50-280
50-281

PART B
VOLUME 4

Vepco

SURRY POWER STATION
UNITS 1 AND 2

FINAL
SAFETY ANALYSIS REPORT

VIRGINIA ELECTRIC AND POWER COMPANY

TABLE OF CONTENTS

OF

THE FINAL SAFETY ANALYSIS REPORT

1 INTRODUCTION AND SUMMARY

2 SITE

3 REACTOR

4 REACTOR COOLANT SYSTEM

5 CONTAINMENT SYSTEM

6 ENGINEERED SAFEGUARDS

7 INSTRUMENTATION AND CONTROL

8 ELECTRICAL SYSTEMS

9 AUXILIARY AND EMERGENCY SYSTEMS

TABLE OF CONTENTS

11 RADIOACTIVE WASTES AND RADIATION PROTECTION

11.1 GENERAL

Waste disposal systems are provided to separate, treat, and dispose of radioactive liquid, gaseous, and solid waste materials. The Liquid, Solid, and Gaseous Waste Disposal Systems are common to both reactor units and designed to serve both units simultaneously. These systems incorporate one or more of the following basic processes:

1. Filtration, to remove particulate matter.

2. Evaporation, to concentrate radioactive constituents into a smaller liquid volume and to separate liquid and gaseous phases.

3. Demineralization, to remove dissolved material.

4. Baling, to reduce the volume of compressible wastes.

5. Natural decay of radioactive isotopes.

6. Dilution, to reduce concentration.

Liquid, gaseous, and solid waste materials originate in the Reactor Coolant System, the auxiliary and emergency systems, the Waste Disposal System, and as a result of operation and maintenance procedures. Waste materials enter the Waste Disposal System directly from their source or via the Vent and Drain System (Section 9.7).

Adequate sampling, analysis, and monitoring of the Waste Disposal System are provided to comply with the design criteria. Process radiation monitors and flow measuring equipment are provided for surveillance of various station waste effluents and process streams to assure compliance with applicable regulations and to provide early indication of possible malfunctions and hazardous conditions.

Sufficient shielding is provided to reduce radiation to acceptable levels for normal operation and incident conditions. Allowable dose rates are based on applicable regulations, expected frequency and duration of exposure to radiation.

Area radiation monitoring equipment, health physics facilities, environmental programs, and administrative controls are provided for surveillance, control of radiation and exposure levels. These ensure radiation protection for plant personnel and the general public in accordance with applicable criteria.

Prior to Unit 1 operation, a radiological study of the environs was initiated (Section 11.3.5). It included an investigation of the background radiology relating to various forms of the aquatic and terrestrial environment. The nature and extent of the post-operational environmental survey will be determined on the basis of results from the pre-operational study.

11.2 RADIOACTIVE WASTE SYSTEMS

11.2.1 DESIGN BASES

The Waste Disposal System will be designed to produce effluents meeting the requirements of 10CFR20.

The Liquid, Solid, and Gaseous Waste Disposal Systems are common to both reactor units. Each Waste Disposal System is designed to accommodate radioactive wastes produced during simultaneous operation of the two units. Both units are assumed to be operating on a daily load follow cycle using boric acid between 100 percent and 50 percent power.

The activity levels are based on the assumption that one percent of the fuel rods in the core will have fuel cladding defects through which fission products may diffuse from the fuel pellets into the Reactor Coolant System.

The systems are also designed to **accommodate the corrosion products** originating in the Reactor Coolant System, and not removed in other systems.

11.2.2 SYSTEM DESIGN

The Waste Disposal System and Radiation Monitoring System are designed to satisfy the applicable sections of the general criteria of Section 1.4. In addition, these systems are designed to limit discharge of radioactive materials from the station so as not to exceed the limits of 10CFR20 or the suggested criteria of 10CFR100, and so as not to endanger the health of station operating personnel. Transportation of radioactive materials from the station is carried out in such a manner that will conform with all Federal, State and Local ordinances applicable.

TABLE 11.2.1

WASTE DISPOSAL SYSTEMS DESIGN DATA

Spent Resin Catch Tank	
Number	1
Capacity, gal	2,019
Design Pressure	30 in. Hg vacuum to 100 psig
Design Temperature, °F	250
Operating Pressure, psig	50
Operating Temperature, °F	120
Material	Stainless Steel Type 316L
Design Code	ASME III C
High Level Waste Drain Tank	
Number	2
Capacity Each, gal	2,390
Design Pressure, psig	25
Design Temperature, °F	200
Operating Pressure	Atmospheric
Operating Temperature, °F	120
Material	Stainless Steel Type 316
Design Code	ASME III C

TABLE 11.2.1 (Continued)

