PRESSURE FOLLOWING STEAM LINE BREAK

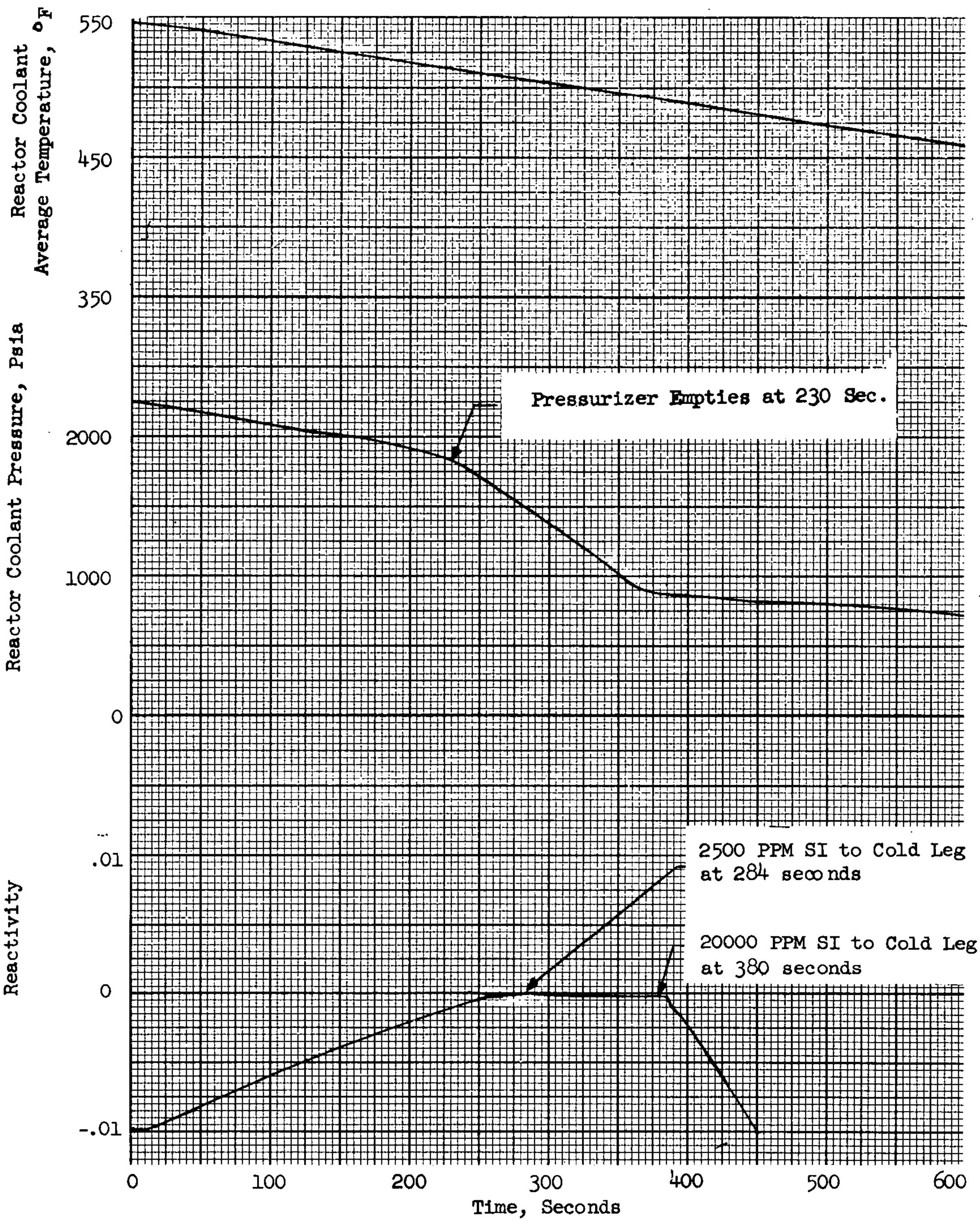
FIGURE 3



STEAM LINE BREAK OUTSIDE CONTAINMENT

FIGURE 4

SUPPLEMENT 1

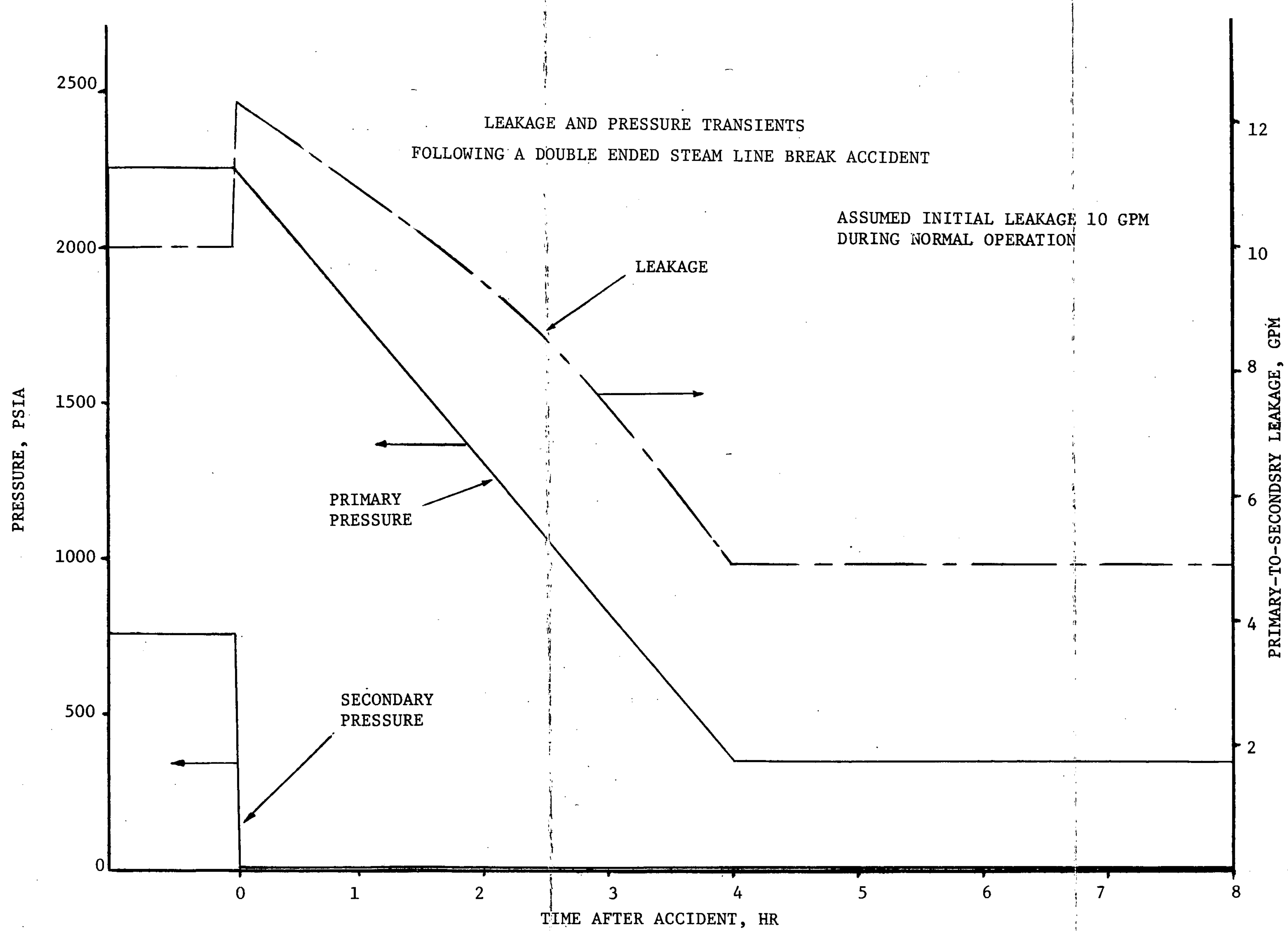


STEAM LINE BREAK EQUIVALENT TO OPENING

ONE STEAM GENERATOR SAFETY VALVE

FIGURE 5

SUPPLEMENT 1



LEAKAGE AND PRESSURE TRANSIENTS FOLLOWING
A STEAM LINE BREAK ACCIDENT
FIGURE 6

SUPPLEMENT 1

Regulatory Suppl File Cy.

U. S. Atomic Energy Commission

Docket No. 50-286

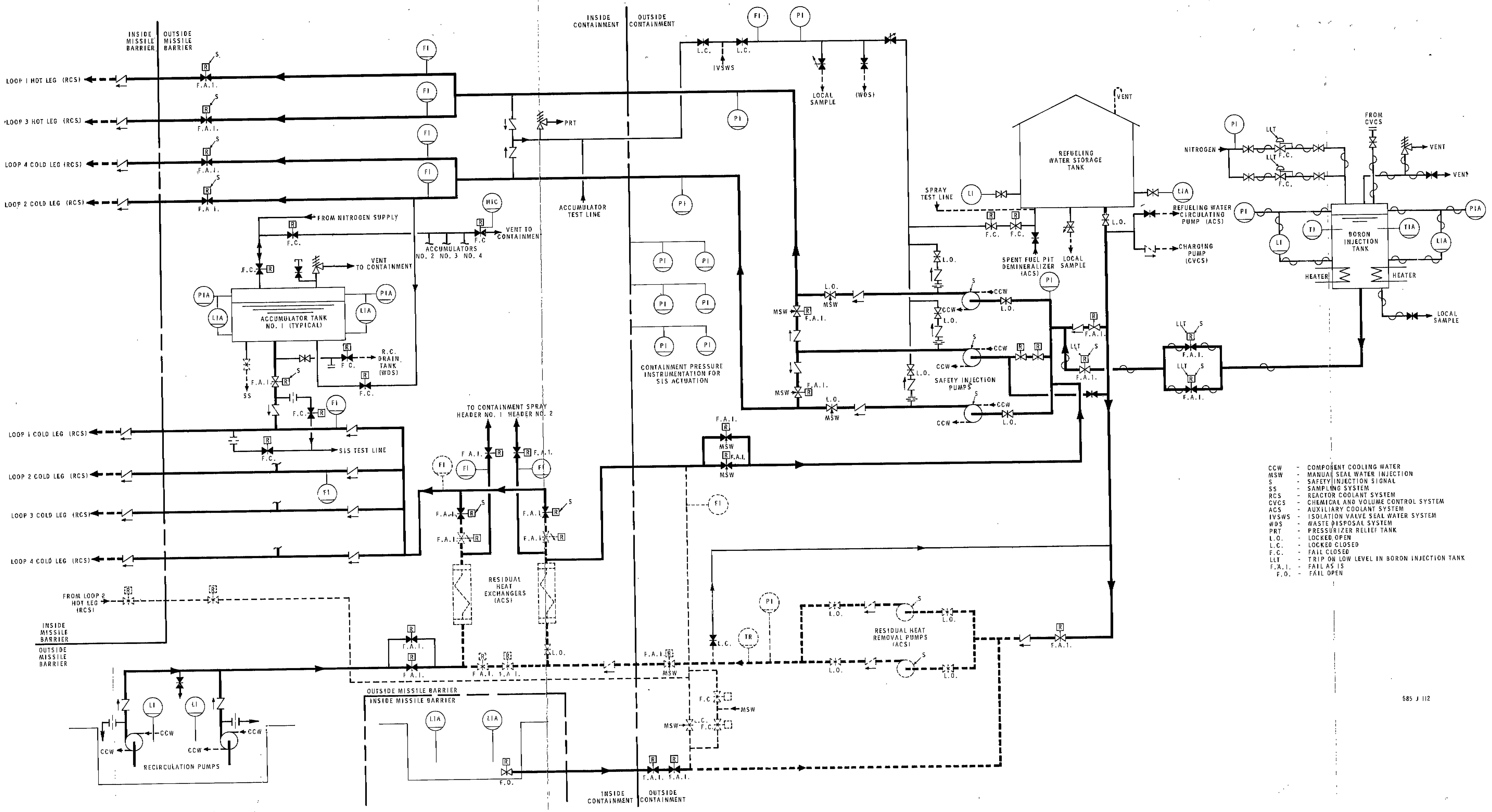
Exhibit B-1

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

FIRST SUPPLEMENT TO:

PRELIMINARY SAFETY ANALYSIS REPORT



CCW - COMPONENT COOLING WATER
MSW - MANUAL SEAL WATER INJECTION
S - SAFETY INJECTION SIGNAL
SS - SAMPLING SYSTEM
RCS - REACTOR COOLANT SYSTEM
CVCS - CHEMICAL AND VOLUME CONTROL SYSTEM
ACS - AUXILIARY COOLANT SYSTEM
IVSWS - ISOLATION VALVE SEAL WATER SYSTEM
WDS - WASTE DISPOSAL SYSTEM
PRT - PRESSURIZER RELIEF TANK
L.O. - LOCKED OPEN
L.C. - LOCKED CLOSED
F.C. - FAIL CLOSED
LLT - TRIP ON LOW LEVEL IN BORON INJECTION TANK
F.A.I. - FAIL AS IS
F.O. - FAIL OPEN

585 J 112

SAFETY INJECTION SYSTEM
FIGURE 1

SUPPLEMENT 1

ITEM 16 (Attachment E Question 3.6)

State the maximum stresses experienced in the tubes and tube sheet during a steam-line rupture transient, including thermal stresses.

ANSWER

An examination of stresses under conditions of a steam line rupture show that for the case of a 2485 psig maximum tube sheet pressure differential the stresses are within acceptable limits. These stresses together with the corresponding stress limits are as follows.

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

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

ITEM 16 (Attachment E Question 3.7)

State the basis for the initial pressurizer level assumed in your analysis of a steam-line rupture. Indicate the margin in pressurizer volume which remains after injection of sufficient boron to render the core subcritical.

ANSWER

The initial pressurizer level assumed in all steam break analyses is that corresponding to the normal level at the load existing before the accident. At no load this value corresponds to a water volume equal to 25% of the pressurizer volume. At loads higher than no load the pressurizer water volume is greater to account for the fact that the density of the water in the Reactor Coolant System is less at the higher average temperatures which exist at higher loads.

The energy removal from the Reactor Coolant System following a steam line rupture causes a reduction of coolant temperature which in turn results in a reduction of the coolant volume and a reduction in system pressure. The reduction in coolant volume causes the pressurizer to empty within the first few seconds. The subsequent addition of the highly borated water by the Safety Injection System renders the core subcritical before the water level is re-established in the pressurizer. The boric acid injection tank is shown in Figure 1 for the revised process flow diagram of the Safety Injection System.

ITEM 16 (Attachment E Question 3.8)

In the PSAR it is stated that in the event of a steam-line rupture accompanied by a stuck control rod and loss of off-site power, there will be ". . . no consequential damage to the primary system and the core will remain in place and intact." Define this criterion in terms of primary system pressure and stress levels; clad stress, strain, and maximum temperature; number of rods undergoing DNB; and metal-water reaction.

ANSWER

As stated in the answer to Question 16 (E-3.5), a steam break inside the containment could lead to DNB and possible clad perforation (no melting or zirconium - water reaction). If any damage were to occur it would be limited to the vicinity of the stuck control rod and would not damage in excess of 10% of the fuel rods.

There is no primary system overpressure transient following a steam line rupture.

Item 16 (Attachment E Question 4.1)

In the PSAR it is stated that in the event of loss of electrical power to all reactor coolant pumps, the plant will be designed in such a manner that the resulting flow coastdown and reactor trip will "... prevent fuel failure and reactor coolant system overpressure." Define this criterion in terms of maximum primary system pressure and minimum DNB ratio experienced.

ANSWER

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 would not result in departure from nucleate boiling (DNB) if the reactor is 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

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; i.e., maintain the DNB ratio above 1.30. Therefore, the fuel will not be damaged as a result of the most severe credible loss-of-coolant flow accident.

Reactor Coolant System pressure control is maintained by the pressurizer syray valves, power operated relief valves and safety valves. Should a rapid increase in Reactor Coolant System temperature cause an increase in pressure these would be actuated in the following order;

1. the spray valves would begin to open at approximately 2275 psia and become fully open at approximately 2325 psia.

2. the power operated relief valves would open at 2350 psia.

3. the pressurizer safety valves would be actuated at 2500 psia.

Since a loss-of-coolant flow is terminated by reactor trip within seconds after it is initiated, the pressure can be maintained below 2350 psia. However, even if the effect of the pressurizer spray and power operated relief valves are ignored, the pressurizer safety valves maintain pressure below 2500 psia.

ITEM 16 (Attachment E Question 4.2)

Indicate the basis for the fission product inventory assumed in the volume control tank and the gas decay tank. List all assumptions made in determining the quantity of radionuclides released to the atmosphere from rupture of these vessels, including air temperature and relative humidity, liquid temperature, halogen vapor pressure, radioactive decay, etc. Discuss the effect that a load-following mode of operation would have on the inventories in these tanks.

ANSWER

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,

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

for daughter nuclides in the coolant,

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

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

C = refers to core

w = refers to coolant

λ = radioactive decay constant

ν = escape rate coefficient for diffusion into coolant

The parameters used in the calculation of the reactor coolant fission product inventory, including pertinent information concerning the expected coolant cleanup flow rate, boron dilution rate, and demineralizer effectiveness, are presented in Table 1. The results of the calculations are presented in Table 1 of Question 16 (E-3.1). In these calculations the defective fuel rods are assumed to be uniformly distributed throughout the core and the fission product escape rate coefficients are therefore based upon an average fuel temperature.

Values for the maximum activity in the vapor phase of the Volume Control Tank is presented in Table 2.

The expression used to evaluate these values is as follows:

$$\frac{dVc}{dt} = \frac{R \times N_w \times Fs}{V_{vapor\ phase}} - \lambda N_v$$

where:

dVc = change of activity in the vapor phase of volume control tank

Fs = stripping fraction defined as fraction of incoming reactor coolant fission products gaseous activity remaining in the volume control tank.

The maximum concentrations of the non-gaseous isotopes in the Volume Control Tank are conservatively assumed to equal the equilibrium concentrations in the reactor coolant. The coolant activities are given in Table 2 of Question 16 (E-3.1).

Since the Volume Control Tank is normally located downstream of the mixed bed demineralizer, concentration of non-gaseous isotopes in the volume control tank is normally a factor of 10 lower than the values mentioned above.

The presence of iodine isotopes in the vapor phase of the volume control tank is considered negligible. Reference is made to Question 16 (E-3.3) of this supplement.

Following a Volume Control Tank Rupture, all noble gases contained within the tank are assumed to be released into the Auxiliary Building atmosphere. Results of evaluation of potential release are shown in Table 2.

The following considerations are given to the possibility of halogen release from "spilled" water into the Auxiliary Building.

a) The Volume Control Tank is a low pressure (20-50psi) low temperature (130°F) system. Therefore, its rupture does not result in "flashing" from its liquid phase. (pH ≈ 6.5).

b) Radioactive halogen concentration in the liquid phase of the Volume Control Tank equals approximately one tenth (.1) of the halogen concentration in the reactor coolant.

In establishing the total gaseous activity in the gas decay tanks, the conservative assumption is made that a complete reactor coolant inventory of gases is discharged into the tanks. The reactor coolant inventory assumed is determined on the basis of equilibrium or maximum specific coolant activities associated with base load operations,which is more conservative than load-follow operation.

In the event of a gas decay tank rupture, the maximum anticipated quantity of waste gas that could be released from any one tank is approximately 1.35×10^4 curies of equivalent Xe-133 which would result in dose of about *8* rem at the site exclusion boundary.

TABLE 1

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	130
	b. Liquid, cu ft	270
8.	Volume Control Tank temperature °F	130
9.	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}
10.	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
11.	Cation bed demineralizer decontamination factor for Cs-134, 136, 137, Y-90 and Mo-99	10.0
12.	Volume control tank noble gas stripping fraction (closed system):	

Isotope	Stripping Fraction
Kr-85	2.3×10^{-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}

TABLE 2

MAXIMUM VOLUME CONTROL TANK NOBLE GAS
CONCENTRATIONS RESULTING FROM OPERATION AT 3216 MWt
WITH SMALL CLADDING DEFECTS IN 1% OF THE FUEL RODS

Isotope	Total Activity (Curies)
Kr-85	2.34
Kr-85m	58.5
Kr-87	23.0
Kr-88	104.2
Xe-133	11,895.0
Xe-133m	135.2
Xe-135	274.0
Xe-135m	1.16
Xe-138	3.5
Total	12,496.56

ITEM 16 (Attachment E Question 4.3)

Justify the assumption that only one row of fuel rods could be damaged in a fuel-handling incident. Consider the effect of dropping one element on another during refueling. Relate your assumption to fuel damage which has occurred during handling at operating reactors.

ANSWER

As discussed in Sections 9 and 12 of the PSAR, extreme care will be given to the handling of all fuel assemblies.

Special precautions will be 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 irradiated fuel handling operations will be conducted under water. The handling tools used in the fuel handling operations will be conservatively designed and the associated devices will be of a fail-safe design.

In the fuel storage area, the fuel assemblies will be spaced in a pattern which will prevent any possibility of a criticality accident. Also, the design of the facility will be such that it will not be possible to carry heavy objects, such as a spent fuel transfer cask, over the fuel assemblies in the storage racks. In addition, the design will be such that only one fuel assembly can be handled at a given time, except of course when a loaded spent fuel cask is being handled.

The maximum velocity of the cranes which move the fuel assemblies will be limited to a relatively low speed. Caution will be 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 will suspend 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 will be such that the fuel rods will be restrained by grid clips which will 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 will be limited due to the restraining force of the grid clips. The force transmitted to the fuel rods during fuel handling will not be of a magnitude great enough 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 have the effect of absorbing a shock and thus limit the force on the individual fuel rods. After the reactor is shut down, the fuel rods will 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.

Preliminary 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, therefore we expect no breeching of 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.

ITEM 16 (Attachment E Question 4.4)

State the basis for the gap activity assumed to be present in the damaged fuel rods during a fuel handling incident and for the halogen partition factor assumed.

ANSWER

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 five concentric rings each with release rate dependent on the mean fuel temperature within that ring. The diffusing isotope is assumed present in the gas gap when it has diffused to the boundary of its ring.

The diffusion coefficient, D', for Xe and Kr in UO_2, varies with temperature through the following expression:

$$D'(T) = D'(1673) \exp\left[-\frac{E}{R}\left(\frac{1}{T} - \frac{1}{1673}\right)\right]$$

where:

E = activation energy

D'(1673) = diffusion coefficient at 1673°K = 1×10^{-11} sec^{-1}

T = temperature in °K

R = gas constant

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[1] 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[2] indicates that iodine diffuses at slightly slower rates than do Xe and Kr.

For a full core cycle at 3216 MWt, the above analysis results in a pellet-clad gap activity of less than 3% of the dose equivalent equilibrium core iodine inventory. The noble gas activity present in the pellet-clad gap and assumed release to the containment is about 2.5% of the core inventory.

The percentage of the total core activity present in the gap for each isotope is listed in Table 1.

The core temperature distribution used in this analysis, based on hot channel factors, $F_{\Delta H} = 1.70$ and $F_q = 2.82$, is presented in Table 2.

TABLE 1

CORE AND GAP ACTIVITY

Assumptions: Operation at 3216 MWt for 500 days
Temperature distribution specified in Table 2

Isotope	Curies in the Core ($X\ 10^7$)	Per Cent of Core Activity in the Gap	Curies in the Gap ($X\ 10^5$)
I-131	8.35	2.3	19.
I-132	12.75	0.26	3.3
I-133	19.09	0.79	15.
I-134	23.01	0.16	3.8
I-135	17.05	0.43	7.5
Kr-85	.1228	17.	2.1
Xe-133	19.02	1.9	55.
Xe-133m	5.16	1.25	6.5
Xe-135	7.23	0.088	.46

(1) Toner, D.F., and Scott, J. S., "Fission Product Release from UO_2," Nuclear Safety, Vol. 3, No. 2, December 1961.

(2) Belle, J., Uranium Dioxide: Properties and Nuclear Applications, Naval Reactors, Division of Reactor Development United States Atomic Energy Commission, 1961.

TABLE 2

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 3 presents a factor which when multiplied by the shutdown I-131 concentration in the core yields the I-131 activity equivalents in the core for various later times.

TABLE 3

EQUIVALENT FACTOR

Time	Equivalent Factor
0	1.88
Average first 2 hours	1.84
Average next 22 hours	1.47
Average next 30 days	0.34

Fuel Handling Accident Source

The noble gas and iodine gap activities 100 hours after shutdown (the earliest time fuel handling would commence) is computed for the 15 highest-rated rods in the highest-rated assembly. The noble gas available for release from these rods is 1370 dose equivalent curies of Xe-133. This is calculated from the information given in Table 4. Each isotopic quantity is multiplied by the ratio of its rem/curie effectiveness and divided by the rem/curie effectiveness of xenon-133. These products are added to determine the total dose-equivalent curies of xenon-133.

The iodine available for release is 760 dose-equivalent curies of I-131.

TABLE 4

NOBLE GAS ACTIVITY RELEASE FROM FUEL HANDLING ACCIDENT
3216 MWt

Isotope	Activity, Curies
Kr-85	77
Xe-133m	73
Xe-133	1220

The validity of the assumed decontamination factor of 10^{-3} applied to the release of iodine with the gases from a breached fuel clad can be assessed from the following experimental result. 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 reagent (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 desired 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. Decontamination factors of 10^{-3} have been measured by others[(3)].

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(3) Diffey, H. R., et.al., "Iodine Clean Up in a Steam Suppression System," AERE-R 4882, 1965.

ITEM 16 (E-4.5)

To illustrate the safety margin which exists due to the inherent design of the facility, identify the maximum hypothetical missile originating from the turbine which would not penetrate the containment vessel. To indicate the sensitivity of casing energy absorption on your analysis, present this information assuming casing energy absorptions of zero, 25, 50, and 100 per cent of your best estimate of absorption.

ANSWER

In the following discussion, it is shown that the maximum missile generated will not penetrate the containment vessel.

As stated in Appendix 16 (E-4.5) attached, no failure of the high pressure turbine is anticipated as a consequence of a postulated maximum overspeed condition assuming failure of the turbine speed control system.

For the low pressure turbine, under the assumed failure of the turbine speed control system and a maximum overspeed of 175 per cent, the postulated missiles are given in Table A, B. Disks 1, 2, 3, 4 and 5 have a relatively large area of turbine casing around them for absorbing energy; thus, disk No. 6 is considered in this evaluation. The impact area for one quarter of disk No. 6 is 4.5 ft^2; it is assumed to have a weight of 3,000 lb and ejection velocity of 722 fps. For each quarter disk the translational kinetic energy is 25.1×10^6 foot-pounds.

The best estimate of absorption of energy by the turbine casing for each quarter of disk No. 6 is 35×10^6 foot-pounds. Table A gives other values relative to the best estimate for casing energy absorption.

TABLE A

DISK NO. 6

Fraction of Energy Per Cent	Best Estimate Absorption Foot-Pounds	Translation Kinetic Energy of Disk Quarter Foot-Pounds	Energy Remaining in Disk Quarter Foot Pounds	Final Velocity of Disk Quarter Ft/Sec
100	35×10^6	25.1×10^6	0.0	0.0
75	26.25×10^6	25.1×10^6	0.0	0.0
50	17.5×10^6	25.1×10^6	7.6×10^6	398
25	8.75×10^6	25.1×10^6	16.35×10^6	538
0	0	25.1×10^6	25.1×10^6	722

If the hypothetical situation of 0 per cent absorption by the casing is postulated for each disk, then for purposes of analysis only, consideration of disk No. 5 is made. The best estimate of energy absorption by the casing per quarter is given in Table B. The impact area for one quarter of disk No. 5 is 5.0 ft^2, its weight is 2,900 lb, and its ejection velocity is 810 fps. For each quarter disk the translational kinetic energy is 29.6×10^6 ft-lb.

TABLE B

DISK NO. 5

Fraction of Energy Absorption Per Cent	Best Estimate Foot-Pounds	Translation Kinetic Energy of Disk Quarter Foot-Pounds	Energy Remaining in Disk Quarter Foot-Pounds	Final Velocity of Disk Quarter Ft/Sec
100	45×10^6	29.6×10^6	0.0	0.0
75	33.75×10^6	29.6×10^6	0.0	0.0
50	22.5×10^6	29.6×10^6	7.1×10^6	398
25	11.25×10^6	29.6×10^6	18.35×10^6	638
0	0	29.6×10^6	29.6×10^6	810

In the hypothetical situation of zero energy absorption by the turbine casing and the turbine building structure, and assuming direct impact normal to the containment structure, maximum missile penetration of the reinforced concrete containment wall would be less than 1.3 ft. for either disk No. 5 or disk No. 6. A factor of safety of at least 2.6 exists against containment penetration by a missile from either disk No. 6 or disk No. 5.

APPENDIX 16 (E-4.5)

As stated in section 12 of the PSAR, 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 guarantees 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 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 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 moisture separator live steam reheater assemblies.

The high-pressure rotor is made of NiCrMoV alloy steel. The specified minimum mechanical properties are given in Table 1.

TABLE 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 2.

TABLE 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 3.

TABLE 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 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 5.

TABLE 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 5.

TABLE 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 5

MINIMUM MECHANICAL PROPERTIES - DISKS (CONT'D)

	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 6.

TABLE 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 7.

TABLE 7

THICKNESS - BACK PLATE OF CYLINDERS

Inner cylinder No. 1	2 inches
Innder 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 8.

TABLE - 8

DISK BURSTING SPEED

Type of Disk	Bursting Speed (per cent of norminal	
	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 - 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 - 10 lists the same parameters for half disks.

TABLE - 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 - 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 11 summarizes the values of the deformation energy per unit volume up to 100%, 75% and 50% of the total elongation, respectively.

TABLE - 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 $\frac{\text{in lb}}{\text{in}^3}$	7,000 $\frac{\text{in lb}}{\text{in}^3}$	10,200 $\frac{\text{in lb}}{\text{in}^3}$
Under "dynamic" loading	7,900 $\frac{\text{in lb}}{\text{in}^3}$	11,900 $\frac{\text{in lb}}{\text{in}^3}$	15,800 $\frac{\text{in lb}}{\text{in}^3}$

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 $\frac{\text{in lb}}{\text{in}^3}$	9,900 $\frac{\text{in lb}}{\text{in}^3}$	14,300 $\frac{\text{in lb}}{\text{in}^3}$
Under "dynamic" loading	9,000 $\frac{\text{in lb}}{\text{in}^3}$	13,500 $\frac{\text{in lb}}{\text{in}^3}$	18,000 $\frac{\text{in lb}}{\text{in}^3}$

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 would have to violate not 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 cause 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 tuptured 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 112×10^6 and 168×10^6 ft lb. Since the anticipated kinetic energy of 4 disk No. 6 quarters, i.e., 100×10^6 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.

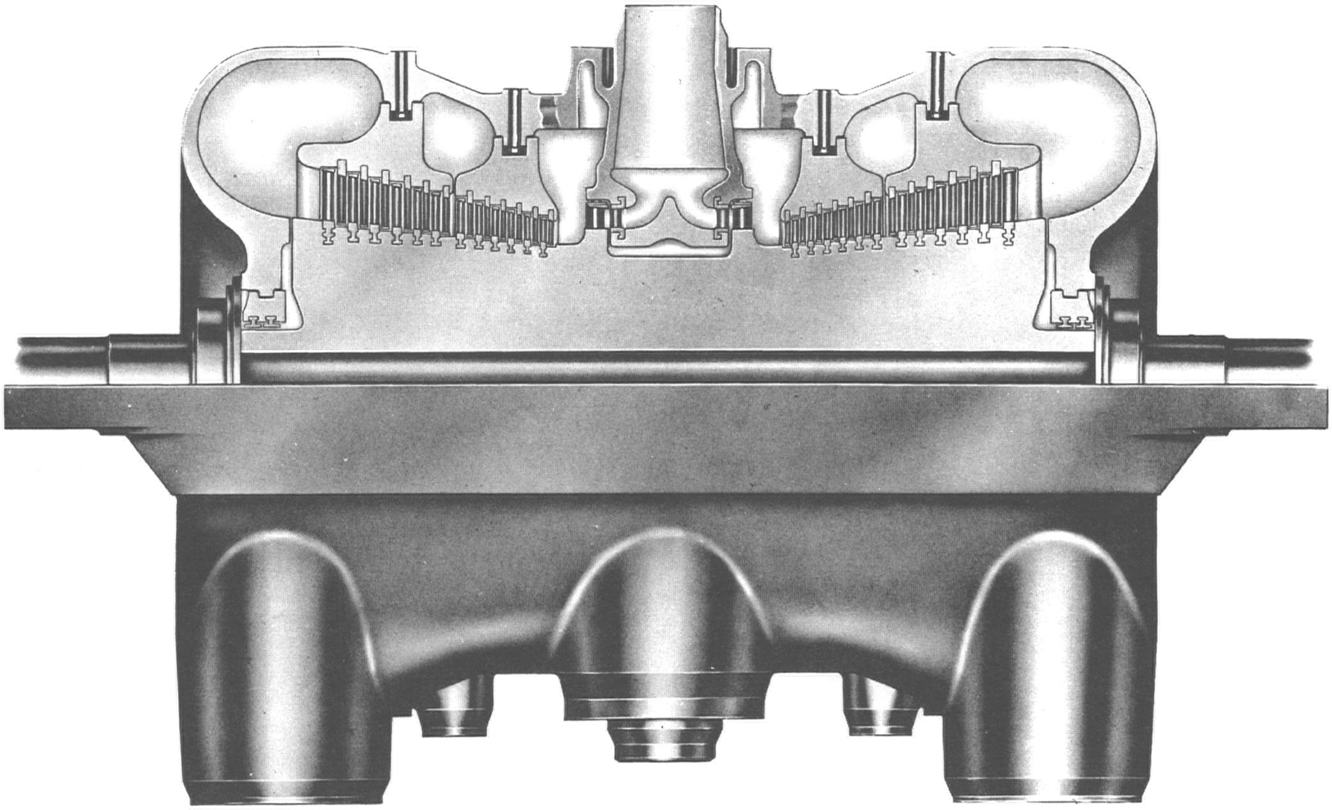
Based on the stress analysis of the low-pressure disks, the maximum actual speed at which the unit may run is 175% of nominal.

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. 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

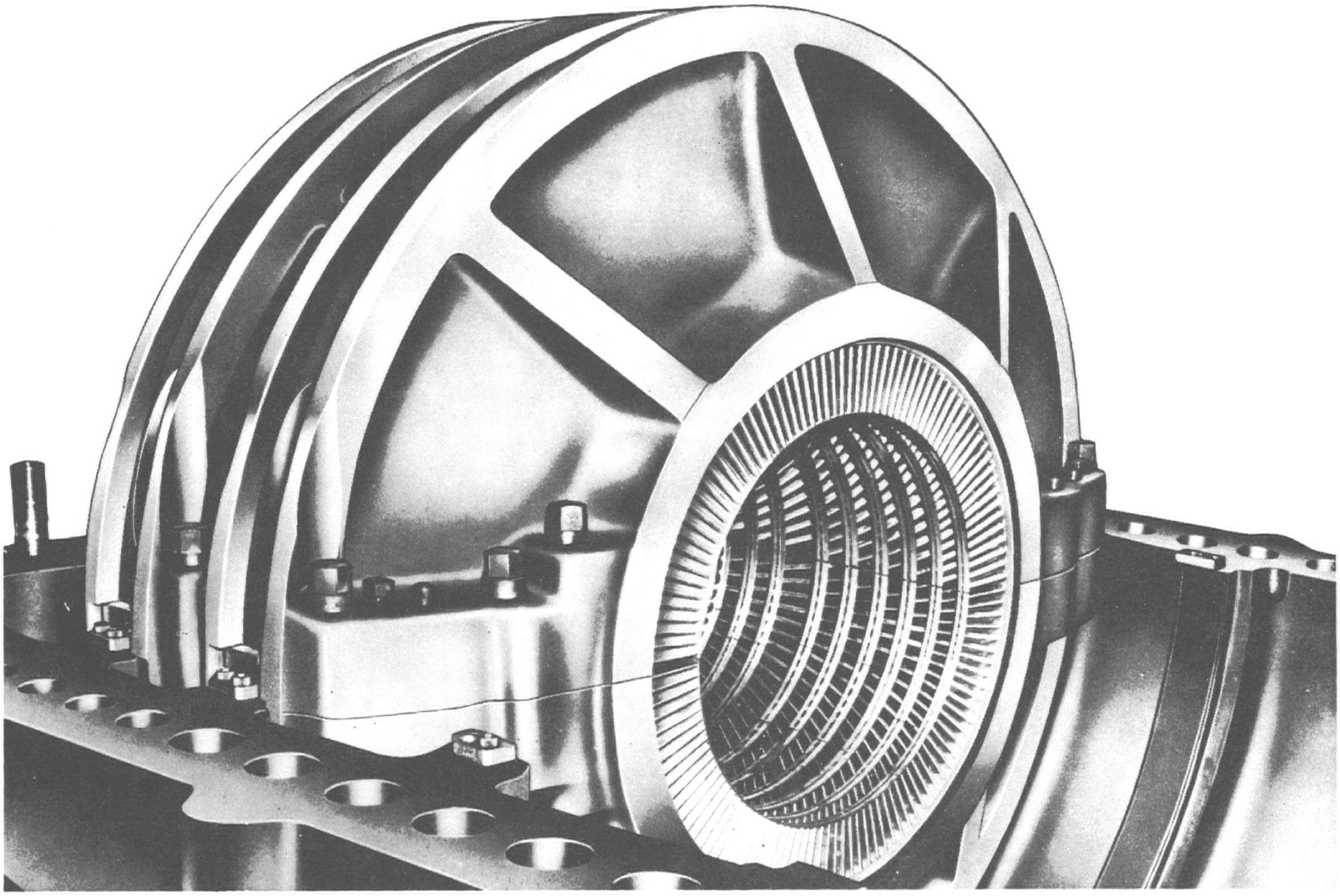


FEATURES

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 1

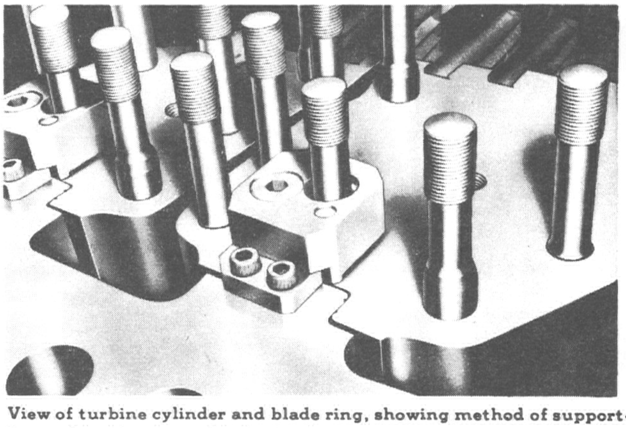
BLADE RINGS



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 2

LOW-PRESSURE ELEMENT

1800-RPM DOUBLE-FLOW DESIGN

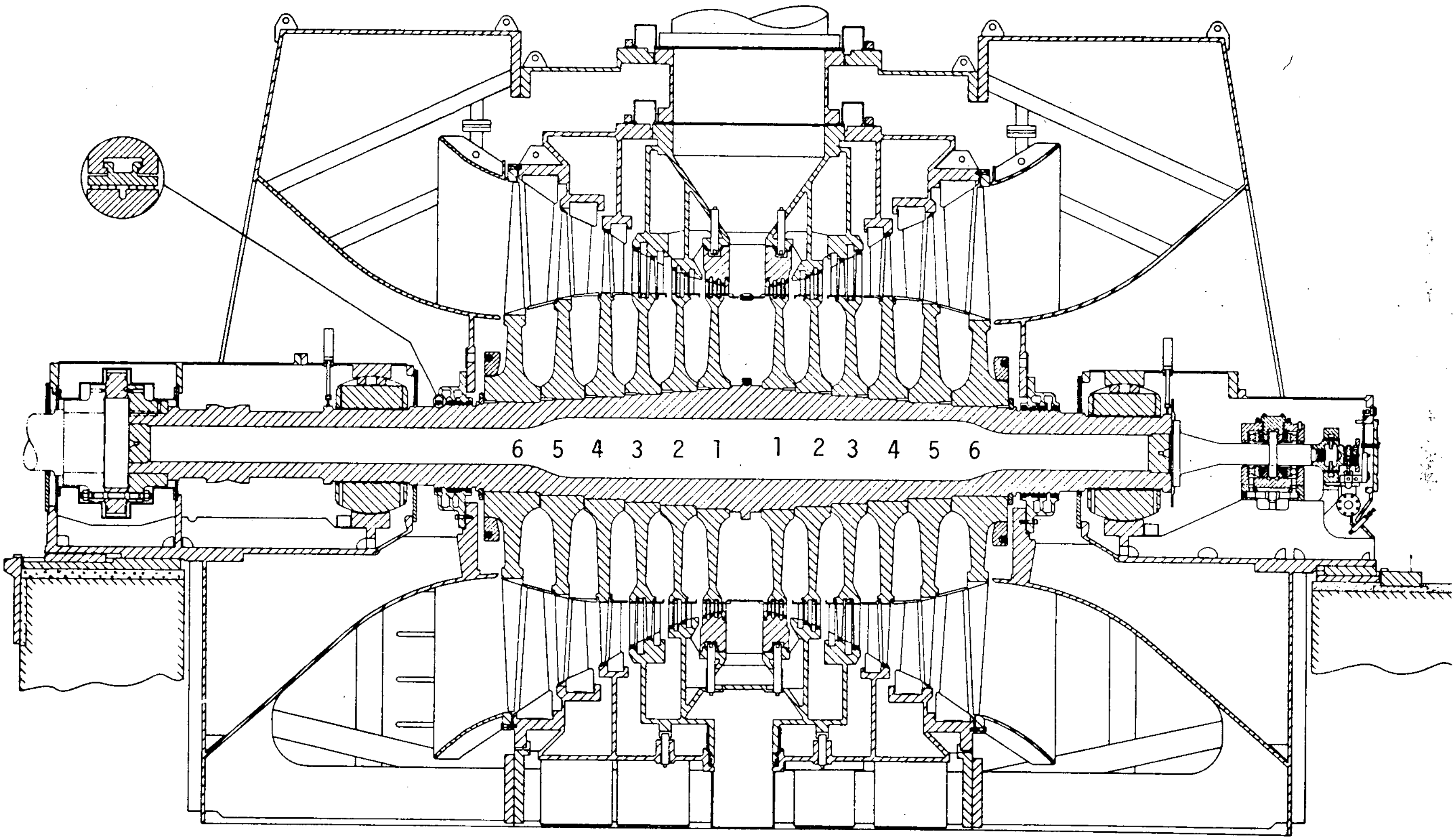
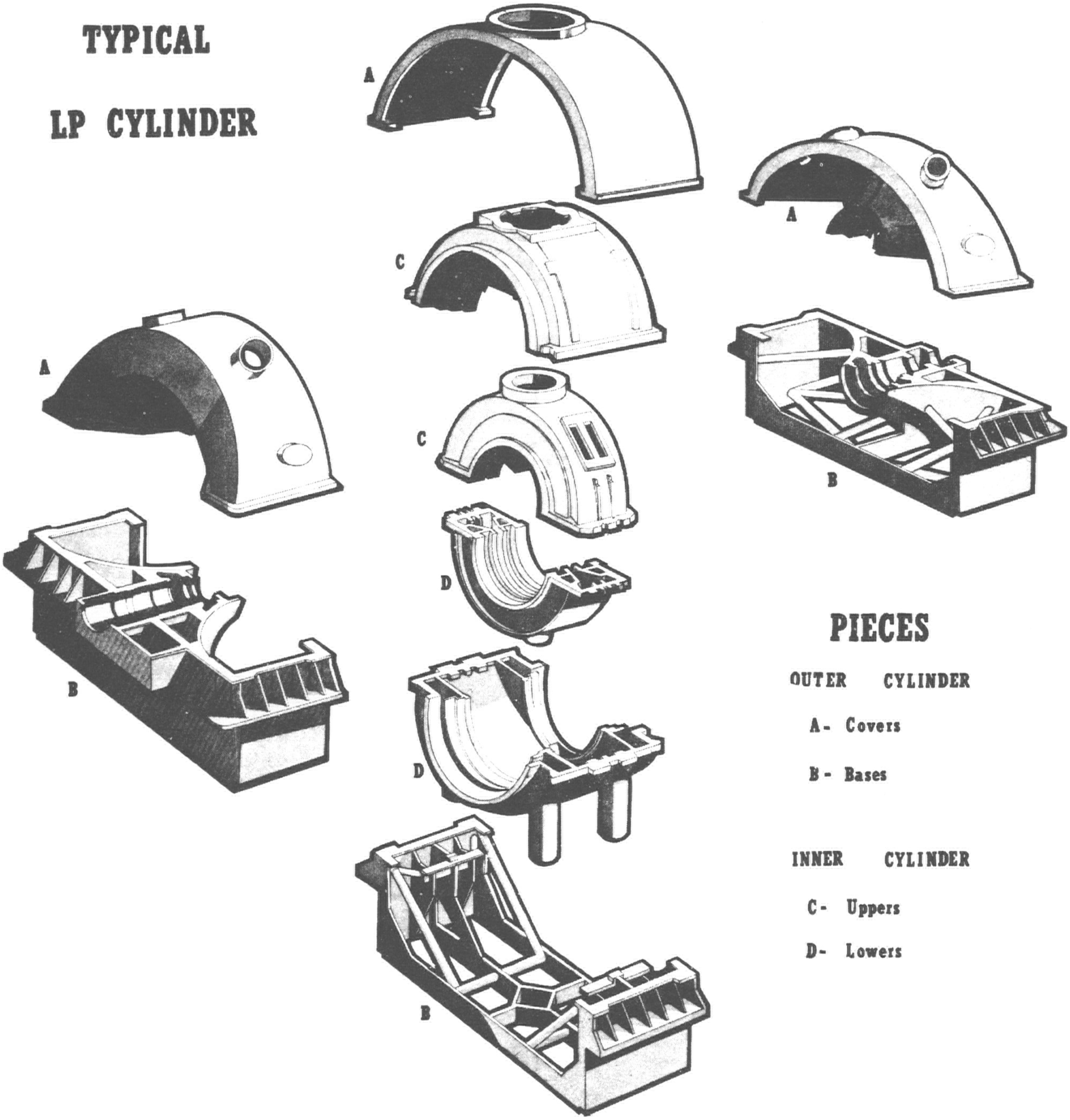


FIGURE 3

SUPPLEMENT 1



Exploded View of Low Pressure Unit

ITEM 16 (Attachment E Question 5.0)

LOSS-OF-COOLANT ACCIDENT

ANSWER

The Loss-of-Coolant Accident Evaluation is presented to provide additional information for the loss-of-coolant accident. This information is contained in Appendix 16 (E-5.0).

APPENDIX 16(E-5.0)

PRIMARY SYSTEM PIPE RUPTURE

SECTION 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 basic design criteria for loss of coolant accident evaluations are that the cladding temperature is to be less than the melting temperature of Zircaloy-4 and the temperature at which gross core geometry distortion, including clad fragmentation may be expected and that the total core metal-water reaction will be limited to less than 1 percent.

These criteria will assure that the core geometry remains in place and substantially intact to such an extent that effective cooling of the core is not impaired.

The safety injection system, even when operating on emergency power will meet the design criteria for reactor coolant piping ruptures up to and including the double ended rupture of a reactor coolant loop. Consequences of these ruptures are welll within those described later in this section for the hypothetical accident and are therfore well with in 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.

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 effects 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 with coastdown and cavitation where appropriate.

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 a 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 single-pass 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 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.

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 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 conductivity of 2.65 Btu/hr-ft-°F for the UO_2, and a 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, thereby minimizing void formation.

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} $\delta k/°F$ 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.

<u>Core Cooling Analysis - LOCTA Code</u>

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.

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:

$$h = 0.023 \frac{k_v}{D_e} \left[\frac{\rho_v D_e}{\mu_v} \left(\frac{Q_1 + Q_v}{A_c} \right) \right]^{0.8} \left[\frac{C_\rho \mu}{k} \right]_v^{0.4}$$

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:

$$\left(\frac{hD}{k}\right)_{iso} = 3.66$$

$$h/h_{iso} = \left[\frac{T_b}{T_w} \right]^{0.25}$$

For turbulent forced convection to steam:

$$\frac{hD}{k} = 0.020 \left[Re_b\right]^{0.8} \left[Pr_b\right]^{0.4} \left[\frac{T_w}{T_b}\right]^{-0.5}$$

where:

h - Heat transfer coefficient on outer surface of fuel rod ($Btu/hr\text{-}ft^2\text{-}°F$)

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

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 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 $Btu/hr\text{-}ft^2$ are used during this period 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[(5)] is evaluated leaving the flooded region of the core.

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 melt, 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.

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 can be specified by input tables. Steam flow as a function of dump valve flow area can be 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
   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
   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 of 4 accumulators operating is also presented. Full flow from the safety injection pumps was assumed at 25 seconds.

Results - Large Area Ruptures

All of the figures presented as results of the loss-of-coolant accident evaluations are those that were submitted for the Indian Point Unit #2 Final Safety Analysis Report. The blowdown and refill transients, the core power transients and the core temperature transients for Indian Point Unit #3 are expected to be very similar to those predicted for the Unit #2 plant. Slight differences in the calculated core parameters may exist due to the increased power rating of Unit #3 at steady state conditions (2758 MWt and 18.4 peak kw/ft for Unit #2 versus 3217 MWt and 18.6 peak kw/ft for Unit #3). Table A.1-1 presents the comparison of the maximum clad temperatures and clad burst predictions for Unit #3 to those calculated and submitted in the Indian Point Unit #2 Final Safety Analysis Report. The trend of the clad temperature transient curves (Figures A.1-23 through A.1-28) for Unit #3 are expected to be similar to those evaluated for Unit #2.

Blowdown and Refill

Figures A.1-1 to A.1-9 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 lead 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.

For the cold leg breaks analyzed, when the core is reflooded by accumulator water, special consideration is given to steam generation in the core which could retard the refill transient. During the refilling of the core with accumulator water, steam will be generated around the hot fuel rods, causing a pressure build-up in the core region. As a result the liquid level in the downcomer will rise at a faster rate than the core level, until the water head between the downcomer and core level is sufficient to push the steam formed in the core through the loops to the break location, where it will escape to the containment. The sketch in A.1-10a shows the path the steam must follow for this cold leg break.

The relationship between this downcomer head and corresponding steam flow rate through the loops is shown in Figure A.1-10. 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.

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 Figure A.1-11. 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.

In the unlikely event that all recirculation loop seals 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 A.1-11. 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.

Core Power Transient During Blowdown

The core power transients calculated are shown in Figures A.1-12 and A.1-13. 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 A.1-14 through A.1-22 present a plot of core pressure and core flow and the calculated heat transfer coefficient used for all breaks. Figures A.1-23 through A.1-26 present the maximum clad temperature transient for the design and adiabatic after blowdown cases associated with the

large cold leg breaks. Figures A.1-27 and A.1-28 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. Table A.1-1 summarizes the important results of the transient.

TABLE A.1-1

Cold Leg Break Size	Maximum Clad Temperature, °F		Total % Clad Burst	
	Unit #2	Unit #3	Unit #2	Unit #3
Double-ended	2120	2225	54	90
6 ft^2	1910	1990	42	85
3 ft^2	1660	1700	28	77
0.5 ft^2	1940	2020	56	89

Hot Leg Break Size	Maximum Clad Temperature, °F	
	Unit #2	Unit #3
Double-ended	2020	2110
6 ft^2	1760	1840
3 ft^2	1480	1510
0.5 ft^2	1315	1350

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.

Rupture 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 and 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 A.1-29.

For cases two and three 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 Figures A.1-30 through A.1-34 and A.1-35 through A.1-39, 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 formation 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 three cases.

1. DNB occurs and no credit is taken for the froth level (Core uncovery times taken from quiet level curve in Figure A.1-39a)

2. DNB occurs and credit is taken for the froth level (Core uncovery times taken from froth level curve in Figure A.1-39a)

3. No DNB and credit taken for the froth level (Core uncovery times as in case 2)

The minimum safety injection case (Case 1) resulted in a peak clad temperature of 2120°F. If credit is taken for the froth level, the two phase mixture results in the hot spot being covered for the entire transient. Assuming DNB occurs (Case 2) a peak clad temperature of 1555°F is evaluated. A more realistic evaluation of no DNB (Case 3) would result in the heat transfer mechanism of nucleate boiling throughout the transient and a hot spot peak clad temperature of approximately 725°F.

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 greater 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 A.1-40 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 are presented in Figure A.1-41. Thus hot spot flooding is maintained up to a six inch break.