Low Level Waste Drain Tank

Number	2
Capacity Each, gal	2,874
Design Pressure, psig	25
Design Temperature, ^{o}F	200
Operating Pressure	Atmospheric
Operating Temperature, ^{o}F	120
Material	Stainless Steel Type 316
Design Code	ASME III C

Waste Disposal Evaporator Distillate Accumulator

Number	1
Capacity, gal	290
Design Pressure, psig	100
Design Temperature, ^{o}F	300
Operating Pressure, psig	15
Operating Temperature, ^{o}F	250
Material	Stainless Steel Type 304
Design Code	ASME III C

TABLE 11.2.1 (Continued)

Waste Disposal Evaporator Test Tanks

Number	2
Capacity Each, gal	548
Design Pressure, psig	25
Design Temperature, ^{o}F	200
Operating Pressure	Atmospheric
Operating Temperature, ^{o}F	140
Material	Stainless Steel Type 304
Design Code	ASME III C

Contaiminated Drains Collection Tanks

Number	2
Capacity Each, gal	1,230
Design Pressure, psig	25
Design Temperature, ^{o}F	200
Operating Pressure	Atmospheric
Operating Temperature, ^{o}F	120
Material	Stainless Steel Type 304
Design Code	ASME III C

TABLE 11.2.1 (Continued)

Contaminated Drains Filter Backwash Tank

Number	1
Capacity, gal	250
Design Pressure	Atmospheric
Design Temperature, ^{o}F	200
Operating Pressure	Atmospheric
Operating Temperature, ^{o}F	120
Material	Stainless Steel Type 304
Design Code	None required

Spent Resin Dewatering Tank

Number	1
Capacity, gal	619
Design Pressure	From 30 in. Hg vacuum to 25 psig
Design Temperature, ^{o}F	200
Operating Pressure	Atmospheric
Operating Temperature, ^{o}F	120
Material	Stainless Steel Type 316L
Design Code	ASME III C

TABLE 11.2.1 (Continued)

Waste Gas Catalytic Recombiner

Number	1		
Capacity, Feed, scfm	1.31		
Design Feed Pressure, psia	22		
Design Feed Temperature, ^{o}F	70-120		
Feed Composition, scfm	Max	Avg	Min
H_2	1.14	0.0805	0
H_2O	0.04	0.026	0
Xe	Trace		
Kr	Trace		
N_2	0.130	0.00922	0
Design Bleed Pressure, psia	14.0		
Design H_2 Bleed Concentration	0.1% Maximum		
Design Bleed Volume	10% of Feed		
Design Code	ASME III C		

TABLE 11.2.1 (Continued)

Waste Gas Decay Tanks

Number	2	
Capacity Each, ft^3	434	
Design Pressure	Outer Tank	Inner Tank
	From 30 In. Hg vacuum to 150 psig	From 30 In. Hg vacuum to 150 psig
Design Temperature, ^oF	200	200
Operating Pressure, psig	Atmospheric	115
Operating Temperature, ^oF	120	140
Material	Carbon	Stainless Steel Type 304L
Design Code	ASME VIII	ASME III C
Earthquake Design	Complies with Class I Requirements	

Waste Gas Surge Tank

Number	1
Capacity, ft^3	15.7
Design Pressure	From 30 In. Hg vacuum to 30 psig
Design Temperature, ^oF	300
Operating Pressure, psig	15
Operating Temperature, ^oF	120
Material	Stainless Steel Type 304
Design Code	ASME III C

TABLE 11.2.1 (Continued)

Low Level Waste Drain Pumps	
Number	2 (one required)
Type	Horizontal Centrifugal
Motor Horsepower, hp	3
Seal Type	Mechanical
Capacity Each, gpm	50
Head at Rated Capacity, ft	77
Design Pressure, psig	150
Materials	
Casing	Stainless Steel Type 316
Shaft	SAE 4140
Impeller	Stainless Steel Type 316
High Level Waste Drain Pumps	
Number	2 (one required)
Type	Horizontal Centrifugal
Motor Horsepower, hp	2
Seal Type	Double Mechanical
Capacity Each, gpm	12
Head at Rated Capacity, ft	60
Design Pressure, psig	150
Materials	
Pump Casing	Stainless Steel Type 316
Shaft	SAE 4140
Impeller	Stainless Steel Type 316

TABLE 11.2.1 (Continued)

Waste Disposal Evaporator Bottoms Pump

Number	1
Type	Horizontal Centrifugal
Motor Horsepower, hp	1 1/2
Seal Type	Canned
Capacity Each, gpm	10
Head at Rated Capacity, ft	44
Design Pressure, psig	150
Materials	
Pump Casing	Stainless Steel Type 316
Shaft	Stainless Steel Type 316
Impeller	Stainless Steel Type 316