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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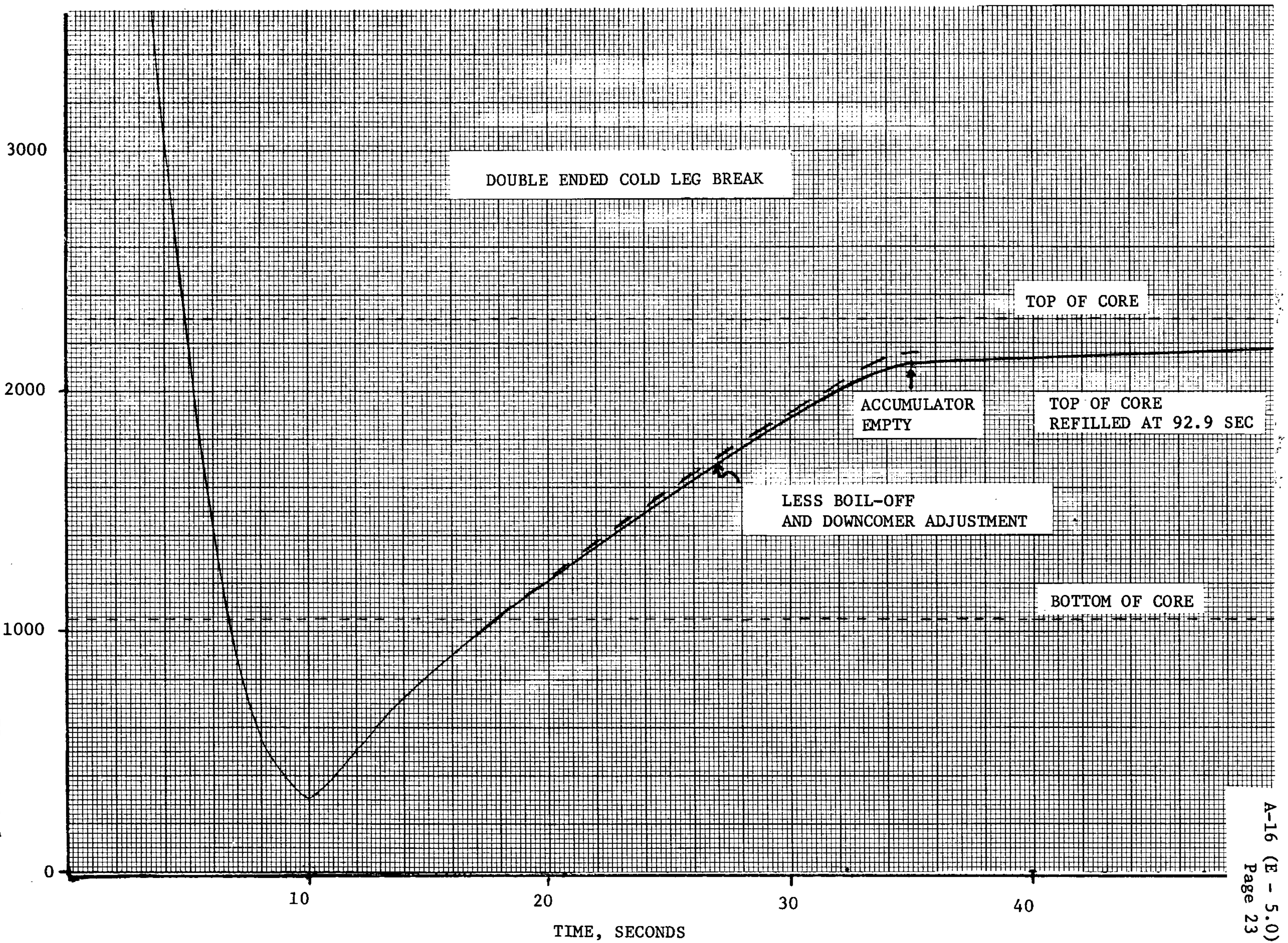


DOUBLE ENDED COLD LEG BREAK
TOP OF CORE
ACCUMULATOR EMPTY
TOP OF CORE REFILLED AT 92.9 SEC
LESS BOIL-OFF AND DOWNCOMER ADJUSTMENT
BOTTOM OF CORE
VOLUME (CU FT)
0
1000
2000
3000
10
20
30
40
TIME, SECONDS

FIGURE A.1-1

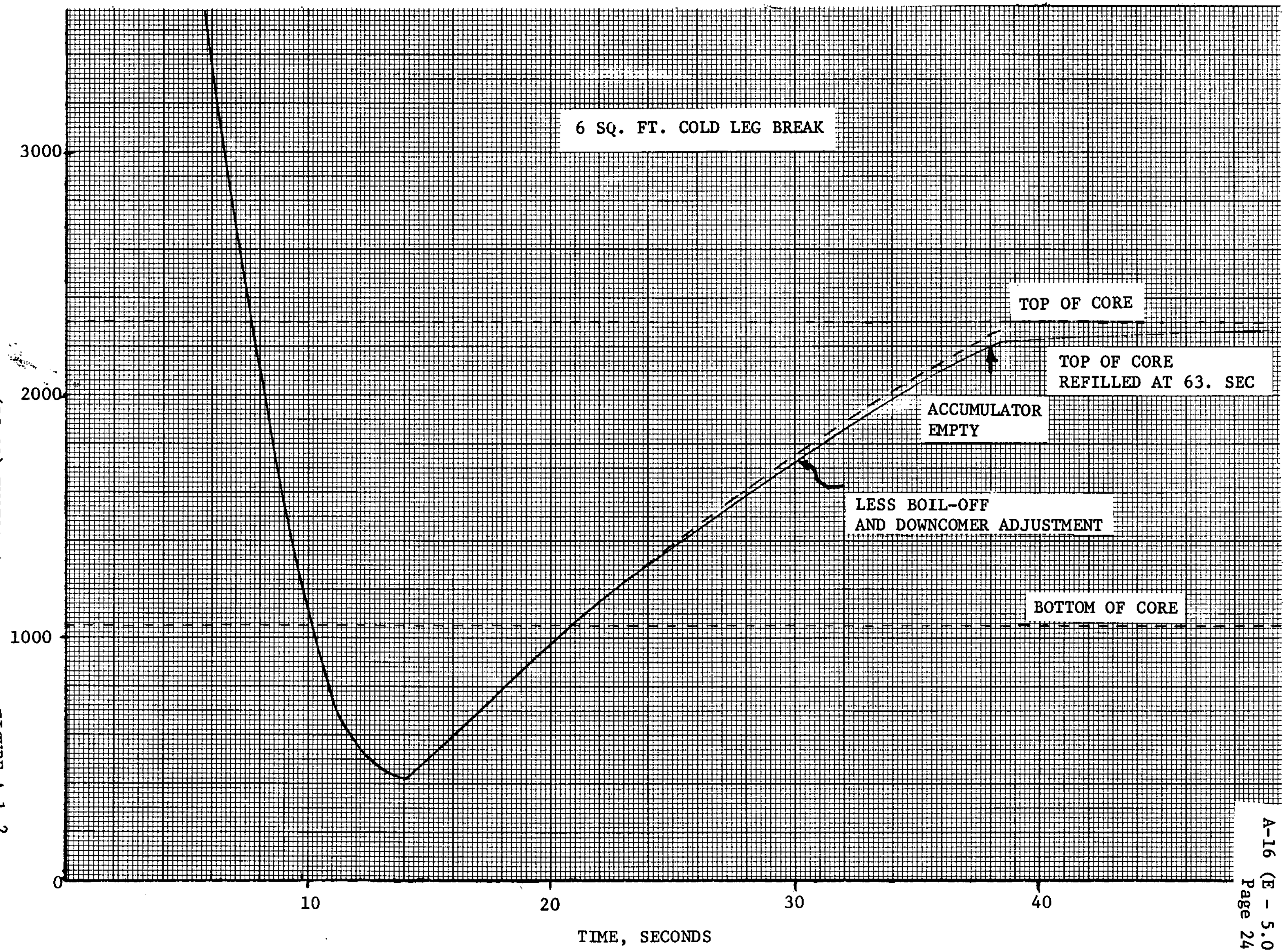

6 SQ. FT. COLD LEG BREAK
TOP OF CORE
TOP OF CORE
REFILLED AT 63. SEC
ACCUMULATOR
EMPTY
LESS BOIL-OFF
AND DOWNCOMER ADJUSTMENT
BOTTOM OF CORE
VOLUME (CU FT)
3000
2000
1000
0
10
20
30
40
TIME, SECONDS

FIGURE A.1-2

Supplement 1

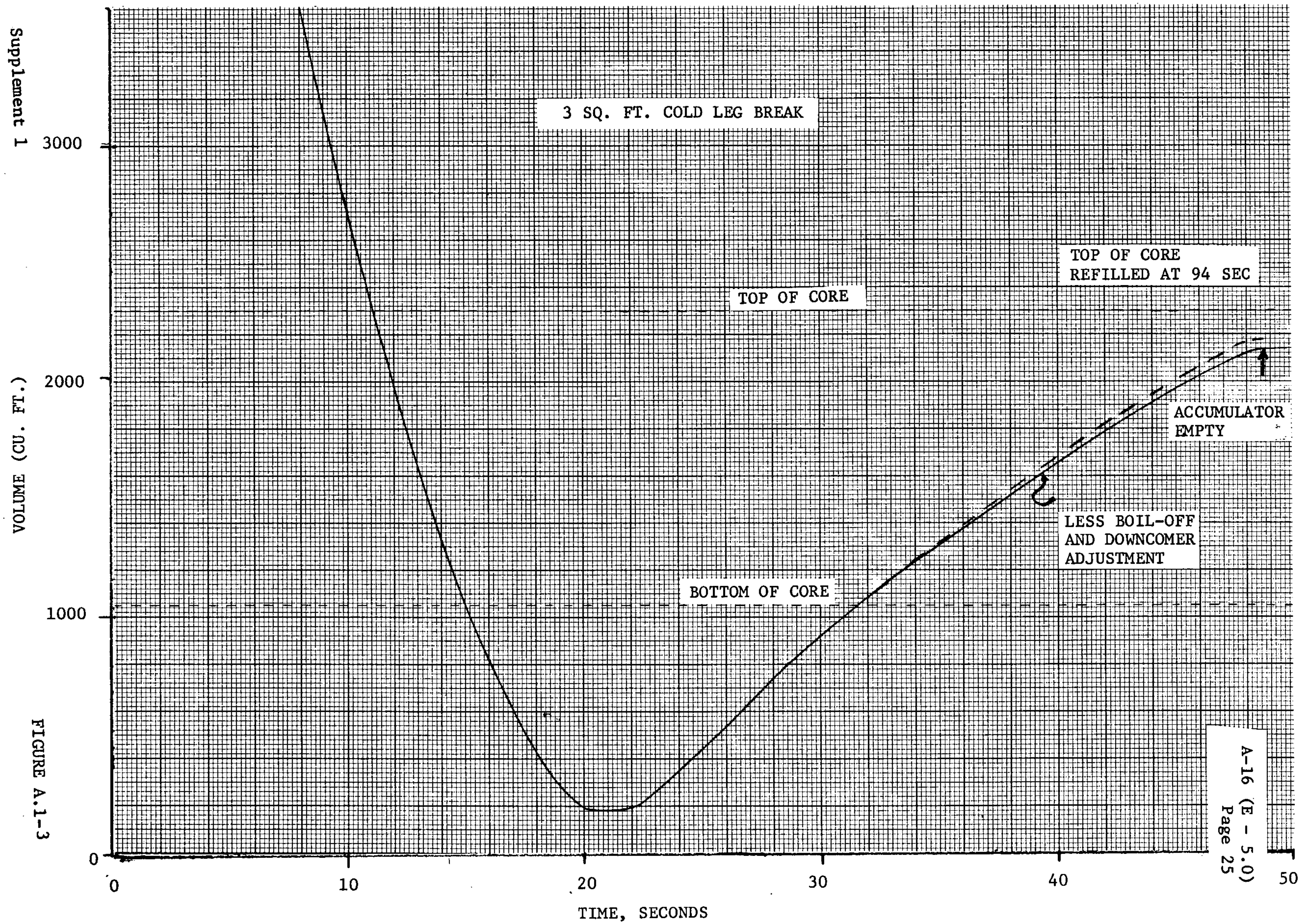


FIGURE A.1-3

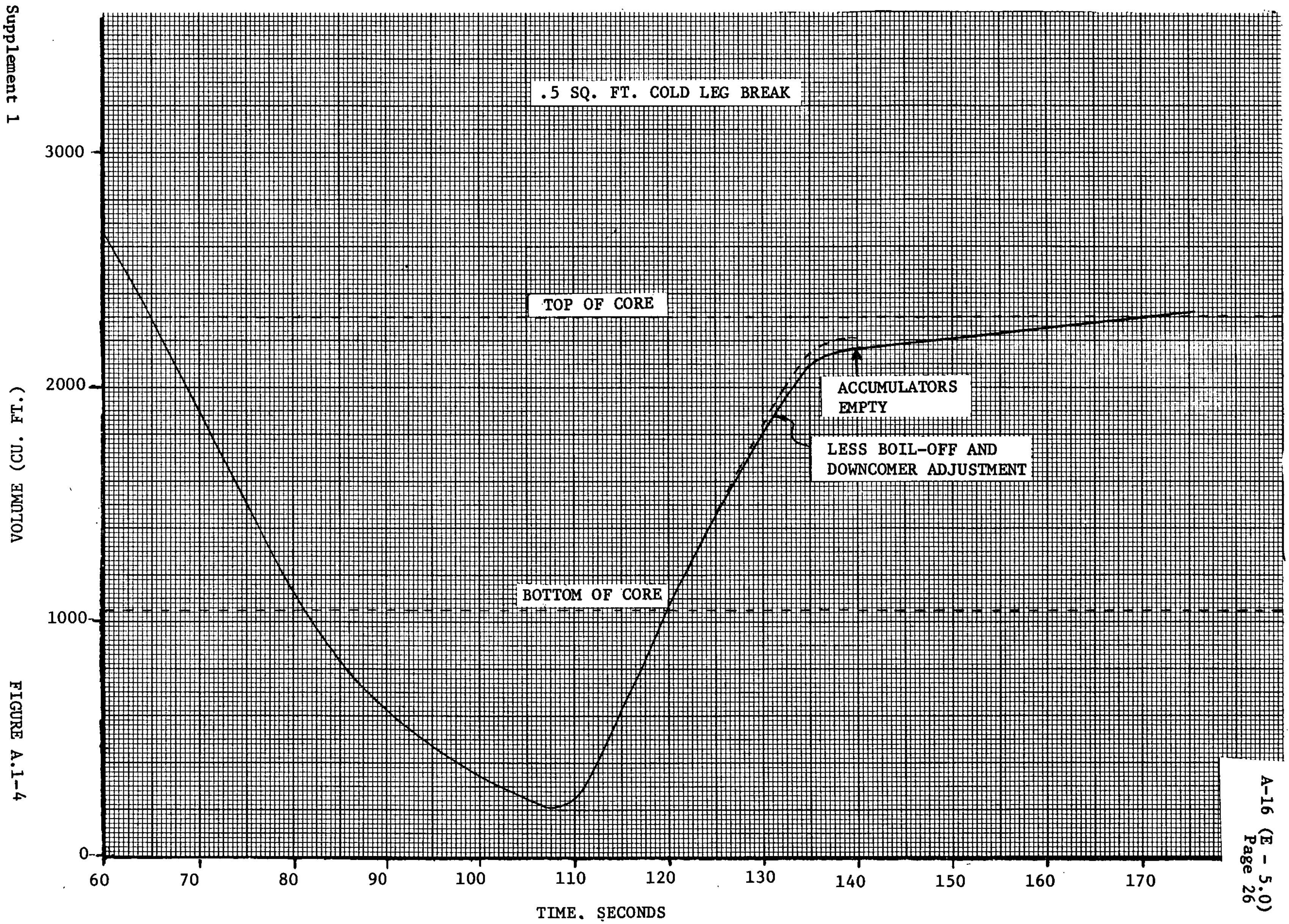

.5 SQ. FT. COLD LEG BREAK
TOP OF CORE
BOTTOM OF CORE
ACCUMULATORS EMPTY
LESS BOIL-OFF AND DOWNCOMER ADJUSTMENT
VOLUME (CU. FT.)
0
1000
2000
3000
60
70
80
90
100
110
120
130
140
150
160
170
TIME, SECONDS

FIGURE A.1-4

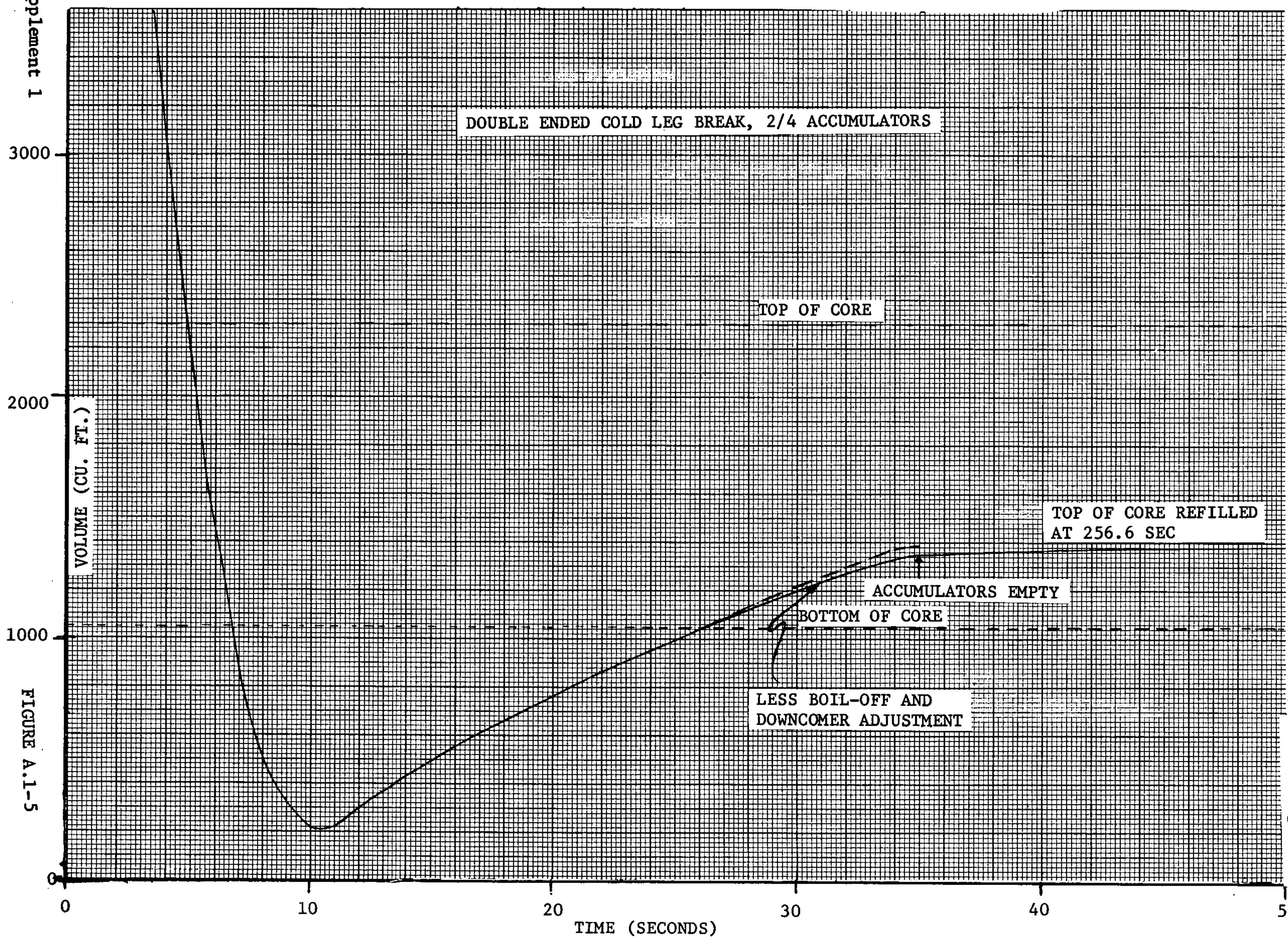

DOUBLE ENDED COLD LEG BREAK, 2/4 ACCUMULATORS
3000
2000
1000
0
VOLUME (CU. FT.)
TOP OF CORE
TOP OF CORE REFILLED
AT 256.6 SEC
ACCUMULATORS EMPTY
BOTTOM OF CORE
LESS BOIL-OFF AND
DOWNCOMER ADJUSTMENT
0
10
20
30
40
50
TIME (SECONDS)

FIGURE A.1-5

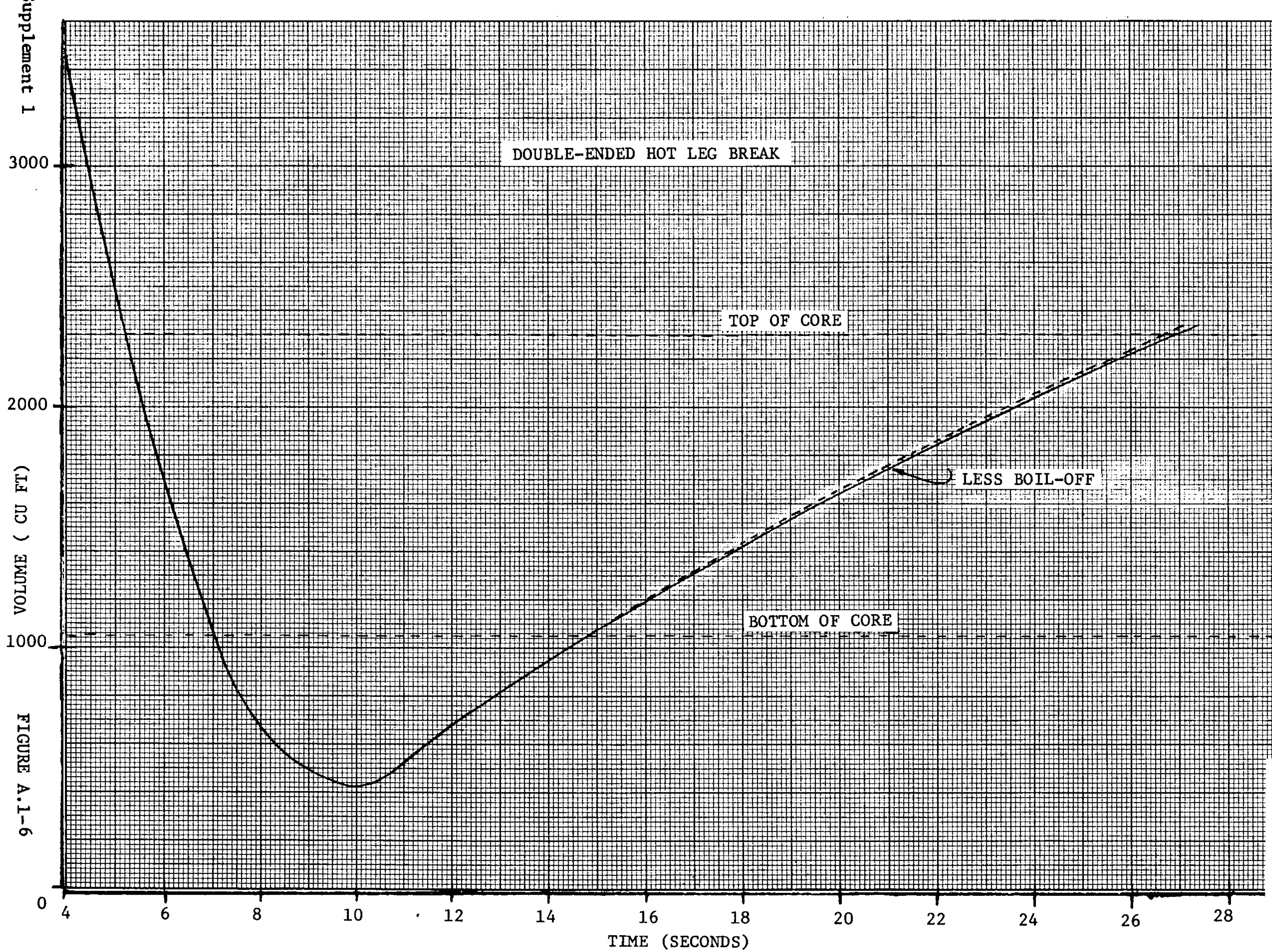

DOUBLE-ENDED HOT LEG BREAK
TOP OF CORE
LESS BOIL-OFF
BOTTOM OF CORE
VOLUME (CU FT)
3000
2000
1000
0
4
6
8
10
12
14
16
18
20
22
24
26
28
TIME (SECONDS)

FIGURE A.1-6

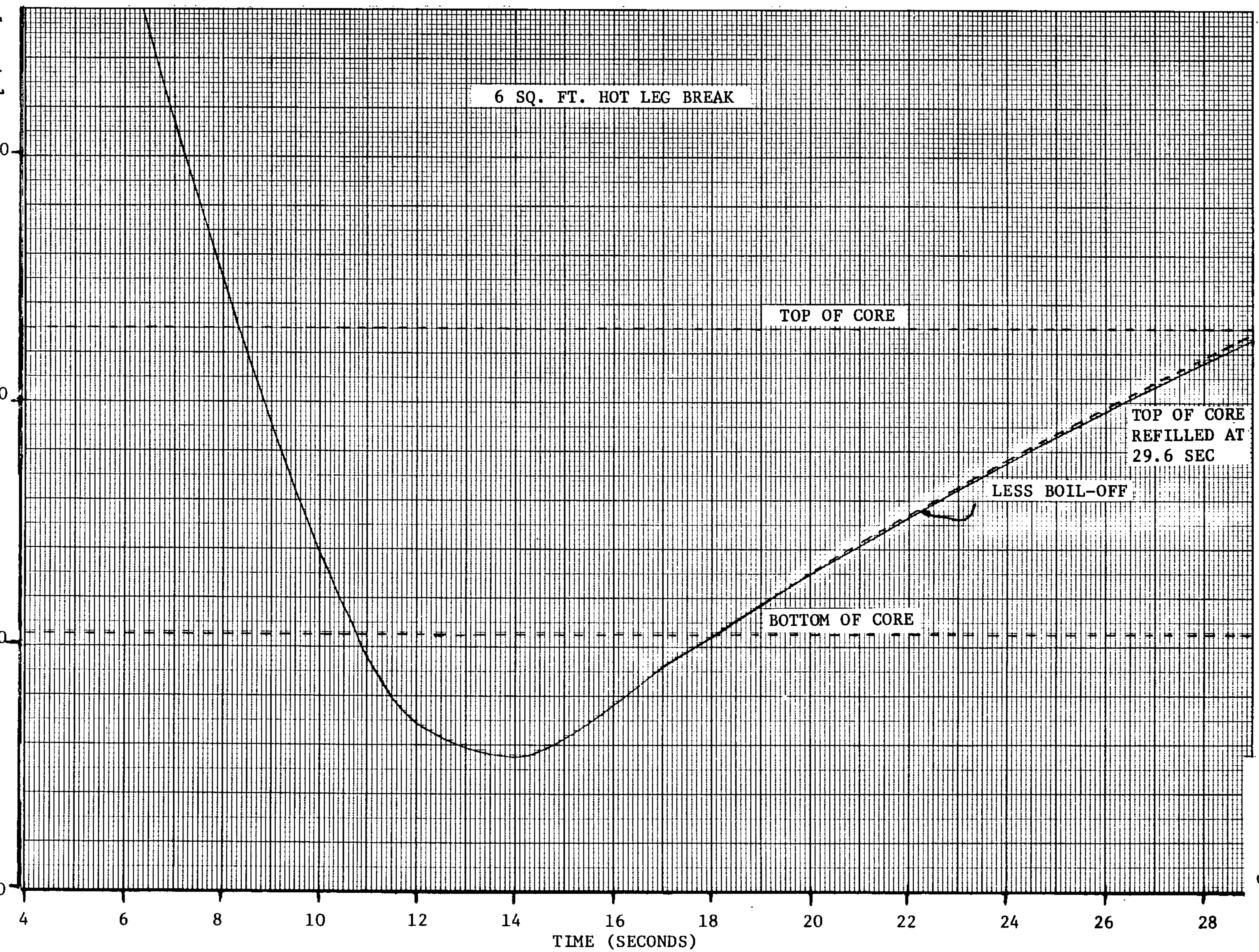

6 SQ. FT. HOT LEG BREAK
TOP OF CORE
TOP OF CORE
REFILLED AT
29.6 SEC
LESS BOIL-OFF
BOTTOM OF CORE
VOLUME (CU FT)
TIME (SECONDS)
3000
2000
1000
0
4
6
8
10
12
14
16
18
20
22
24
26
28

FIGURE A.1-7

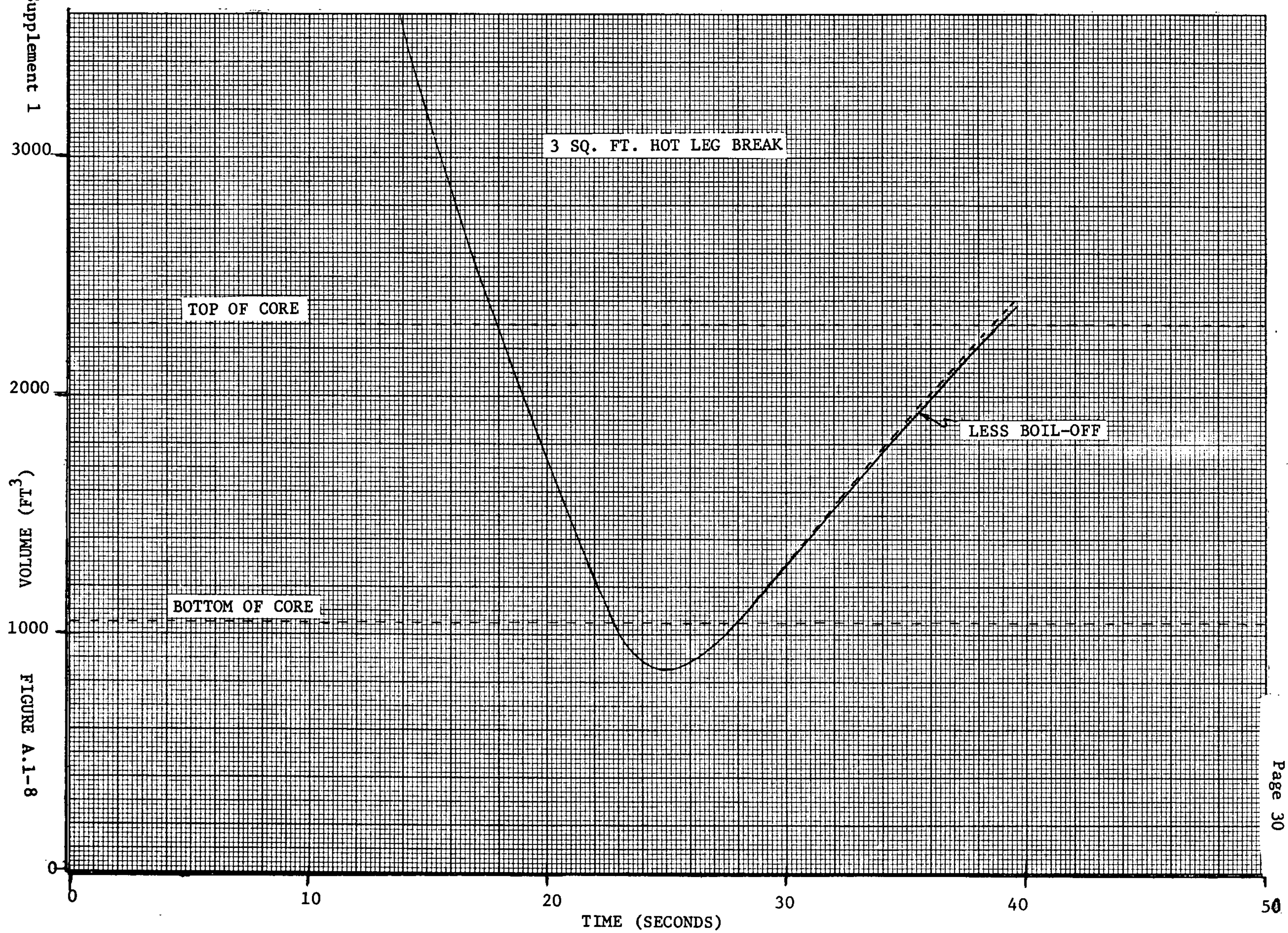

3 SQ. FT. HOT LEG BREAK
TOP OF CORE
BOTTOM OF CORE
LESS BOIL-OFF
VOLUME (FT^3)
3000
2000
1000
0
TIME (SECONDS)
0
10
20
30
40
50

FIGURE A.1-8

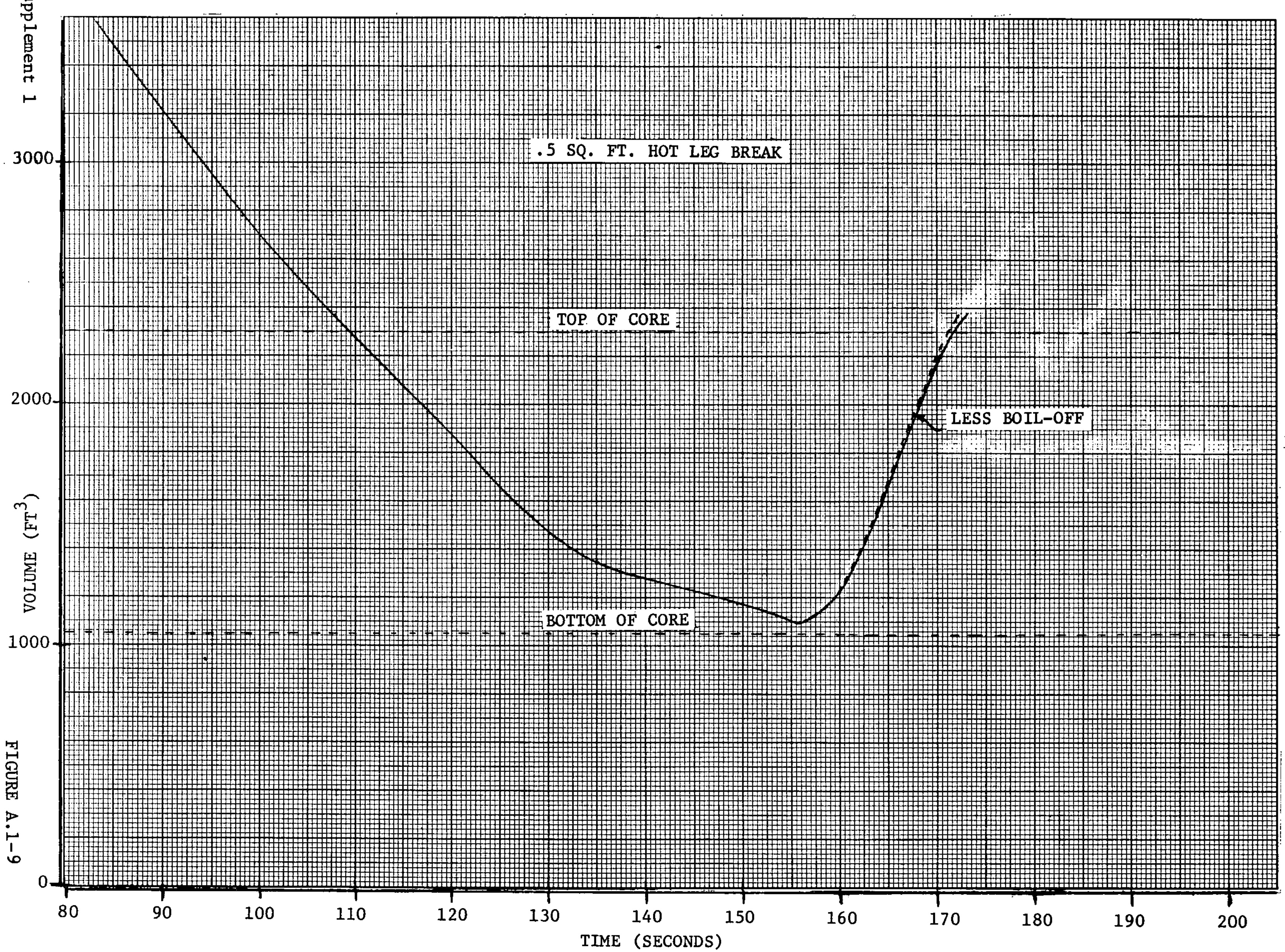

.5 SQ. FT. HOT LEG BREAK
TOP OF CORE
LESS BOIL-OFF
BOTTOM OF CORE
VOLUME (FT^3)
0
1000
2000
3000
80
90
100
110
120
130
140
150
160
170
180
190
200
TIME (SECONDS)

FIGURE A.1-9

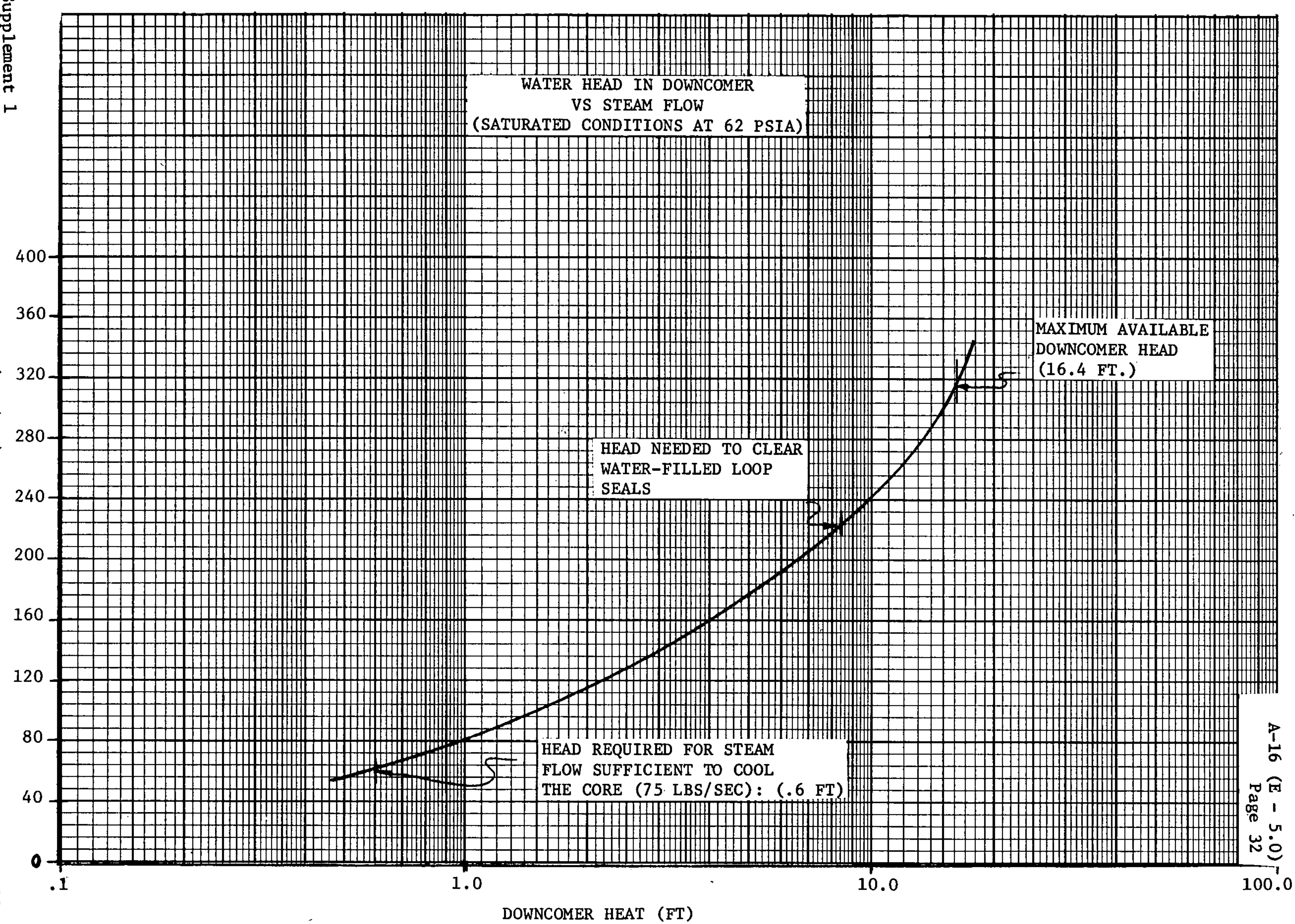
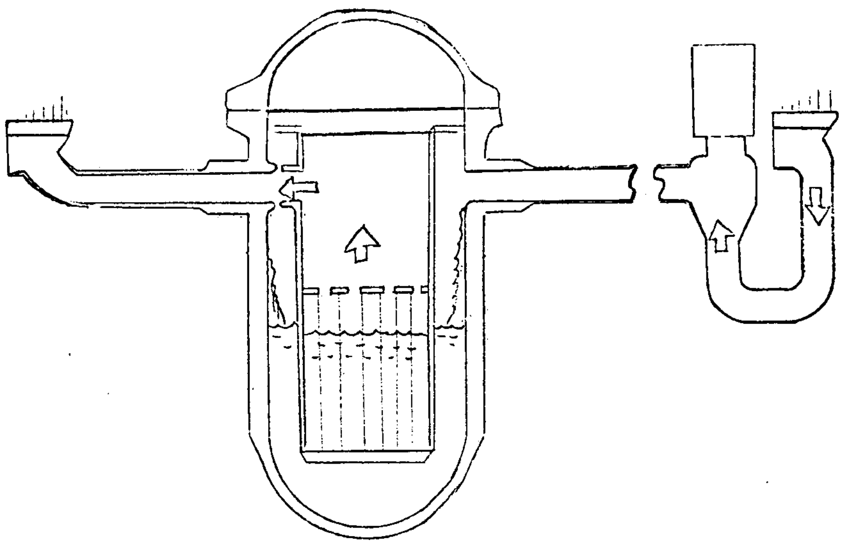

WATER HEAD IN DOWNCOMER
VS STEAM FLOW
(SATURATED CONDITIONS AT 62 PSIA)
MAXIMUM AVAILABLE
DOWNCOMER HEAD
(16.4 FT.)
HEAD NEEDED TO CLEAR
WATER-FILLED LOOP
SEALS
HEAD REQUIRED FOR STEAM
FLOW SUFFICIENT TO COOL
THE CORE (75 LBS/SEC): (.6 FT)
STEAM FLOW THRU FOUR LOOPS (LBS/SEC)
400
360
320
280
240
200
160
120
80
40
0
.1
1.0
10.0
100.0
DOWNCOMER HEAT (FT)

FIGURE A.1-10



STEAM FLOW PATH

FIGURE A.1-10a

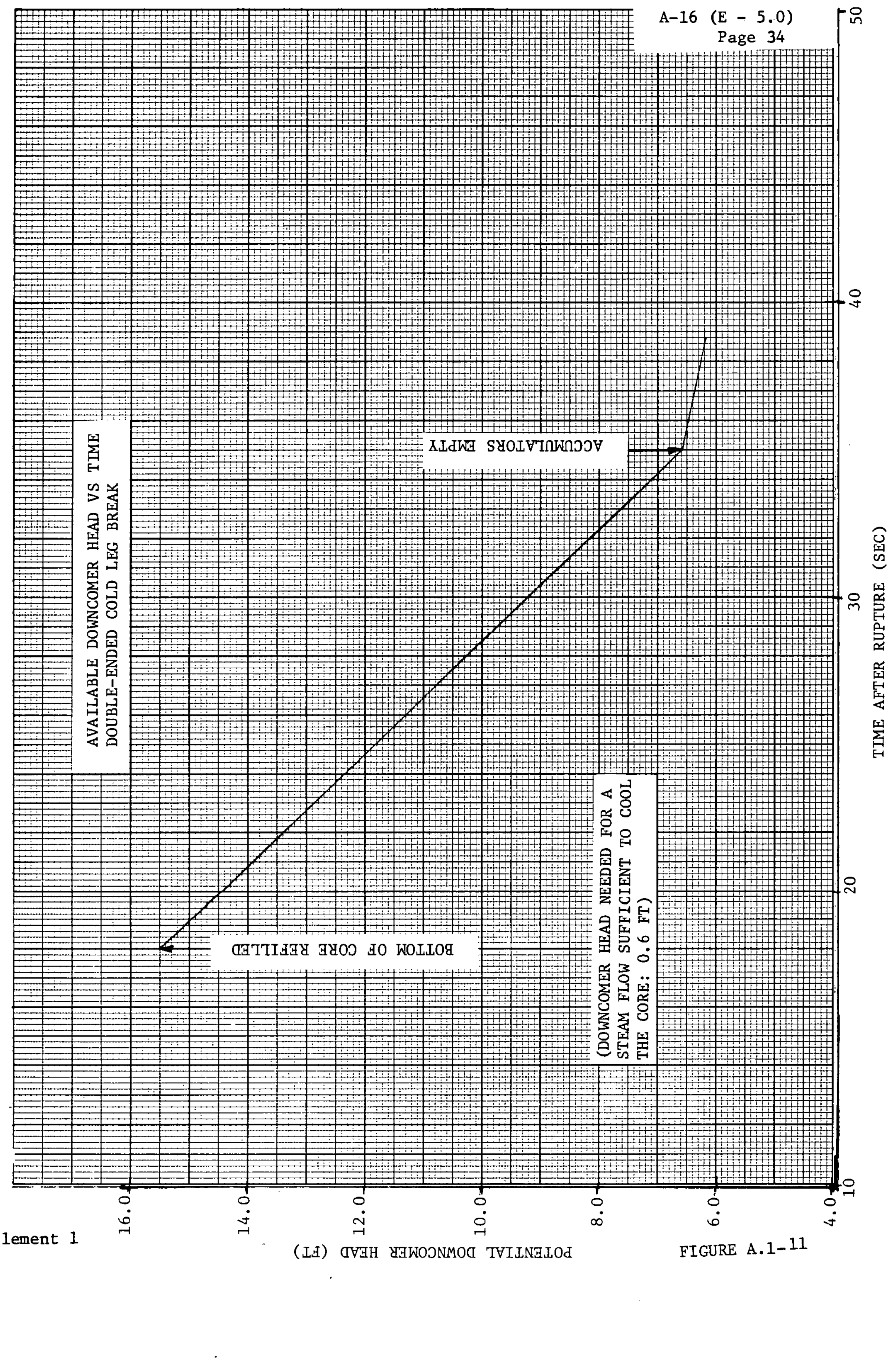
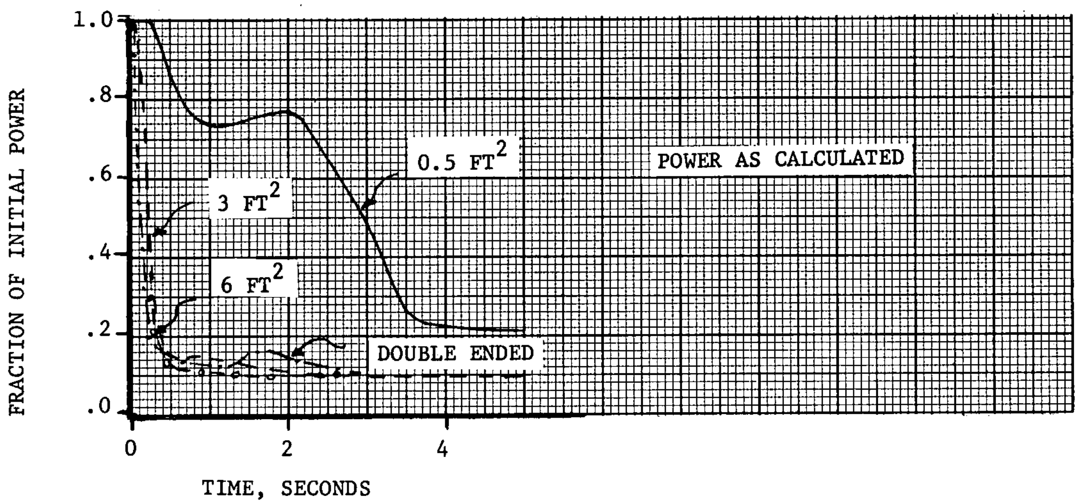
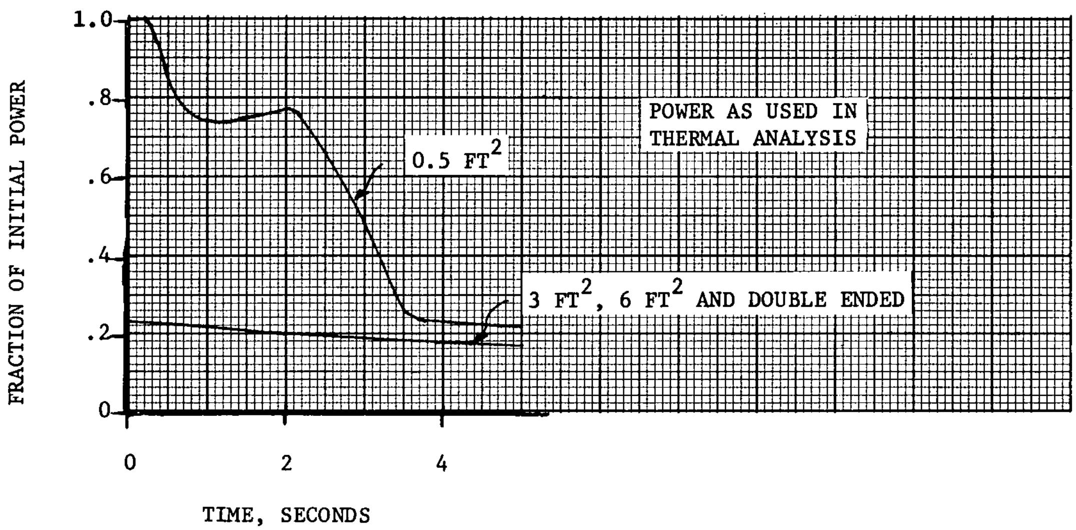
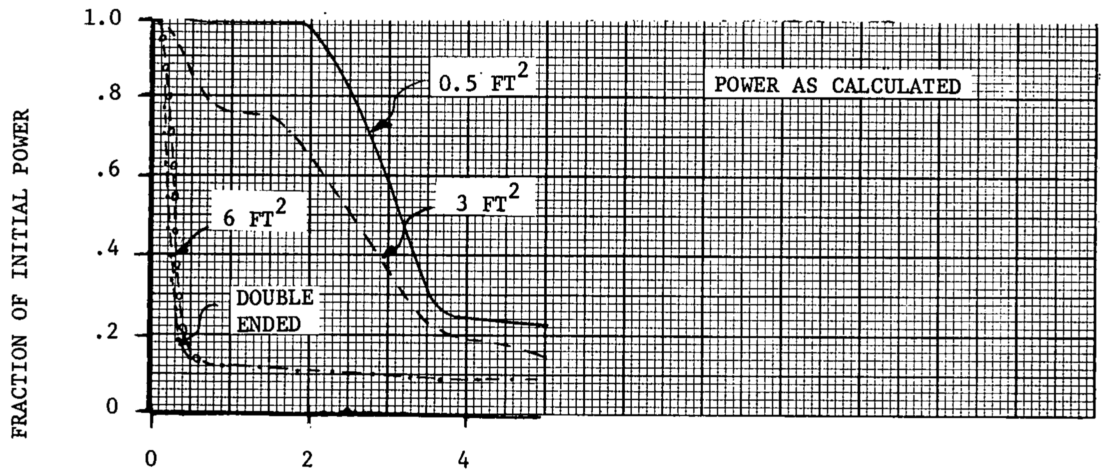

AVAILABLE DOWNCOMER HEAD VS TIME
DOUBLE-ENDED COLD LEG BREAK
ACCUMULATORS EMPTY
BOTTOM OF CORE REFILLED
(DOWNCOMER HEAD NEEDED FOR A
STEAM FLOW SUFFICIENT TO COOL
THE CORE: 0.6 FT)
TIME AFTER RUPTURE (SEC)
10
20
30
40
50
POTENTIAL DOWNCOMER HEAD (FT)
16.0
14.0
12.0
10.0
8.0
6.0
4.0

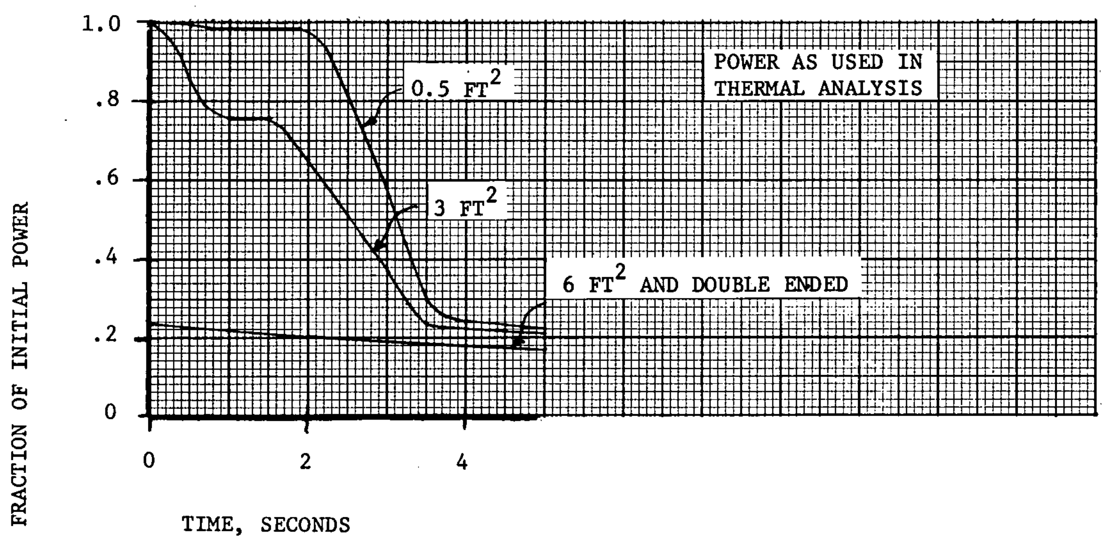
FIGURE A.1-11



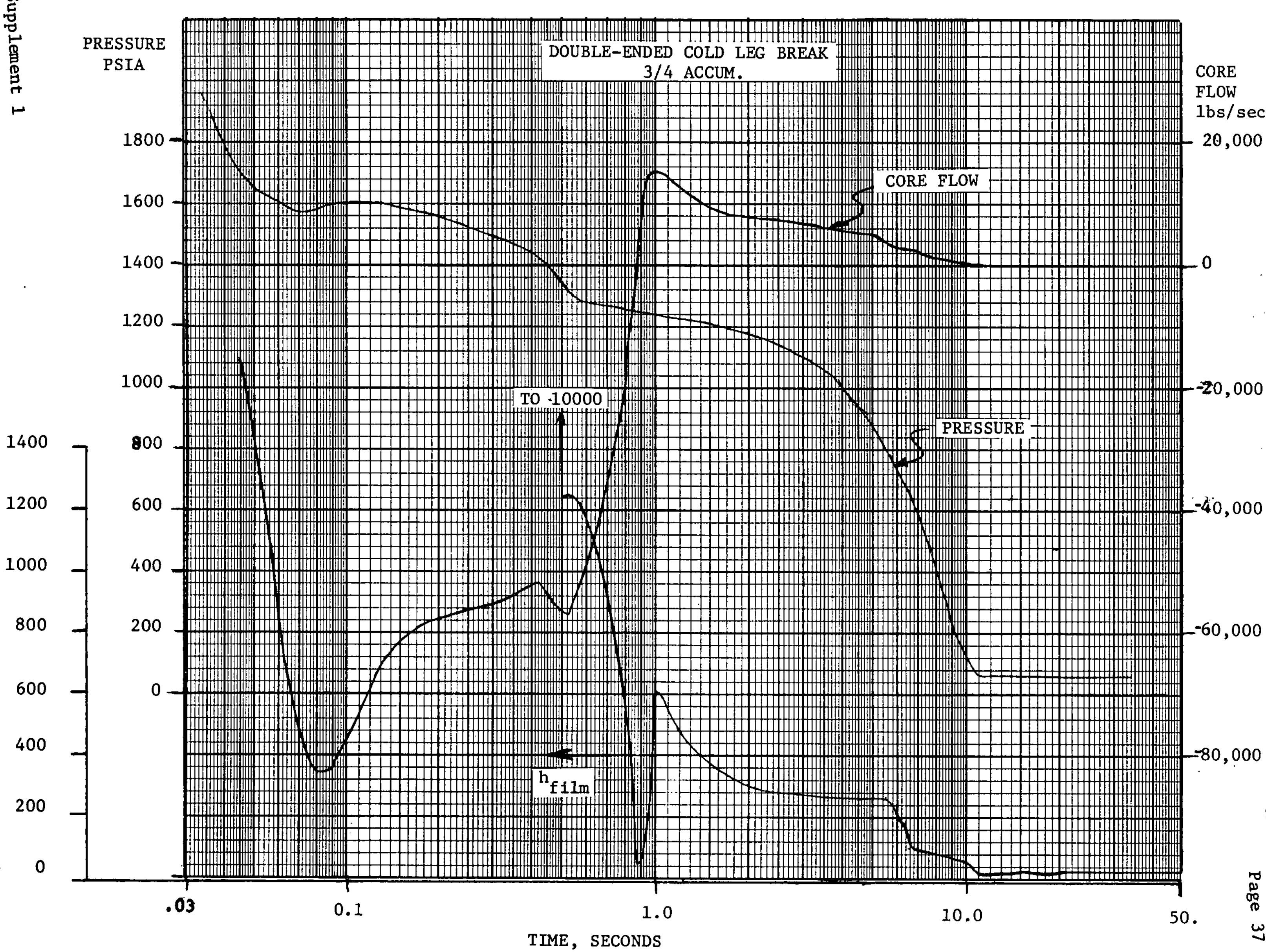


POWER TRANSIENT DURING BLOWDOWN
COLD LEG BREAKS



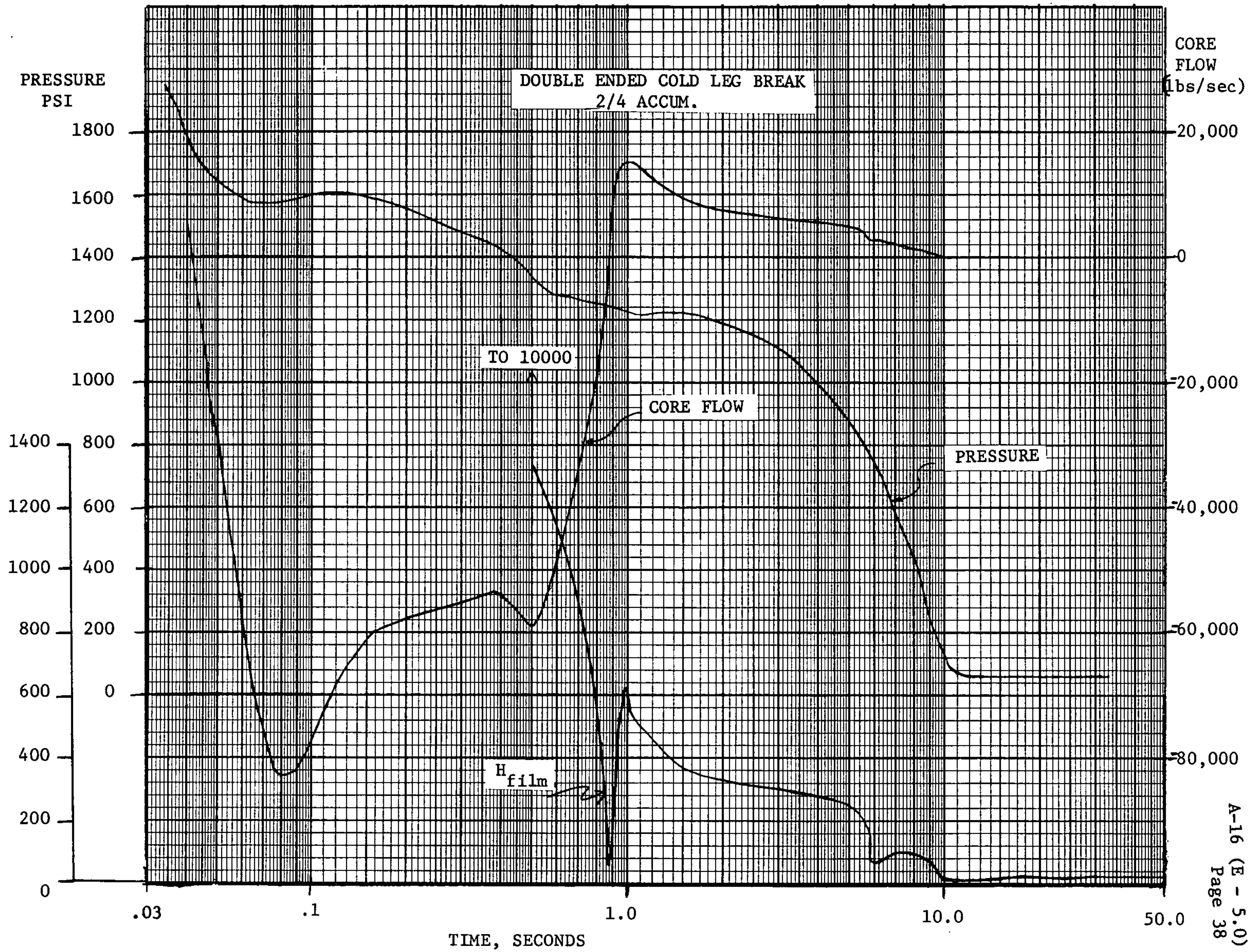


POWER TRANSIENT DURING BLOWDOWN
HOT LEG BREAKS



DOUBLE-ENDED COLD LEG BREAK
3/4 ACCUM.
PRESSURE
PSIA
1800
1600
1400
1200
1000
800
600
400
200
0
CORE
FLOW
lbs/sec
20,000
0
-20,000
-40,000
-60,000
-80,000
h_{film} - BTU/HR/FT^2/°F
1400
1200
1000
800
600
400
200
0
CORE FLOW
PRESSURE
TO 10000
h_{film}
.03
0.1
1.0
10.0
50.
TIME, SECONDS

FIGURE A.1-14



DOUBLE ENDED COLD LEG BREAK
2/4 ACCUM.
PRESSURE
PSI
1800
1600
1400
1200
1000
800
600
400
200
0
1400
1200
1000
800
600
400
200
0
h_{film} - BTU/HR/FT^2/°F
CORE
FLOW
(lbs/sec)
20,000
0
-20,000
-40,000
-60,000
-80,000
TO 10000
CORE FLOW
PRESSURE
H_{film}
.03
.1
1.0
10.0
50.0
TIME, SECONDS

FIGURE A.1-15

Supplement 1

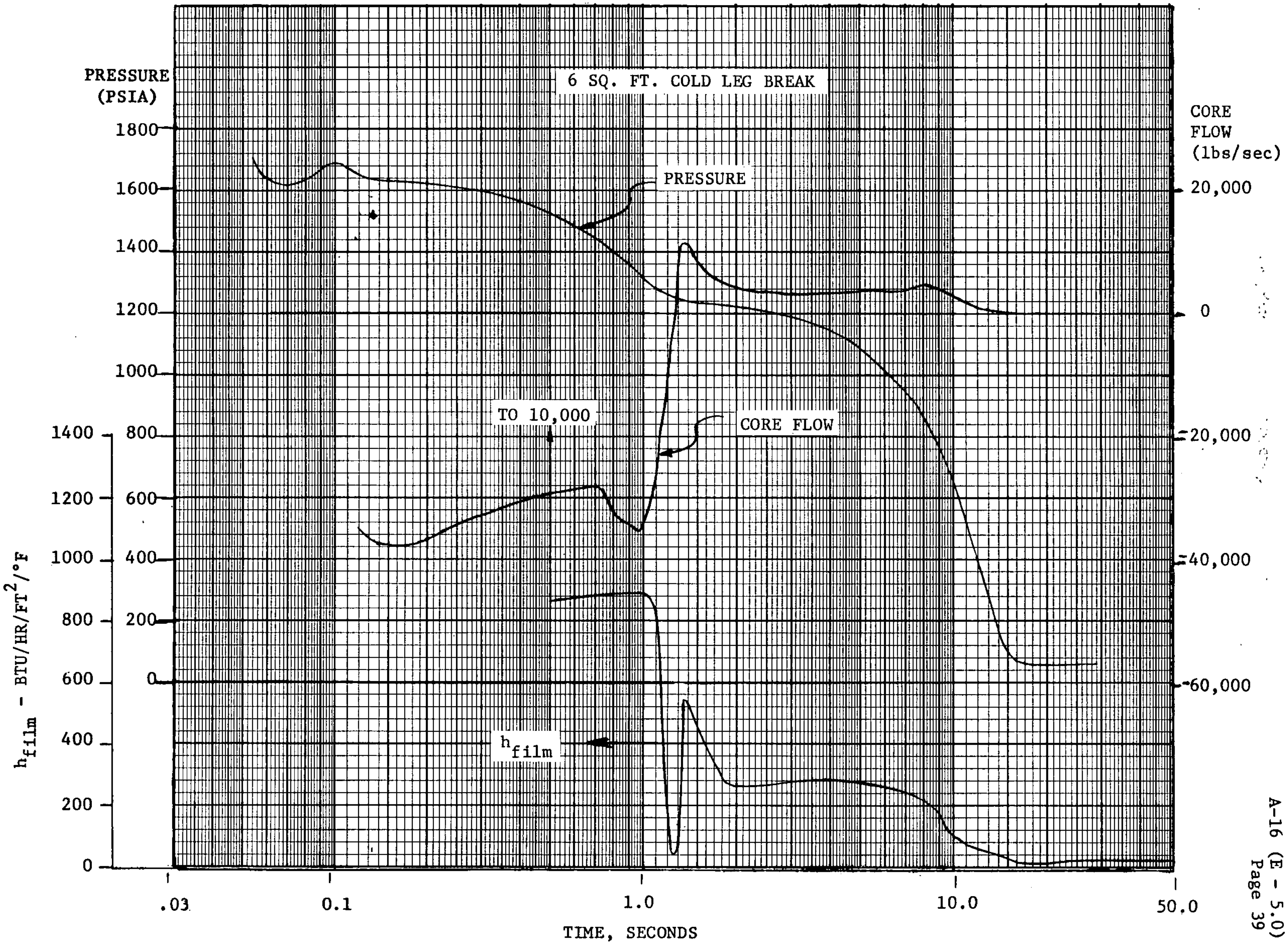


FIGURE A.1-16

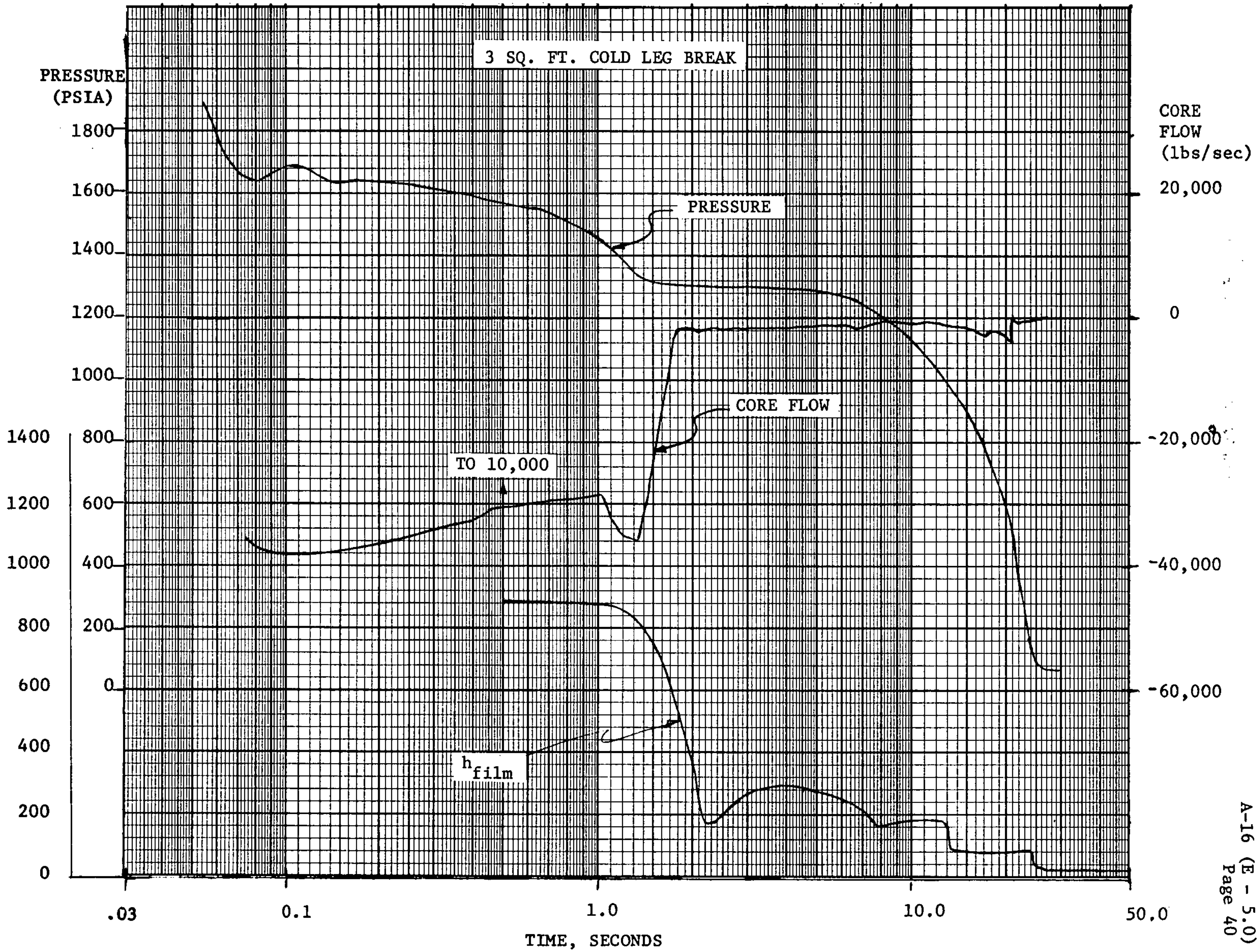
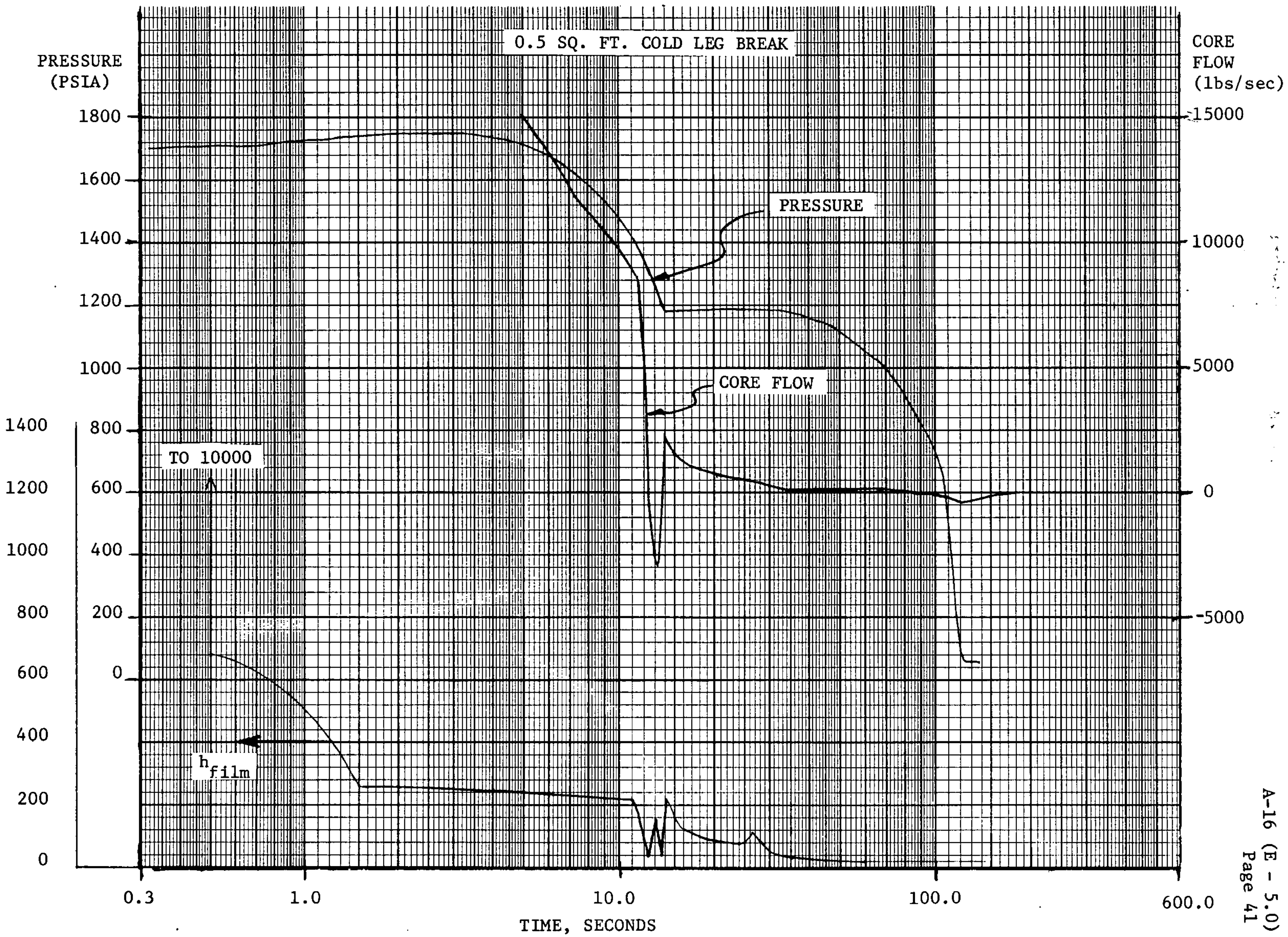

3 SQ. FT. COLD LEG BREAK
PRESSURE (PSIA)
1800
1600
1400
1200
1000
800
600
400
200
0
CORE FLOW (lbs/sec)
20,000
0
-20,000
-40,000
-60,000
PRESSURE
CORE FLOW
TO 10,000
h_{film}
h_{film} - BTU/HR/FT^2/°F
1400
1200
1000
800
600
400
200
0
.03
0.1
1.0
10.0
50.0
TIME, SECONDS

FIGURE A.1-17



0.5 SQ. FT. COLD LEG BREAK
PRESSURE (PSIA)
1800
1600
1400
1200
1000
800
600
400
200
0
CORE FLOW (lbs/sec)
15000
10000
5000
0
-5000
1400
1200
1000
800
600
400
200
0
h film - BTU/HR/FT2/°F
PRESSURE
CORE FLOW
TO 10000
h film
0.3
1.0
10.0
100.0
600.0
TIME, SECONDS

FIGURE A.1-18

Supplement 1

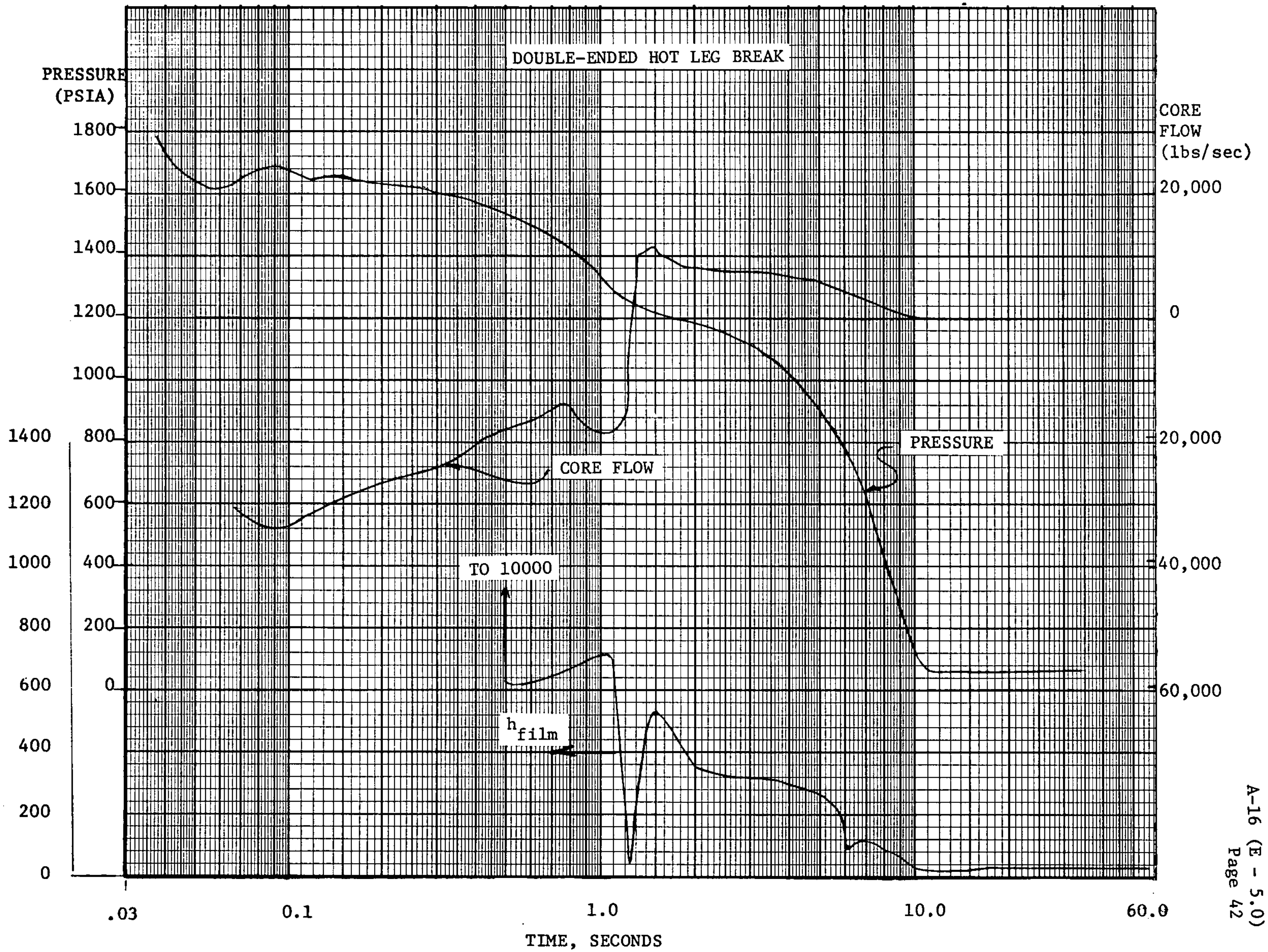


FIGURE A.1-19

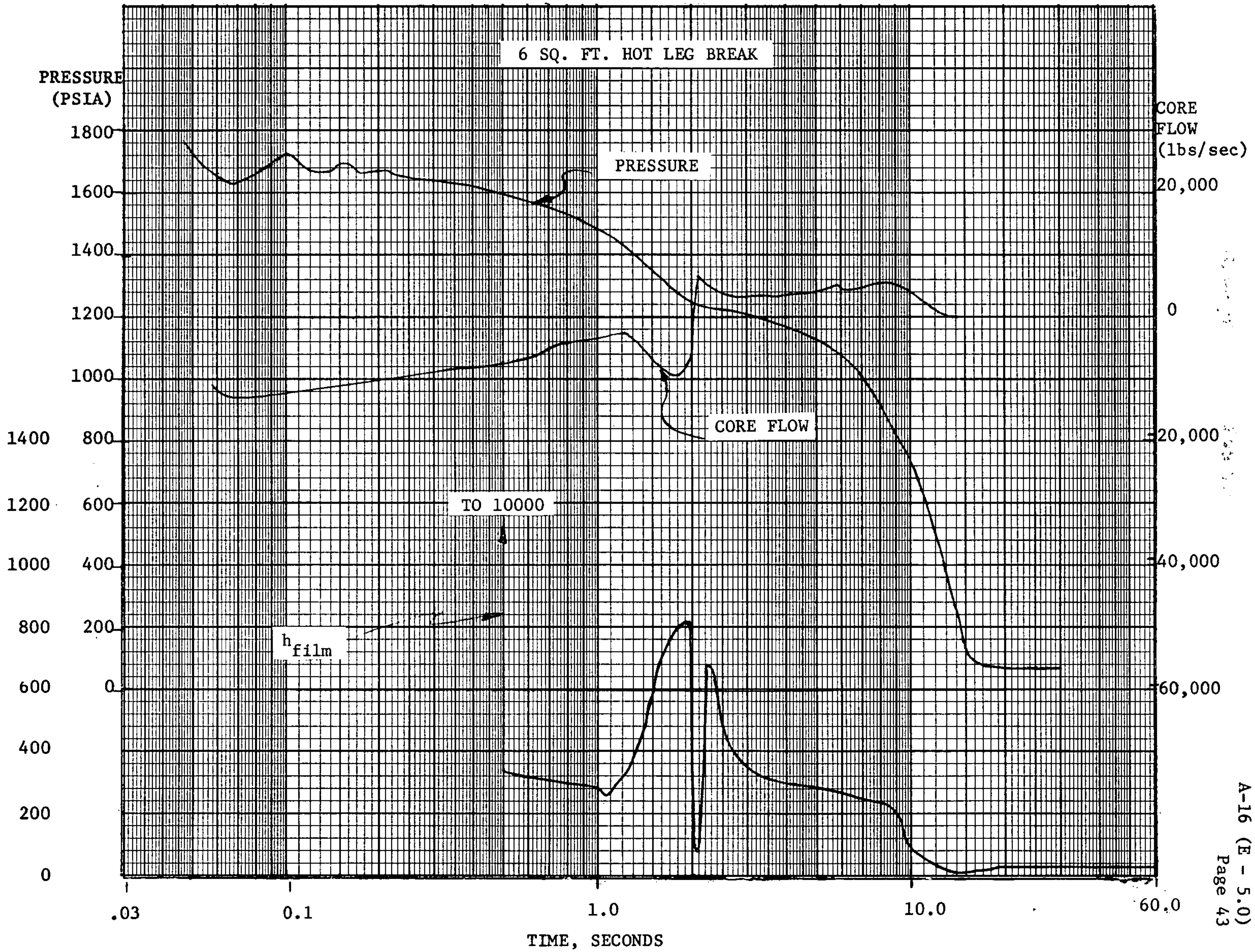

6 SQ. FT. HOT LEG BREAK
PRESSURE (PSIA)
1800
1600
1400
1200
1000
800
600
400
200
0
CORE FLOW (lbs/sec)
20,000
0
-20,000
-40,000
-60,000
h_{film} - BTU/HR/FT^2/°F
1400
1200
1000
800
600
400
200
0
PRESSURE
CORE FLOW
TO 10000
h_{film}
.03
0.1
1.0
10.0
60.0
TIME, SECONDS

FIGURE A.1-20

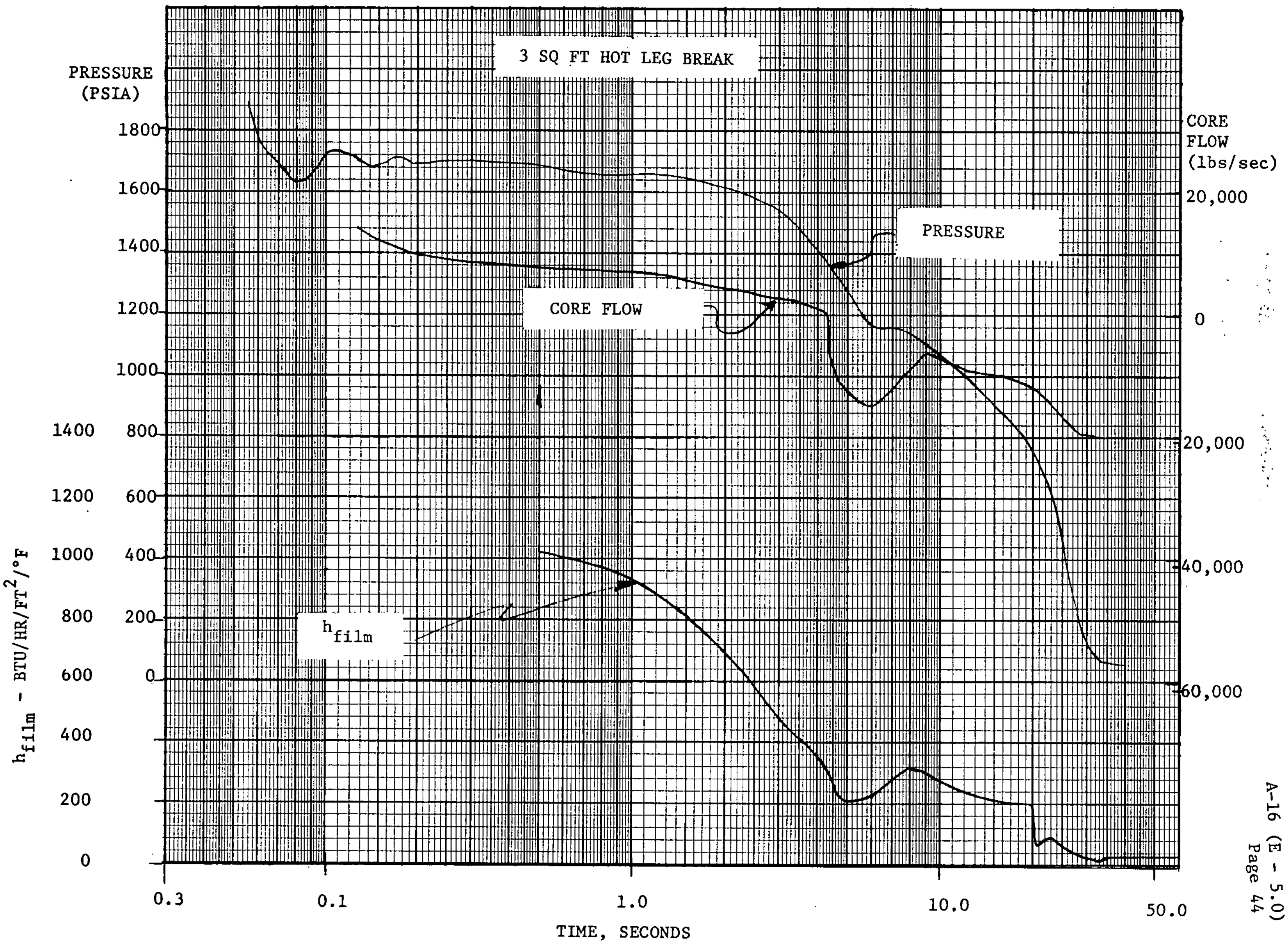

3 SQ FT HOT LEG BREAK
PRESSURE (PSIA)
1800
1600
1400
1200
1000
800
600
400
200
0
CORE FLOW (lbs/sec)
20,000
0
20,000
40,000
60,000
PRESSURE
CORE FLOW
h film
h film - BTU/HR/FT2/°F
1400
1200
1000
800
600
400
200
0
0.3
0.1
1.0
10.0
50.0
TIME, SECONDS

FIGURE A.1-21

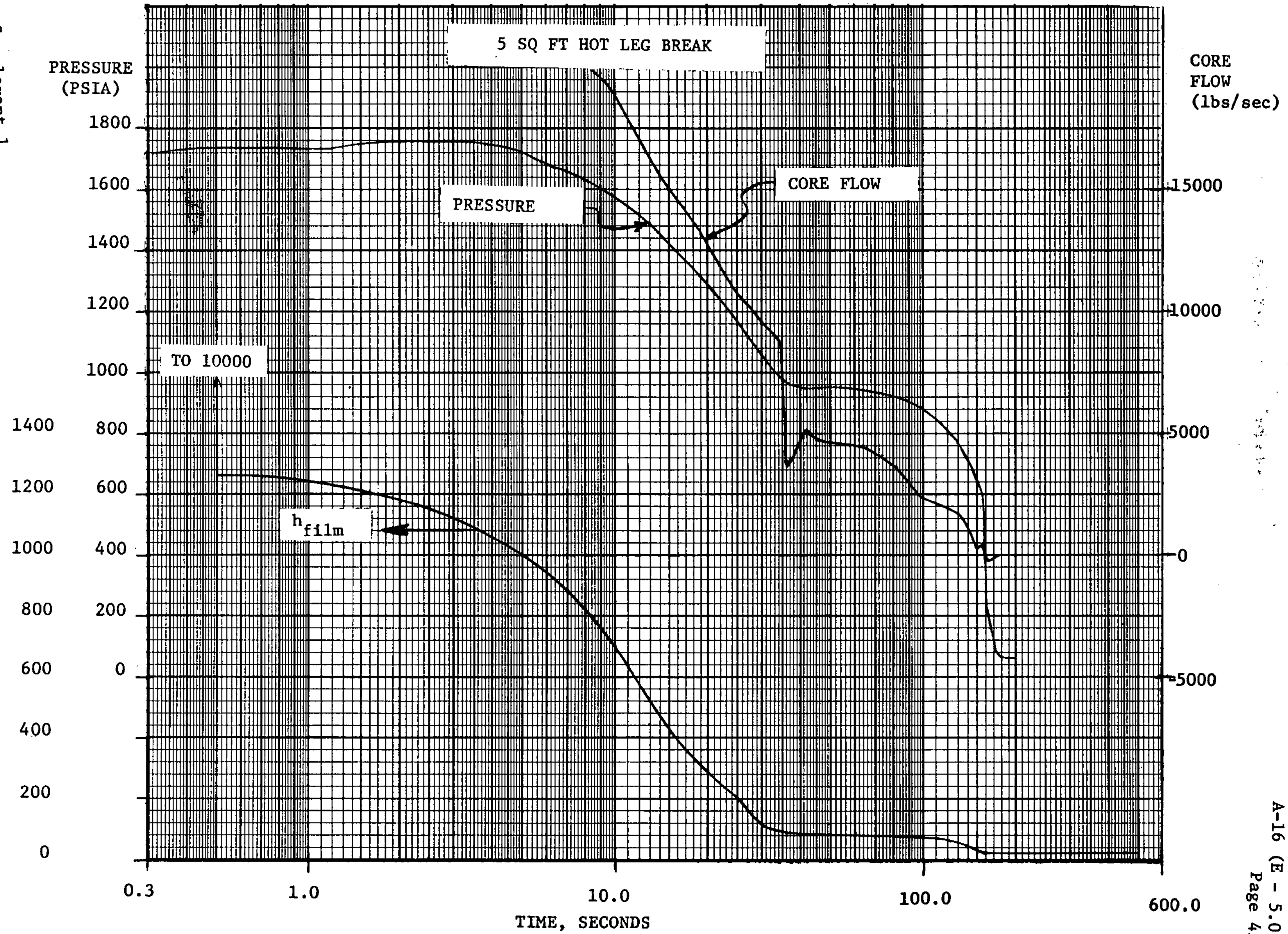

5 SQ FT HOT LEG BREAK
PRESSURE (PSIA)
CORE FLOW (lbs/sec)
PRESSURE
CORE FLOW
TO 10000
h_{film}
h_{film} - BTU/HR/FT^2/°F
1800
1600
1400
1200
1000
800
600
400
200
0
1400
1200
1000
800
600
400
200
0
15000
10000
5000
0
-5000
0.3
1.0
10.0
100.0
600.0
TIME, SECONDS

FIGURE A.1-22

DOUBLE ENDED COLD LEG BREAK - 2758 MWt

CLAD TERMPERATURE TRANSIENT V.S. TIME AFTER BREAK

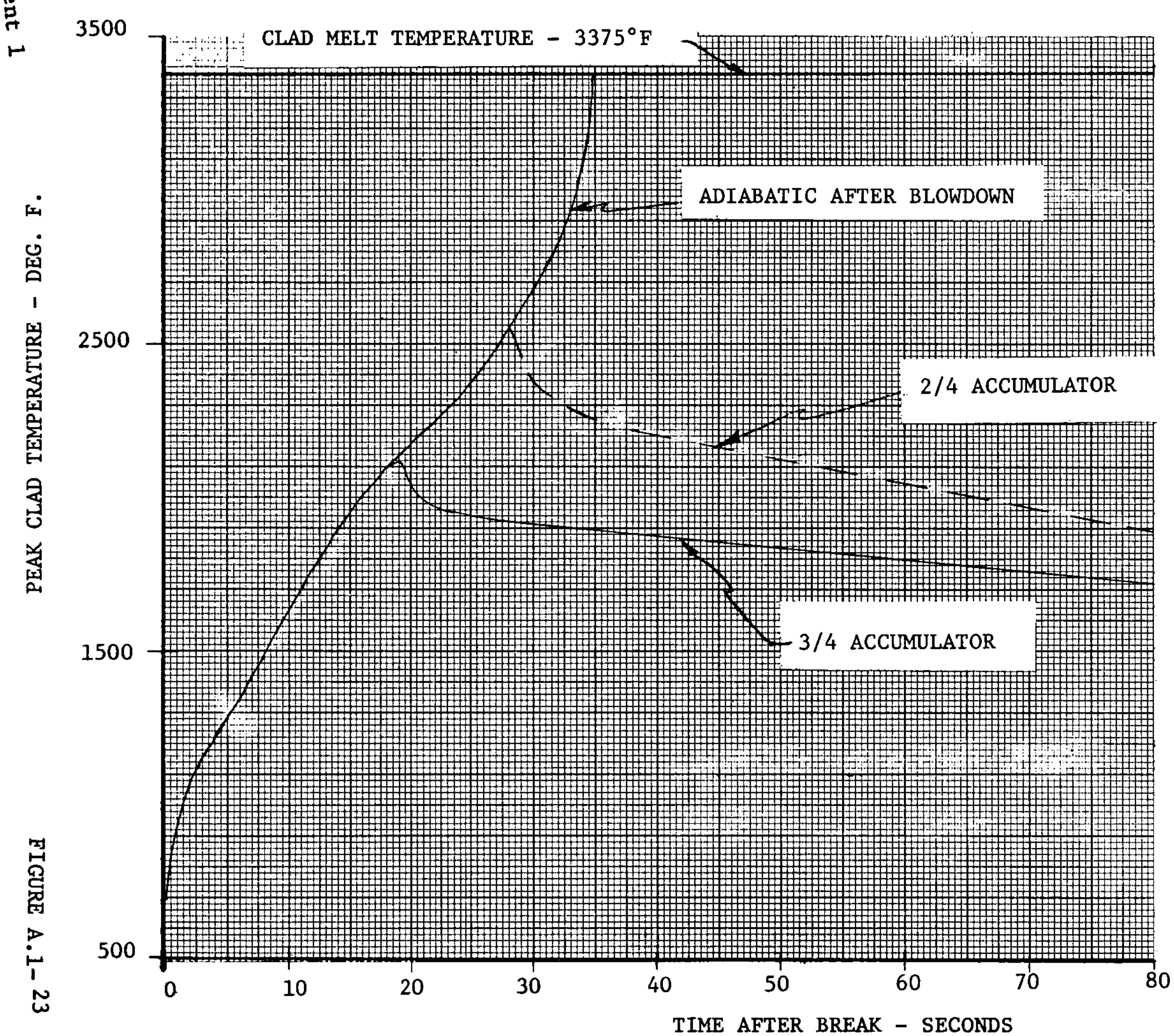


FIGURE A.1- 23

6 FT^2 COLD LEG BREAK - 2758 MWt
CLAD TEMPERATURE TRANSIENT
VS TEMPERATURE AFTER BREAK

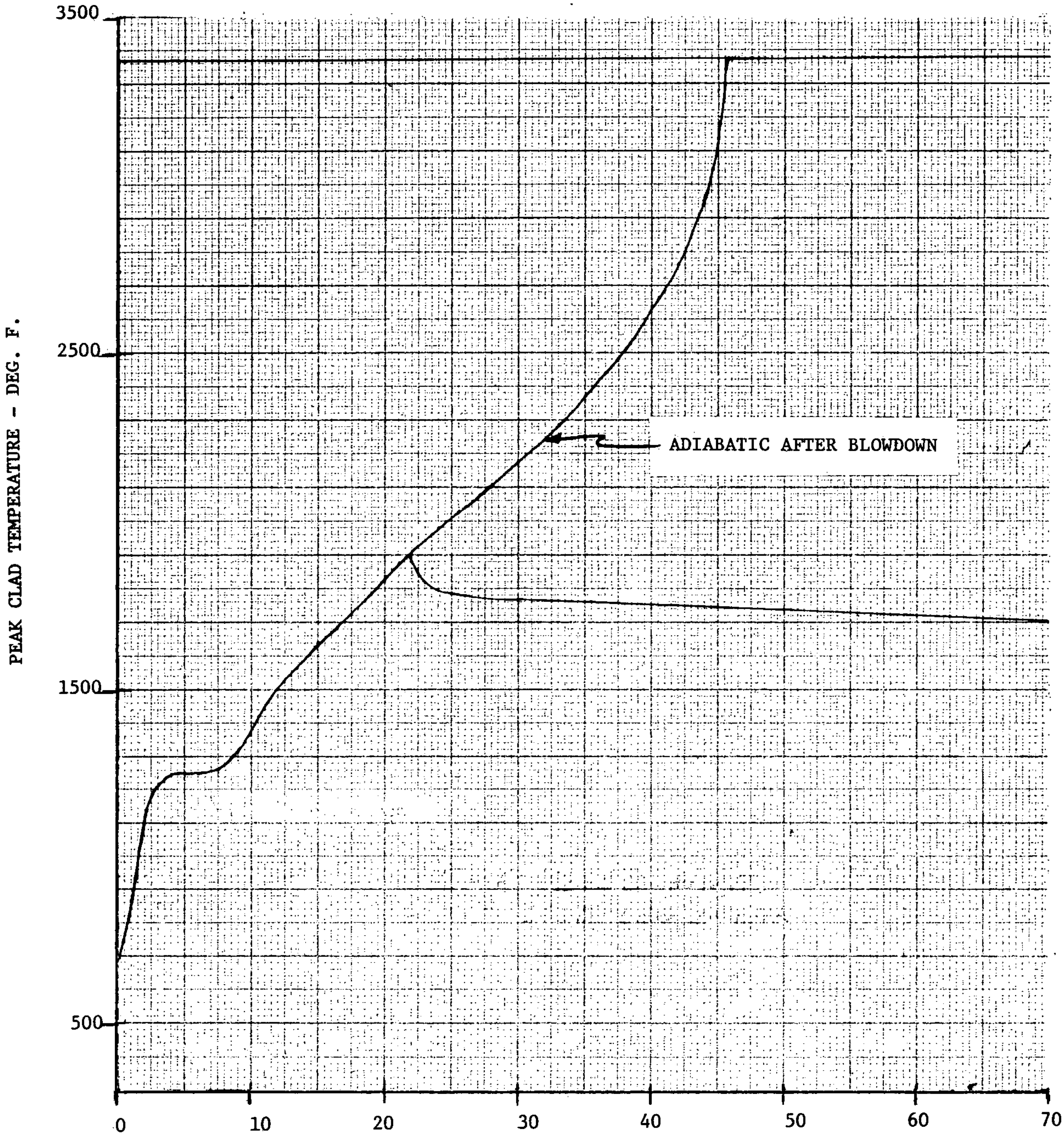


FIGURE A.1- 24

$3FT^2$ COLD LEG BREAK - 2758 MWt
CLAD TEMPERATURE TRANSIENT VS
TIME AFTER BREAK

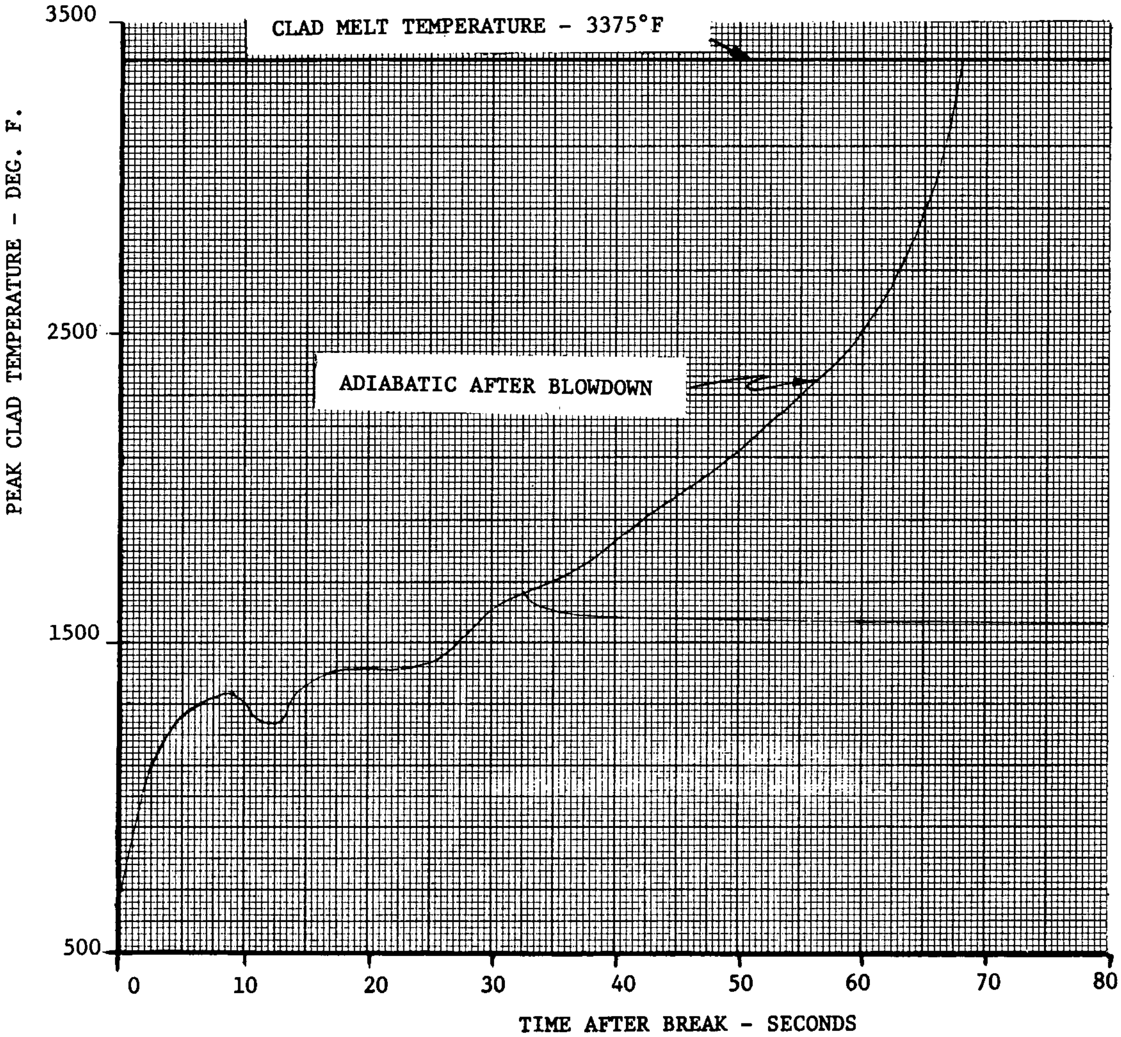


FIGURE A.1- 25

.5 FT^2 COLD LEG BREAK - 2758 MWt

CLAD TEMPERATURE TRANSIENT VS TIME AFTER BREAK

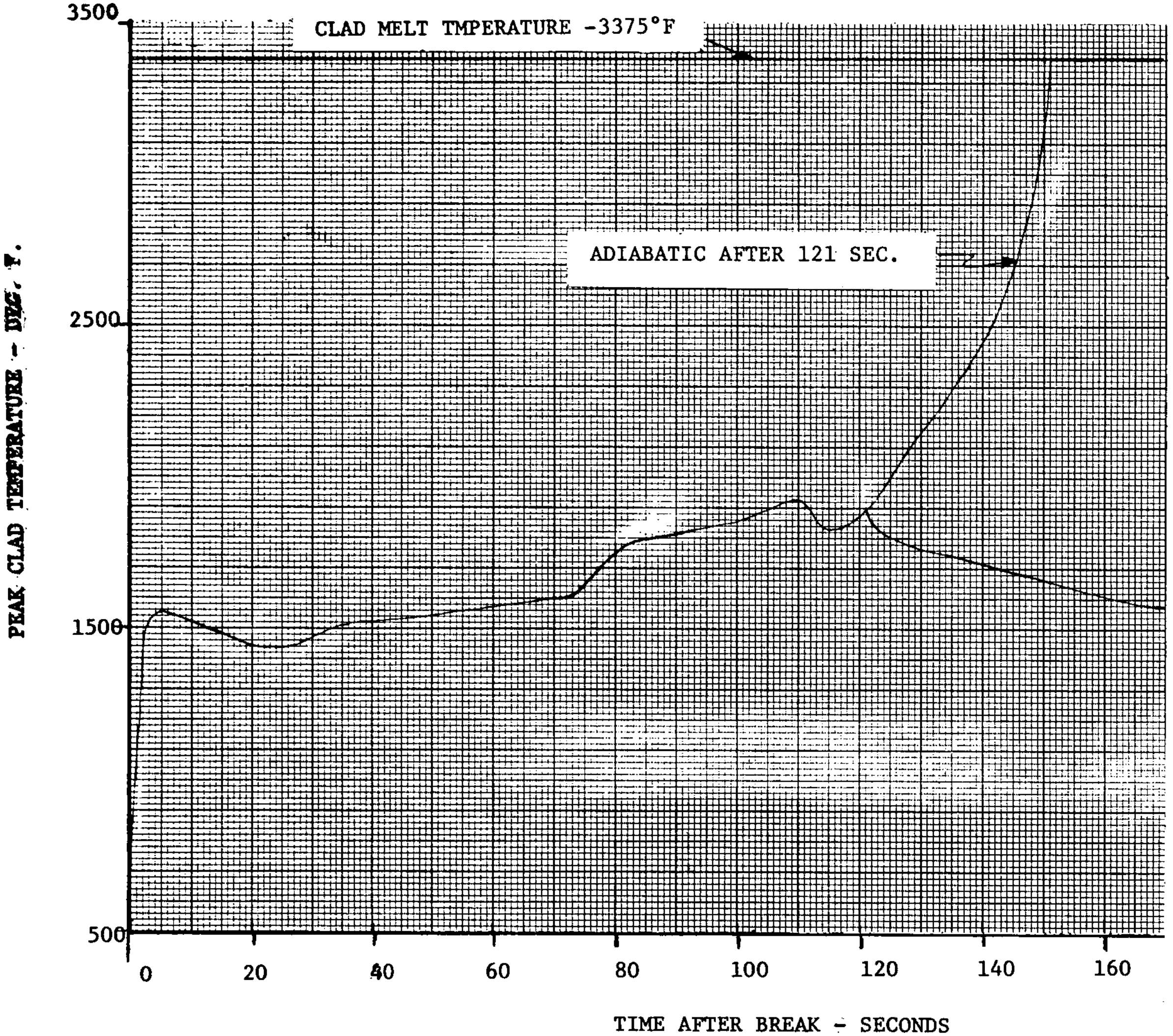


FIGURE A.1-26

Supplement 1

CLAD TEMPERATURE TRANSIENT VS TIME AFTER BREAK

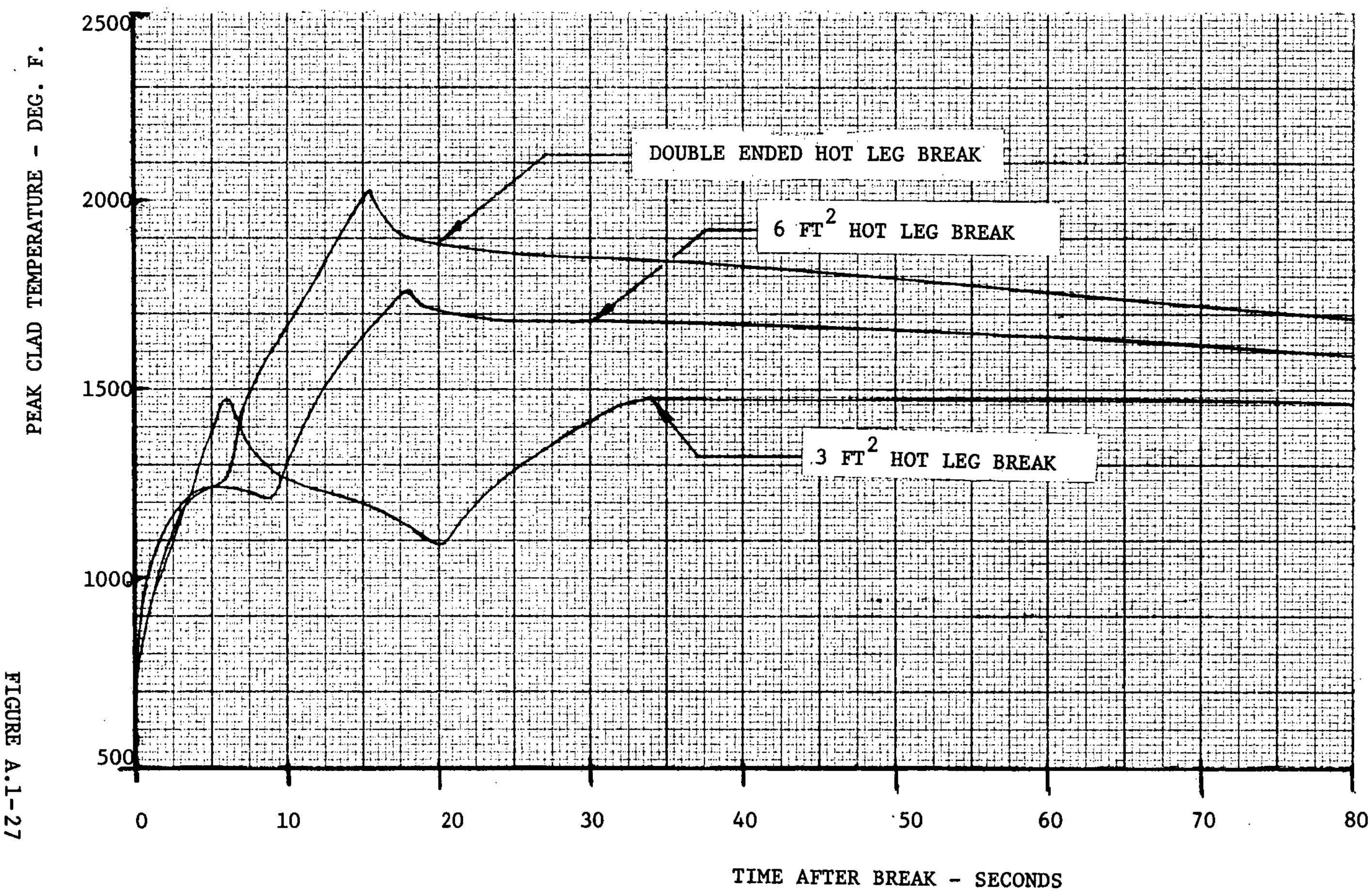


FIGURE A.1-27

Supplement 1

.5 FT^2 HOT LEG BREAK -2758 MWt

CLAD TEMPERATURE TRANSIENT VS TIME AFTER BREAK

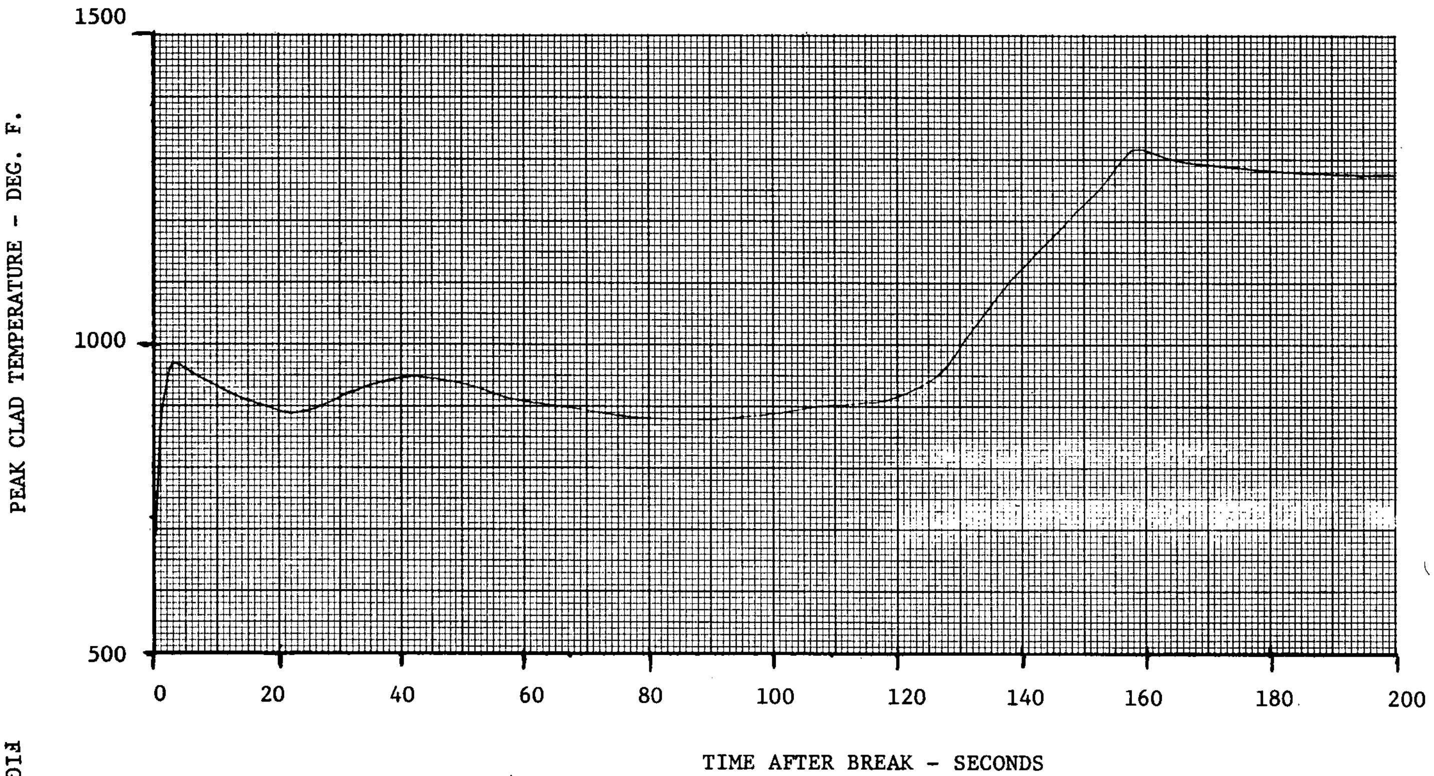


FIGURE A.1-28

SAFETY INJECTION PUMP CURVES

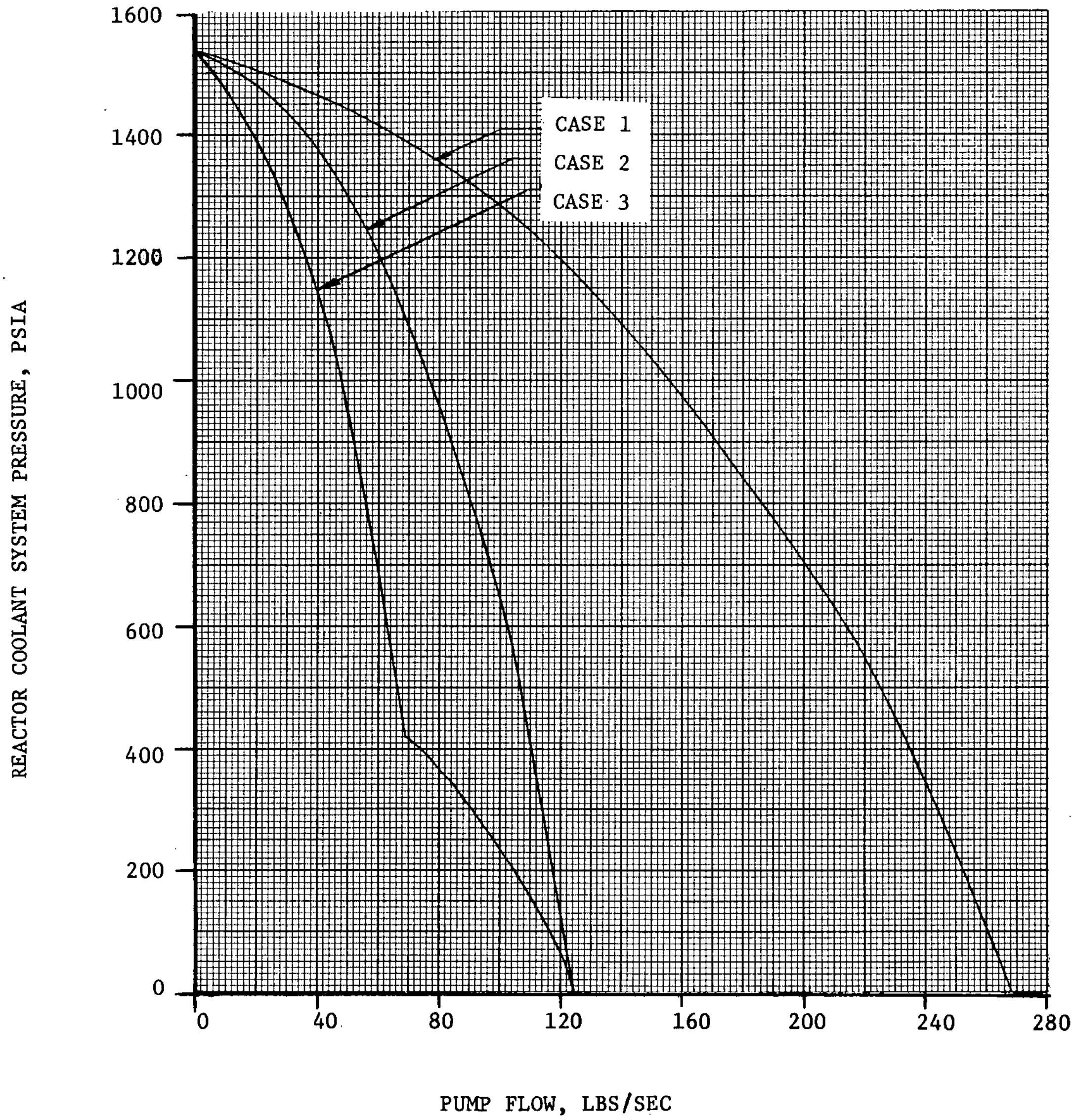


FIGURE A.1-29

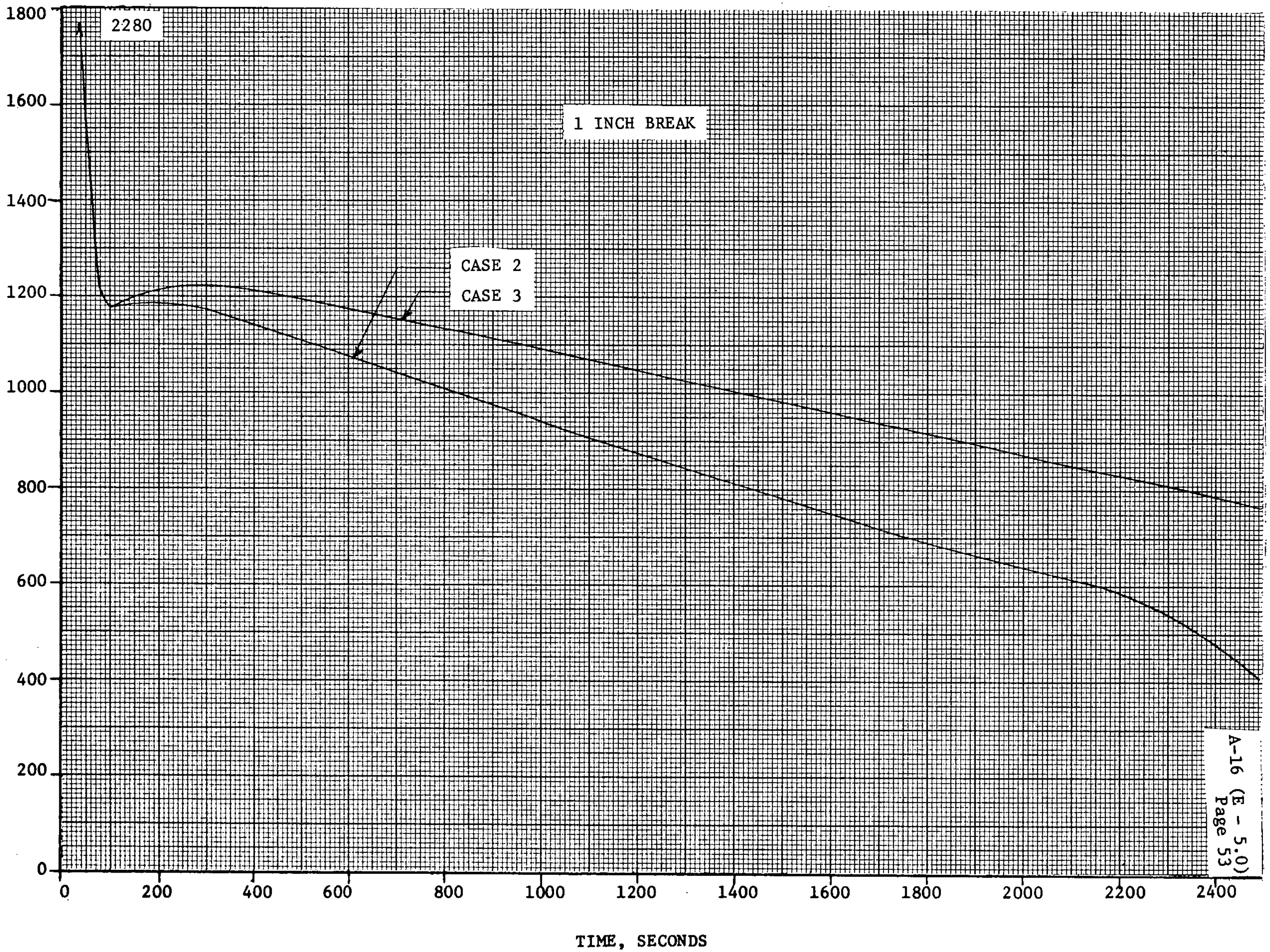
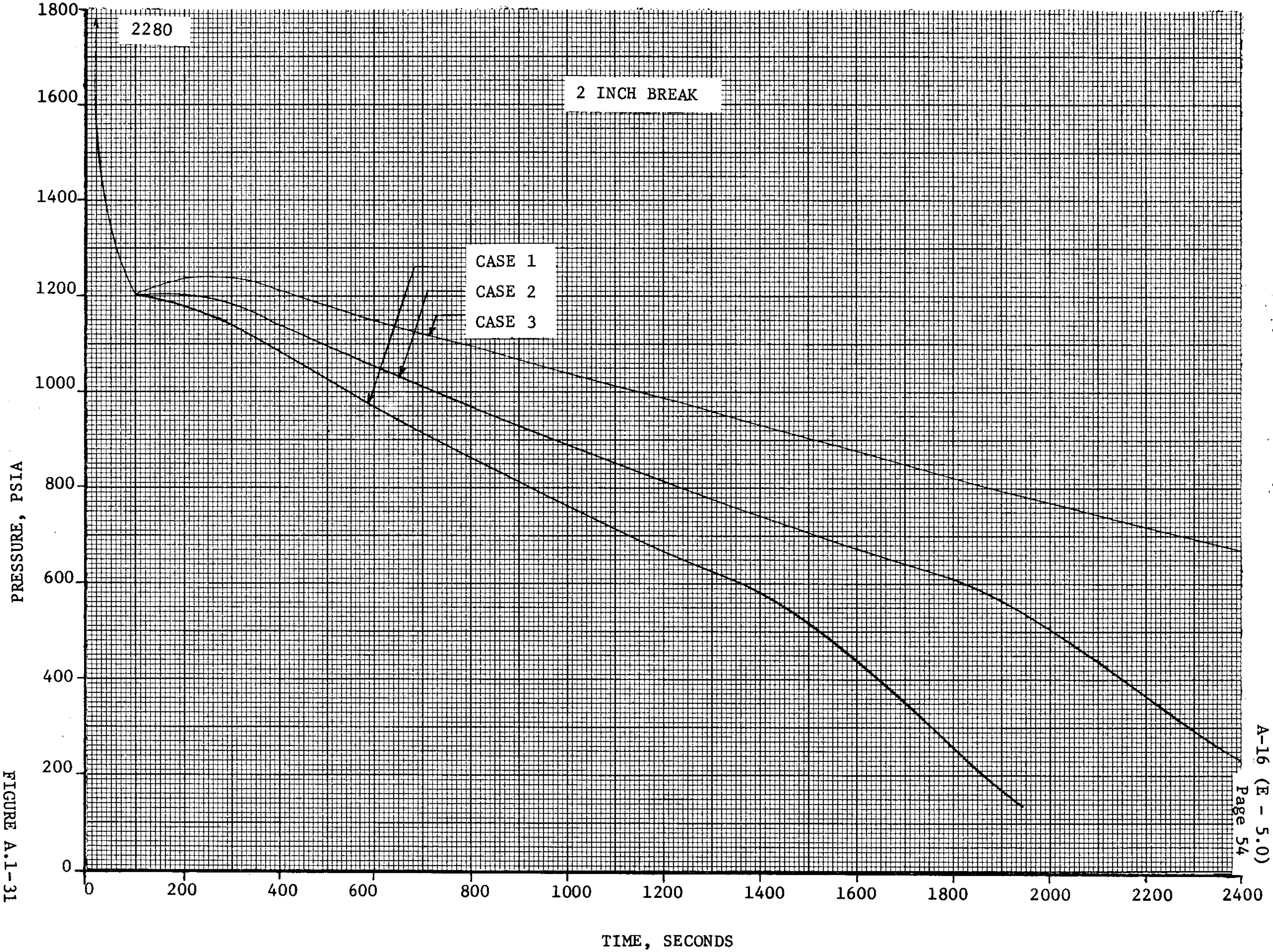

2280
1 INCH BREAK
CASE 2
CASE 3
PRESSURE, PSIA
TIME, SECONDS
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400

FIGURE A.1-30



2 INCH BREAK
2280
CASE 1
CASE 2
CASE 3
PRESSURE, PSIA
TIME, SECONDS
1800
1600
1400
1200
1000
800
600
400
200
0
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400

FIGURE A.1-31

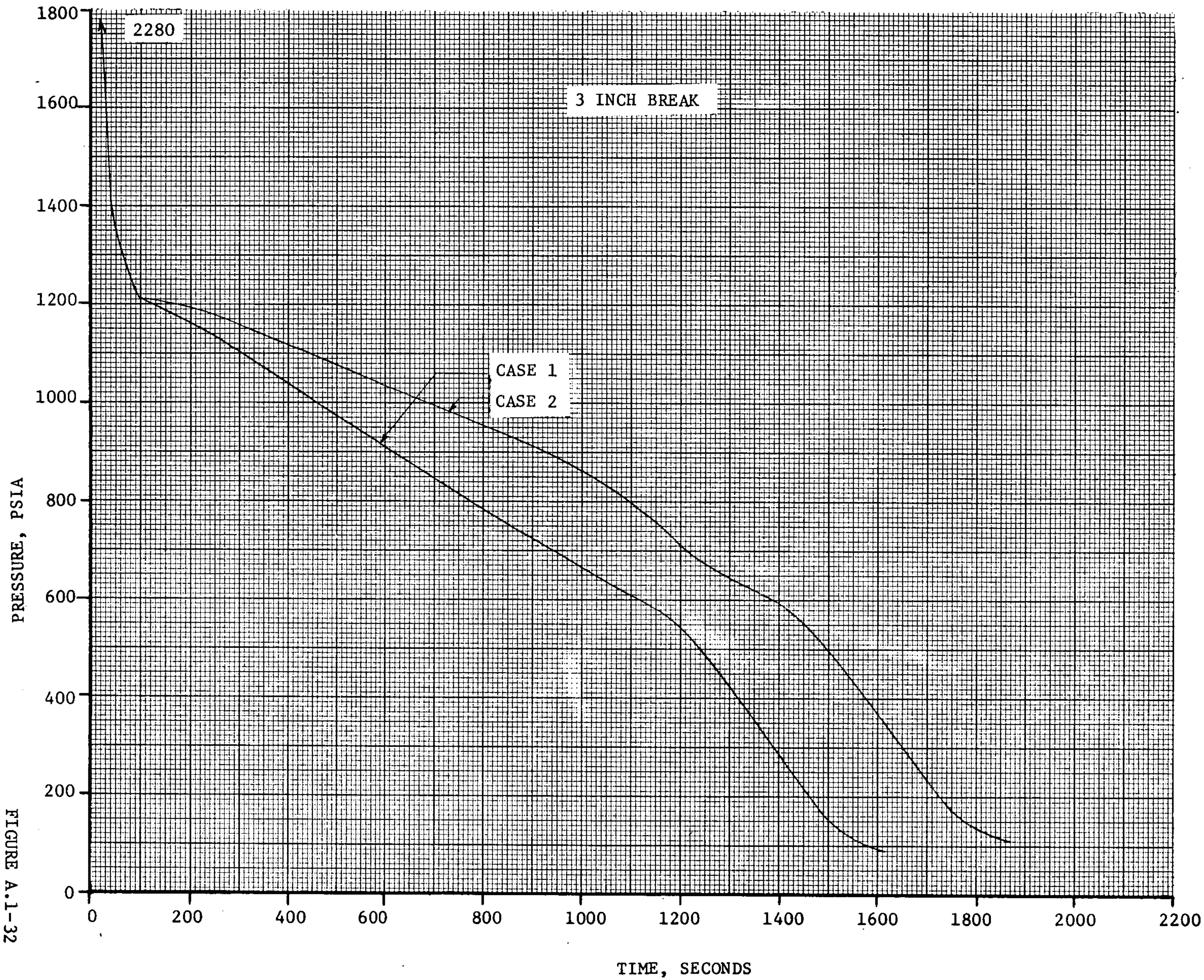

2280
3 INCH BREAK
CASE 1
CASE 2
PRESSURE, PSIA
TIME, SECONDS
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200

FIGURE A.1-32

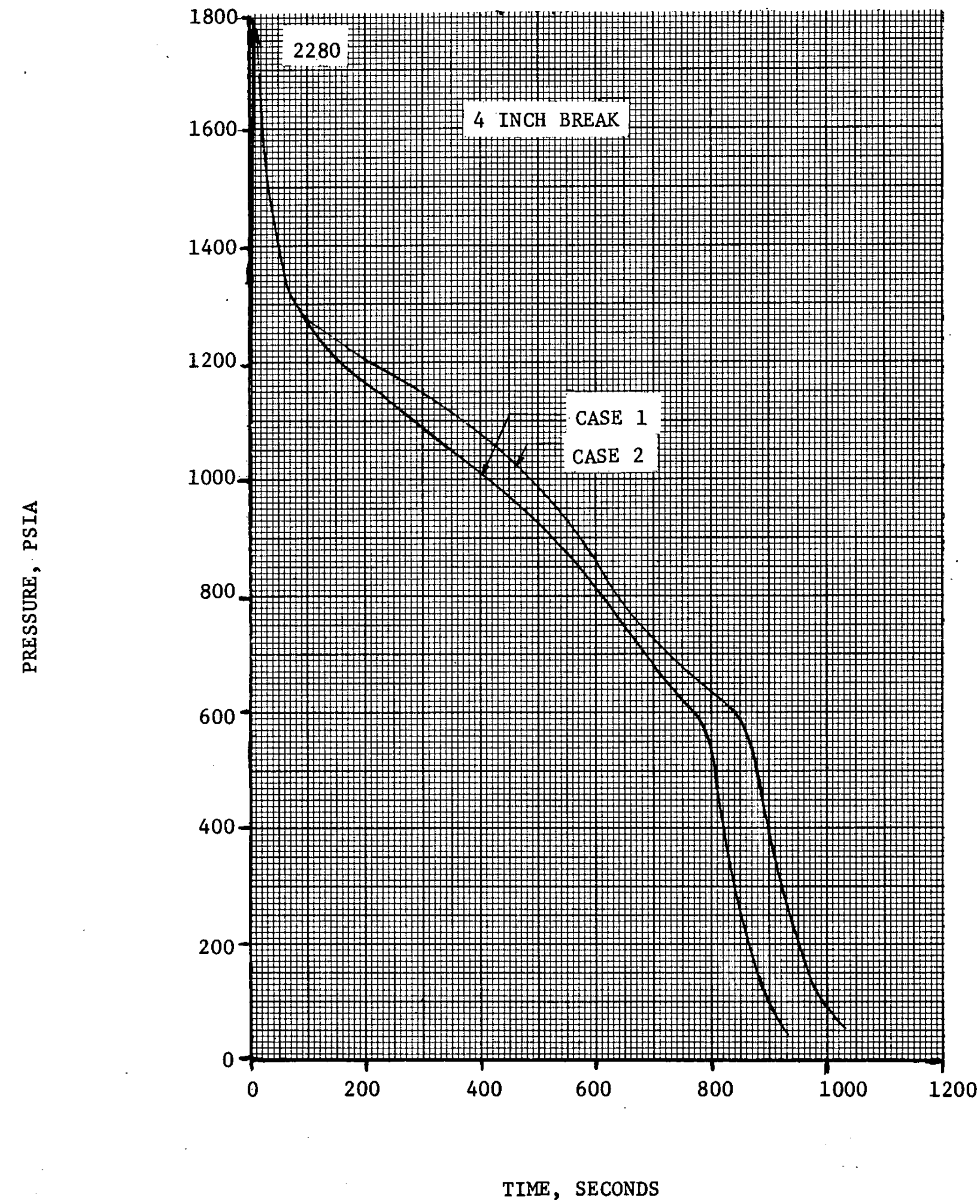

2280
4 INCH BREAK
CASE 1
CASE 2
PRESSURE, PSIA
1800
1600
1400
1200
1000
800
600
400
200
0
200
400
600
800
1000
1200
TIME, SECONDS

FIGURE A.1-33

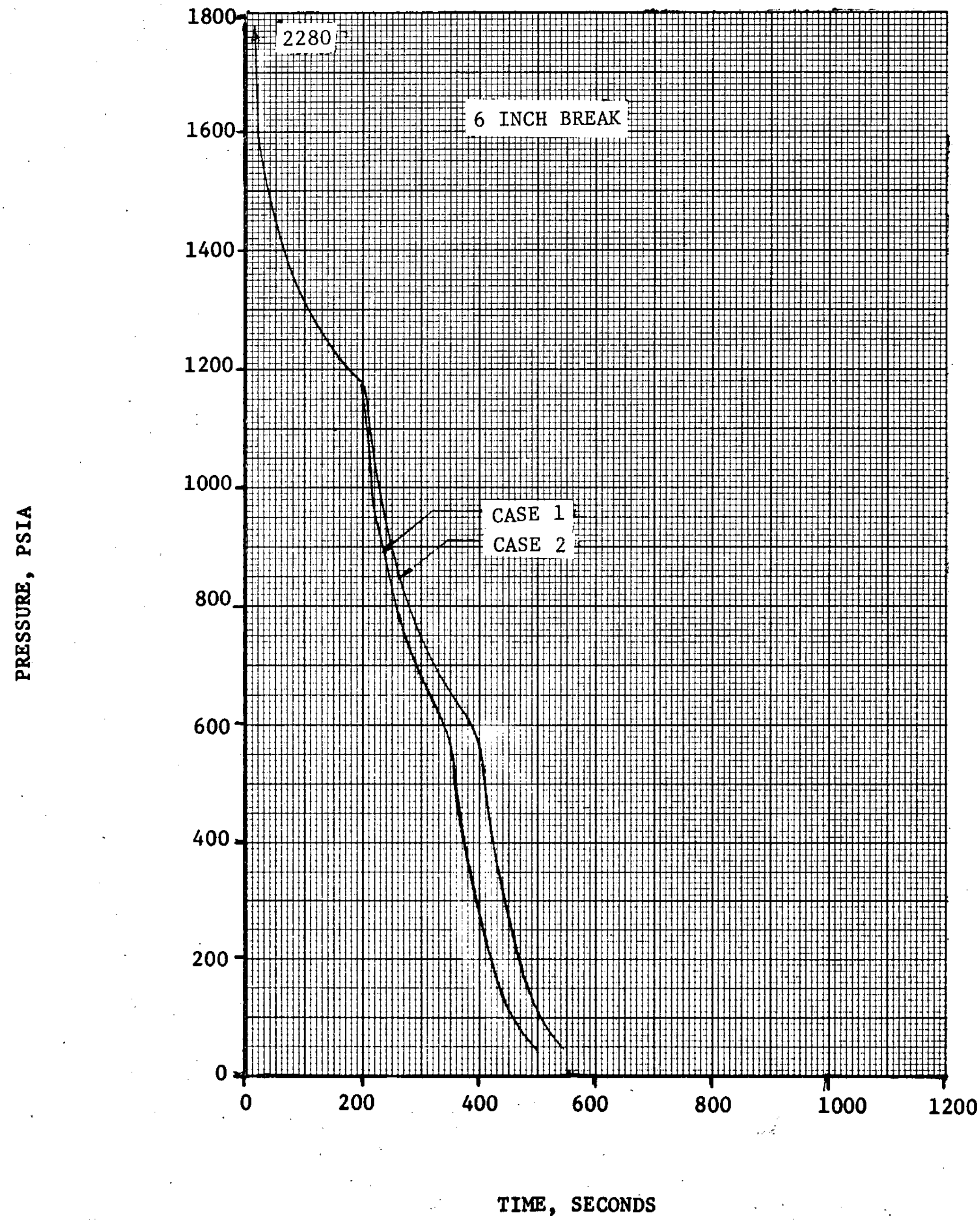

2280
6 INCH BREAK
CASE 1
CASE 2
1800
1600
1400
1200
1000
800
600
400
200
0
PRESSURE, PSIA
0
200
400
600
800
1000
1200
TIME, SECONDS

FIGURE A.1-34

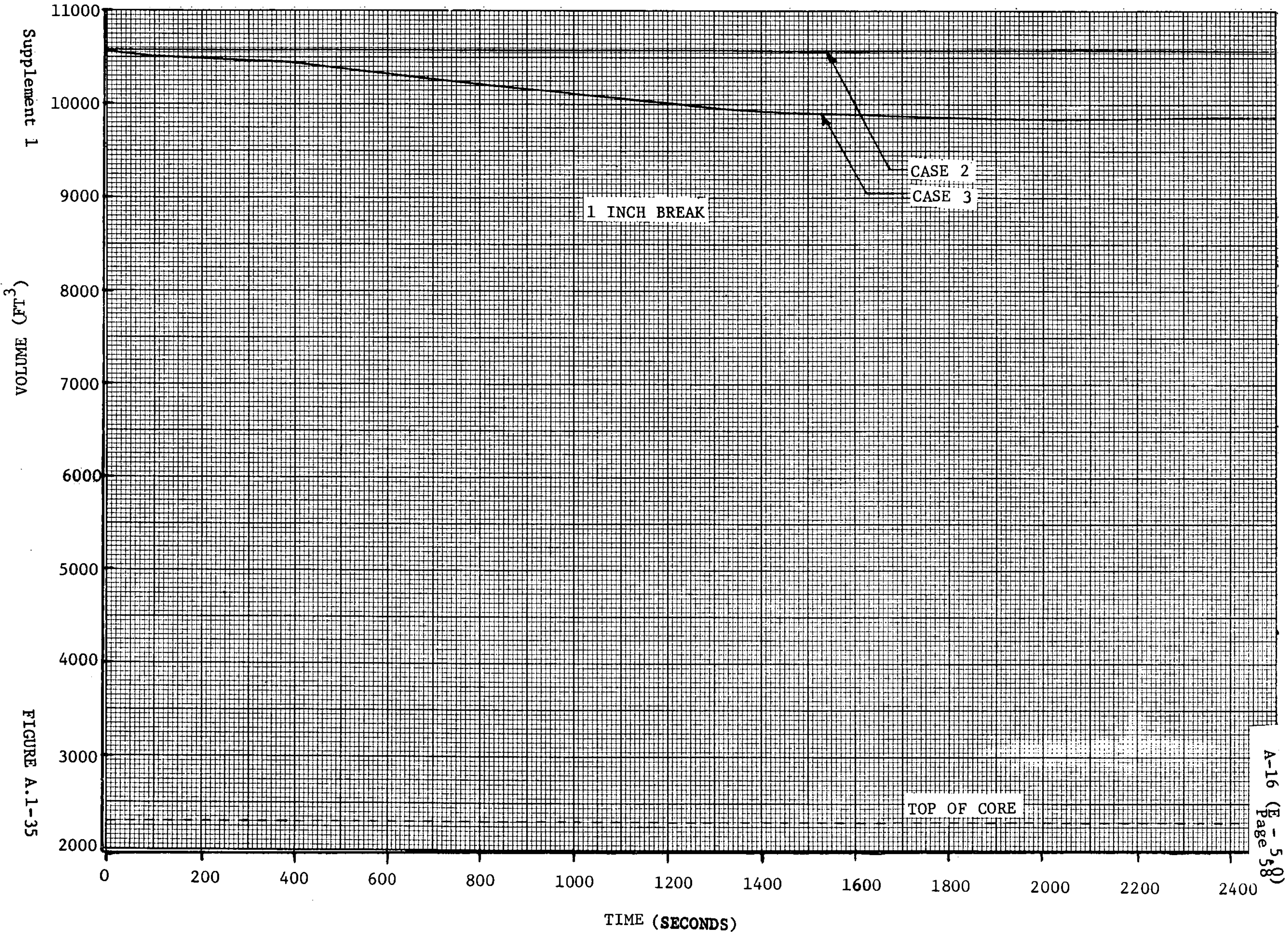

11000
10000
9000
8000
7000
6000
5000
4000
3000
2000
VOLUME (FT^3)
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
TIME (SECONDS)
CASE 2
CASE 3
1 INCH BREAK
TOP OF CORE

FIGURE A.1-35

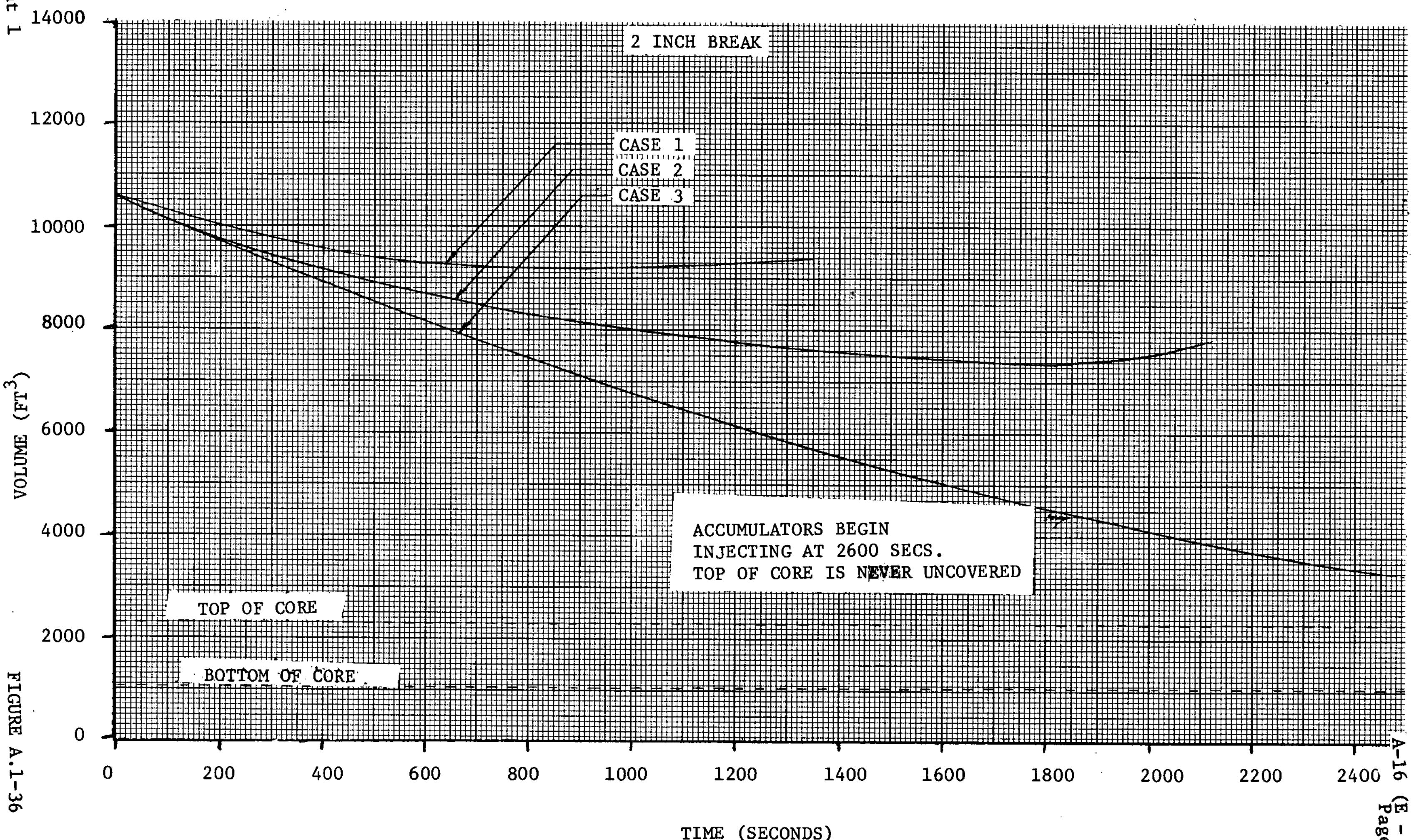

2 INCH BREAK
CASE 1
CASE 2
CASE 3
ACCUMULATORS BEGIN
INJECTING AT 2600 SECS.
TOP OF CORE IS NEVER UNCOVERED
TOP OF CORE
BOTTOM OF CORE
VOLUME (FT^3)
TIME (SECONDS)
0
2000
4000
6000
8000
10000
12000
14000
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400

FIGURE A.1-36

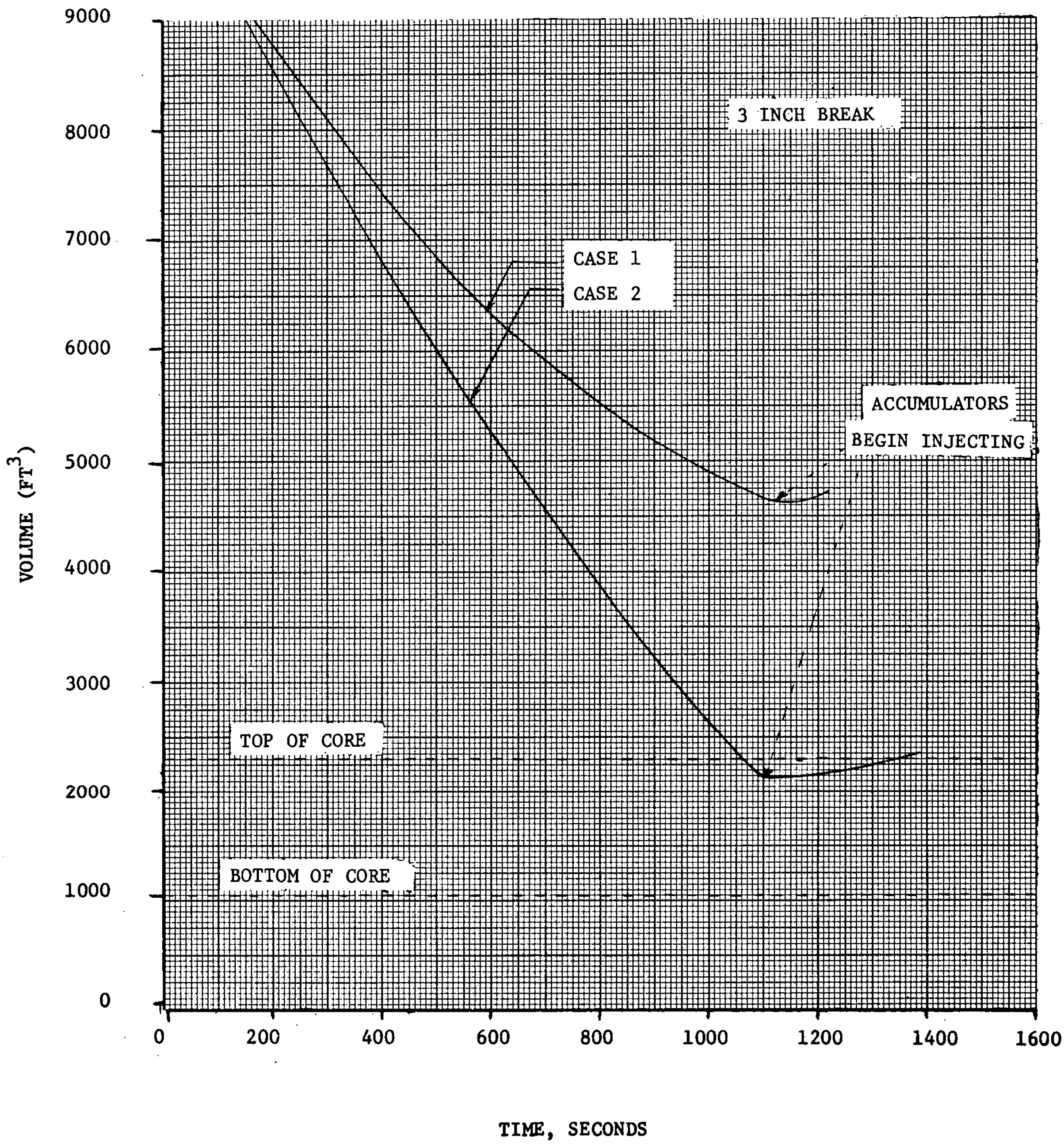

9000
8000
7000
6000
5000
4000
3000
2000
1000
0
VOLUME (FT^3)
0
200
400
600
800
1000
1200
1400
1600
TIME, SECONDS
3 INCH BREAK
CASE 1
CASE 2
ACCUMULATORS
BEGIN INJECTING
TOP OF CORE
BOTTOM OF CORE

FIGURE A.1-37

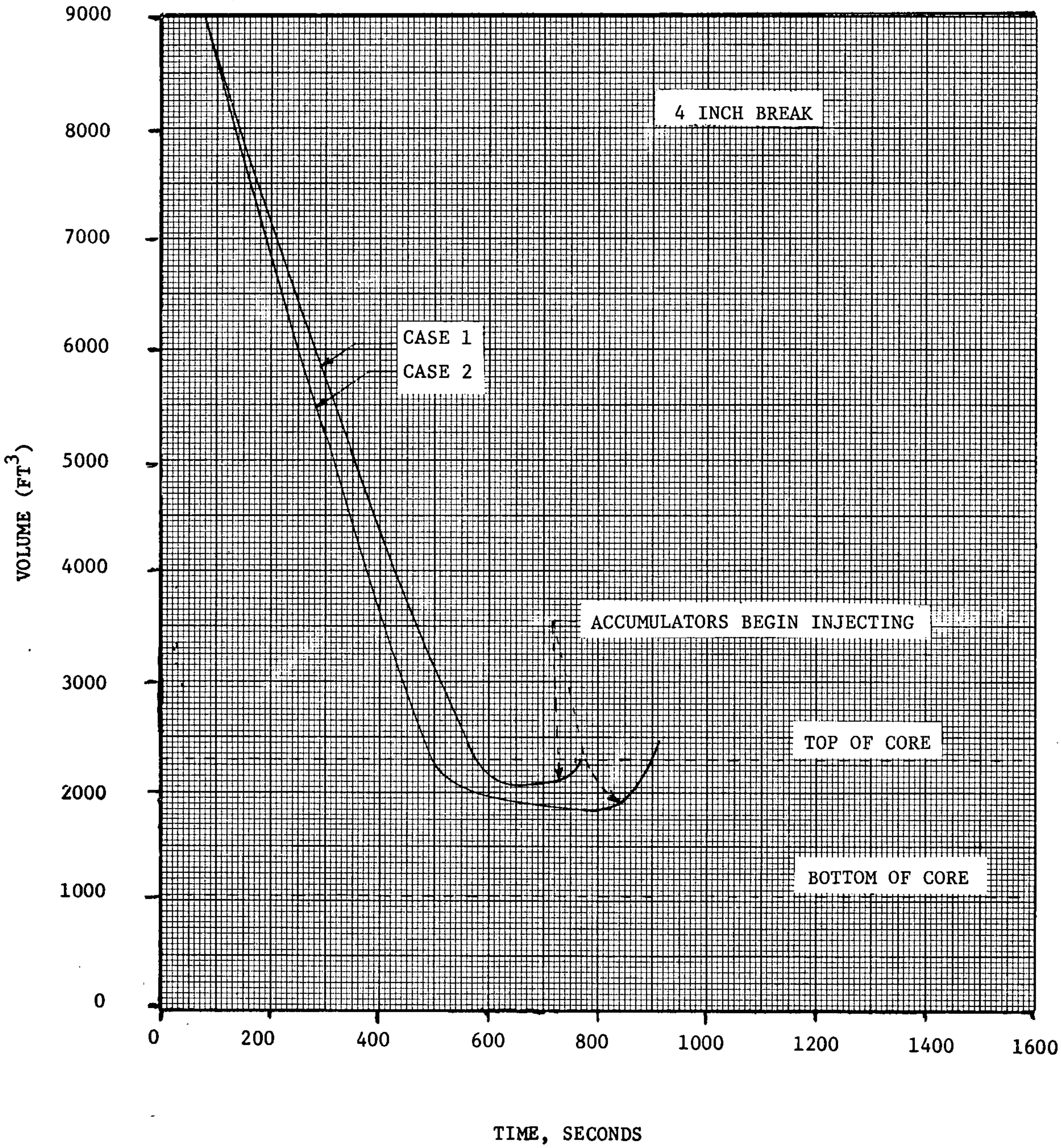

4 INCH BREAK
CASE 1
CASE 2
ACCUMULATORS BEGIN INJECTING
TOP OF CORE
BOTTOM OF CORE
VOLUME (FT^3)
9000
8000
7000
6000
5000
4000
3000
2000
1000
0
200
400
600
800
1000
1200
1400
1600
TIME, SECONDS

FIGURE A.1-38

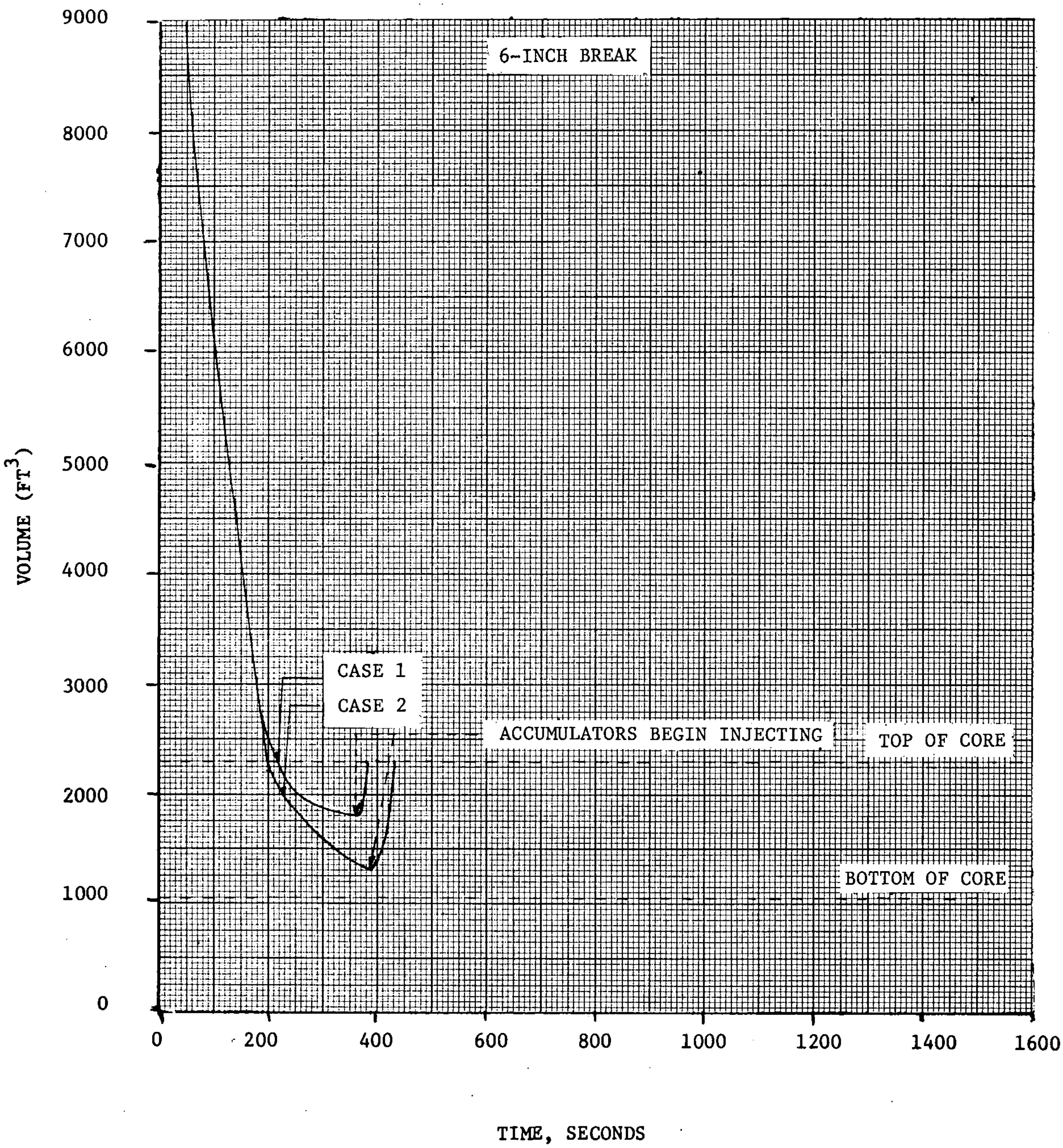

6-INCH BREAK
CASE 1
CASE 2
ACCUMULATORS BEGIN INJECTING
TOP OF CORE
BOTTOM OF CORE
VOLUME (FT^3)
9000
8000
7000
6000
5000
4000
3000
2000
1000
0
0
200
400
600
800
1000
1200
1400
1600
TIME, SECONDS

FIGURE A.1-39

6 INCH BREAK

SAFETY INJECTION
TWO HIGH HEAD PUMPS, 3/4 FLOW

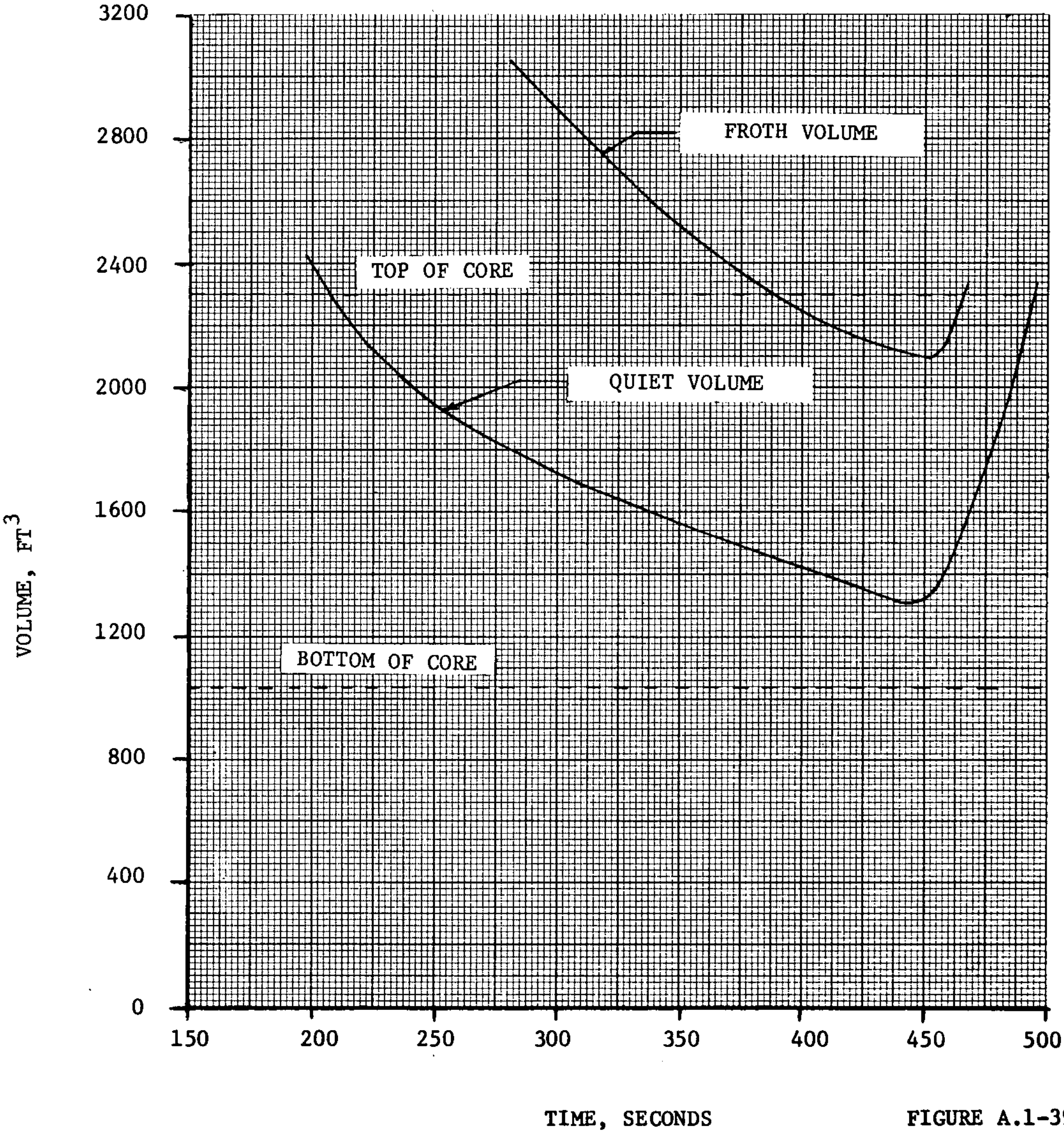


FIGURE A.1-39a

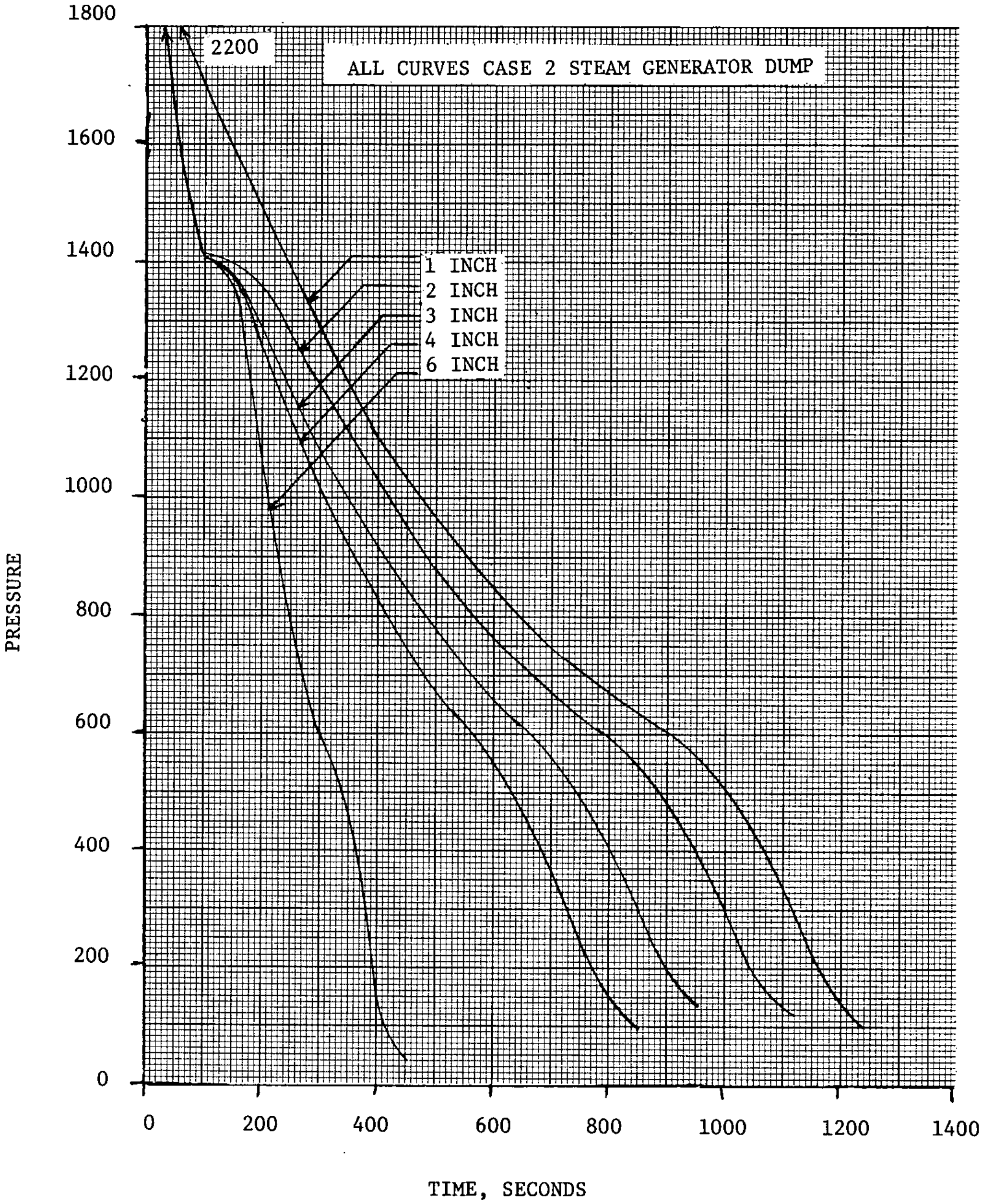

1800
1600
1400
1200
1000
800
600
400
200
0
2200
ALL CURVES CASE 2 STEAM GENERATOR DUMP
1 INCH
2 INCH
3 INCH
4 INCH
6 INCH
PRESSURE
0
200
400
600
800
1000
1200
1400
TIME, SECONDS

FIGURE A.1-40

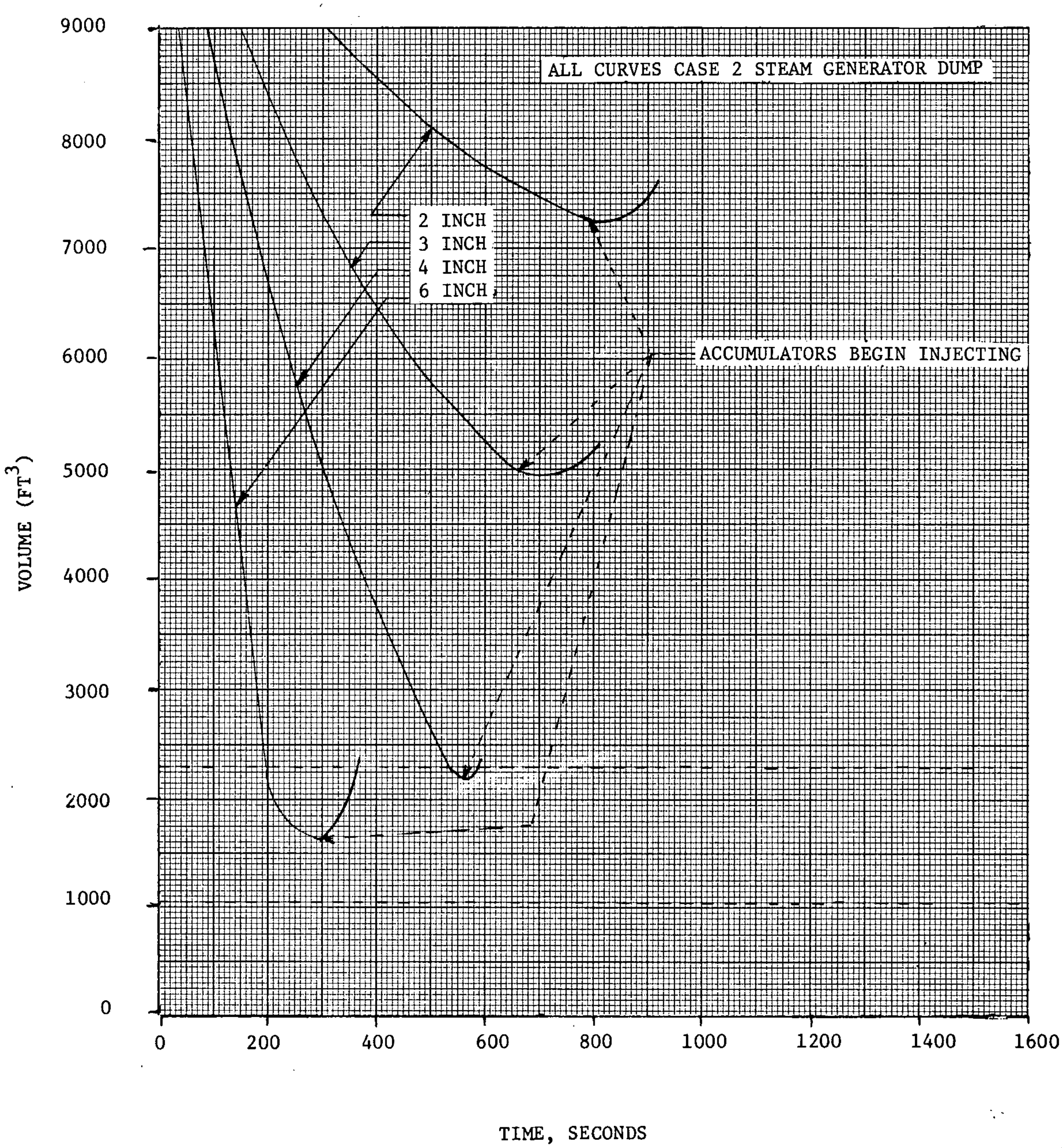

ALL CURVES CASE 2 STEAM GENERATOR DUMP
2 INCH
3 INCH
4 INCH
6 INCH
ACCUMULATORS BEGIN INJECTING
9000
8000
7000
6000
5000
4000
3000
2000
1000
0
VOLUME (FT^3)
0
200
400
600
800
1000
1200
1400
1600
TIME, SECONDS

FIGURE A.1-41

SECTION 2 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 6.3% greater than the application at 3025 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 through, 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 A.2-1 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 for Indian Point #2. 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:

$$\frac{\text{total coolant energy transferred into containment}}{\text{(containment vessel volume)}\ \ \text{(time interval to peak pressure)}}$$

From this the maximum of h for steel is calculated:

$$h_{max} = 75 \left(\frac{E}{t_p V} \right) 0.60$$

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:

$$h_s = h_{max} \sqrt{\frac{t}{t_p}} \quad 0 \leq t \leq t_p \qquad (2)$$

h_s = heat transfer coefficient for steel (Btu/hr-ft^2-^{o}F)

t = time from start of accident (sec)

The exponential decrease of the heat transfer coefficent is given by:

$$h_s = h_{stag} + (h_{max} - h_{stag})\ e^{-.05(t-t_p)} \quad t > t_p \qquad (3)$$

where

$$h_{stag} = 2 + 50x \qquad 0 \leq x \leq 1.4 \qquad (4)$$

h_{stag} = h for stagnant conditions (Btu/hr-ft^2-^{o}F)

χ = 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 now assumes an overall heat transfer rate Q_{tot}. 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 value 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;

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 effects 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 now 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

$$\dot{m} = \bar{h}_D (\rho_{sg} - \rho_{sw}) \tag{1}$$

From ref. (3), pp. 471-473, experimental data for mass and heat transfer will correlate 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

$$q_1 = \frac{\dot{m}\ \lambda h P}{\rho C P_{am} t} (\rho_{sg} - \rho_{sw})$$

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 transfer 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 pressure of non-condensible gases is taken into consideration by virtue of the fact that the theory behind the analyses 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 A.2-1.

Nomenclature

$\dot{m}$ 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 temperature, lbm/ft^3

u_s Free steam gas velocity,ft/min.

Sc Schmidt number, M/pD, 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/pcu_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 the 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:*

$$V^2 = \frac{4Dg\ (\rho - \rho_m)}{3C_D\rho_m} \qquad (1)$$

For the 700 micron drop size expected from the nozzles, the terminal velocity is less than 7 ft/sec. For a 1000 micron drop, the velocity would be less than 10 ft/sec. The Nusselt number for heat transfer, Nu, and the Nusselt number for mass transfer, Nu' (Sherwood Number), can be calculated from the empirical relations given by Ranz and Marshall.[8]

$$\underline{Nu} = 2 + 0.6\ (\underline{Re})^{1/2}\ \ (\underline{Pr})^{1/3} \qquad (2)$$

$$\underline{Nu}' = 2 + 0.6\ (\underline{Re})^{1/2}\ \ (\underline{Sc})^{1/3} \qquad (3)$$

---

* 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 <u>Nu</u> and <u>Nu'</u> respectively. The equations describing the temperature rise of a falling drop are:

$$\frac{d}{dt} (Mu) = mh_g + q \tag{4}$$

$$\frac{d}{dt} (M) = m \tag{5}$$

where

$$q = h_c A (T_s - T) \tag{6}$$

$$m = k_G A (P_s - P_v) \tag{7}$$

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 and Typical Units

A	area	ft^2
C_D	drag coefficient	-
D	droplet diameter	ft
g	acceleration of gravity	$lbm\text{-}ft/lbf\text{-}sec^2$
h_c	coefficient of heat transfer	$B/ft^2\text{-}sec\text{-}F^o$
h_s	steam enthalpy	B/lbm
k_G	coefficient of mass transfer	$lbm/sec\text{-}ft^2$ psi
M	droplet mass	lbm
m	diffusion rate	lbm/sec
$\underline{Nu}$	Nusselt number for heat transfer	-
$\underline{Nu}'$	Nusselt number for mass transfer	-
P_s	steam partial pressure	psia
P_v	droplet vapor pressure	psia
$\underline{Pr}$	Prandtl number	-
q	heat flow rate	B/sec
$\underline{Re}$	Reynolds number	-
$\underline{Sc}$	Schmidt number	-
T	droplet temperature	°F
T_s	steam temperature	°F
t	time	sec
u	droplet enternal energy	B/lbm
V	velocity	ft/sec
ρ	droplet density	lbm/ft^3
ρ_m	steam-air mixture density	lbm/ft^3

Containment Pressure Transients

The containment pressure was calculated for a range of large area ruptures of the Reactor Coolant System. The rupture sizes considered were:

a. Double Ended Rupture
b. 6 ft^2 break
c. 3 ft^2 break
d. .5 ft^2 break

Figure A.2-2 presents the results of the transients. For all cases a pressure peak of less than 40 psig was calculated. Since the design pressure for the IPP #3 plant is 47 psig, a margin more than 17% above the conservative value of the blowdown 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 A.2-2 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 Table A.2-3. 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 reactor vessel during the relatively fast blowdown.

Energy Sink

Figure A.2-3 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, Btu
U_i = Initial external energy in the containment, Btu
$\Sigma\ (mh)_{in}$ = Enthalpy added by blowdown sources, Btu
$\Sigma\ Q_{in}$ = Energy added directly to containment atmosphere by hydrogen-oxygen recombination, Btu
$\Sigma\ Q_{out}$ = Heat removal by containment structure and cooling system, Btu

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)	= 2.43×10^6 Btu
Air (m) (C_v) (T) = (178,205) (0.172) (120)	= 3.67×10^6 Btu
Sump (m) (u) = (12,343) (87.9)	= 1.08×10^6 Btu
	7.18×10^6 Btu

The internal energy balance at the end of blowdown is given in Table A.2-4. 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 A.2-4 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 A.2-5 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:

$$Q_{core} = \frac{h_{fg}}{hg} Q_{Allowed} \quad = \quad 48 \times 10^6 \times \frac{914.7}{1177.9} \quad = 37.2 \times 10^6 \text{ Btu}$$

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 A.2-5 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 x 10^6 Btu of energy after blowdown 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 blowdown would be required to generate 13 x 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 blowdown 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 A.2-5 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 A.2-6 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 blowdown 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 capability 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 A.2-7 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 of 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 A.2-8. 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 A.2-9 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 A.2-10 and A.2-11, 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 A.2-12 and A.2-13 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.

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

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TABLE A.2-1

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	60,000	3/8	511	0.11	26
Containment Dome	Steel lined concrete	28,500	1/2	511	0.11	26
Containment Floor	Unlined concrete	15,000	12	150	0.186	0.08
Refueling Canal	Lined concrete	10,000	3/8	511	0.11	26
Misc. Concrete Structure	Unlined Concrete	57,000	12	150	0.186	0.08
Misc. Steel Structure	Steel					
a. Thin		13,000	1/8	511	0.11	26
b. Thick		12,000	1/2	511	0.11	26

TABLE A.2-2

ENERGY SOURCES

1.	Reactor Coolant System Internal Energy	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 Metal (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^2	387.2×10^6
	c. 3 ft^2	389.8×10^6
	d. .5 ft^2	403.8×10^6

TABLE A.2-3

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 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 A.2-4

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

Supplement 1

FAN COOLER HEAT REMOVAL AS A FUNCTION OF CONTAINMENT PRESSURE

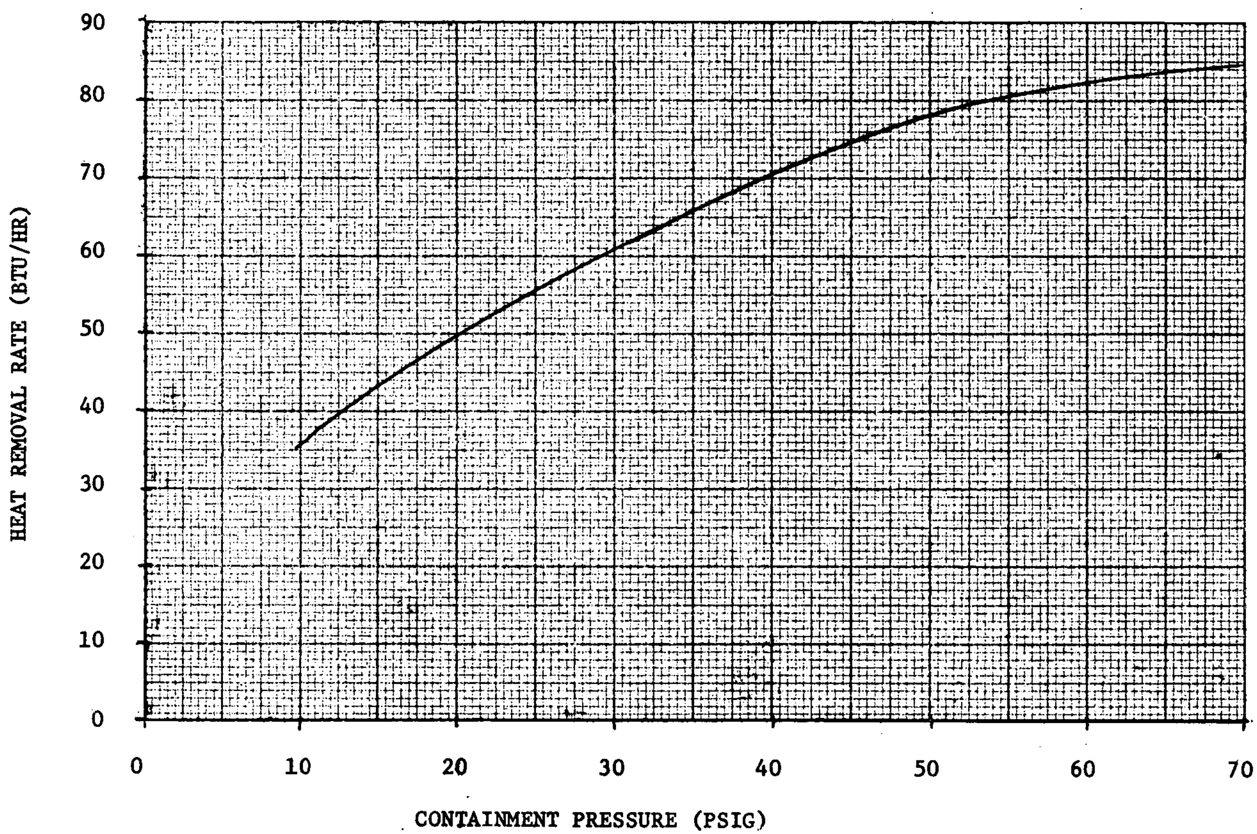

FIGURE A.2-1

CONTAINMENT PRESSURE TRANSIENT

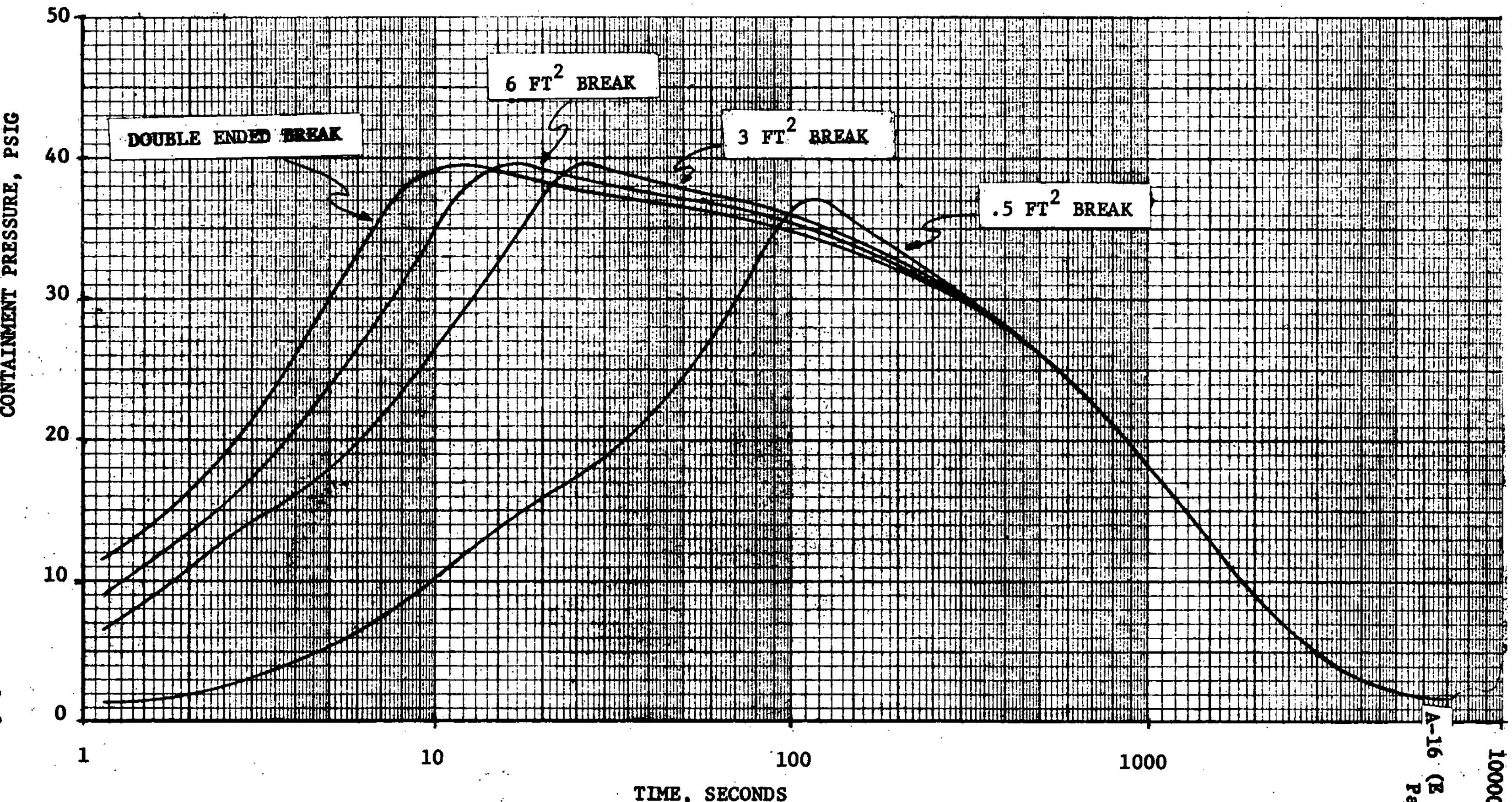

FIGURE A.2-2

CONTAINMENT CAPABILITY STUDY

CONTAINMENT PRESSURE VS STEAM-AIR INTERNAL ENERGY

VOLUME: 2.6×10^6 FT^3

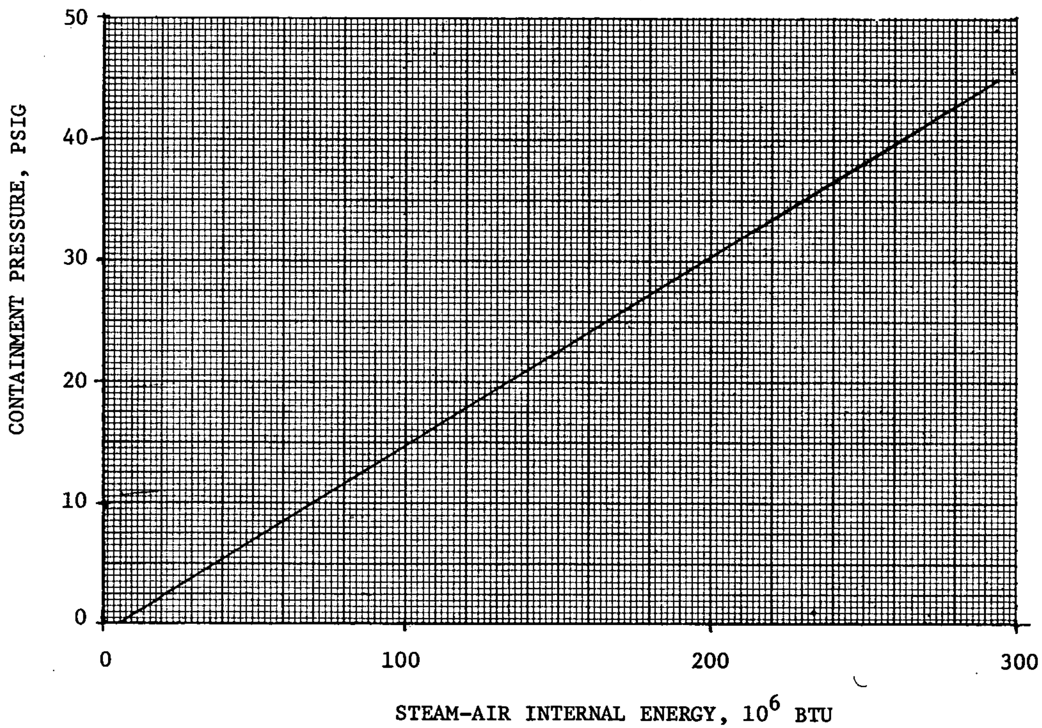

FIGURE A.2-3

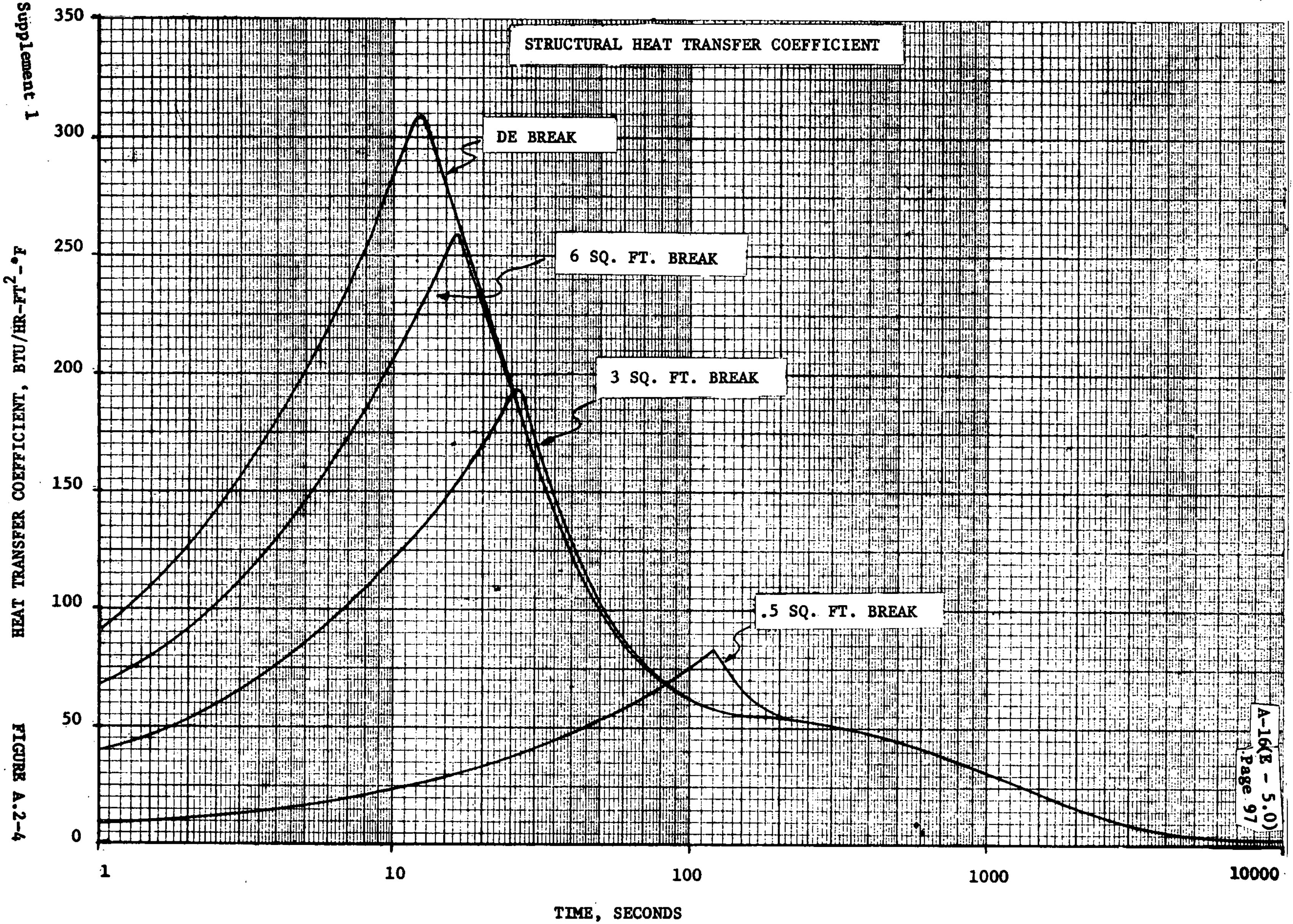
STRUCTURAL HEAT TRANSFER COEFFICIENT
DE BREAK
6 SQ. FT. BREAK
3 SQ. FT. BREAK
.5 SQ. FT. BREAK
HEAT TRANSFER COEFFICIENT, BTU/HR-FT^2-°F
350
300
250
200
150
100
50
0
1
10
100
1000
10000
TIME, SECONDS

FIGURE A.2-4

CONTAINMENT CAPABILITY STUDY

CONTAINMENT PRESSURE VS STEAM-AIR INTERNAL ENERGY

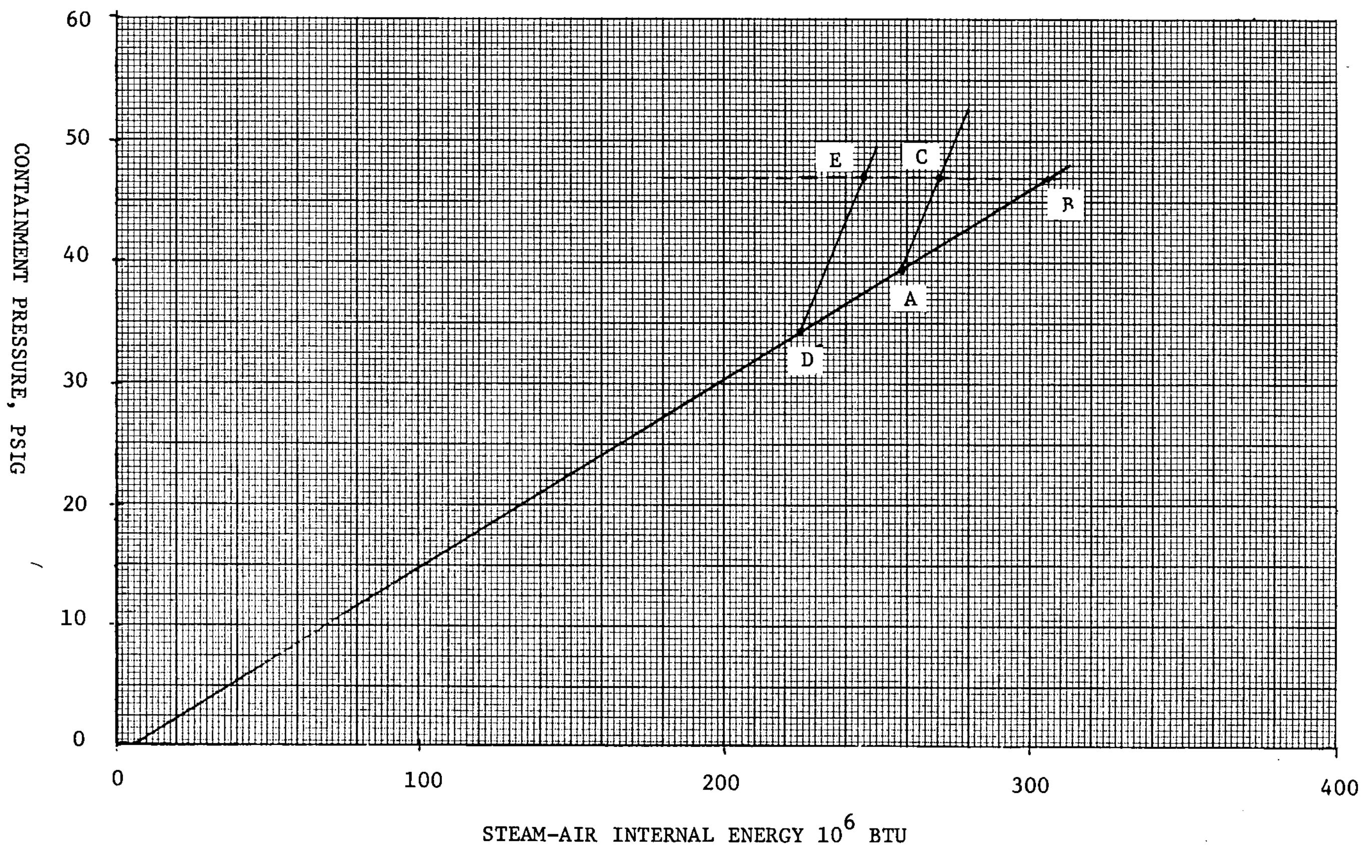

FIGURE A.2-5

CONTAINMENT CAPABILITY STUDY
ALL AVAILABLE ENERGIES

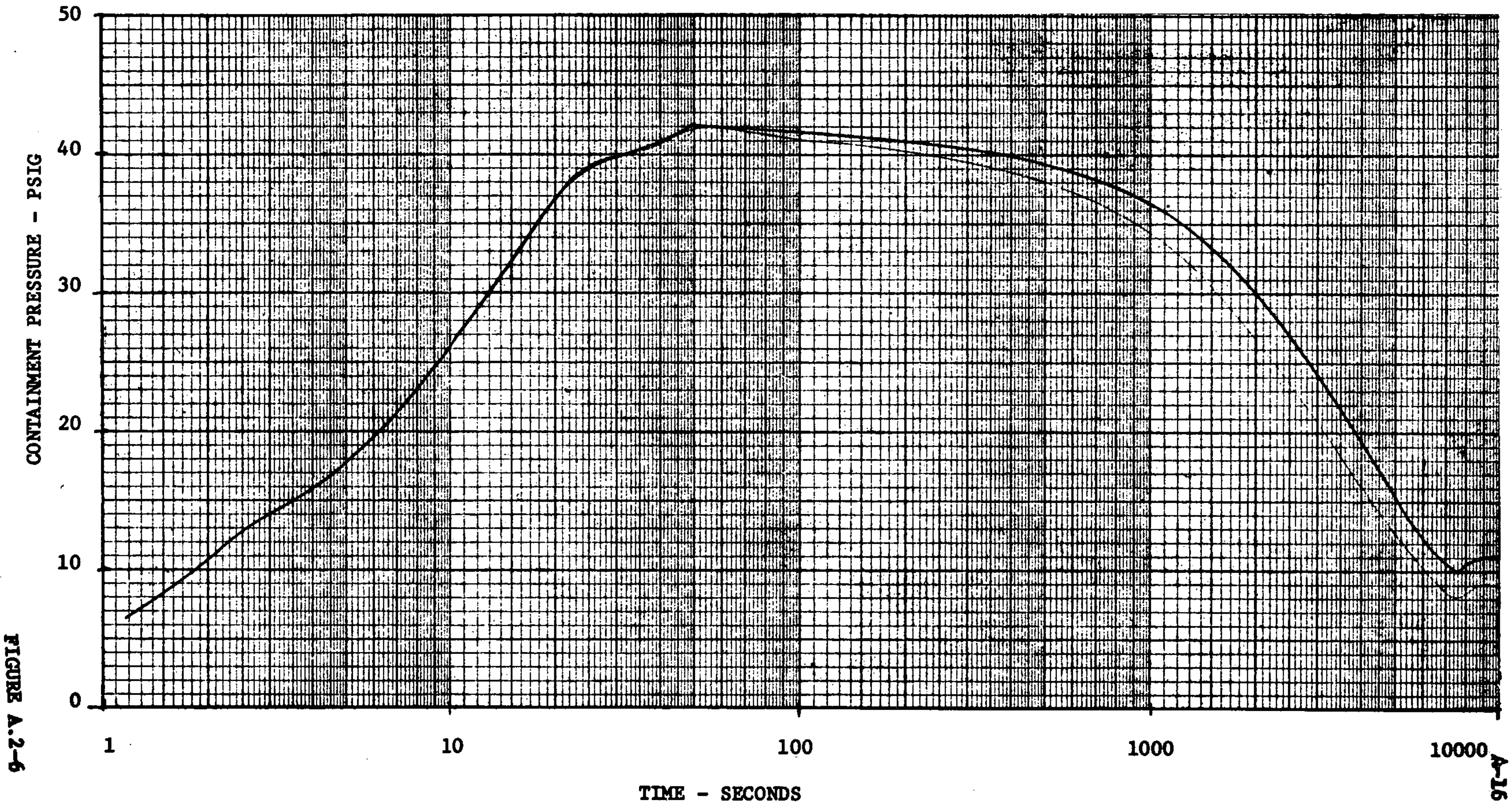

FIGURE A.2-6

CONTAINMENT CAPABILITY STUDY

ZR-H_2O REACTION - (33%)

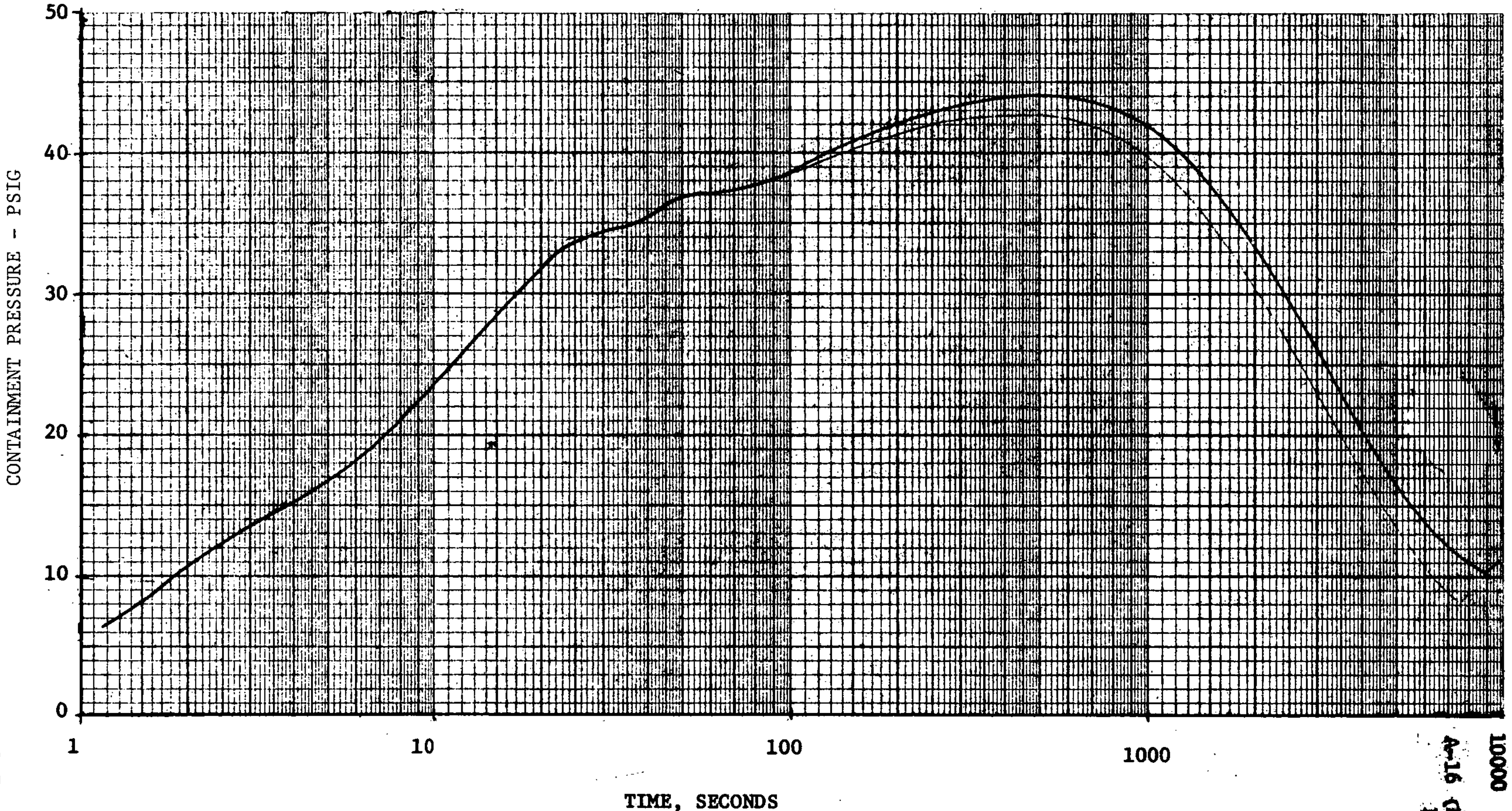

FIGURE A.2-7

CONTAINMENT PRESSURE TRANSIENT

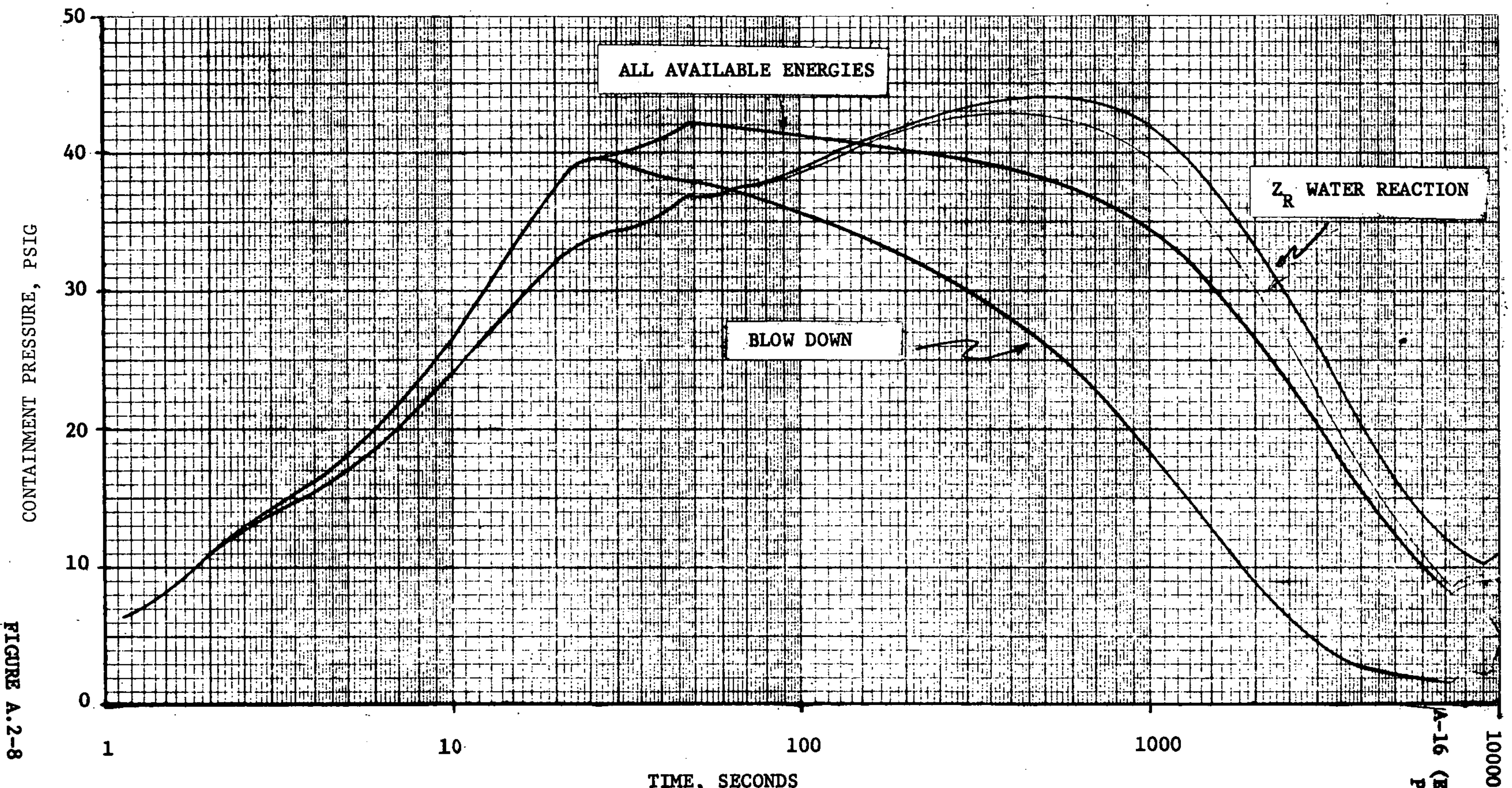

FIGURE A.2-8

CONTAINMENT CAPABILITY STUDY
RATE OF ENERGY ADDITION

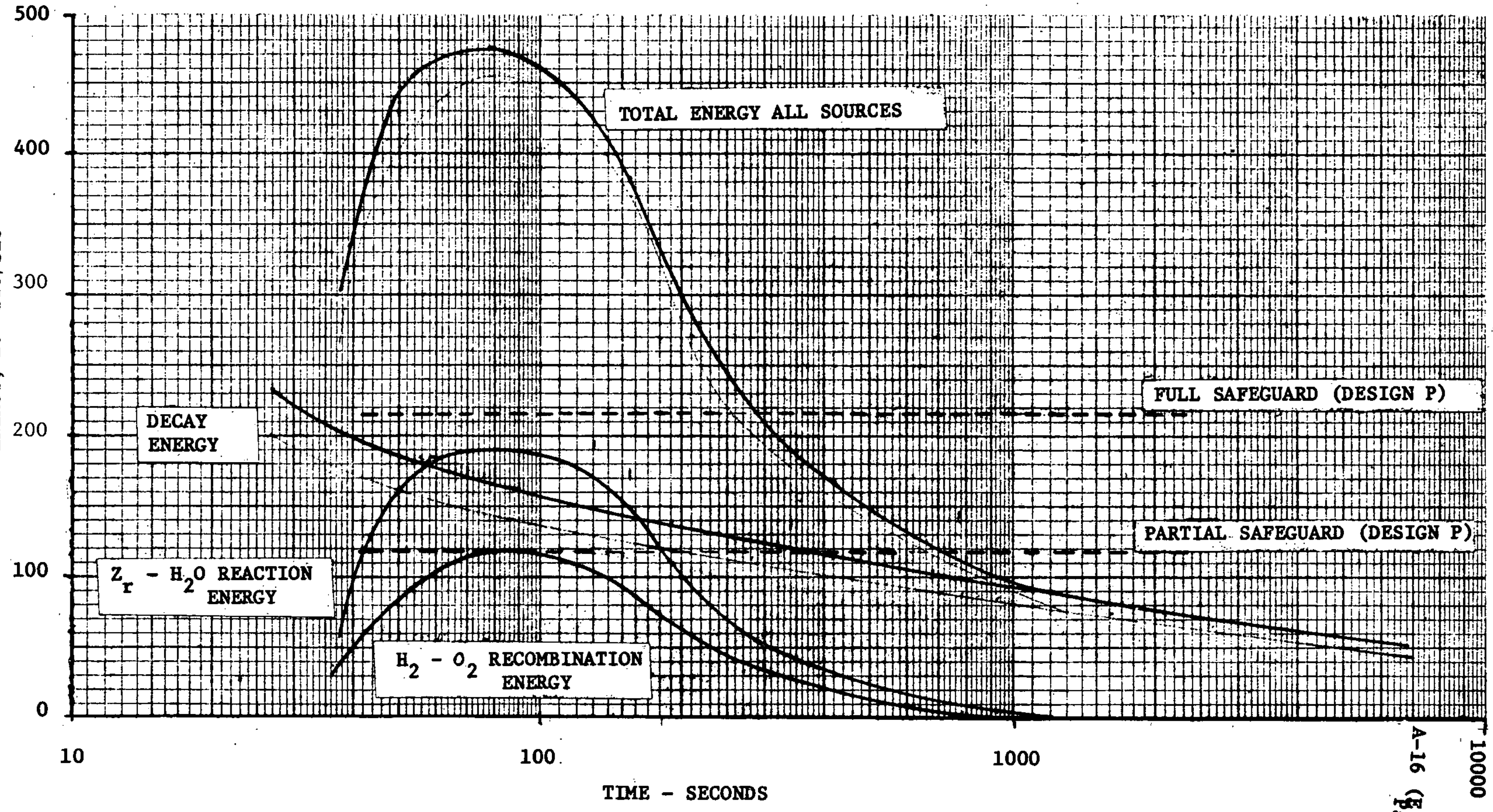

FIGURE A.2-9

CONTAINMENT CAPACITY

CASE 1

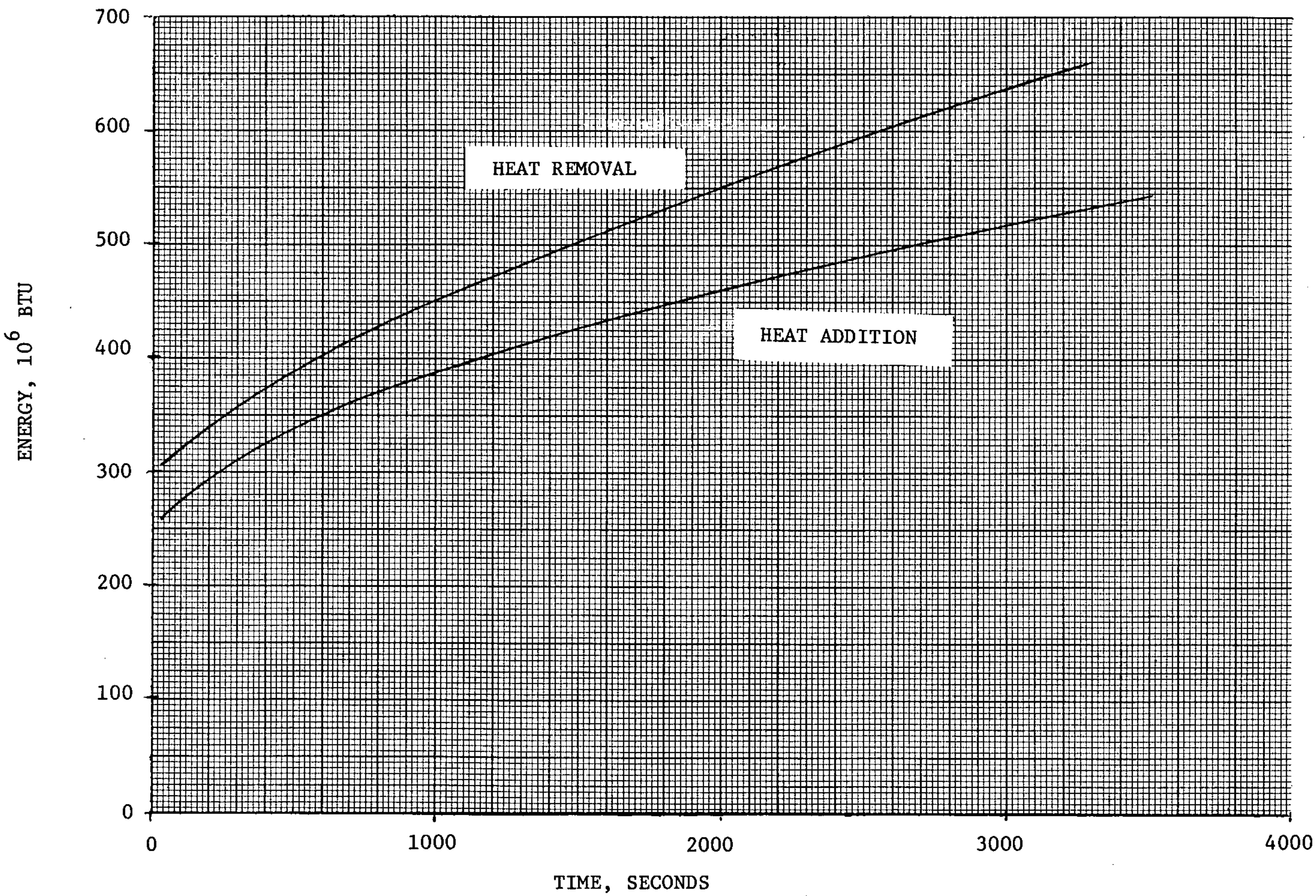

FIGURE A.2-10

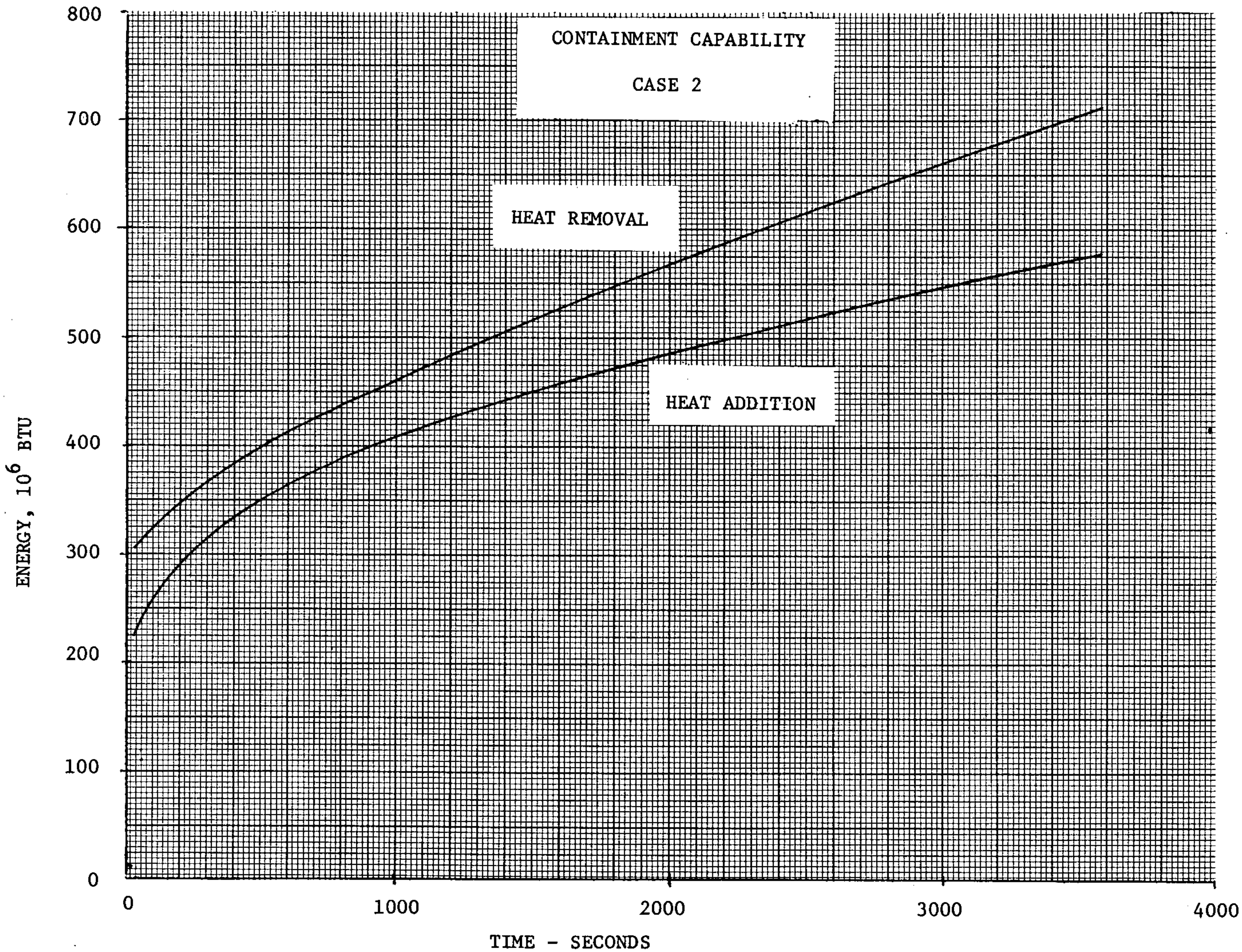
CONTAINMENT CAPABILITY
CASE 2
HEAT REMOVAL
HEAT ADDITION
ENERGY, 10^6 BTU
800
700
600
500
400
300
200
100
0
0
1000
2000
3000
4000
TIME - SECONDS

FIGURE A.2-11

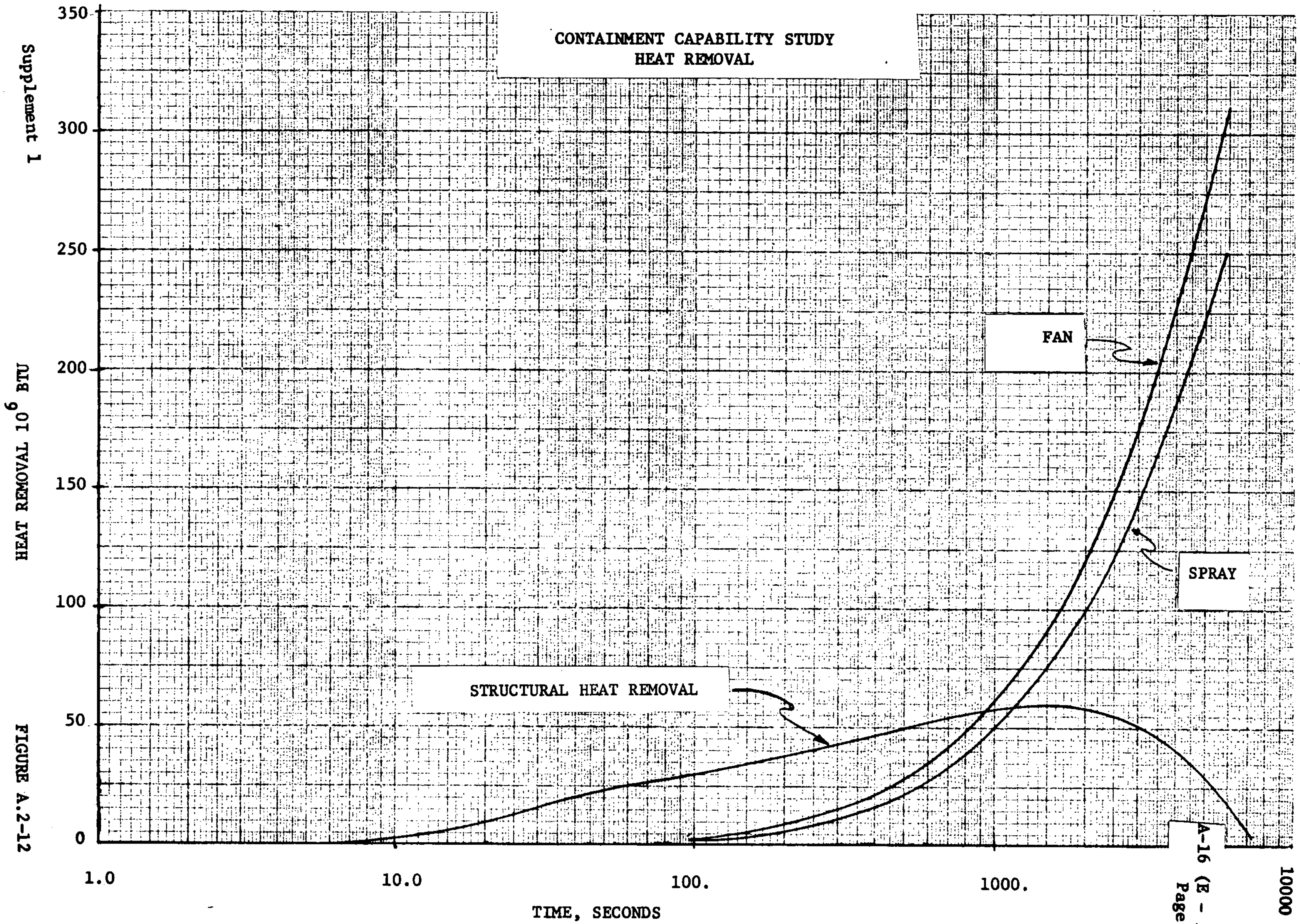
CONTAINMENT CAPABILITY STUDY
HEAT REMOVAL
HEAT REMOVAL 10^6 BTU
350
300
250
200
150
100
50
0
FAN
SPRAY
STRUCTURAL HEAT REMOVAL
1.0
10.0
100.
1000.
10000
TIME, SECONDS

FIGURE A.2-12

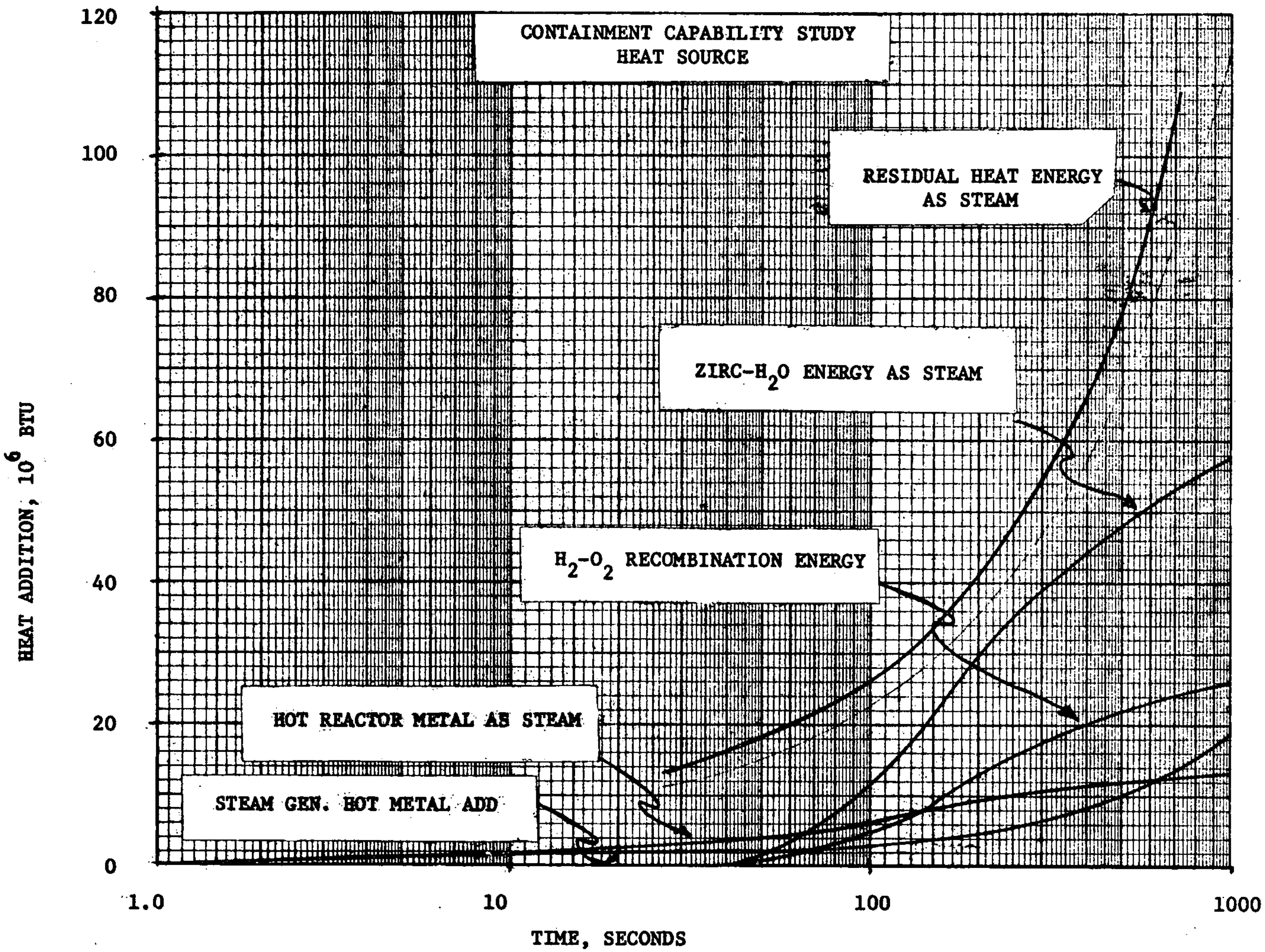
CONTAINMENT CAPABILITY STUDY
HEAT SOURCE
RESIDUAL HEAT ENERGY
AS STEAM
ZIRC-H_2O ENERGY AS STEAM
H_2-O_2 RECOMBINATION ENERGY
HOT REACTOR METAL AS STEAM
STEAM GEN. HOT METAL ADD
HEAT ADDITION, 10^6 BTU
120
100
80
60
40
20
0
1.0
10
100
1000
TIME, SECONDS

FIGURE A.2-13

ITEM 16 (Attachment E Question 5.1)

Discuss the effect of the loss-of-coolant blowdown on steam generator tube and tube sheet integrity and state the maximum leakage rate from secondary-to-primary which could occur following blowdown. Indicate the stresses experienced due to the differential pressure following blowdown and the thermal stresses resulting from blowdown. Consider the plant to be approaching the end of life with the maximum primary-to-secondary leakage with which it would be operated.

ANSWER

In the case of a primary pressure loss accident, the secondary-to-primary pressure differential can reach 1100 psig. This pressure differential is less than the primary-secondary design pressure differential (1520 psi) for normal operating conditions. Hence, no stresses in excess of those covered in Section III rules for normal operation are experienced on the tube sheet for this accident case. For the tubes, actual pressure tests of 3/4 in. O.D./.058 inch wall Inconel tubing show collapse under external pressure of 5700-5900 psi. Extrapolating this data to 7/8 in. O.D./0.50 inch wall tubes, collapse would occur at about 2630 psi at 650°F. This gives a factor of safety of 2.4 against collapse under the 1100 psig accidental application of external pressure to tubes. A check of the ASME Section VIII design curves for Iron-Chromium-Nickel Steel cylinders under external pressure indicates a predicted collapse pressure for the tubes of 2310 psi, which is close to the extrapolated value for the experimental results.

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

Thus, a loss-of-coolant accident will not cause rupture of either the tubes or tube sheet, and no primary to secondary leakage will occur as a result of the accident.

The maximum leakage rate from the Reactor Coolant System to other sources such as the containment or a steam generator allowable during normal operation will be determined during the final design stage.

ITEM 16 (Attachment E Question 5.2)

Provide plots of water volume in the pressure vessel, clad hot spot temperature, per cent metal-water reaction, and per cent of clad experiencing perforations vs. time for cold leg and hot leg breaks. Assume break areas of 0.005, 0.1, 0.5, and 3 square feet and a double-ended break of the main coolant line. Assume that (1) the emergency core cooling system does not function, (2) minimum engineered safety features operate with two accumulators discharging to the vessel, (3) minimum engineered safety features operate with three accumulators discharging to the vessel, and (4) the system functions at 100% of capacity. For reference, include the adiabatic (from time of rupture) clad temperature transient on the temperature plots.

ANSWER

The analysis performed in answer to this question is presented in Appendix 16 (E-5.0), Section 1. In this section is presented the water volume in the pressure vessel and clad hot spot temperature transients along with the total number of clad perforations expected. The zirconium-metal water reaction was computed to be less than 1% for all design break sizes. In addition to the above information being presented for the large break sizes, a clad temperature study was also conducted for the 6" diameter hole ($\sim 0.2\ ft^2$). For smaller breaks (1", 2", 3" and 4") it is shown that the clad hot spot is covered for the entire transient, and temperatures less than that evaluated for the 6" break are expected. Water volume and pressure curves are presented for these small breaks.

ITEM 16 (Attachment E Question 5.3)

Provide a plot of time to recover the lower half of the core vs. break area assuming minimum engineered safety features and time to reach 1% clad metal-water reaction and time to reach incipient clad melt vs. break area assuming the emergency core cooling system does not function. Consider a range of break areas from 0.005 square feet to that equivalent to a double-ended cold leg rupture.

ANSWER

An analysis was performed for the Unit #3 reactor to determine the margin of time available between the occurrence of incipient clad melting when it is assumed that the core cooling system does not function and the occurrence of effective core cooling when the emergency core cooling system does operate properly. Adiabatic heat transfer was assumed to take place after completion of blowdown for the case of no safety injection system operating. The results of the analysis are presented in the attached Figure 1 of cold leg break area versus time to incipient clad melting. The margin of time available to reach 1% clad metal-water reaction on a core-wide basis would be greater than that shown for incipient clad melting to occur.

COLD LEG BREAK AREA VS TIME TO INCIPIENT CLAD MELTING

ASSUMING AN ADIABATIC FUEL ROD AFTER BLOWDOWN

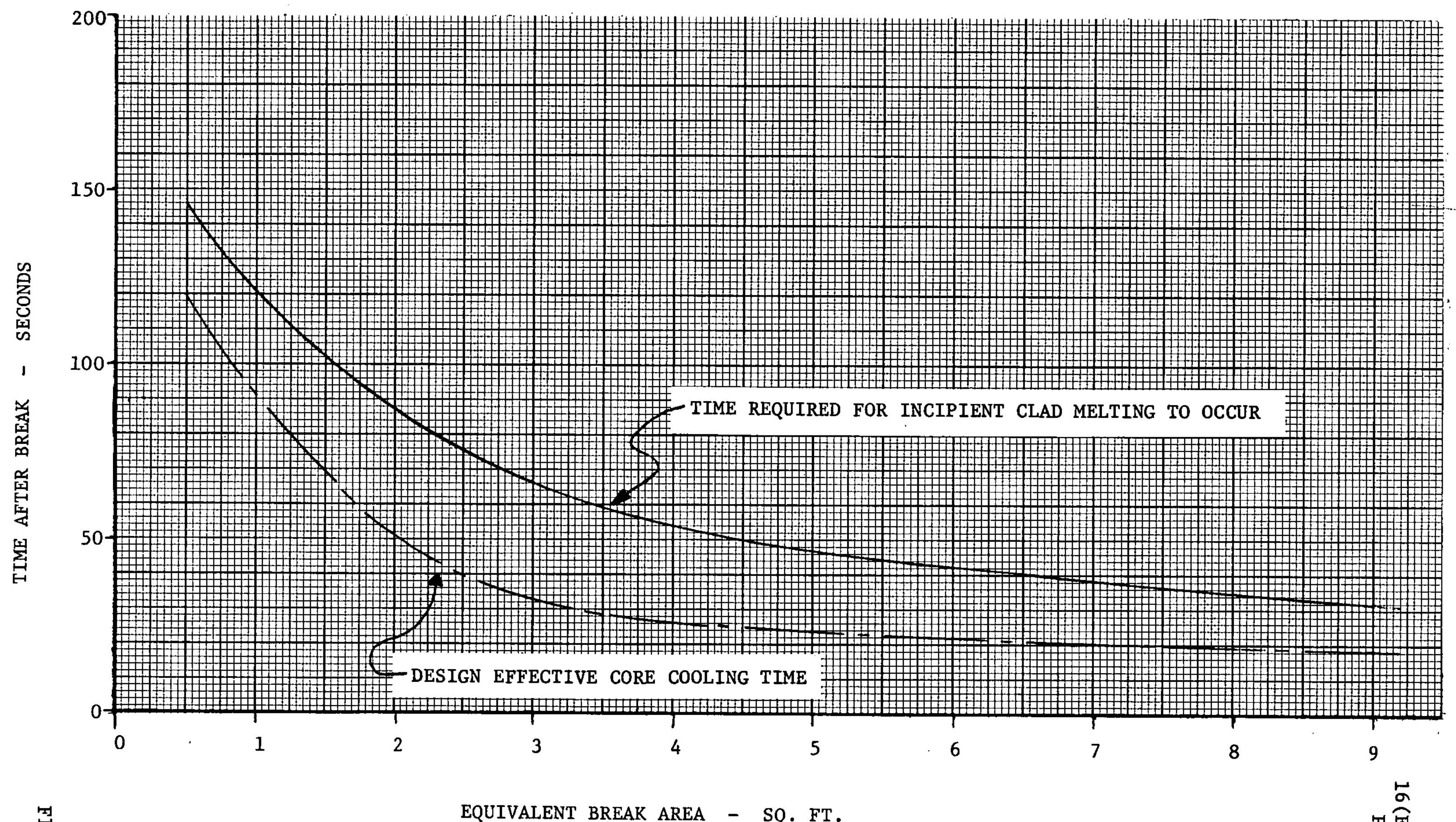

FIGURE 1

SUPPLEMENT 1

ITEM 16 (Attachment E Question 5.4)

On one graph for each break size, plot hot-spot heat transfer coefficient, average vessel pressure, core flow rate, and energy input to the hot element vs. time. Provide this information on hot and cold leg breaks of 0.005, 0.1, 0.5, and 3 square feet and also for double-ended ruptures of the main coolant line.

ANSWER

The answer to this question is presented in Section 1 of Appendix 16(E-5.0).

ITEM 16 (Attachment E Question 5.5)

It appears that if a steam bubble were formed in the pressure vessel head following a loss-of-coolant accident due to establishment of a water seal between the main coolant pump and the steam generator, sufficient back-pressure might result to limit the amount of cooling water entering the core and thus delay core cooldown. Describe the method whereby this is considered. Present a curve of maximum clad temperature vs. time assuming (1) two accumulators discharge to the reactor vessel, and (2) three accumulators discharge to the vessel. Continue these curves until the peak clad temperature is below the saturation temperature associated with containment pressure. In deriving these curves, assume minimum engineered safety features are operable.

ANSWER

The steam bubble question is discussed in Section 1 of Appendix 16(E-5.0). The clad temperature transients presented in this section for the condition of either two or three accumulators operating are shown only until the clad temperature transient is turned around and the temperature falls below 1800°F. At this time no more energy due to the metal-water reaction is being generated. The clad temperature transient at this point in time is under control and is decreasing in an orderly manner.

Item 16 (Attachment E Question 6.1)

LOSS-OF-COOLANT ACCIDENT CONTAINMENT PRESSURE RESPONSE

For the analysis performed of the pressure response of the containment to a loss-of-coolant accident during blowdown, provide the following:

a. Assumptions of flow conditions during both subcooled and two-phase blowdown.

b. Basis for the flow coefficients used.

c. Basis for the heat transfer coefficients employed in transferrring core sensible heat during blowdown.

d. Determination and method of addition of energy generated by fission within the core and decay.

e. Assumption regarding relative humidity and basis for the value employed.

ANSWER

Loss-of-Coolant Accident Containment Pressure Response. The Containment Integrity Evaluation is presented to provide additional information of the response of the containment following a loss-of-coolant accident. This information is contained in Appendix 16 (E-5.0), Section 2. The first part of the evaluation contains the method of analysis which describes the codes used to compute the containment transient, the principal assumptions of the calculation, and a justification of the heat removal capacity of the fan cooler and spray systems. The results of the analysis for several sizes of loss-of-coolant accident is presented. Particular emphasis is given the energy sources and sinks in the study, as well as a quantitative discussion of the margins available. Finally, the capability of the containment to withstand additional sources of energy is presented.

The type of analysis presented in the evaluation is intended to supersede the transients presented in Supplement #1 to the IPP #2 PSAR (Questions 4(a)-4(f)). (Note however, that the answer to Questions 4 (a)-4 (f) still apply to IPP #3). It is felt that the current analysis provides a more meaningful method to evaluate the containment capability.

a. The flow conditions during blowdown are calculated by the FLASH Code. Flow through a leak is calculated in subcooled conditions, using FAUSKE'S model for metastable flow for short pipes or MOODY's model for homogeneous equilibrium flow for long pipes. Once the leaking region reaches saturation, MOODY's correlation is used for both cases.

b. The flow coefficient is conservatively set equal to 1.0. No credit is taken for the sharp edge orifice resulting from pipe splits or breaks of connecting pipes. This maximizes the flow through the break and into the containment. Comparison of the FLASH blowdown results with the LOFT semi mode blowdown test verifies that FLASH overpredicts the mass flow during blowdown.

c. To conservatively account for the core sensible energy transferred to the coolant and ultimately to the containment during blowdown, it is assumed that DNB does not occur. The heat transfer coefficients are determined by the correlations of Lottes-Latz Dithes-Boelter during forced convection and Jenns during nucleate boiling. The sensitivity to this assumption will be discussed in answer to question 16 (E-6.4-a).

d. All of the energy generated by fission within the core and decay during blowdown is assumed to be transferred to the coolant.

e. The relative humidity assumed to be inside the containment, at the beginning of the blowdown is the same as existing during normal operation, i.e., about 35 percent. The value employed is 2260 lbs. of steam, which corresponds to:

(i) Partial Pressure of Steam: $P_{PSTM} = \frac{RTM}{144V} = 0.595$ PSIA.

(ii) Relative Humidity: $RH = \frac{0.595}{1.692} = 35\%$ at 120°F

Item 16 (Attachment E Question 6.2)

For the period after the initiation of blowdown, provide the following:

a. A listing of the effective surface areas and thickness of the static sinks and a justification for the heat transfer coefficients applied.

b. A description of the thermal resistance assumed between the liner and containment concrete and a justification of the amount of heat assumed to be absorbed by the concrete during blowdown.

c. A discussion of the adequacy of the heat transfer coefficients assumed for transfer of heat from hot metal.

ANSWER

a. A listing of the surface areas of the static heat sinks is contained in Table A.2-1 of the above Containment Integrity Evaluation, and the justification for the heat transfer coefficient is on page 69 of Appendix 16 (E-5.0). The sensitivity to use of these assumptions is discussed in the answer to 16 (E-6.4-C).

b. A gap coefficient of 10 Btu/hr ft^2°F was assumed between the liner and containment concrete. It is felt that this represents a much larger gap between the liner and concrete during the transient than would actually exist, and therefore is conservative with respect to the containment pressure transients. In any event, the answer to 16 (E-6.4-C) shows that very little energy is transferred to the containment concrete during blowdown.

c. A small heat transfer coefficient: 2.0 Btu/hr-°F-ft^2 is employed to reflect actual conditions since these surfaces are covered by stagnant steam inside the reactor coolant system.

Item 16 (Attachment E Question 6.3)

Provide a plot of peak containment pressure vs. rupture diameter to permit determination of the most severe loss-of-coolant accident.

ANSWER

The Figure 1 shows the containment pressure versus the break area. The most severe break area is assumed to be 3 FT^2.

CONTAINMENT PRESSURE

VS

BREAK AREA

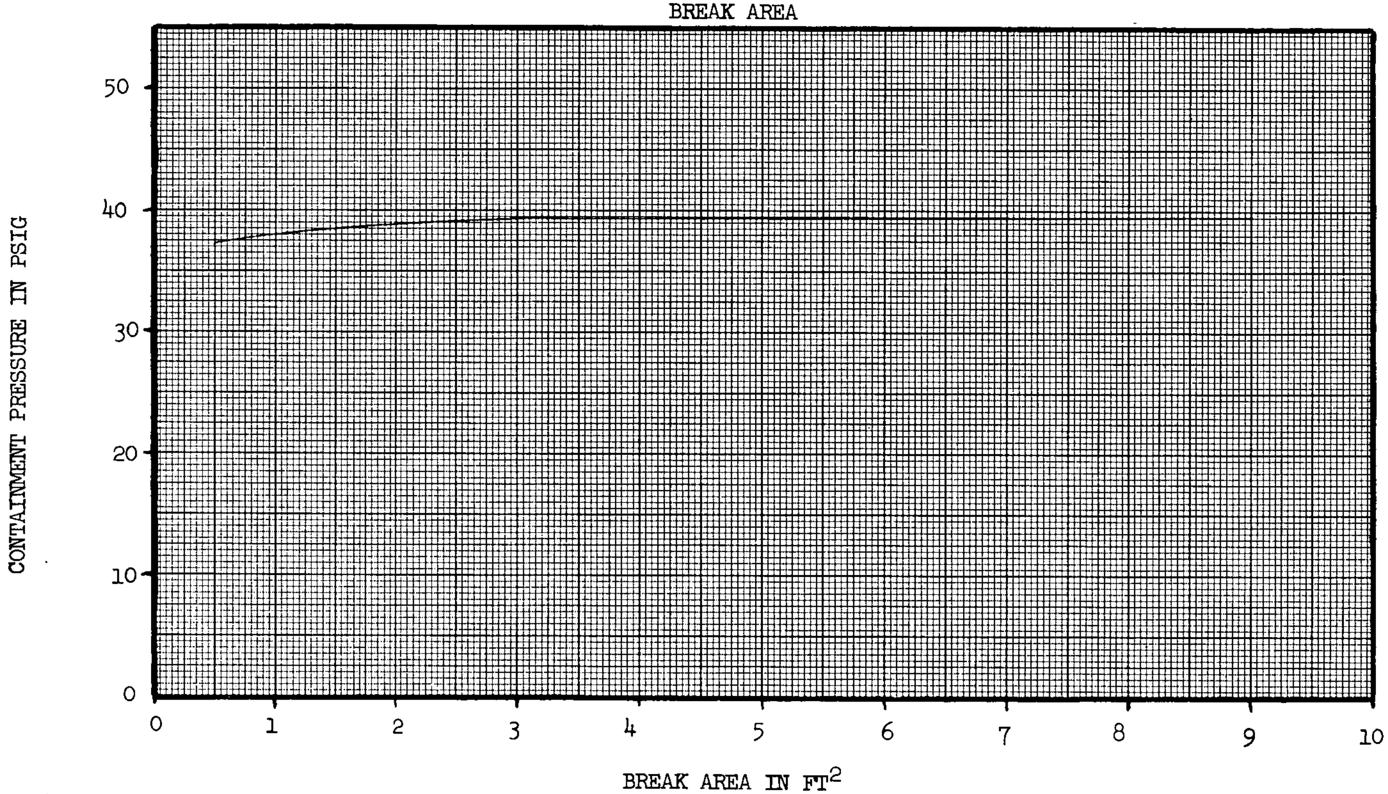

FIGURE 1

Item 16 (Attachment E Question 6.4)

In order that we might understand the magnitude and accuracy of the individual energy sources and sinks, provide the curves of the following parameters as a function of time: (Assume the rupture area associated with the peak of the curve in 6.3 above and that minimum safeguards operate). Include error limits indicating maximum and minimum values of these parameters on these curves.

a. Energy sources

(1) Energy transferred to containment from core sensible heat.

(2) Energy transferred to containment from hot metal.

(3) Decay heat energy transferred to containment.

(4) Energy of zirconium-water reaction and associated hydrogen oxidation.

b. The derivative with respect to time of the parameters in "a."

c. Energy sinks.

(1) Energy absorbed by static sinks.

(2) Energy absorbed by fan-cooler units.

(3) Energy absorbed by spray water.

(4) Energy absorbed by containment liner.

(5) Energy absorbed by containment concrete.

(6) Energy absorbed by the miscellaneous metal within the containment.

(7) Energy absorbed by internal concrete structures.

d. The derivative with respect to time of the parameters in "c".

ANSWER

a. A parameter study was run to determine the sensitivity of the peak containment pressure to different rate of heat transfer from the core and intervals during blowdown. Three cases were run.

1. All the core energy stored, energy related with internals hot metal, and decay energy transferred to containment via the coolant during blowdown.

2. The same energy sources transferred to the containment according to the FLASH Code calculations. Note that credit is taken for heat removal in the steam generators in the FLASH Code.

3. The same energy sources transferred to the containment at the rate predicted by the LOCTAR Code.

The results of the computations are presented in Figure a-1.

Case 2 is considered the design basis blowdown for the containment. This case can be compared to the LOCTAR case where energy is conservatively stored in the core to account for the minimum fuel temperature. Finally, if all the energy sources were transferred to the containment during blowdown (Case 1) a peak pressure of 43.2 psig is calculated, which is much less than the containment design pressure of 47 psig.

The energy release for each source is presented in Figures a-2 through a-4.

During blowdown a metal water reaction of less than 0.1% is calculated. However, the capability of the containment to withstand much larger amounts is presented in Appendix 16 (E-5.0), Section 2, The Containment Evaluation.

b. The derivatives with respect to time of the parameters in "a" are presented in Figure b-1.

c. A parameter study was run to determine the sensitivity of the peak containment pressure to different rates of heat removal in the containment static heat sinks.

As discussed in the Containment Integrity Evaluation the structural heat transfer coefficients used in the evaluation were determined by the Tagami method. This replaced the coefficients determined by Kalflat which were previously used. Figure c-1 presents a comparison of the containment pressures calculated by the two methods. The peak containment pressure by the Tagami method is about 1 psi more conservative than the Kalflat method. Figure c-1

also presents the containment pressures for heat transfer coefficients equal to 1/2 of the Tagami values and for an adiabatic containment. Note that even for the case of zero heat removal by the static heat sinks the containment pressure is limited to less than design pressure.

The integrated energy absorbed by the various structural heat sinks is presented in Figure c-2.

The active heat removal source (fans and sprays) do not contribute to the heat removal before the peak containment pressure is reached. The total integrated heat removal for the fans and spray is presented in Figure c-3.

d. The derivatives with respect to time of the parameters in "c" are presented in Figures d-1 and d-2.

CONTAINMENT PRESSURE VS TIME
FOR DIFFERENT TRANSFER COEFFICIENTS EMPLOYED
IN TRANSFERRING CORE SENSIBLE HEAT DURING BLOWDOWN
3 FT^2 BREAK

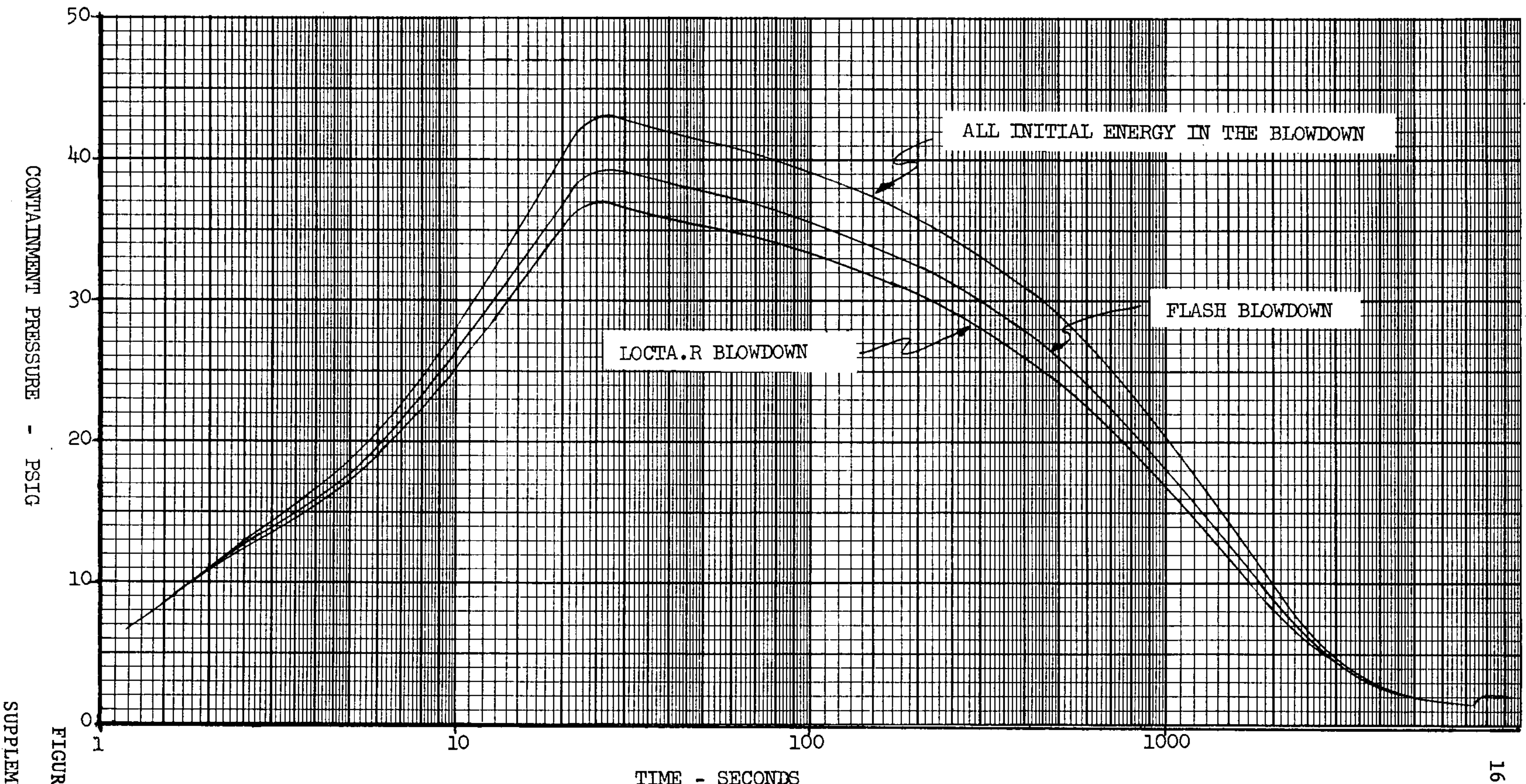

FIGURE a-1
SUPPLEMENT 1

ENERGY RELEASED TO CONTAINMENT

DURING BLOWDOWN

ALL INITIAL ENERGY RELEASED

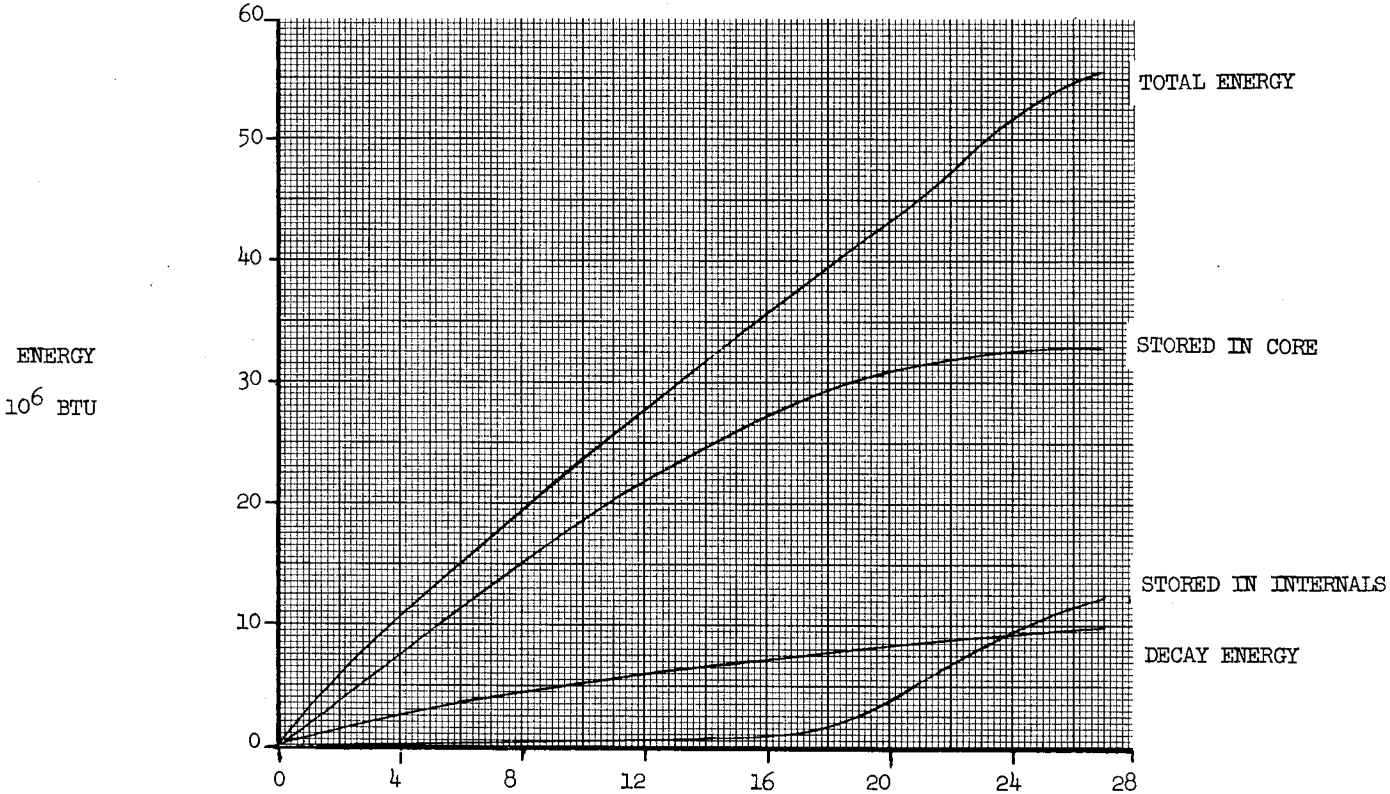

FIGURE a-2

SUPPLEMENT 1

16(E-6.4)

ENERGY RELEASED TO CONTAINMET
DURING BLOWDOWN
ENERGY RELEASED COMPUTED BY FLASH

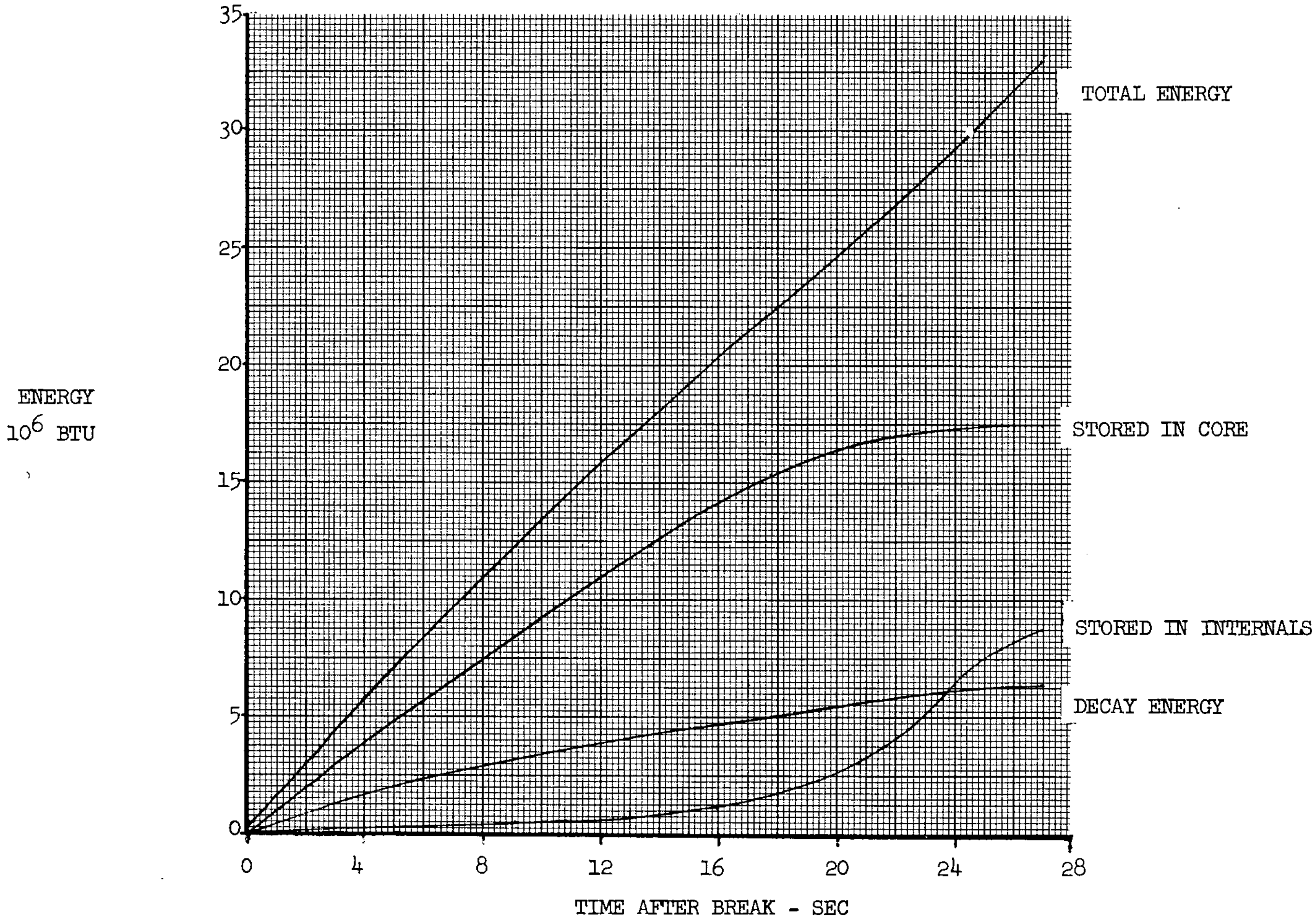

FIGURE a-3
SUPPLEMENT 1

ENERGY RELEASED TO CONTAINMENT

DURING BLOWDOWN

ENERGY RELEASED COMPUTED BY LOCTAR

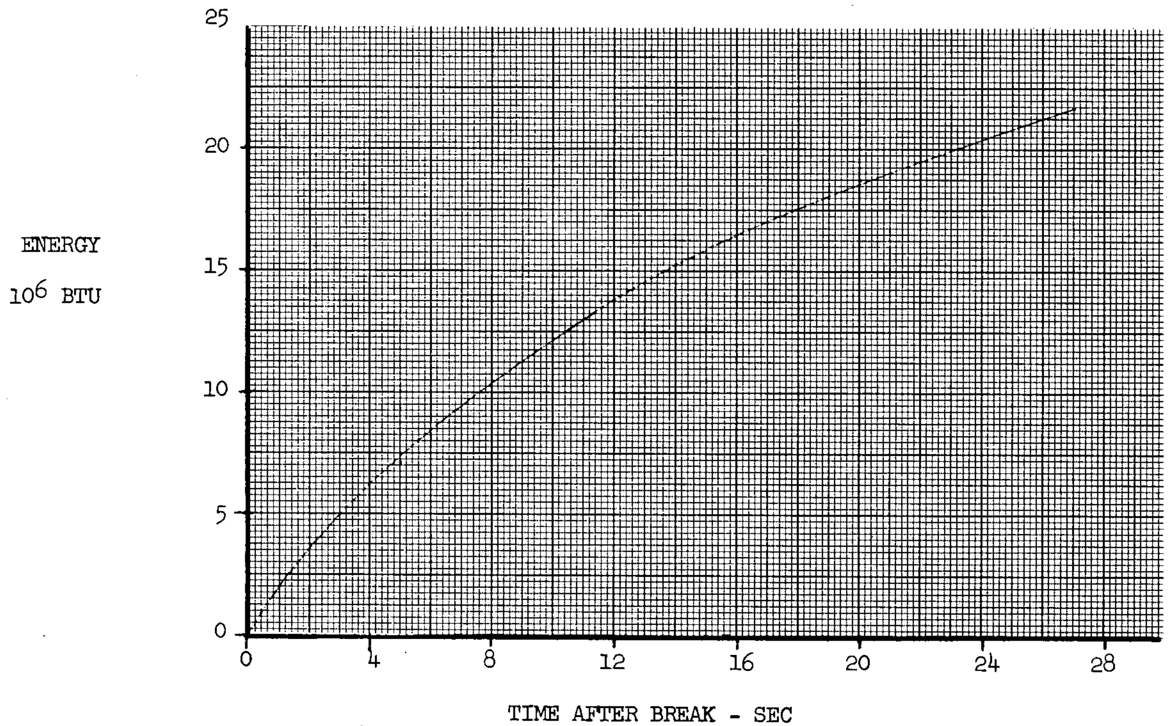

FIGURE a-4
SUPPLEMENT 1

ENERGY RELEASED TO THE CONTAINMENT
DURING BLOWDOWN

(1) = (2) + $.74 \times 10^6$ BTU/SEC

(3) = (2) - $.5 \times 10^6$ BTU/SEC

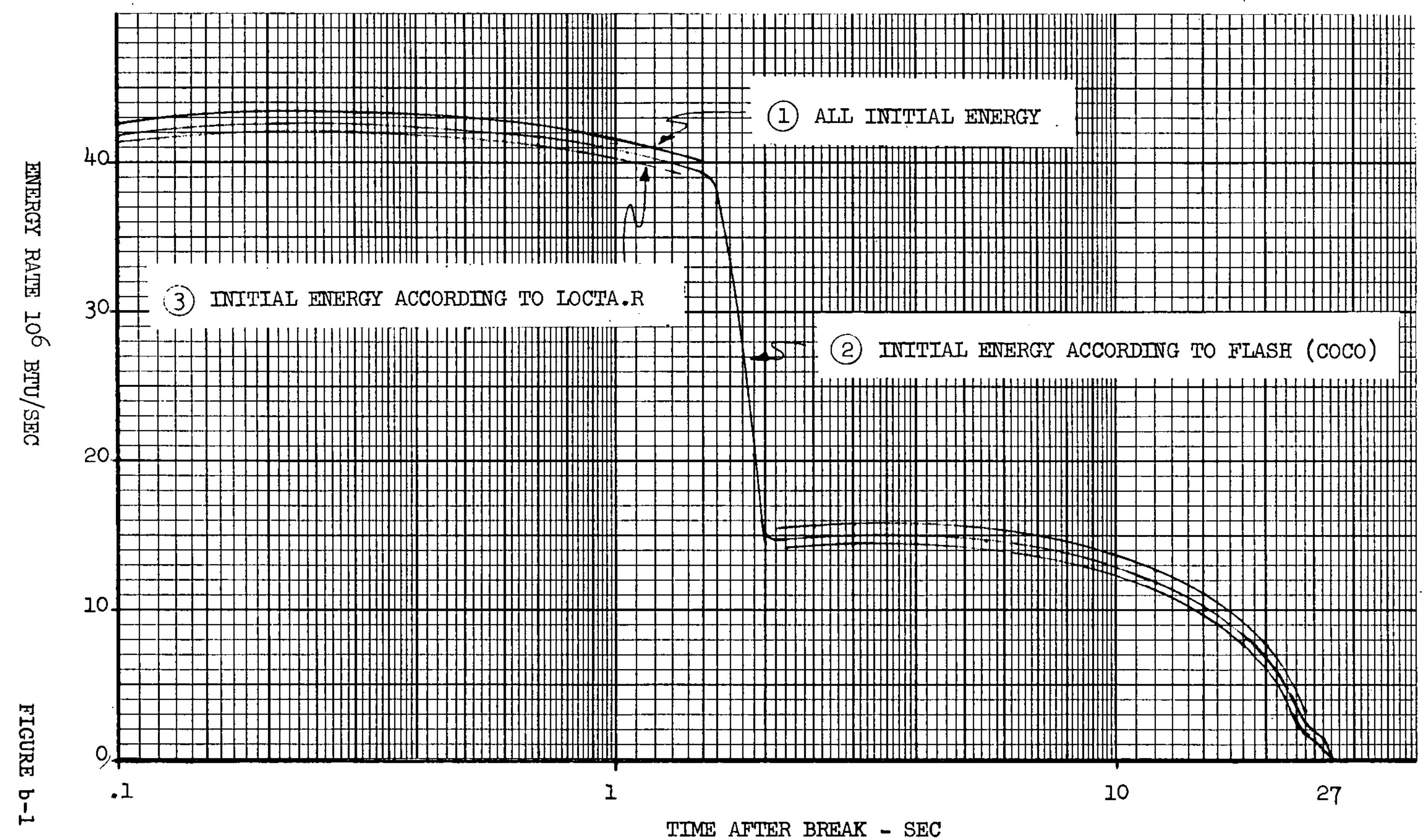

FIGURE b-1
SUPPLEMENT 1

CONTAINMENT PRESSURE VS TIME

FOR VARIOUS HEAT TRANSFER COEFFICIENTS

3 FT^2

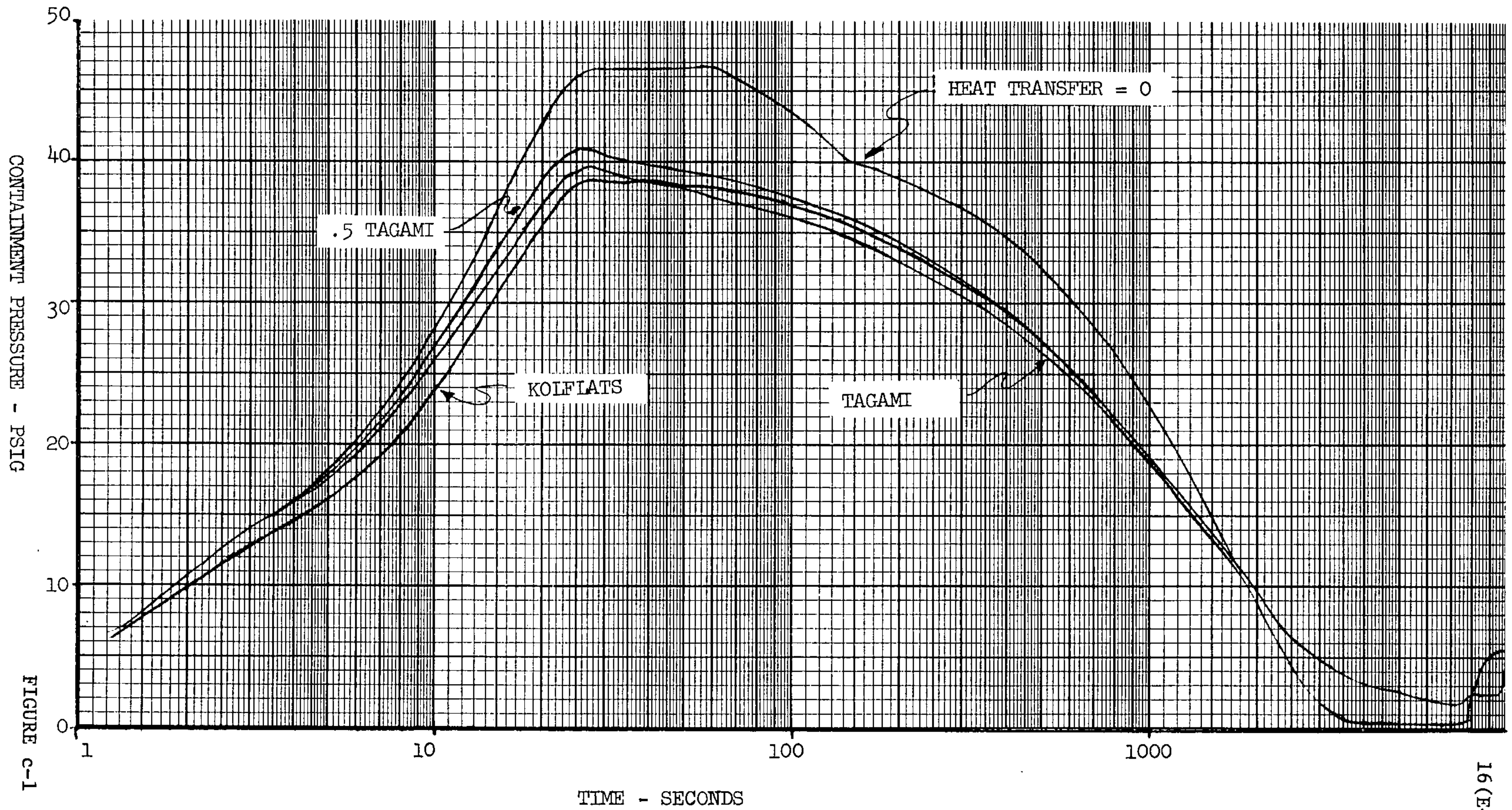

FIGURE C-1

SUPPLEMENT 1

STATIC SINKS TOTAL ENERGY

VS TIME

3 FT^2

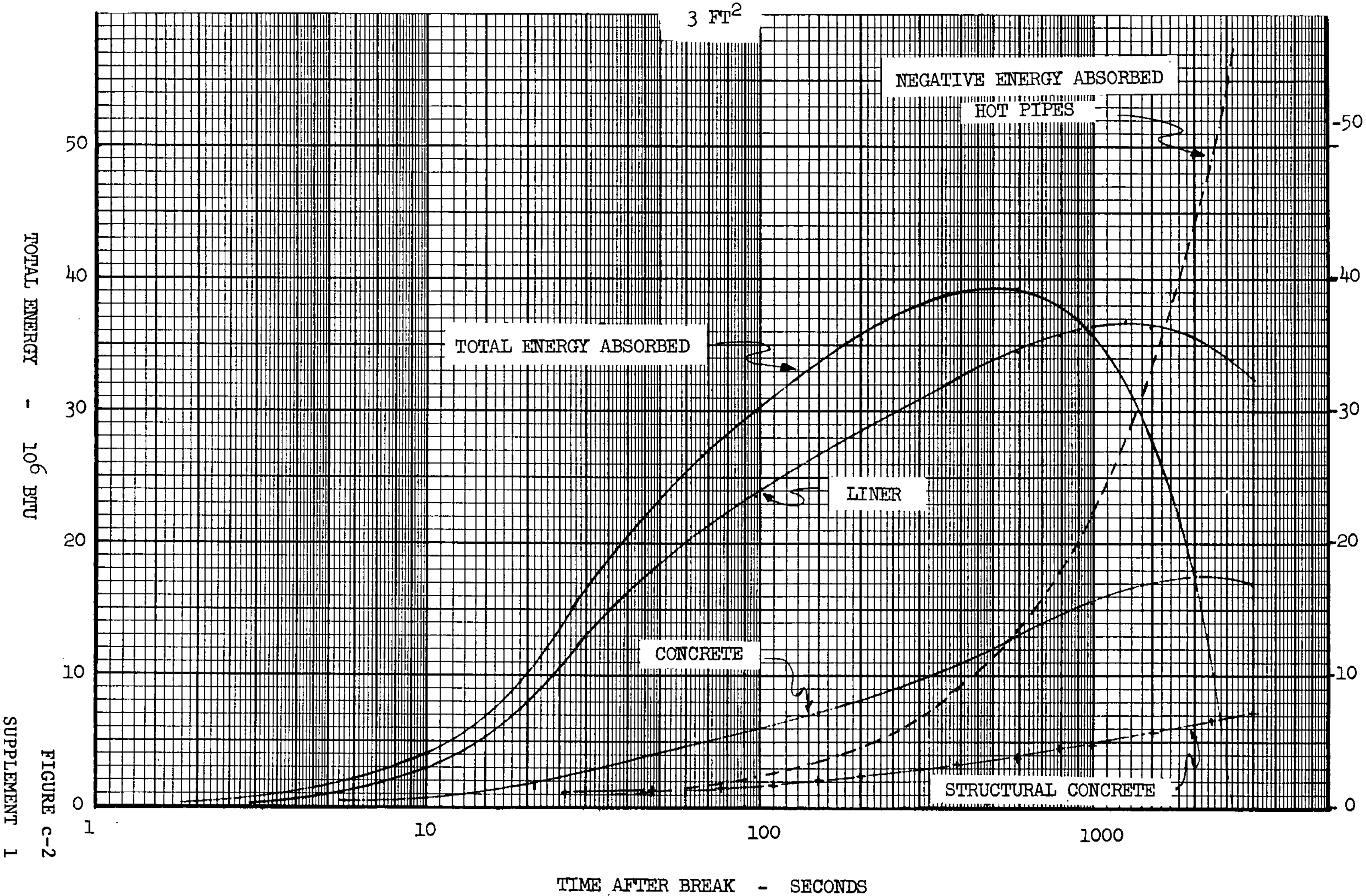

FIGURE c-2

SUPPLEMENT 1

TOTAL HEAT REMOVAL VS. TIME - 3 FT^2

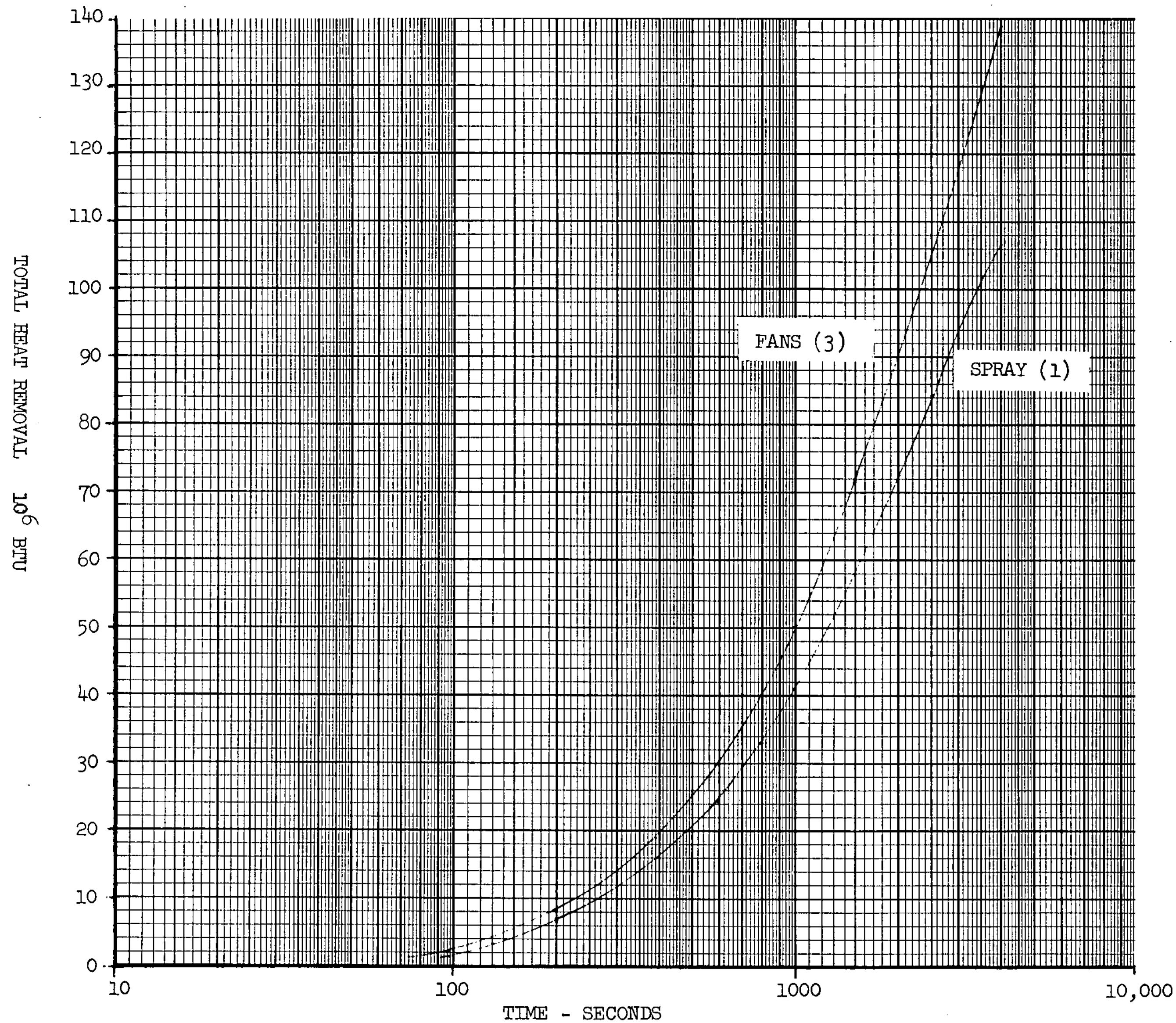

FIGURE c-3
SUPPLEMENT 1

STATIC SINKS ENERGY RATE

VS

TIME

3 FT^2

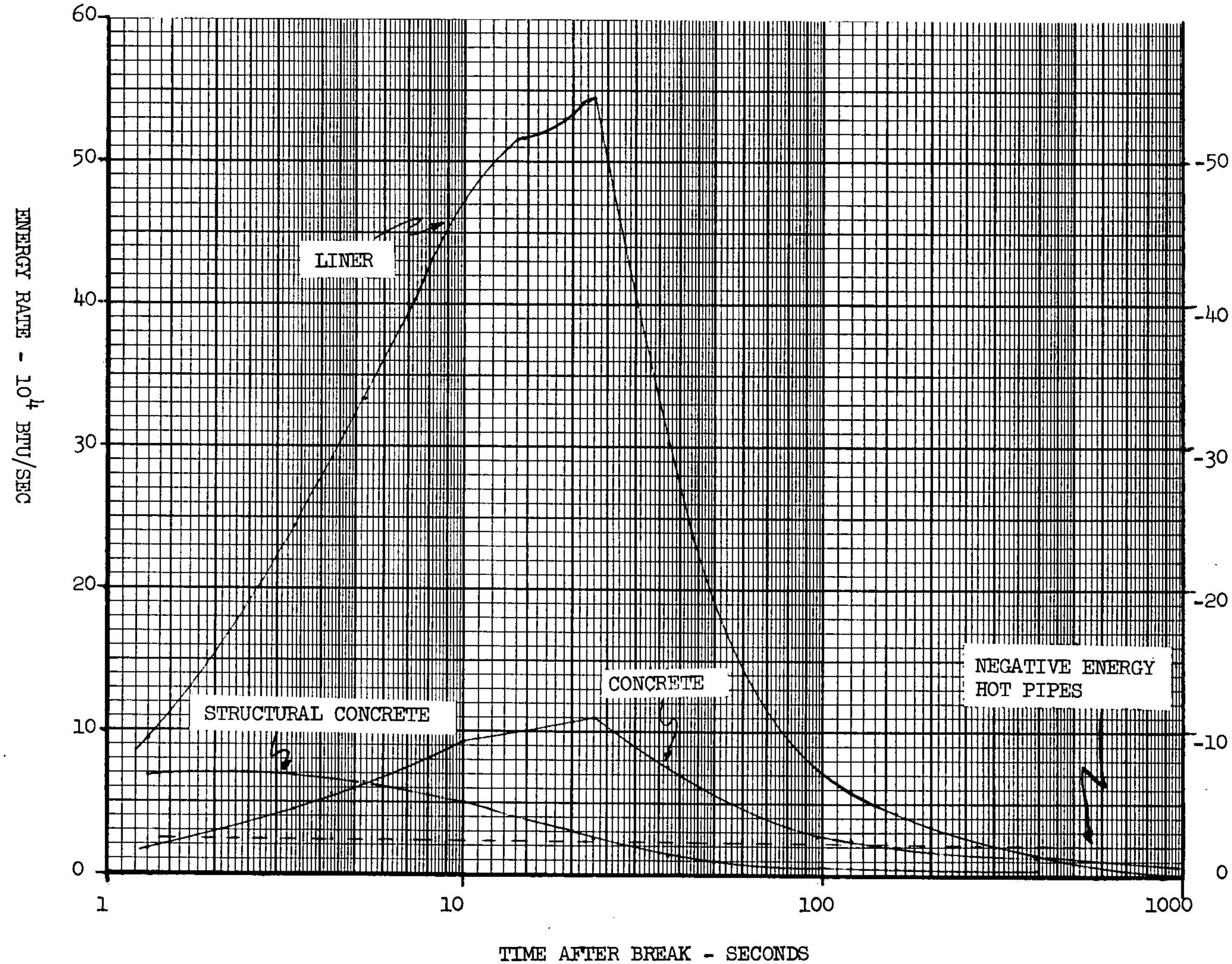

FIGURE d-1

SUPPLEMENT 1

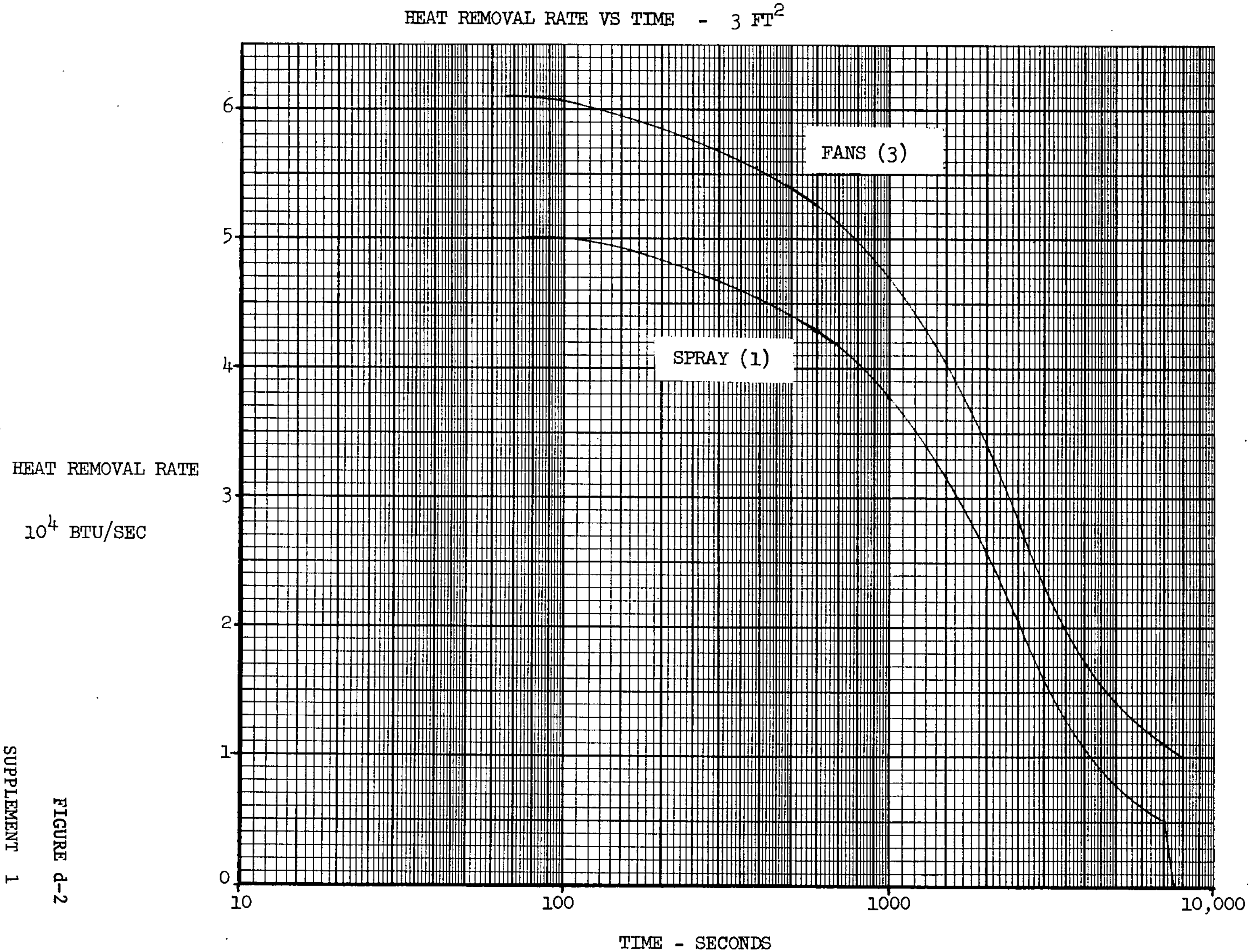

FIGURE d-2

SUPPLEMENT 1

Item 16 (Attachment E Questions 6.5-6.7)

16 (E-6.5) Present curves of containment pressure vs. time for the following cases, assuming the safety injection system, including its associated accumulator subsystem, does not operate, metal-water reaction occurs as determined by core heatup and steam availability, and the rupture area is that associated with the peak of the curve in 6.3 above:

(1) Minimum containment engineered safety features operate.

(2) One residual heat removal pump and three containment fans operate, containment spray system inoperative.

16 (E-6.6) In order to evaluate the margin provided in the containment design for metal-water reaction, provide a plot of peak containment pressure vs. percent metal-water reaction. Assume the rupture area associated with the peak of the curve requested in 6.3 above, minimum engineered safety features operate, and linear metal-water reaction rates of 0.05%/sec. and 0.10% sec.

16 (E-6.7) Discuss the possibility of a localized pressure pulse originating within the containment which could damage structural members if pipe rupture were to occur in closed areas (e.g., the space between the nozzles and the shield wall, the primary pump and steam generator compartments, etc.).

ANSWER

The Containment Integrity Evaluation, Section 2 of Appendix 16 (E-5.0), presents a complete study of the capability of the containment to withstand additional sources of energy. It is felt that this study is responsive to the intent of 16 (E-6.5), 16 (E-6.6), and 16 (E-6.7).

ITEM 16 (Attachment E Question 7.1)

State your criterion regarding acceptable thyroid and whole body doses to operators in the control room following a fission release equivalent to a 100% meltdown with TID-14844 release fractions. Assume no credit is taken for operation of the isolation valve seal water system; i.e., assume a containment leakage rate of 0.1%/day.

ANSWER

The control room is designed and equipped to assure continued occupancy under all operating and accident conditions.

Sufficient shielding, distance, and containment integrity are provided to assure that control room personnel during occupancy of the control room are not subject to doses under postulated accident conditions with minimum operation of the engineered safeguards, which in the agregate would exceed ten percent of the suggested limits in 10 CFR 100.

Reference is made to Item 2 Criterion 11.

ITEM 16 (Attachment E Question 7.2)

Considering the relatively high population density in the vicinity of the proposed Indian Point site, and in order that we might assess the sensitivity of the off-site consequences of the design basis accident to assumptions made concerning the chemical form of the iodines and the removal coefficient (λ) of the reagent spray, you are requested to submit plots of the following parametric studies:

1) Average iodine reduction factor as a function of removal coefficient (λ) for time periods of 0-2, 0-12, and 0-24 hours and for 0-30 days. Assume organic iodine content of 0, 5, 10, 15, and 20% of the total radio-iodine.

2) Two-hour and 30-day thyroid and whole body doses vs. distance, assuming 0, 5, 10, 15, and 20% organic iodine and assuming (1) no iodine removal by the sprays and (2) spray efficiency as predicted by your calculations.

3) Number of people exposed vs. whole body and thyroid dose received in 2 hours, 12 hours, and 30 days, assuming organic iodine fractions of 0, 5, 10, 15, and 20%. Present this for each of the two 22-1/2° sectors having the highest population.

Assume a TID-14844 fission product release with a containment leak rate of 0.1%/day for the first 24 hours and 0.045%/day thereafter.

ANSWER

Figures 1 through 4 show iodine reduction factor as a function of inorganic iodine removal coefficient λ of the reagent spray for the requested time periods and percent of organic iodine up to 20% of total core inventory. No credit is taken for removal of organic iodine due to condensation or radiodecomposition.

Figures 5 through 9 show thyroid and whole body doses respectively vs distance for various percentages of organic iodine with no credit given to organic iodine removal by the sprays. In the case of Figure 5 no credit is given to the removal of any iodine form.

Thyroid dose calculated with the design value removal coefficient, (i.e. $\lambda_i = 32\ hr^{-1}$ for inorganic iodine[1]) meets 10 CFR 100 limit when the initial fraction of methyl-iodide is approximately 4% of total radioiodine with one spray operating for the 0-2 hr. site boundary dose, while an initial methyl-iodine fraction of approximately 4% with one spray operating reaches that limit at the low population zone for the 0-30 day period of time.

---

(1) RGE - FSAR Appendix 6-A.

Whole body gamma dose is seen to be within 10 CFR 100 limits for the requested range of parameters both for no removal of iodine and with the sprays in operation. The curves for 5, 10, and 15% methyl will be found in order between the curves for 0 and 20% methyl.

Man-rem values for the two 22 1/2° sectors having the highest population are listed in Tables 1 through 12. For all the cases listed above a TID-14844 fission product release was assumed with a containment leak rate of 0.1% day for the first 24 hours and 0.045% day thereafter. In the table for Man-Rem-Whole Body Gamma dose, 15, 10, and 5% Man-Rem values lie between those of 20 and 0% methyl.

0-2 hr

TABLE 1

MAN REM - THYROID DOSE
SECTOR I

$\lambda_i = 32.$
$\lambda_o = 0.$

ZONE	MILES EXTREMES OF ZONE AT WHICH AVERAGE DOSE HAS BEEN CALCULATED		POP. 1960	Man Rem % β 0%	5%	10%	15%	20%
1	.2	1	160	5.11×10^3	3.60×10^4	6.70×10^4	9.80×10^4	1.29×10^5
2	1	2	800	7.06×10^3	4.98×10^4	9.26×10^4	1.35×10^5	1.78×10^5
3	2	3	20	8.13×10^1	5.73×10^2	1.06×10^3	1.55×10^3	2.05×10^3
4	3	4	200	4.99×10^3	3.52×10^3	6.54×10^3	9.56×10^3	1.25×10^4
5	4	5	23,000	3.96×10^3	2.79×10^4	5.19×10^4	7.58×10^4	9.98×10^4
6	5	10	23,000	2.27×10^4	1.60×10^5	2.98×10^5	4.36×10^5	5.74×10^5
7	10	15	30,000	1.24×10^4	8.76×10^4	1.62×10^5	2.38×10^5	3.13×10^5
8	15	25	250,000	5.37×10^4	3.79×10^5	7.04×10^5	1.03×10^6	1.35×10^6
9	25	35	1,200,000	1.30×10^5	9.20×10^5	1.71×10^6	2.50×10^6	3.29×10^6
10	35	45	2,200,000	1.52×10^5	1.07×10^6	1.99×10^6	2.91×10^6	3.83×10^6
11	45	55	500,000	2.44×10^4	1.72×10^5	3.20×10^5	4.68×10^5	6.16×10^5

TABLE 2

MAN REM - THYROID DOSE
SECTOR II

$\lambda_i = 32$
$\lambda_0 = 0$

ZONE	MILES EXTREMES OF ZONE AT WHICH AVERAGE DOSE HAS BEEN CALCULATED		POP. 1960	Man Rem % β 0%	5%	10%	15%	20%
1	.2	1	55	1.75×10^3	1.24×10^4	2.30×10^4	3.36×10^4	4.43×10^4
2	1	2	750	6.62×10^3	4.67×10^4	8.68×10^4	1.26×10^5	1.67×10^5
3	2	3	1,500	6.10×10^3	4.30×10^4	7.99×10^4	1.16×10^5	1.53×10^5
4	3	4	20	4.99×10^1	3.52×10^2	6.54×10^2	9.56×10^2	1.25×10^3
5	4	5	0	0	0	0	0	0
6	5	10	9,000	9.21×10^3	6.49×10^4	1.20×10^5	1.76×10^5	2.32×10^5
7	10	15	25,000	1.03×10^4	7.30×10^4	1.35×10^5	1.98×10^5	2.61×10^5
8	15	25	400,000	8.60×10^4	6.06×10^5	1.12×10^6	1.64×10^6	2.16×10^6
9	25	35	2,000,000	2.17×10^5	1.53×10^6	2.85×10^6	4.16×10^6	5.48×10^6
10	35	45	2.500,000	1.72×10^5	1.21×10^6	2.26×10^6	3.31×10^6	4.35×10^6
11	45	55	500,000	2.44×10^4	1.72×10^5	3.20×10^5	4.68×10^5	6.16×10^5

0-12 hr

TABLE 3

MAN REM - THYROID DOSE
SECTOR I

$\lambda_i = 32$
$\lambda_o = 0$

ZONE	MILES EXTREMES OF ZONE AT WHICH AVERAGE DOSE HAS BEEN CALCULATED		POP. 1960	MAN REM % β 0%	5%	10%	15%	20%
1	.2	1	160	5.11×10^3	8.03×10^4	1.55×10^5	2.30×10^5	3.06×10^5
2	1	2	800	7.06×10^3	1.09×10^5	2.10×10^5	3.12×10^5	4.14×10^5
3	2	3	20	8.13×10^2	1.27×10^3	2.46×10^3	3.66×10^3	4.85×10^3
4	3	4	200	4.99×10^2	2.83×10^3	1.51×10^4	2.25×10^4	2.98×10^4
5	4	5	23,000	3.96×10^3	6.23×10^4	1.20×10^5	1.79×10^5	2.37×10^5
6	5	10	23,000	2.27×10^4	3.58×10^5	6.94×10^5	1.03×10^6	1.36×10^6
7	10	15	30,000	1.24×10^4	1.92×10^5	3.73×10^5	5.53×10^5	7.34×10^5
8	15	25	250,000	5.37×10^4	8.27×10^5	1.60×10^6	2.37×10^6	3.14×10^6
9	25	35	1,200,000	1.30×10^5	1.99×10^6	3.85×10^6	5.71×10^6	7.57×10^6
10	35	45	2,200,000	1.52×10^5	2.30×10^6	4.45×10^6	6.61×10^6	8.76×10^6
11	45	55	500,000	2.44×10^4	3.68×10^5	7.13×10^5	1.05×10^6	1.40×10^6

TABLE 4

0-12 hr

$\lambda_i = 32$

$\lambda_o = 0$

MAN REM - THYROID DOSE
SECTOR II

ZONE	MILES EXTREMES OF ZONE AT WHICH AVERAGE DOSE HAS BEEN CALCULATED		POP. 1960	MAN REM % β 0%	5%	10%	15%	20%
1	.2	1	55	1.75×10^3	2.76×10^4	5.35×10^4	7.93×10^4	1.05×10^5
2	1	2	750	6.62×10^3	1.02×10^5	1.97×10^5	2.93×10^5	3.88×10^5
3	2	3	1,500	6.10×10^3	9.55×10^4	1.85×10^5	2.74×10^5	3.64×10^5
4	3	4	20	4.99×10^1	7.83×10^2	1.51×10^3	2.25×10^3	2.98×10^3
5	4	5	0	0	0	0	0	0
6	5	10	9,300	9.21×10^3	1.45×10^5	2.80×10^5	4.16×10^5	1.52×10^5
7	10	15	25,000	1.03×10^3	1.60×10^5	3.11×10^5	4.61×10^5	6.11×10^5
8	15	25	400,000	8.60×10^4	1.32×10^5	2.56×10^6	3.79×10^6	5.03×10^6
9	25	35	2,000,000	2.17×10^5	3.31×10^6	6.41×10^6	4.52×10^6	1.26×10^6
10	35	45	2,500,000	1.72×10^5	2.62×10^6	5.06×10^6	7.51×10^6	9.96×10^6
11	45	55	500,000	2.44×10^4	3.68×10^5	7.13×10^5	1.05×10^6	1.40×10^6

0-720 hr

TABLE 5

MAN REM - THYROID DOSE
SECTOR I

$\lambda_i = 32$
$\lambda_o = 0$

ZONE	MILES EXTREMES OF ZONE AT WHICH AVERAGE DOSE HAS BEEN CALCULATED		POP. 1960	MAN REM % β 0%	5%	10%	15%	20%
1	.2	1	160	5.11×10^3	1.51×10^5	2.96×10^5	4.42×10^5	5.88×10^5
2	1	2	800	7.06×10^3	1.39×10^5	2.71×10^5	4.04×10^5	5.36×10^5
3	2	3	20	8.13×10^1	1.52×10^3	2.97×10^3	4.42×10^3	5.86×10^3
4	3	4	200	4.99×10^2	9.14×10^3	1.77×10^4	2.64×10^4	3.50×10^4
5	4	5	23,000	3.96×10^3	7.16×10^4	1.39×10^5	2.07×10^5	2.74×10^5
6	5	10	23,000	2.27×10^4	4.06×10^5	7.89×10^5	1.17×10^6	1.55×10^6
7	10	15	30,000	1.24×10^4	2.12×10^5	4.12×10^5	6.12×10^5	8.11×10^5
8	15	25	250,000	5.37×10^4	8.97×10^5	1.74×10^6	6.12×10^6	8.12×10^6
9	25	35	1,200,000	1.30×10^5	2.12×10^6	4.12×10^6	6.12×10^6	8.12×10^6
10	35	45	2,200,000	1.52×10^5	2.44×10^6	4.73×10^6	7.02×10^6	9.32×10^6
11	45	55	500,000	2.44×10^4	3.88×10^5	7.53×10^5	1.11×10^6	1.48×10^6

TABLE 6

0-720 hr

MAN REM - THYROID DOSE
SECTOR II

$\lambda_i = 32$
$\lambda_o = 0$

ZONE	MILES EXTREMES OF ZONE AT WHICH AVERAGE DOSE HAS BEEN CALCULATED		POP. 1960	MAN REM % β 0%	5%	10%	15%	20%
1	.2	1	85	1.75×10^3	5.19×10^4	1.02×10^5	1.52×10^5	2.02×10^5
2	1	2	750	6.62×10^3	1.30×10^5	2.54×10^5	3.78×10^5	5.02×10^5
3	2	3	1,500	6.10×10^3	1.14×10^5	2.23×10^5	3.31×10^5	4.40×10^5
4	3	4	20	4.99×10^1	9.14×10^2	1.77×10^3	2.64×10^3	3.50×10^2
5	4	4	0	0	0	0	0	
6	5	10	9,300	9.21×10^3	1.64×10^5	3.19×10^5	4.74×10^5	6.76×10^5
7	10	15	25,000	1.03×10^4	1.76×10^5	3.43×10^5	5.10×10^5	5.48×10^5
8	15	25	400,000	8.60×10^4	1.43×10^6	2.78×10^6	4.13×10^6	1.35×10^6
9	25	35	2,000,000	2.17×10^5	3.54×10^6	6.88×10^6	1.02×10^7	1.05×10^7
10	35	45	2,500,000	1.72×10^5	2.77×10^6	5.38×10^6	7.98×10^6	1.48×10^7
11	45	55	500,000	2.44×10^4	3.88×10^5	7.53×10^5	1.11×10^6	1.40×10^6

TABLE 7

0-2 hr

MAN REM - WHOLE BODY DOSE

ZONE	MILES EXTREMES OF ZONE AT WHICH AVERAGE DOSE HAS BEEN CALCULATED		POP. 1960	MAN REM NO SPRAY	MAN REM 20% METHYL	MAN REM 0% METHYL
1	.2	1	160	1.23×10^3	7.42×10^2	5.38×10^2
2	1	2	800	1.7×10^3	1.02×10^3	7.48×10^2
3	2	3	20	1.96×10^1	1.10×10^1	8.54×10^0
4	3	4	200	1.2×10^2	6.03×10^1	3.35×10^1
5	4	5	23,000	9.55×10^3	4.78×10^3	4.16×10^3
6	5	10	23,000	5.5×10^3	2.75×10^3	2.39×10^3
7	10	15	30,000	3.0×10^3	1.5×10^3	1.30×10^3
8	15	25	250,000	1.29×10^4	6.5×10^3	5.66×10^3
9	25	35	1,200,000	3.14×10^4	1.57×10^4	1.37×10^4
10	35	45	2,200,000	3.66×10^4	1.83×10^4	1.59×10^4
11	45	55	500,000	5.85×10^3	2.95×10^3	2.56×10^3

0-2 hr

TABLE 8

MAN REM - WHOLE BODY DOSE
SECTOR II

ZONE	MILES EXTREMES OF ZONE AT WHICH AVERAGE DOSE HAS BEEN CALCULATED		POP. 1960	MAN REM NO SPRAY	MAN REM 20% METHYL	MAN REM 0% METHYL
1	.2	1	55	4.24×10^{2}	2.55×10^{5}	1.85×10^{2}
2	1	2	750	1.6×10^{3}	9.62×10^{2}	7.01×10^{2}
3	2	3	1,500	1.47×10^{3}	8.31×10^{2}	6.40×10^{2}
4	3	4	20	1.2×10^{1}	6.03×10^{0}	3.35×10^{0}
5	4	5	0	0	0	0
6	5	10	9,300	2.22×10^{3}	1.11×10^{3}	9.67×10^{2}
7	10	15	25,000	2.5×10^{3}	1.25×10^{3}	1.08×10^{3}
8	15	25	400,000	2.07×10^{4}	1.04×10^{4}	9.06×10^{3}
9	25	35	2,000,000	5.24×10^{4}	2.62×10^{4}	2.28×10^{4}
10	35	45	2,500,000	4.15×10^{4}	2.07×10^{4}	1.81×10^{4}
11	45	55	500,000	5.85×10^{3}	2.95×10^{3}	2.65×10^{3}

0-12 hr

TABLE 9

MAN REM - WHOLE BODY DOSE
SECTOR I

ZONE	MILES EXTREMES OF ZONE AT WHICH AVERAGE DOSE HAS BEEN CALCULATED		POP. 1960	MAN REM NO SPRAY	MAN REM 20% METHYL	MAN REM 0% METHYL
1	.2	1	160	2.03×10^3	1.30×10^3	9.56×10^2
2	1	2	800	2.77×10^3	1.77×10^3	1.30×10^3
3	2	3	20	3.23×10^1	1.99×10^1	1.51×10^1
4	3	4	200	1.98×10^2	1.15×10^2	9.32×10^1
5	4	5	23,000	1.59×10^4	9.15×10^3	7.4×10^3
6	5	10	23,000	9.29×10^3	5.26×10^3	4.25×10^3
7	10	15	30,000	4.90×10^3	2.83×10^3	2.29×10^3
8	15	25	250,000	2.10×10^4	1.21×10^4	9.88×10^3
9	25	35	1,200,000	5.08×10^4	2.93×10^4	2.38×10^4
10	35	45	2,000,000	5.89×10^4	3.31×10^4	2.76×10^4
11	45	55	500,000	9.44×10^3	5.26×10^3	4.4×10^3

0-12 hr

TABLE 10

MAN REM - WHOLE BODY DOSE
SECTOR II

ZONE	MILES EXTREMES OF ZONE AT WHICH AVERAGE DOSE HAS BEEN CALCULATED		POP. 1960	MAN REM NO SPRAY	MAN REM 20% METHYL	MAN REM 0% METHYL
1	.2	1	55	7.00×10^2	4.48×10^2	3.28×10^2
2	1	2	750	2.60×10^3	1.66×10^3	1.22×10^3
3	2	3	1,500	2.42×10^3	1.49×10^3	1.13×10^3
4	3	4	20	1.98×10^1	1.15×10^1	9.3×10^0
5	4	5	0	0	0	0
6	5	10	9,300	3.75×10^3	2.13×10^3	1.72×10^3
7	10	15	25,000	4.08×10^3	2.36×10^3	1.91×10^3
8	15	25	400,000	3.37×10^4	1.94×10^4	1.58×10^4
9	25	35	2,000,000	8.47×10^4	4.88×10^4	3.96×10^4
10	35	45	2,500,000	6.7×10^4	3.77×10^4	3.13×10^4
11	45	55	500,000	9.44×10^3	5.26×10^3	4.4×10^3

0-720 hr

TABLE 11

MAN REM - WHOLE BODY DOSE
SECTOR I

ZONE	MILES EXTREMES OF ZONE AT WHICH AVERAGE DOSE HAS BEEN CALCULATED		POP. 1960	MAN REM NO SPRAY	MAN REM 20% METHYL	MAN REM 0% METHYL
1	.2	1	160	3.92×10^3	2.72×10^3	2.03×10^3
2	1	2	800	4.45×10^3	3.0×10^3	2.22×10^3
3	2	3	20	5.08×10^1	3.33×10^1	2.51×10^1
4	3	4	200	3.09×10^2	1.94×10^2	1.52×10^2
5	4	5	23,000	2.45×10^4	1.53×10^4	1.20×10^4
6	5	10	23,000	1.4×10^4	8.92×10^3	6.87×10^3
7	10	15	30,000	7.43×10^3	4.67×10^3	3.64×10^3
8	15	25	250,000	3.17×10^4	1.98×10^4	1.55×10^4
9	25	35	1,200,000	7.58×10^4	4.7×10^4	3.69×10^4
10	35	45	2,200,000	8.78×10^4	5.27×10^4	4.28×10^4
11	45	55	500,000	1.40×10^4	8.27×10^3	6.82×10^3

TABLE 12

0-720 hr

MAN REM - WHOLE BODY DOSE
SECTOR II

ZONE	MILES EXTREMES OF ZONE AT WHICH AVERAGE DOSE HAS BEEN CALCULATED		POP. 1960	MAN REM NO SPRAY	MAN REM 20% METHYL	MAN REM 0% METHYL
1	.2	1	55	1.34×10^3	9.33×10^2	6.98×10^2
2	1	2	750	4.18×10^3	2.81×10^3	2.08×10^3
3	2	3	1,500	3.8×10^3	2.5×10^3	1.88×10^3
4	3	4	20	3.09×10^1	1.94×10^1	1.52×10^1
5	4	5	0	0	0	0
6	5	10	9,300	5.66×10^3	3.55×10^3	2.78×10^3
7	10	15	25,000	6.45×10^3	3.89×10^3	3.03×10^3
8	15	25	400,000	5.06×10^4	3.17×10^4	2.48×10^4
9	25	35	2,000,000	1.26×10^5	7.84×10^4	6.15×10^4
10	35	45	2,500,000	9.99×10^4	5.98×10^4	4.86×10^4
11	45	55	500,000	1.4×10^4	8.27×10^3	6.82×10^3

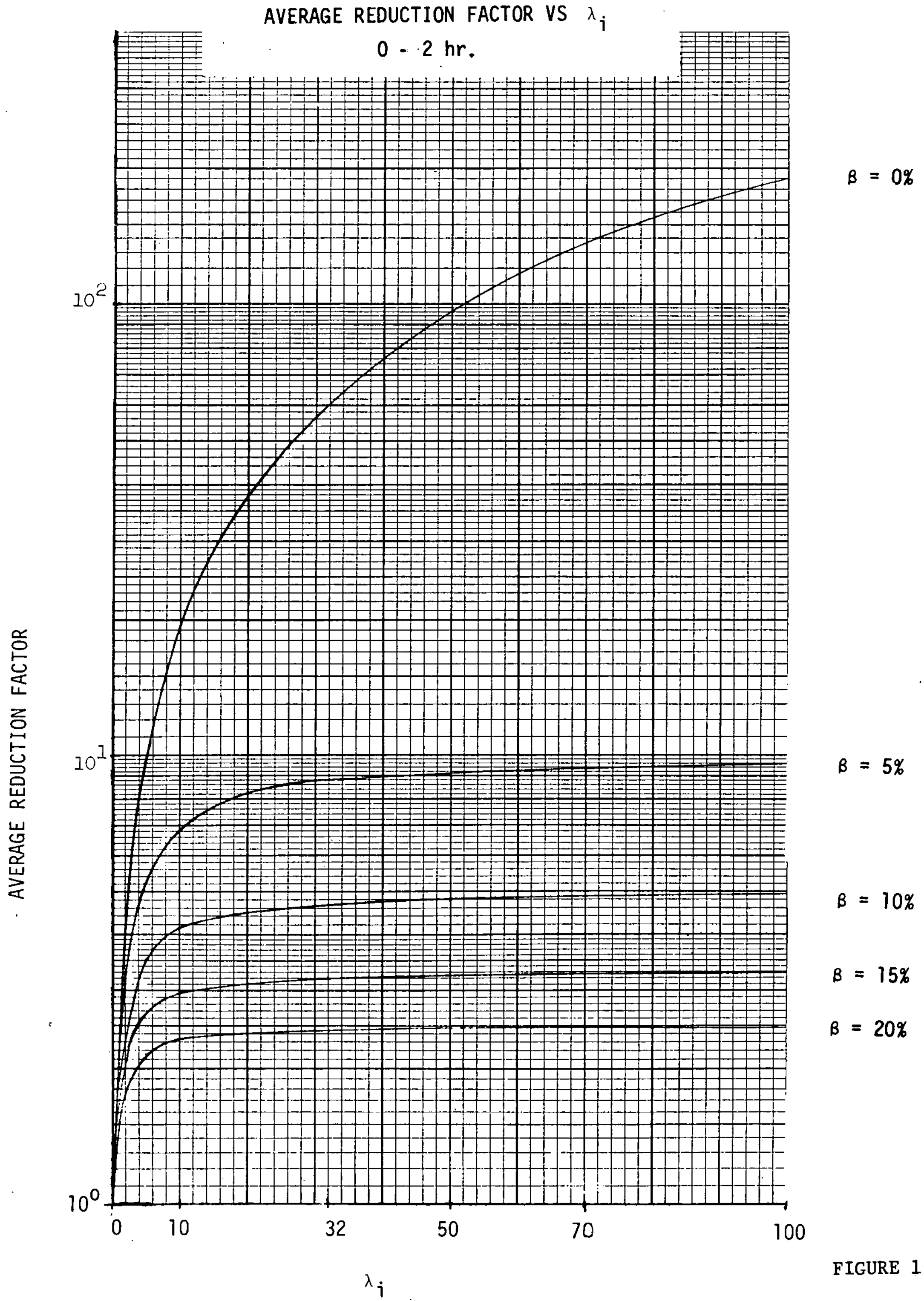
AVERAGE REDUCTION FACTOR VS λ_i
0 - 2 hr.
AVERAGE REDUCTION FACTOR
10^2
10^1
10^0
0
10
32
50
70
100
λ_i
β = 0%
β = 5%
β = 10%
β = 15%
β = 20%

FIGURE 1

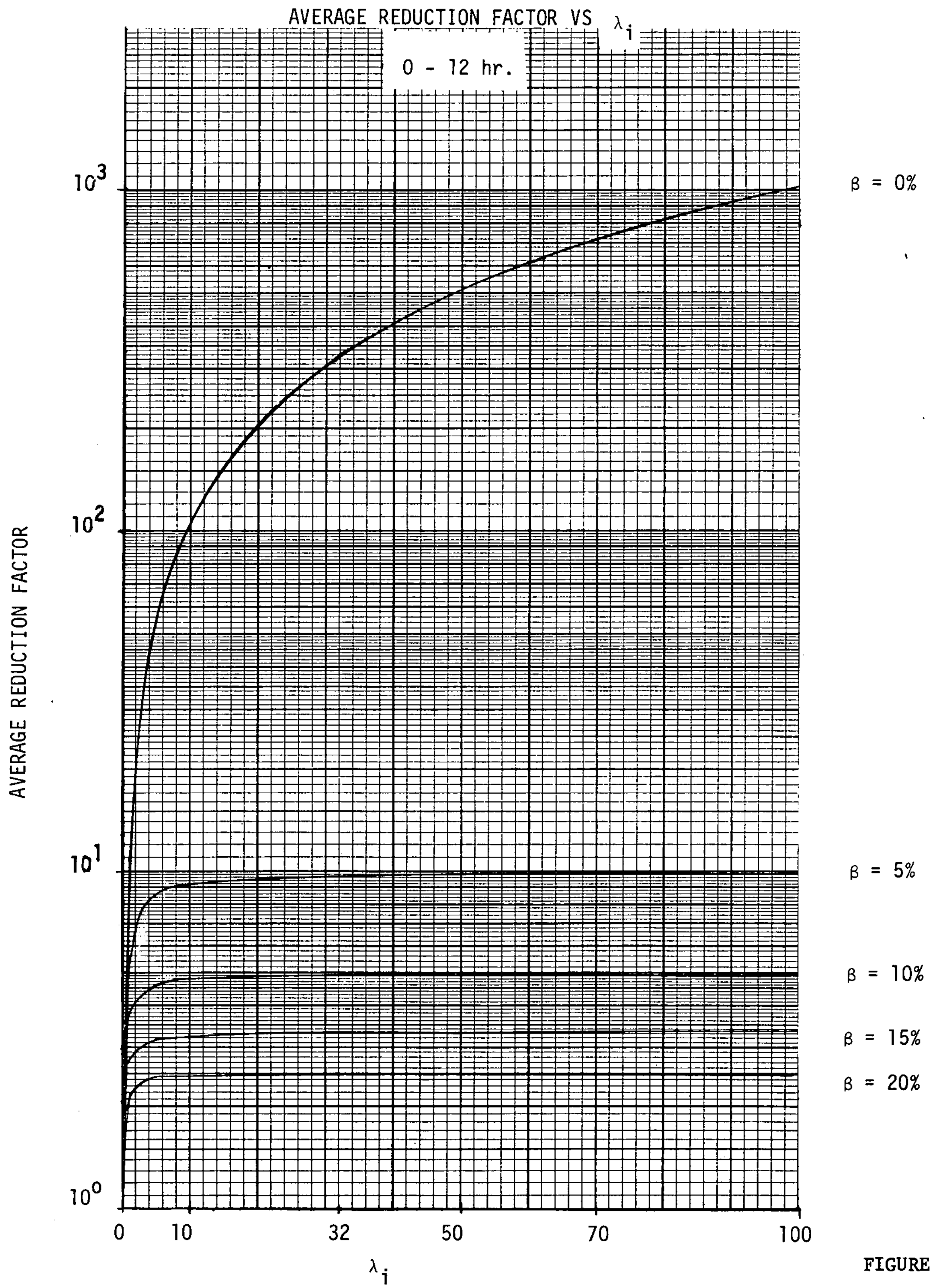

FIGURE 2

SUPPLEMENT 1

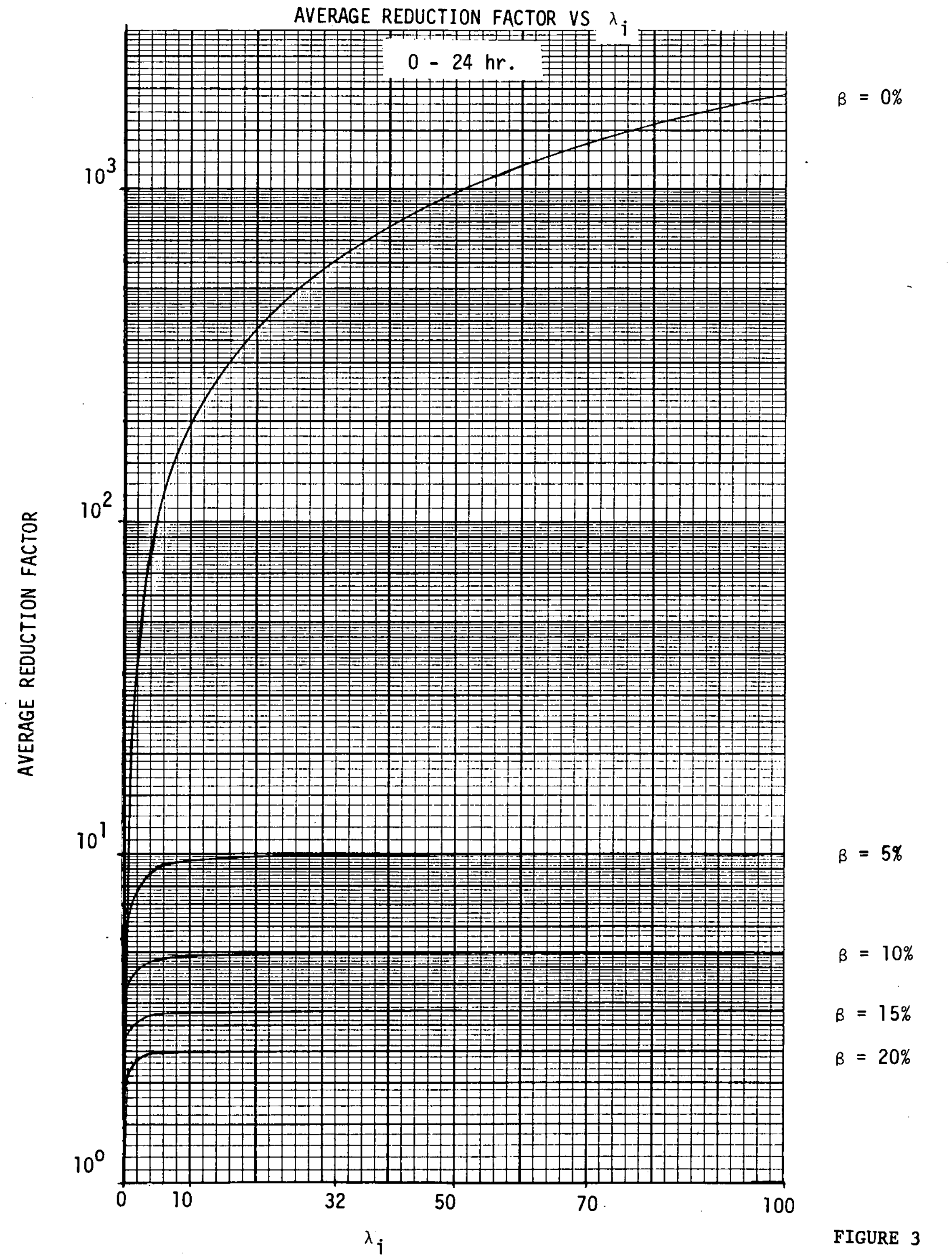
AVERAGE REDUCTION FACTOR VS λ_i
0 - 24 hr.
AVERAGE REDUCTION FACTOR
10^3
10^2
10^1
10^0
0
10
32
50
70
100
λ_i
β = 0%
β = 5%
β = 10%
β = 15%
β = 20%

FIGURE 3

AVERAGE REDUCTION FACTOR VS λ_i

0 - 720 hr.

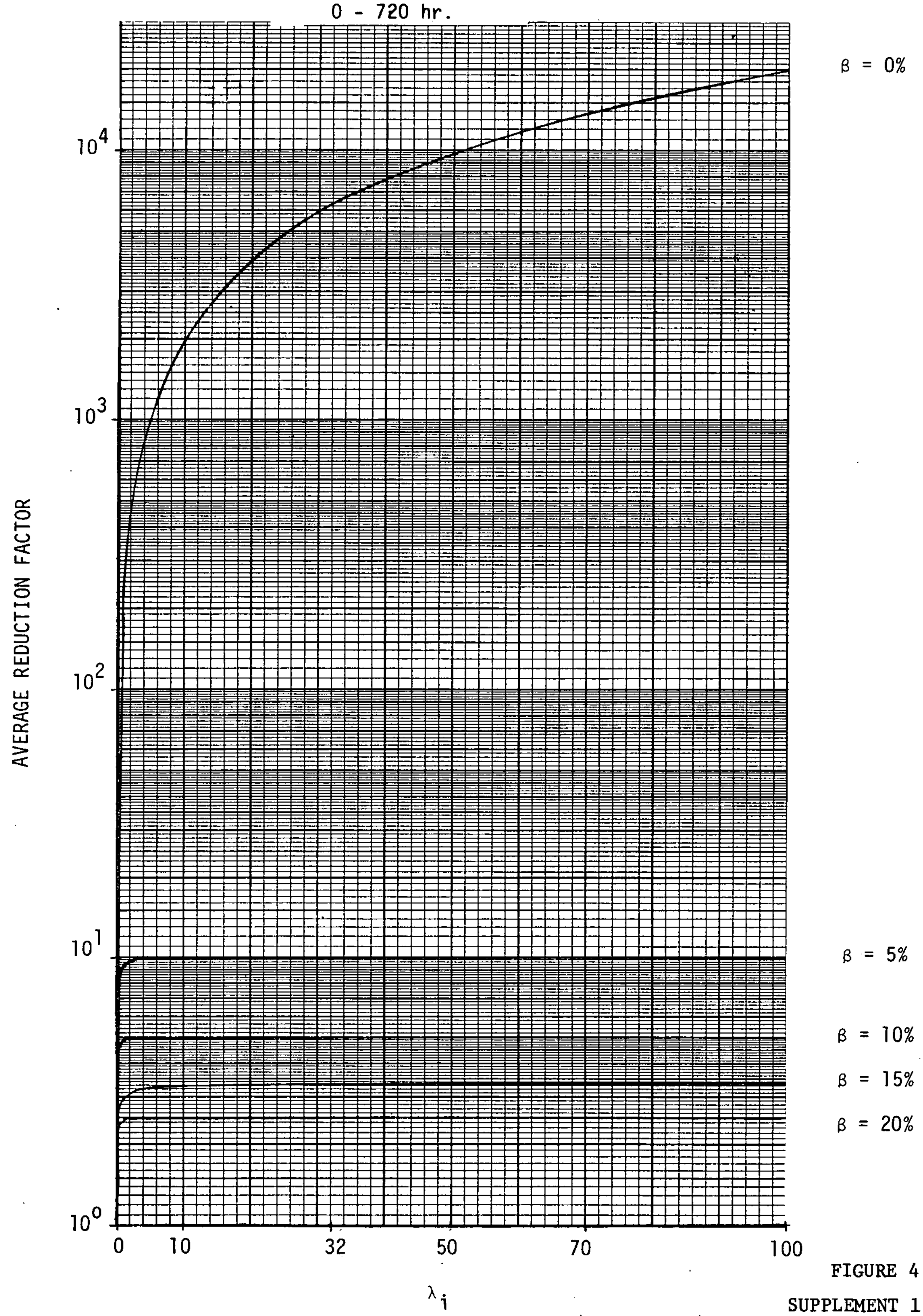

FIGURE 4

SUPPLEMENT 1

THYROID DOSE VS DISTANCE

0 - 2 hr and 0 - 30 days

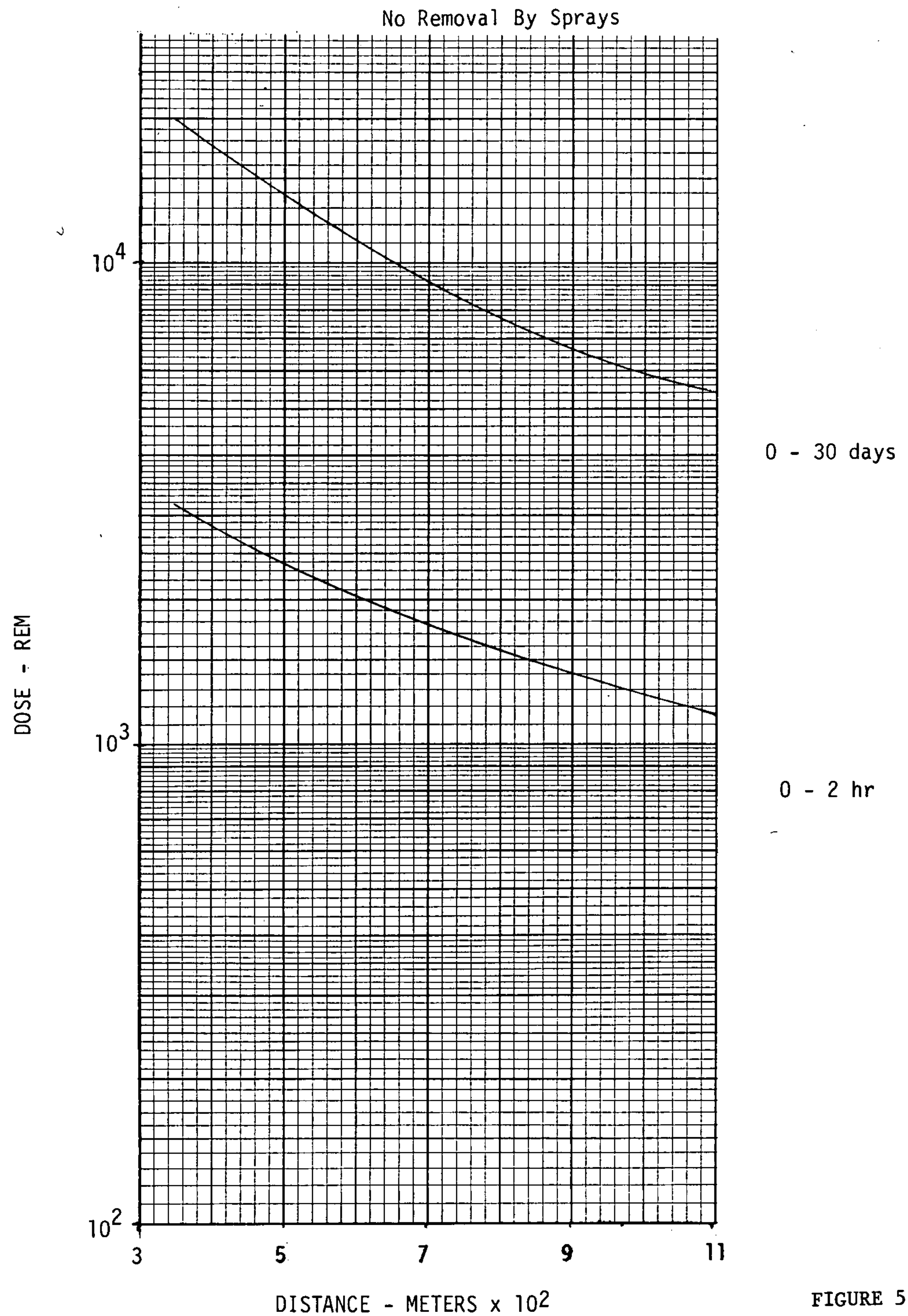

FIGURE 5

SUPPLEMENT 1

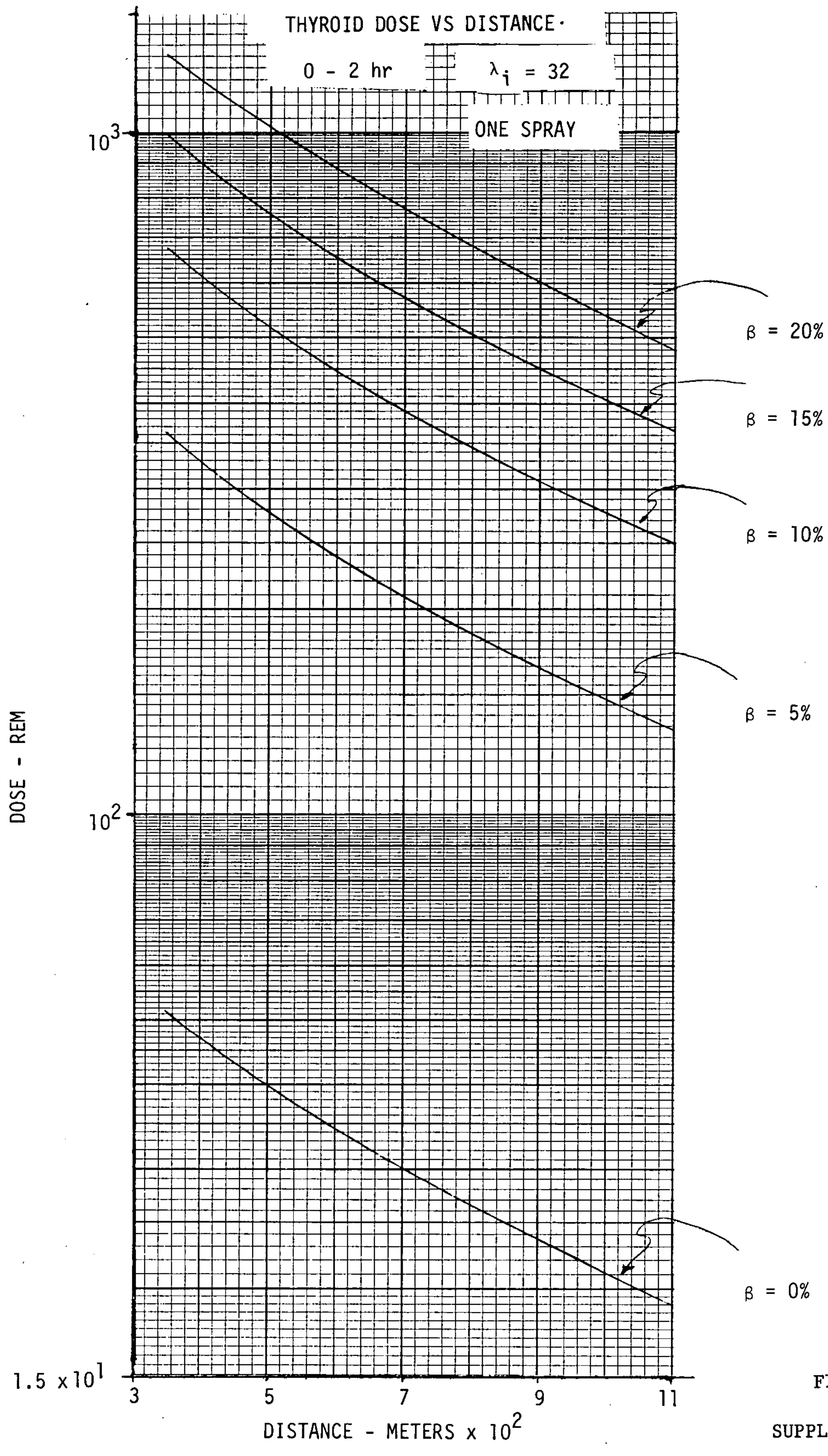

FIGURE 6

SUPPLEMENT 1

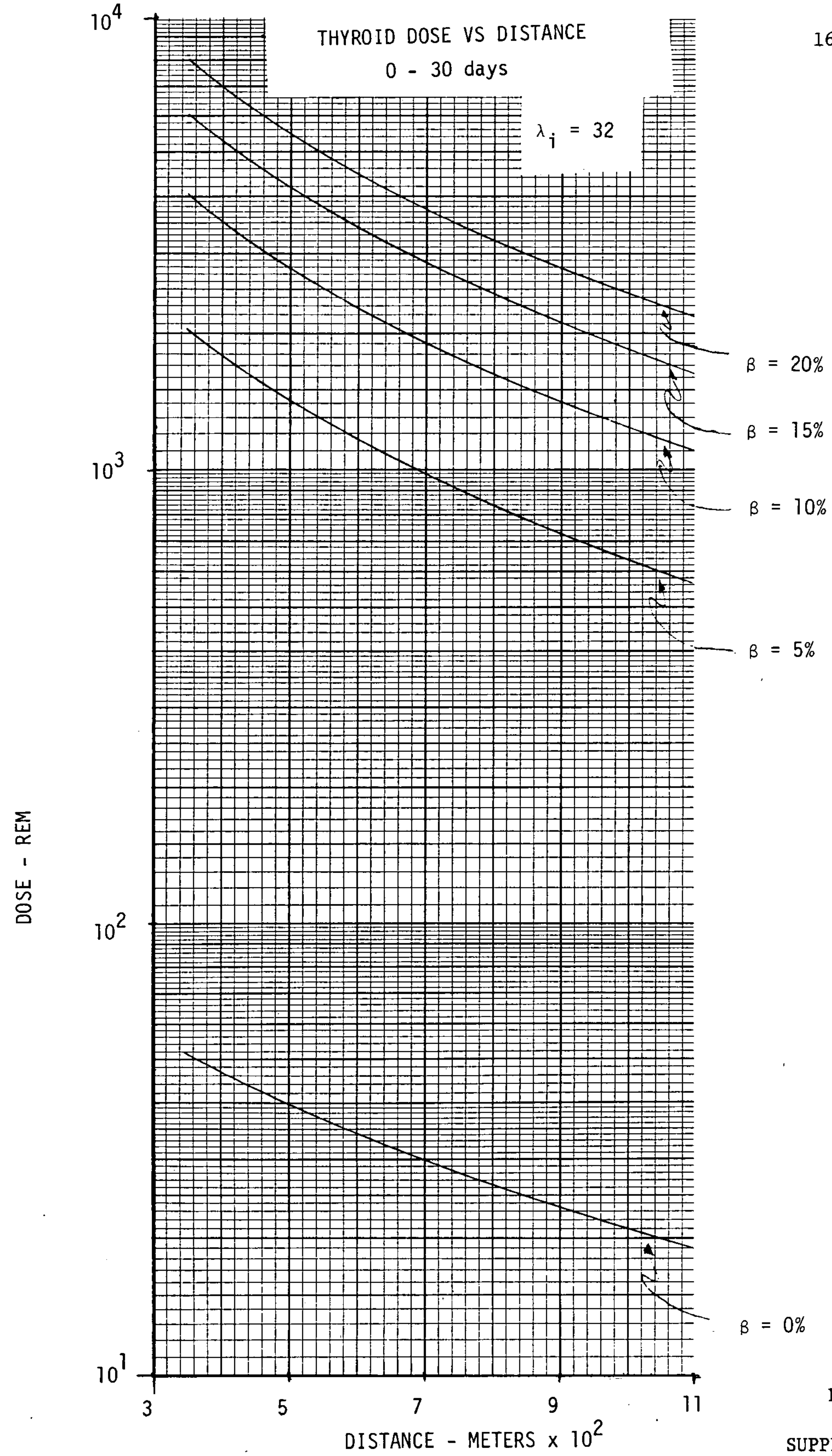
THYROID DOSE VS DISTANCE
0 - 30 days
$\lambda_i = 32$
10^4
10^3
10^2
10^1
DOSE - REM
β = 20%
β = 15%
β = 10%
β = 5%
β = 0%
3
5
7
9
11
DISTANCE - METERS x 10^2

FIGURE 7

WHOLE BODY γ DOSE VS DISTANCE

0 - 2 HR

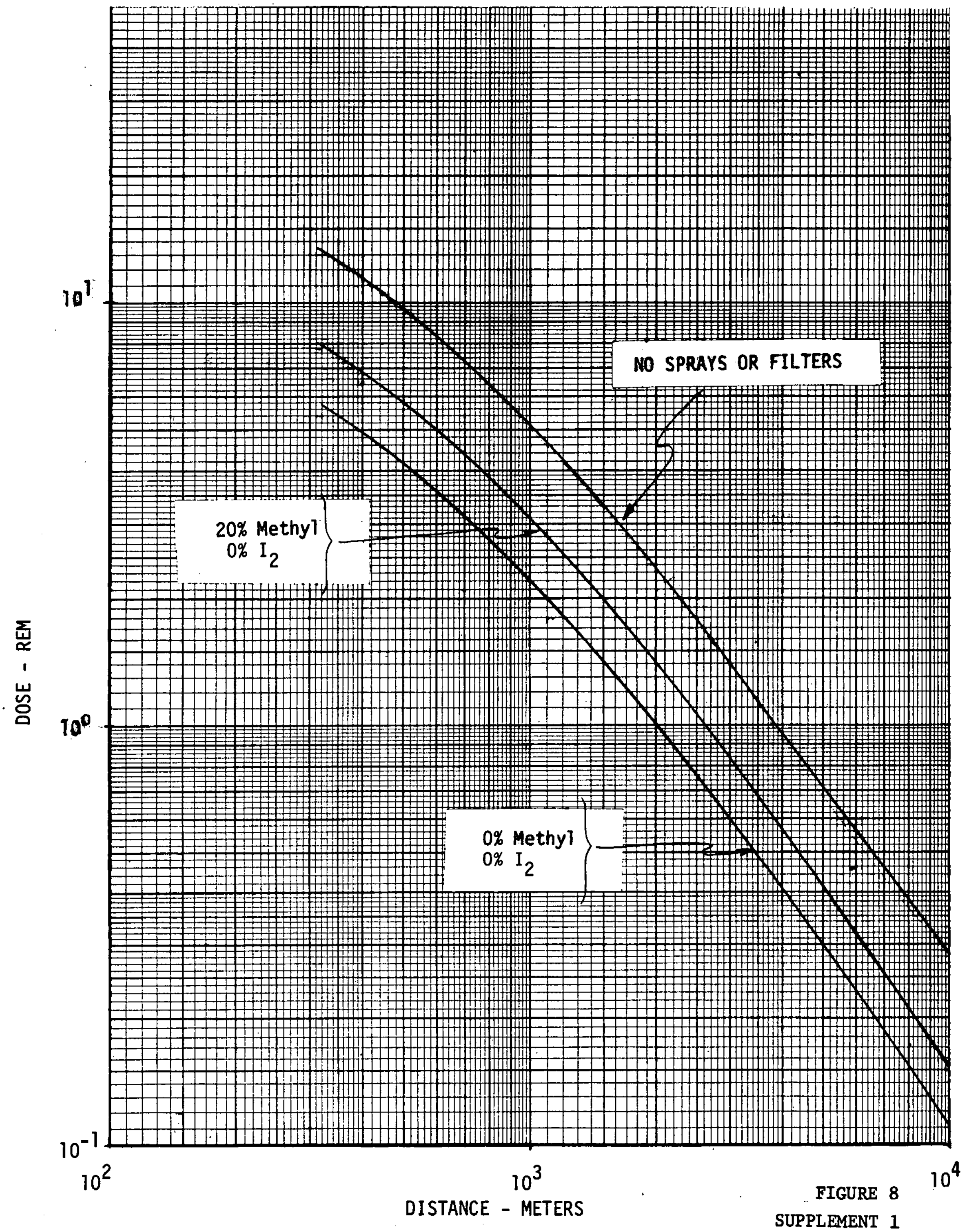

FIGURE 8
SUPPLEMENT 1

WHOLE BODY γ DOSE VS DISTANCE

0 - 30 DAYS

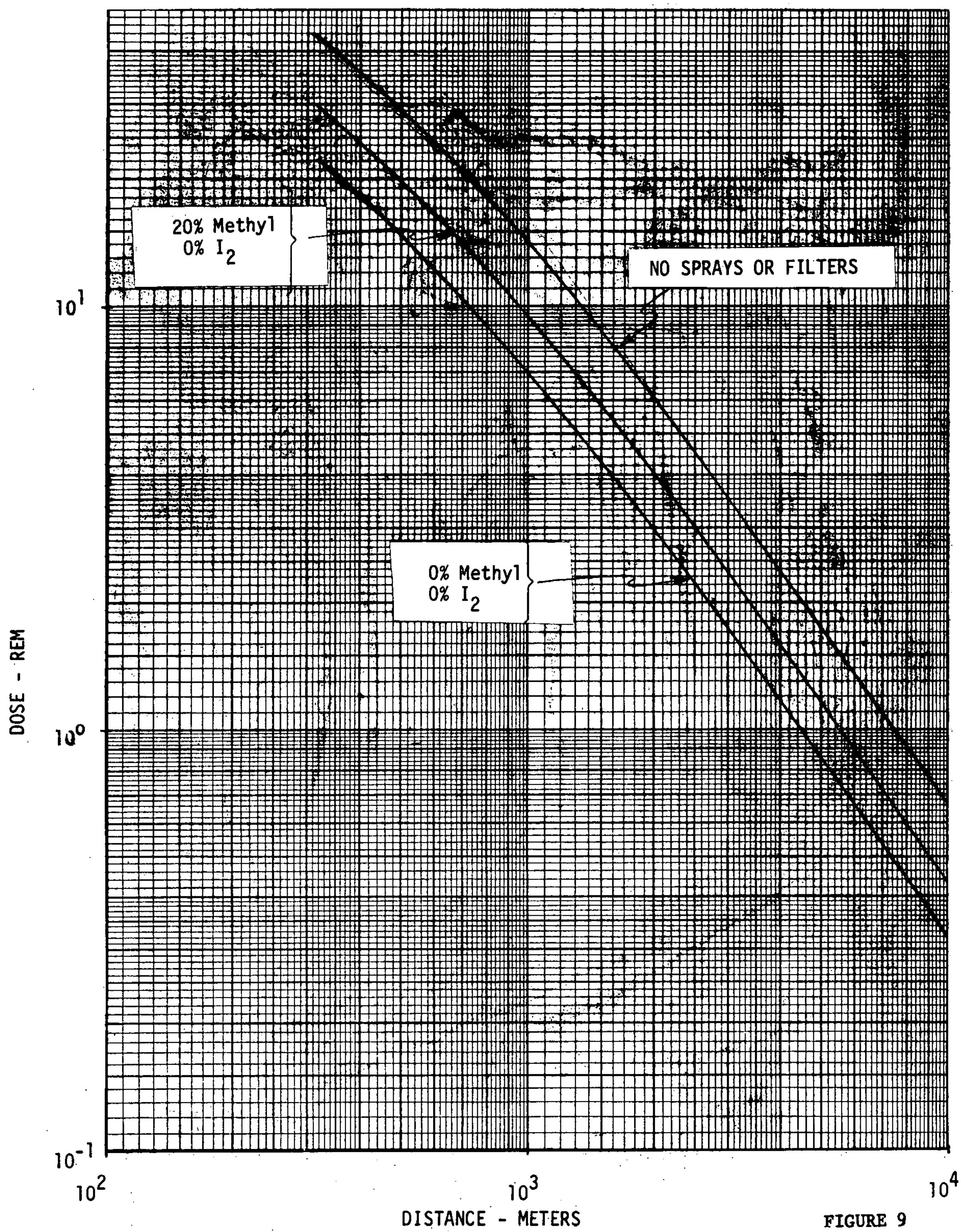

FIGURE 9

SUPPLEMENT 1

Item 17 (Attachment F Question 1.0)

Discuss the operation of the emergency diesel power supply system under accident conditions with no normal power sources available. Indicate the sequence of equipment that is automatically started during the injection phase (including designation and horsepower of each), and the loads (designation and horsepower) on each diesel for the recirculation phase. Give basis for the rating of the diesel proposed to furnish these loads. Confirm that after any single fault or failure, including the failures of any one diesel to start, sufficient power is available for engineered safety features. Describe equipment used for the automatic sequence loading and consider a failure in one automatic sequence. Confirm that all equipments, including diesels, fuel,auxiliaries, controls, wiring, etc., have the physical separation required to prevent a single accident (including fire) from disabling more than one diesel power supply.

ANSWER

The first source of emergency power is the 138 - 6.9 kV Station Auxiliary Transformer. If this first source should fail, the final 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 source, 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 up to a 45 second cranking cycle. The emergency units are capable of being started and sequential loading begun within 10 seconds after the initial signal.

To ensure rapid start the units are equipped with water jacket and lube oil 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 diesel-generator facility includes a fuel storage tank at each unit with a capacity for two hours of operation. The main fuel storage provided with the facility consists of three tanks each with a capacity of 7,500 gallons. This storage capacity will provide sufficient fuel to allow two diesels to operate continuously at 3500 kw total for approximately 5 days. Transfer of oil from these storage tanks which are underground to automatically maintain level in each unit tank is accomplished by a motor driven pump, one for each unit tank. Any oil transfer pump is capable of pumping from any of the three storage tanks through manual valving. In addition to this fuel supply there are two 30,000 gallon storage tanks on site as a part of the gas turbine unit. Fuel can be drawn from these tanks and transferred in containers to the diesel storage tanks. It can be seen that less than 50% of this capacity would provide fuel for an additional 5 days. There are also 2 - 6,000 gallon fuel oil storage tanks on site for the Unit No. 1 super-heater ignition.

Additional supplies of diesel oil are available locally. Under normal conditions, 25,000 gallons can be delivered on a one or two day notice. Additional supplies are also maintained in the New Rochelle - Mount Vernon area (about 40 miles from the plant) and are available for use during emergencies, subject to extreme cold weather conditions (increased domestic heating usage) and available transportation.

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

1. Initiation of safety injection operation.

2. Undervoltage on either of the two 6900 volt buses connected to the outside power.

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.

2. Connect the diesel generators to their respective buses.

In the event one of the emergency generators does not start when called for, and there is no fault on the 480 volt bus associated with that generator, tie breakers are automatically closed to distribute the buses between the two other emergency generators and the following sequence is initiated automatically.

a. Start 2 out of 2 Auxiliary Component Cooling Pumps*
b. Start 2 out of 3 Safety Injection Pumps
c. Start 1 out of 2 Residual Heat Removal Pumps
d. Start 1 out of 2 Spray Pumps if the containment pressure signal is present
e. Start 2 out of 3 Nuclear Service Water Pumps
f. Start 3 out of 5 Containment Air Recirculation Cooling Fans

* Both pumps are started, but only one is required.

g. Start 1 out of 2 Auxiliary Feedwater Pumps

h. Start 1 out of 2 Spray Pumps if the containment pressure signal is present (action taken in case high-high containment pressure signal did not exist at step (d))

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 1. The loads are given for a post-blowdown containment pressure transient.

3. In the event all three emergency diesels start, the following sequence is initiated automatically.

   a. Start 2 out of 2 Auxiliary Component Cooling Pumps
   b. Start 3 Safety Injection Pumps
   c. Start 2 Residual Heat Removal Pumps
   d. Start 2 Spray Pumps if containment pressure signal is present
   e. Start 2 out of 3 Nuclear Service Water Pumps
   f. Start 5 out of 5 Containment Air Recirculation Cooling Fans
   g. Start 2 Auxiliary Feedwater Pumps
   h. Start 2 Spray Pumps if the containment pressure signal is present (action taken in case high-high containment pressure signal did not exist at step (d))

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.

Provision for a single failure plus failure of redundant pump or other engineered safeguards load to start is accomplished by the addition of circuit breaker interlocks in the redundant load "close" or start control circuits.

Separation

Diesel-Generator

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 smoke and heat in order to quickly actuate the detection and spray devices. In addition, manual fire fighting equipment is located at each personnel access door.

A switchgear panel is located on the west end of the building containing breakers, relays and electrical bus structures for all three diesel engine generators. The panels are compartmentalized with switchgear for each engine separated from each other. The tie breakers to the emergency buses are redundant and also located in separate cubicles. During accident conditions, the tie breakers are open (non-synchronized emergency bus modus operandi) which will preclude short circuit interaction on other 480 v emergency power buses. In the event of an electrical fire, smoke detectors are provided to annunciate in the main control building. With the compartmentalized panel design, conflagration to other electrical components is minimized.

Each diesel generator has its own fuel oil storage tank and its own fuel oil supply pump. These tanks are located in a pit outside the diesel generator building which is filled with clean sand and provided with a pit cover. It is, therefore, unlikely that a fire associated with any one of the fuel oil storage tanks would prevent fuel oil from being supplied to the remaining two diesels or cause any damage to the diesel generator building.

The automatic spray system previously described and the curbing would allow operation of the remaining two diesels in the event of fire in any one of the three diesel generators.

Cables and Instrumentation

1. Power Cabling

   a. Safeguard power cables are to have as many physical channels as required for the minimum safeguard function. For example, the five (5) containment fan power circuits will have a minimum of three (3) channels since a minimum of three (3) fans are required to operate.

b. The underground cabling to the intake structure supplying the service water pumps will pass through the same manhole in two (2) locations. These lead and PVC jacketed cables will be separated either vertically or horizontally on racks within the manholes and covered with fireproof material.

c. The main 480 Volt Diesel Generator output buses will be for the most part horizontally separated and enclosed in ventilated metallic enclosures. Low voltage power circuits within the Diesel Building will be run in rigid steel conduit.

2. Control Cabling

a. Control cables will be separated into three (3) basic channels in open cable trays as required for redundant and safeguard control circuits. These groups of cables will serve systems of above 65 volts and less than 500 volts and will include multiconductor control cable and other cable as required. Safeguard control circuits are to have as many channels as required for the minimum safeguard function. The cables in the main circuits are as follows:

(1) Motor Operated Valves - Two (2) channels for the redundant valves. All valves cables from the redundant Motor Control Center will be in one channel and all valves cables from the other Motor Control Center will be in the second channel.

(2) Solenoid Valves - Two (2) channels will be furnished where required for redundant valves and safeguards. Solenoid valve circuits and controls will be supplied from redundant D.C. sources of power.

(3) Motor Control - Motor control cabling is to be run in as many physical channels as required for the minimum safeguard function.

3. Instrumentation Cabling

a. Instrumentation cables will be separated into four (4) basic channels. These groups of cables will be for systems of 65 volts and less and will include twisted shielded pair and quad cables.

The instrumentation circuits will be separated to accomplish the minimum safeguard function, and to prevent loss of more than one complete section of a system from a single fault or accident.

4. General

a. All cables assigned to separation channels will be run in one assigned channel and will be so designated from the beginning of the cable to the final termination.

b. Minimum physical separation between safety feature instrumentation, control and power cable channels will be one foot where runs are made in trays or without separation where channels are run in individual metal conduits. At spreading areas near terminations the separation may consist of a metal barrier.

c. Minimum physical separation between redundant cables will be a metal barrier.

d. Where power circuits are located below protective circuits in open cable trays, fire barriers will be provided beneath the protective circuit trays unless the vertical separation is greater than three (3) feet.

e. A fire detection system utilizing the ionization type detectors will be provided in the cable tunnel and other unattended areas where large concentrations of cables exist.

5. Connecting Tubing and Transmitters

a. The connecting tubing installation from within containment measurement locations on piping systems and vessels to the transmitters will also provide for the minimum safeguard function. Each of the redundant channels will be physically separated and protected from the others and from externally generated missiles.

Each stainless steel tube is routed in 3" structural channel. Where the line is located outdoors, a pre-insulated, pre-traced tubing bundle is used. Each tubing channel within the reactor compartment is routed separately, located as far as possible from each of the other redundant channels, so as to prevent simultaneously damage to all channels from an internally generated missile. Where structural channels must be located closer than 12", they will be arranged to provide dual barriers or additional barriers will be provided.

b. Transmitters for converting mechanical measurements into electrical signals are located on structural steel racks.

Transmitters within the containment building are mounted on racks outside of the Radiation Barrier in the annulus so that maintenance can be performed during power operation.

Each transmitter and its respective piping and wiring is separated from adjacent transmitters by a steel plate barrier. The rack manufacturer has also physically protected each instrument cable by separate conduit installation.

6. Control Room Instrument Racks

a. Within the control room, redundant receivers, bistables, relays, etc. are separated into channel sets. The channel sets are physically separated by space (aisles), metal barriers, or other equipment.

Cable Tray Loading

Physical loading of cable trays is carefully controlled by means of a conduit and cable schedule computer program. Program inputs include limits as recommended by cable manufacturers and those limits obtained thru good engineering practice and experience. Input data is additionally checked for errors before processing by means of a supplemental program.

Electrical loading and consequent heat dissipation of cables in ladder trays throughout will be carefully studied and controlled to insure no overloading or excess heating. The criteria for electrical loading has been developed using IPCEA standards, manufacturer's recommendations, and good engineering practice.

Since the installation will contain several different sizes of cable installed on open ladder racks with no spacing between cables, there is no existing industry standard current rating table which applies directly. In making this type of calculation, it has been the custom to use published ampacity tables (all based upon calbes of the same size) and to analyze and extrapolate this data using sound engineering principles to fit as nearly as possible the actual conditions we have. It was on this basis that the following method of determining current ratings for cables was developed.

The foundation for all calculations is given in IPCEA Publication P-46-426, "Power Cable Ampactities-Copper Conductors", section D.2 of the introduction. Instructions are given as to how to treat cables on latter supports where there is no spacing. The procedure is:

> "For single conductor shielded or non-shielded, 3-conductor triplex-shielded or non-shielded, and 3-conductor shielded cables, apply the appropriate factor from Table VIII to the ampacity of a 3-conductor shielded cable of the same conductor size, operating temperature, and voltage rating given in the tables for cables in air."

When applying this criteria to a single conductor non-shielded cable, a conservative ampacity rating will be obtained. This is because the mutual heating between conductors and the additional (I^2R) heat losses due to circulating currents in the shield must be compensated for by reducing the ampacity of the cable. Following the criteria, this reduction is also applied to single conductor non-shielded cables, giving a factor of safety with regard to ampacity.

Table VIII of the IPCEA Publication P-46-246 gives derating factors for cables without maintained spacing:

Total Number of Conductors	Grouping Derating Factor
3	1.00
4-6	0.80
7-9	0.70
10-24*	0.70
25-42*	0.60
43 and up*	0.50

*These factors include the effects of load diversity

The IPCEA ampacities for three conductor insulated cables in air at 40°C ambient and 75°C conductor temperature are found in P-46-426. Using the voltage rating column as recommended by the cable supplier, we obtain the following ratings:

Size	Ampacity
350 MCM	372 amps
500 MCM	547 amps
750 MCM	568 amps

These ampacities must be corrected further because the IPCEA ampacity tables (and the grouping derating factor) are based upon all cables in the rack being the same size. This correction is determined by the variation and number of different size cables in a tray. It takes into account the fact that the larger size cables are dissipating more (I^2R) heat than would be dissipated by the average size conductor in the tray, while the smaller sizes are dissipating less. Consequently, the larger cables can be slightly increased in rating, while the smaller sizes must be derated even more. Applying these additional factors to the ampacities for cables in a tray with no maintained spacing, we arrive at a formula for allowable loading of cables in ladder racks:

Allowable loading of a single cable in tray with varied size cables and no maintained spacing	=	Ampacity of 3/c shielded cable in air of same size, operating temp., and voltage rating	X	Grouping derating factor	X	Size variation factor

The calculations will be made and compared against actual loadings for all power cable in trays in the control room, electrical tunnel, PA Building, and Turbine Hall.

TABLE 1

EMERGENCY DIESEL GENERATOR LOSS OF
COOLANT ACCIDENT LOADS
2 OF 3 DIESELS RUNNING

Component	Quantity	Rated hp, Each	Injection Phase≈15 min. Quantity Required	Injection Phase≈15 min. kw	Recirculation Phase Quantity Required	Recirculation Phase 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

TABLE 1 (CONT'D)

Component	Quantity	Rated hp, Each	Injection Phase≈15 min. Quality Required	Injection Phase≈15 min. kw	Recirculation Phase Quantity Required	Recirculation Phase kw
Battery Charger	2	30	0	0	1	25
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
Auxiliary Building Fans	2	30	0	0	1	25
				3156		3497*

* 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.

ITEM 17 (Attachment F Question 2.0)

Confirm that a fault on any bus will lock out all possible sources of power to that bus until the fault is cleared, and that lockouts will be provided for bus ties to prevent any two power sources from being tied together. Confirm that circuit breakers connecting Emergency Generators 1, 2, and 3 to buses 5A, 2A, 3A, and 6A will not close if there is a voltage on the bus from any source that is not synchronized, and that Emergency Generators that are synchronized with power from Station Auxiliary Transformer supply are left connected to that power only as long as necessary for load testing purposes.

ANSWER

Interlocks are provided so that a fault on any bus will lock out all possible sources of power to that bus until the fault is cleared and so that no two power sources will be tied together.

One emergency diesel-generator set is connected to bus 5A, one to 6A and the other to buses 2A and 3A. Each set will be automatically started upon under-voltage on one of the 6900 volt buses associated with outside power.

Interlocks are provided to prevent circuit breakers connecting emergency diesel-generator 1, 2 and 3 to buses 2A, 3A, 5A and 6A from closing if there is a voltage on the bus from any source that is not synchronized.

During load testing the emergency diesel-generators are left connected to the Station Auxiliary Transformer supply only as long as required to complete the test.

The 480 volt system and the emergency diesel generators connected thereto are designed to supply power to and start the minimum engineered safeguards electrical equipment even though the system has experienced a single fault or failure including the failure of one of the 3 diesels to start. In addition, the system provides for the failure to start if any one or all of the redundant injection phase loads outlined in Table 1 of Item 17 (F-1.0).

An analysis of the system and schemes required to accomplish this additional failure feature shows that the principle system element involved has to do with the closing of the two tie breakers 3A - 6A and 2A - 5A. Essentially all of the other control features are required to assure starting the minimum safeguards equipment under the single failure criteria. A failure analysis need therefore only concern the control of these breakers to show that the design does not violate the single failure criteria.

The close circuit of each of the breakers is supervised or actuated by the following contacts or devices: (Each category is in series and must be closed to actuate or close the breaker.)

A. Redundant (parallel) master tie relay (2-1/X1) or (2-11/X1) which detects loss of voltage on the 6900 volt buses and the need for safety injection or boiler feed water. The coil circuit is supervised by timers which in turn detect presence of voltage from the three diesel generators. If three generator voltages are present the timer does not pick up and the master tie relays in turn are not energized. If two or less generator voltages are detected for a preselected time the master tie relay is energized and the tie breakers are closed through the following interlocks.

B. Bus 2A and Bus 5A (bus 3A and bus 6A) undervoltage relays. (redundant) Either one of the busses to be tied together must be deenergized to permit closing of the tie breaker.

C. Bus 2A fault detector relay. (86/2A) Detects and latches in on occurance of bus fault as actuated by the engine generator or station service transformer breaker short circuit detector devices.

D. Bus 5A fault detector relay. (86/5A) Same as C above except actuated by breakers associated with bus 5A.

E. Tie breaker overload or short circuit detector. Contacts latch open.

F. Tie breaker trip bar switch. Lock out coil must be energized to close this switch.

Since all control actuation elements of the tie breakers are redundant an examination of a failure mode which would close the breakers and thus jeopardize two emergency sources of power would envolve of review of the single or simulataneous failure that could cause completion of the control circuitry.

To energize the category A relays above through failure:

1. A master safety injection relay or boiler feed requirement relay would have to close correctly or fail closed plus:

2. Two out of 3 voltage ralays on either bus 5 or bus 6 would have to close thus loss of voltage or fail closed plus:

3. 3 diesels would have to start and anyone of the voltage relays on the diesel output fail to pick up or:

4. Sequence relay (3-1) would have to fail to pick up thus permitting the timer to time out and close a circuit to the master tie relay (2-1/X1)

Assuming the muliple failures outlined above occur, the category B condition must also be satisfied to permit closing of the tie breaker. If diesel generators are connected to say bus 2A and 5A the voltage relays (catagory B) connected to the respective busses would detect the presence of voltage and would block closing of the tie breaker. Thus a multiple failure of category A relays plus 3 diesel generators connected to the busses plus a failure of one of category B relays would be required to initiate unintentional closing of a tie breaker. Should the breaker close it is very likely that category E devices would trip the tie breaker and lock it open.

The tie breakers as well as the generator and load breakers are of the trip free variety which means that with simulataneous close and trip signals present the breaker will trip and lock open. Contacts of category A, B and C relays are also added to the trip circuitry of the tie breakers to assure that they are tripped at the beginning of sequencing and to act as secondary interlocking to prevent unintentional tie breaker closure. Thus still another failure would be required to permit unintentional closure.

Tie breakers are also supplied with a lockout device which thru spring action actuates the trip bar (prevents electrical or manual closure of the breaker) unless the device is energized. Thus two breaker coils, one closing, and one lockout, must be energized to close the breaker. Manual closure at the breaker cannot be executed unless the electric close interlocking is satisfied.

ITEM 17 (Attachment F Question 3.0)

Evaluate the ability of the system to supply power to safety loads under accident conditions with a loss of outside power, and with any single fault or failure in the d-c system. Additional information is needed, including diagrams, of the 125V d-c system and the 120V a-c instrument supply system. This information should include assurance that buses, batteries, and inverter sets are physically separated so that a single accident could not take out both sources of supply to controls, instruments, and other important loads.

ANSWER

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.

3. Three diesel generator sets are connected to the engineered safety features buses to supply emergency shutdown power in the event of loss of all other a-c auxiliary power.

4. Emergency power supply for vital instruments and control and supplied for emergency lighting is from the two 125 Volt d-c station batteries.

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.

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 d-c power will be available for starting the emergency generators and other emergency uses.

The d-c system is redundant from battery source to actuation devices which are powered from the batteries. Two batteries feed two main distribution power panels which in turn feed two relay and instrument d-c buses. Safeguards relays and safety devices which use d-c as a power source are redundant and receive their power from one of the two sources. Safegurads pump starters which are d-c actuated receive power through automatic throw-over equipments which can receive power from either d-c source.

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.

The physical locations of electrical distribution system equipment is such as to minimize vulnerability of vital circuits to physical damage and prevent concurrent loss of all auxiliary power as a result of accidents.

Diagrams of the 125 V d-c and 120 V a-c systems are shown in Figures 1 and 2, respectively.

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.

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.

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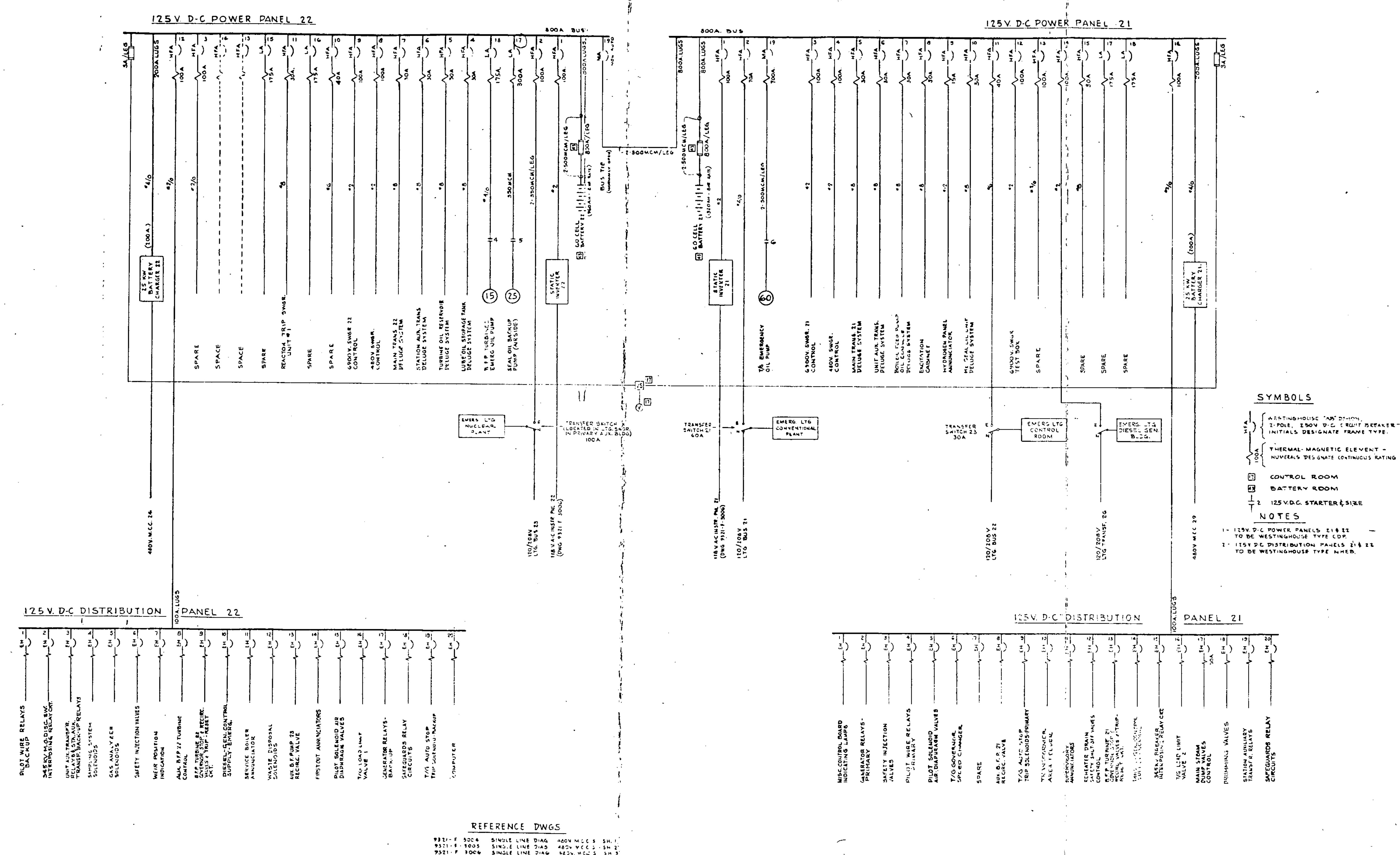
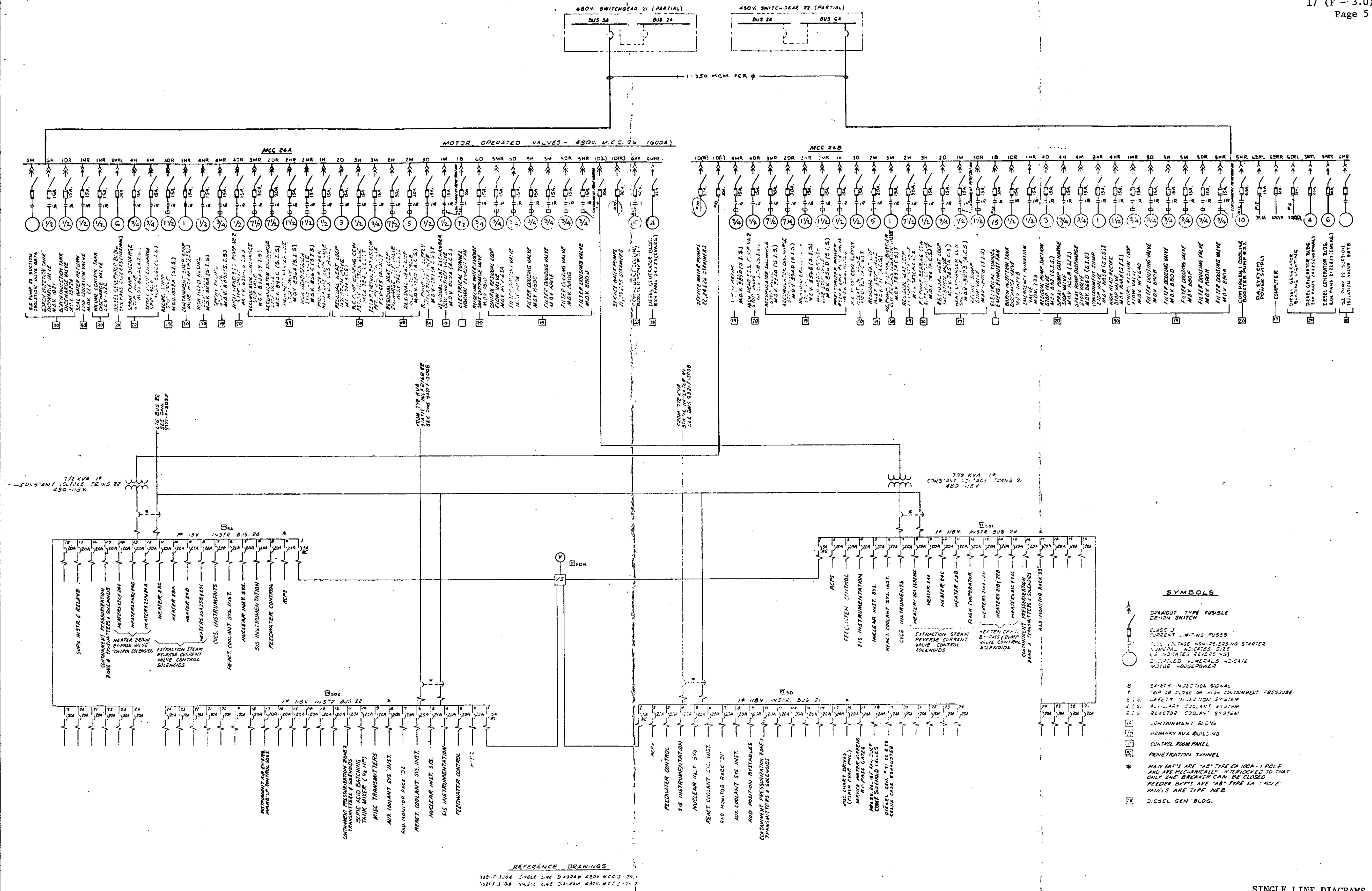


SINGLE LINE DIAGRAM
D-C SYSTEM
FIGURE 1

Supplement 1



SINGLE LINE DIAGRAMS
480V MOTOR CONTROL CENTER AND
INSTRUMENT BUSES

FIGURE 2

Supplement 1

ITEM 17 (Attachment F Question 4)

Evaluate the ability to supply electric power from the incoming power lines to the engineered safety features. Include, as a minimum, the effect of sudden trip of the unit, fault on the incoming lines, fault or equipment failure in the Buchanan substation, or fault or equipment failure within the plant. Supply pertinent statistics showing the ability of the system to withstand the tripping of large units.

ANSWER

The incoming 138 KV line from Buchanan Substation feeds two 6900 volt buses within the plant. Each of these buses feeds a 2000/2666 KVA transformer which in turn is the normal source for a safeguards power bus. Two transformers have adequate capacity to run the complete minimum safeguards systems. The signal which generates the initiation of engineered safeguards also trips the turbine generator. On any trip of the unit, four 6900 volt buses are automatically transferred to the outside source of power and two more 2000/2666 KVA transformers are thereby connected to serve safeguards buses. Connections and ties are such that a failure of any one circuit or supply bus or component will not prevent the four safeguards power buses from being fed from outside power through at least three 6900 - 480 volt transformers.

The failure of the incoming lines, or an equipment failure at Buchanan substation will appear at the plant as a loss of outside power and the onsite diesel supply system will function as described above.

Item 17 (Attachment F Question 5.0)

Evaluate the ability of all electrical components required for safety to withstand the accident environment. Include an identification of the equipment, and an estimation of the length of time that each piece of equipment must function. Discuss those special design provisions which enable the motors, valves, wiring, and any other components to function in the accident environment.

ANSWER

The totally enclosed, internally cooled recirculation pump motors are designed to operate in an ambient condition (external to the motor) of saturated steam of 271°F and 47 psig pressure for one day, followed by unattended operation of 155°F and 5 psig in a steam atmosphere for one year.

Those motors for the valves inside containment that must operate following the loss-of-coolant accident are designed so that the valves can perform the required function during the recovery period.

Some of the motors are provided only to drive engineered safety features equipment. Although these motors are normally run only for test, the design loading and temperature rise limits are based on accident conditions. Normal design margins are specified for these motors to make sure the expected lifetime includes allowance for the occurrence of accident conditions.

Should suitable equipment not be available, the detail plant design will incorporate features to modify the equipment environment to be compatible with the equipment. An example of this is the requirement to operate continuously a large induction motor during accident and post-accident conditons where a totally enclosed motor with a water to air cooler will be used. With this design, the high temperature air will be cooled before entering the motor and the winding insulation will not be subjected to high temperature.

The Containment Air Recirculation Cooling and Filtration System is required to possess sufficient margin to withstand an over-rated condition of 70.5 psig and 298°F for one hour without loss of operability. No specific criteria for heat removal or air cleaning capability are applied at the over-rated condition. The equipment is designed to operate at the post-accident conditions at 47 psig and 271°F for three hours, followed by operation in an air-steam atmosphere at 47 psig and 271°F for three hours, followed by operation in an air-steam atmosphere at 20 psig, 219°F for an additional 21 hours. The equipment design will permit subsequent operation in an air-steam atmosphere at 5 psig, 152°F for an indefinite period.

All components are capable of withstanding or are protected from differential pressure which may occur during the rapid pressure rise to 47 psig in ten (10) seconds. The materials of construction are suitable for use in sodium hydroxide and mild boric acid solutions, such as stainless steel or equivalent corrosion resistant material.

Table 1 gives an estimation of the length of time essential pieces of equipment in the primary containment are expected to operate under accident conditions.

TABLE 1

EXPECTED TIMES ELECTRICAL COMPONENTS MUST FUNCTION

Safety injection system and containment isolation actuation sensors (first five minutes after accident).

Safety injection system motor operated valves and flow instrumentation (first five minutes after accident).

Accumulator level instrumentation (first five minutes after accident).

Containment sump level instrumentation (three hours).

Air and motor operated containment isolation valves (operation completed in first five minutes after accident).

Power and instrumentation cables for the above listed equipment.

Item 17 (Attachment F, Question 6.0)

We believe the component cooling water and the service water systems should be designed to accommodate a single failure, either active or passive, as a means of insuring long-term function following a loss-of-coolant accident. Describe and discuss the modifications required to meet this goal.

ANSWER

In order to insure the long term function of the component cooling water system following a loss-of-coolant accident the component cooling system and service water system have been designed to accommodate a single failure which may be either active or passive.

The component cooling water system is to be provided with two main loop headers. The loads are divided between the two headers such that each header is capable of supplying the necessary service to enable continued containment sump/core recirculation following a loss-of-coolant accident i.e., one residual heat removal heat exchanger, one recirculation pump, one residual heat removal pump and at least one high head safety injection pump to each loop header.

In normal operation the two loop headers are fed via a common pump/heat exchanger system. During recirculation following a loss-of-coolant accident the two headers are operated independently as two separate component cooling water systems. Isolation valves are provided to enable each loop to be isolated at the pump/heat exchanger system so that each loop is finished with an independent pump/heat exchanger unit.

Similarly provision is made to enable each of the component cooling water heat exchangers to be fed from independent service water pump/header systems.

Item 17 (Attachment F Question 7.0)

Provide a detailed diagram of the steam supply system line which leads from the main steam header to the auxiliary feedwater pumps. Evaluate the consequences of a single failure in the steam supply to these pumps.

ANSWER

Figure 1 shows typical steam supply to auxiliary feed pump. The Auxiliary Feed System is provided with three feed pumps, two motor driven and one turbine driven. The turbine driven pump is twice the capacity of a motor driven pump. One motor driven pump has sufficient capacity to maintain a water inventory in the associated steam generators and prevent relief through the primary cycle pressurizer relief valve following a reactor turbine trip.

Steam is obtained from two separate steam mains ahead of the main isolation valve. Each supply is provided with a stop check valve suitable for local isolation.

Steam is ensured for the turbine under all conditions except

1) A failure in the single supply to the turbine. This would be a break in the line anywhere downstream of the stop check valves.

2) A failure of the pressure relief valve at low pressure in the steam generators. This valve is a conventional spring load valve.

The system has been designed to ensure a steam supply in the event of:

1) A failure of one steam main upstream of the stop check valves. The stop check valves prevent flow to the broken main.

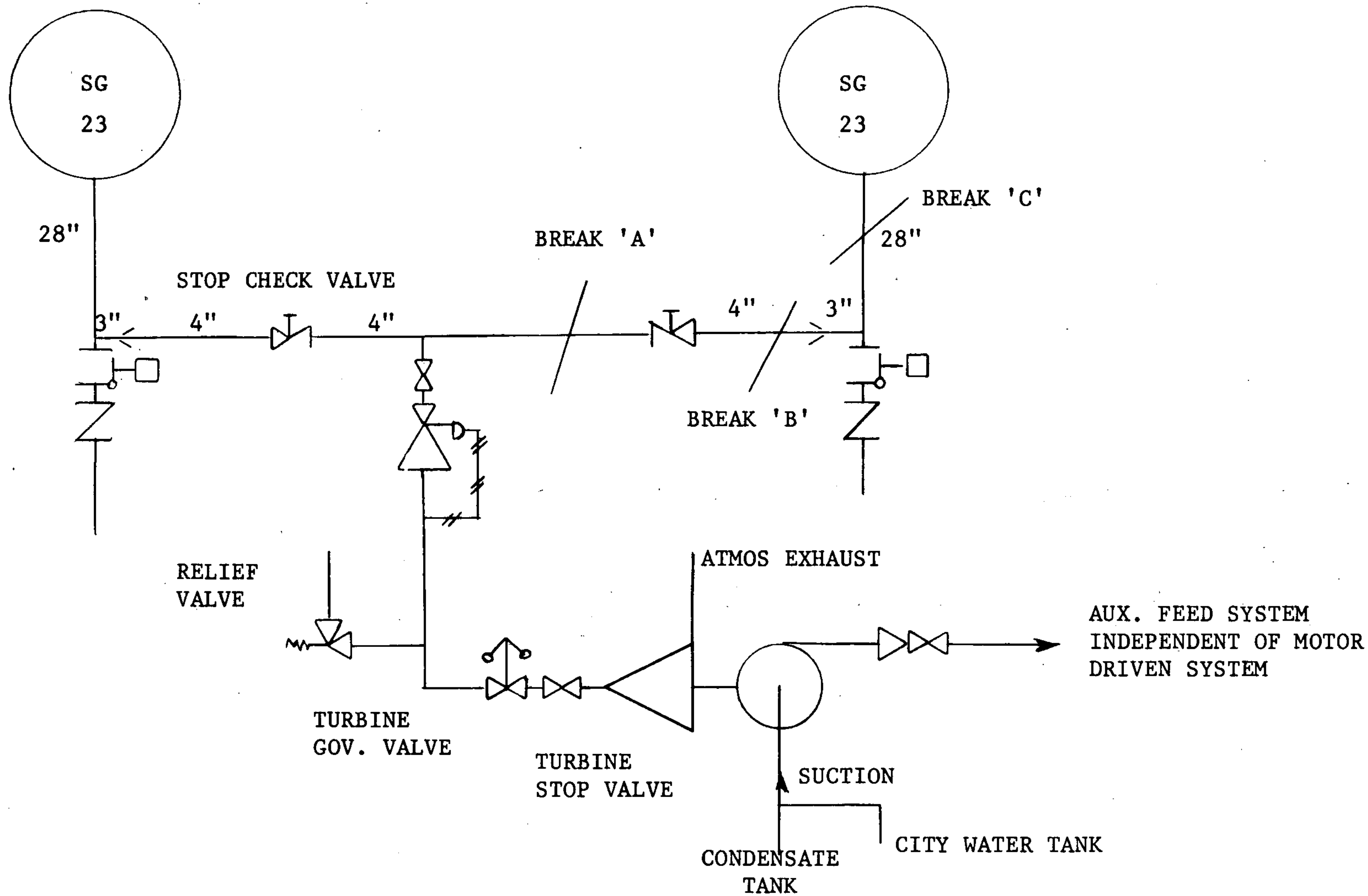
2) Failure of the pressure reducing valve. This valve fails open on loss of control (either through loss of electrical power or air supply). The relief valve is designed to ensure sufficient steam flow through the steam system while maintaining a safe pressure at the turbine. This applies at safety valve pressure in the steam generator.

3) Protection has been provided to the steam system in the event of a failure downstream of the stop check valves by limiting the take off points at the steam main to 3" nominal pipe size. This restricts the consequences of the rupture of this pipe to a release of steam less severe than that resulting from the sticking open of a safety valve. The latter case is discussed in Item 16.

In the remote chance of loss of the turbine driven pump then one motor pump is adequate to ensure the public safety.

FIGURE 1

Supplement 1



STEAM SUPPLY TO AUX. FEED PUMP

Item 17 (Attachment F, Question 8.0)

Describe the pressure relief protection provided for the gas decay tanks associated with the Waste Disposal System.

ANSWER

Each of the gas decay tanks is provided with a code relief valve. Valves are soft seated to minimize any tendency to leak radioactive gas across the seat, and bellows sealed to eliminate leakage through the top works. Piping is arranged so that there is always an open flow path from the tank to the relief valve and from the relief valve to the plant vent. Any flow through the relief valve is discharged from the site through the plant vent, and passes through the same filtering and monitoring systems that normal controlled gaseous discharges do. Relief valve set pressure is well above the normal maximum operating pressure which should further minimize any tendency to leak across the seat.

Item 17 (Attachment F, Question 9.0)

Describe the ventilation provisions provided for the areas near the waste storage tanks and the engineered safety feature equipment.

ANSWER

WASTE STORAGE TANKS	ENGINEERED SAFETY FEATURES
Waste Holdup Tank	Safety Injection Pumps
Laundry Tank	Containment Sump Pumps
Reactor Coolant Drain Tank	Residual Heat Removal Pumps
Chemical Drain Tank	
Waste Condensate Tank	
Spent Resin Storage Tank	
Gas Decay Tank	

The room containing the waste storage tank and the engineered safety features equipment are ventilated by the Primary Auxiliary Building ventilation system. This system maintains a partial vacuum in the rooms by operation of the building exhaust system, which draws air from the rooms through a closed ductwork system and directs it through HEPA filters before discharge to the plant vent. Two exhaust fans (30 hp) can be powered by the diesel. Makeup air flows into the rooms from the corridor and open areas of the building, maintaining the general flow from regions of low radioactivity levels progressively toward regions of higher radioactivity levels. The air flow through each room is adequate for removal of heat from the equipment located in the room. There is one supply fan for the auxiliary building which supplies approximately 48,000 CFM of offsite air.

ITEM 18

The following information is required for the completion of our review of your application for construction permit for Indian Point Nuclear Generating Unit No. 3. This matter was stated in the AEC letter of July 1, 1968 from Peter A. Morris to Consolidated Edison of New York.

1. Please describe and analyze those features of the Indian Point Unit No. 3 protection instrument system which differ from those of the Diablo Canyon design. Include the instrumentation which initiates engineered safety feature, containment isolation, pressure reduction, and air cleaning systems.

2. The ACRS "Report on the Pacific Gas and Electric Company Nuclear Unit - Diablo Canyon Site," dated December 20, 1967, expressed concern with respect to (a) control rod position indication, and (b) separation of certain control and safety functions. Please address these two items by describing and analyzing any design changes you may wish to accomplish, or by justifying your present position.

3. Provide an analysis of the flooding potential of the Indian Point site considering both maximum probable hurricane storm surges and flooding from excessive rainfall or dam failures upriver.

4. Considering maximum flooding elevations at the site, analyze the protection requirements of these various critical plant structures, system, or components required for safe shutdown of the facility or for the operation of the consequence limiting engineered safety features following the occurrence of the design basis loss-of-coolant accident.

ANSWER

1. The Indian Point Unit No. 3 protection instrument system is of the same design as that of Pacific Gas and Electric Company Nuclear Unit -- Diablo Canyon Site.

2. (a) The control rod position indication for Indian Point Unit No. 3 will consist of individual readouts for each control rod.

   (b) The protection system design proposed is in accordance with the IEEE criteria such that no single failure will prevent protective action or cause spurious tripping of the reactor at power. This includes design criteria for the use of instrumentation for both control and protection. The design also reflects

an evolution in design established from extensive evaluation of the normal operating and postulated accident conditions in Westinghouse PWR plants.

Consolidated Edison is cognizant of the concern expressed by the ACRS regarding separation of control and protection instrumentation as noted in the ACRS "Report on the Pacific Gas and Electric Company Nuclear Unit - Diablo Canyon Site," dated December 20, 1967, and also of the ACRS concern relative to common mode failures as expressed in the ACRS "Report on Public Service Electric and Gas Company - Salem Nuclear Generating Station," dated June 21, 1968.

The Indian Point design does not have complete separation of control and protection but rather makes effective use of all process signals for both control and protection thereby achieving a high degree of functional diversity. The proposed design with respect to the above mentioned control and safety functions is currently being reviewed on a general basis with the AEC Reactor Technology Branch.

This review is being directed toward determining the balance of redundancy, separation, functional diversity, equipment diversity, surveillance and qualification testing. Westinghouse is participating in this detailed review and is preparing additional analyses in support of the review areas noted above. The final design will reflect the results of this detailed review.

3. In the hydrology section of the PSAR for both Units 2 and 3, the consultants, Metcalf and Eddy, concluded that flooding at the site is a highly unlikely possibility.

The potential flood from an upstream dam failure poses no threat to Indian Point. The only dam on the Hudson River is a navigational lock and dam located at Troy, N. Y., approximately 100 miles upstream. There is only a 14' head behind this dam and any release of this water would have an insignificant effect on the water level at Indian Point.

The potential flood from an excessive rainfall does not appear to present a threat to the plant site. No calculations have been made for a maximum probable rainfall in this area, but it is relevant to note that the high runoff in the Hudson River Basin during the Spring months causes only an one-half foot increase in the normal 3' tidal range at Indian Point. In view of the fact that the mean high water is at elevation +2.2' and the flood control elevation of the plant is +13.9', the flooding of Unit 3 due to excessive runoff would be highly improbable.*

Since the elevation of the Hudson River is determined primarily by the tides, the most probable cause of flooding at Indian Point would come from an unusual tidal condition, such as a tidal surge during a severe hurricane. The extreme high water mark of 7.4' was recorded on November 25, 1950 during an exceptionally severe hurricane.

In an attempt to postulate the maximum probable tidal surge at Indian Point, reference is made to a hurricane study of New York Harbor prepared by Samuel Gofseyeff and Frank L. Panuzio of the Army Corps of Engineers. This study, which appeared in the Journal of the Waterways and Harbors Division, Proceedings of the American Society of Civil Engineers, is a report on the history of every major storm in New York Harbor since 1635. The report includes a prediction of the maximum probable hurricane that could hit the New York Harbor.

Although the report does not include the Hudson River above New York City, a reasonable approximation of the rise in the river level at Indian Point due to the tidal surges in New York Harbor discussed in the report, can be obtained by multiplying the value range at each location. Therefore, using the Battery as a reference point, the ratio of the surge in the harbor to the surge at Indian Point would be 4.4 to 3.

*NOTE: All elevations referred to in this report use the mean sea level elevation at Sandy Nook as the datum point.

The three most severe hurricanes to hit New York Harbor, September 21, 1821; November 25, 1950; and September 12, 1960, produced tidal surges at the Battery of 11', 8.1' and 6.3' respectively. Accordingly these surges would appear as 7.5', 5.5' and 4.3' surges at Indian Point. The 5.5' surge predicted for the November 25, 1950 hurricane agrees with the actual surge that produced the 7.4' high water mark recorded for Indian Point on that date.

The maximum probable hurricane predicted for New York Harbor would produce a 15.3' surge at the Battery. Using the 3 to 4.4 ratio, the corresponding surge at Indian Point would be 10.4 feet and would result in a high water mark of 13.1 feet, at maximum spring high tide of +2.7', still below the flood control elevation of +13.9'.

In veiw of the recorded history of the Hudson River and New York Harbor and the predicted maximum hurricane surge at Indian Point, the previous statement that flooding at the site is a highly unlikely possibility, still remains valid.

4. The maximum probable flood presents no danger to the Indian Point Unit No. 3 as shown in the analysis presented in Item 18, Question 3, since critical plant structures, systems and components required for safe shutdown of the facility or for operation of the consequence limiting engineered safety features are located above the flood control elevation.

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