Waste Disposal Evaporator Distillate Pump

Number	1
Type	Horizontal Centrifugal
Motor Horsepower, hp	1 1/2
Seal Type	Mechanical
Capacity Each, gpm	7
Head at Rated Capacity, ft	44
Design Pressure, psig	150
Materials	
Pump Casing	Stainless Steel Type 316
Shaft	SAE 4140
Impeller	Stainless Steel Type 316

TABLE 11.2.1 (Continued)

Waste Disposal Evaporator Test Tank Pumps

Number	2 (one required)
Type	Horizontal Centrifugal
Motor Horsepower, hp	5
Seal Type	Mechanical
Capacity Each, gpm	50
Head at Rated Capacity, ft	97
Design Pressure, psig	150
Materials	
Pump Casing	Stainless Steel Type 316
Shaft	SAE 4140
Impeller	Stainless Steel Type 316

Contaminated Drains Transfer Pumps

Number	2 (one required)
Type	Horizontal Centrifugal
Motor Horsepower, hp	3
Seal Type	Mechanical
Capacity Each, gpm	50
Head at Rated Capacity, ft	79
Design Pressure, psig	150
Materials	
Pump Casing	Stainless Steel Type 316
Shaft	SAE 4140
Impeller	Stainless Steel Type 316

TABLE 11.2.1 (Continued)

Spent Resin Dewatering Tank Pump

Number	1
Type	Horizontal Centrifugal
Motor Horsepower, hp	10
Seal Type	Mechanical
Capacity, gpm	100
Head at Rated Capacity, ft	144
Design Pressure, psig	375
Materials	
Pump Casing	Stainless Steel Type 316
Shaft	SAE 4140
Impeller	Stainless Steel Type 316

Waste Disposal Evaporator Circulating Pump

Number	1
Type	Horizontal Centrifugal
Motor Horsepower, hp	7 1/2
Seal Type	Double Mechanical
Capacity, gpm	700
Head at Rated Capacity, ft	13
Design Pressure, psig	400
Materials	
Pump Casing	Gould Alloy 20
Shaft	Stainless Steel Type 303
Impeller	Gould Alloy 20

TABLE 11.2.1 (Continued)

Waste Disposal Evaporator Distillate Condenser

Number	1	
Total Duty, Btu/hr	3.31×10^6	
	Shell	Tube
Design Pressure, psig	100	150
Design Temperature, °F	338	200
Operating Pressure, psig	15	75
Operating Temperature, In/Out, °F	250/250	105/131
Material	304SS	304SS
Fluid	Steam	Water
Design Code	ASME III C	ASME III C

TABLE 11.2.1 (Continued)

Waste Disposal Evaporator Reboiler

Number	1	
Total Duty, Btu/hr	4.0×10^6	
	Shell	Tube
Design Pressure, psig	200	100
Design Temperature, °F	388	350
Operating Pressure, psig	100	15
Operating Temperature, In/Out, °F	338/338	250/262
Material	Carbon Steel	Incoloy 825
Fluid	Steam	Waste Liquid with up to 12% boric acid and up to 25% total solids
Design Code	ASME VIII	ASME III C

Waste Disposal Evaporator Distillate Cooler

Number	1	
Total Duty, Btu/hr	330,000	
	Shell	Tube
Design Pressure, psig	100	100
Design Temperature, °F	300	300
Operating Pressure, psig	75	30
Operating Temperature, In/Out, °F	105/135	250/140
Material	Carbon Steel	304SS
Fluid	Water	Water
Design Code	ASME III C	ASME III C

TABLE 11.2.1 (Continued)

Waste Disposal Evaporator Bottoms Cooler

Number	1	
Total Duty, Btu/hr	$4.5x10^5$	
	Shell	Tube
Design Pressure, psig	150	100
Design Temperature, °F	350	350
Operating Pressure, psig	75	25
Operating Temperature, In/Out, °F	150/160	250/160
Material	Carbon Steel	304SS
Fluid	Water	Waste Liquid with up to 12% boric acid and up to 25% total solids
Design Code	ASME VIII	ASME III C

Process Vent Blower

Number	2 (one required)
Type	Multi-stage Centrifugal
Motor Horsepower, hp	7.5
Capacity Each, scfm	300
Differential Pressure, psi	2
Suction Pressure, psia	14.0
Design Pressure, psig	15
Materials	
Casing	Cast Iron
Impeller	Aluminum
Shaft	Stainless Steel Type 304

TABLE 11.2.1 (Continued)

Waste Disposal Distillate Filter

Number	1
Retention Size, Microns	5
Filter Element Material	Fibre
Capacity Normal, gpm	40
Capacity Maximum, gpm	75
Material	304SS
Design Pressure, psig	150
Design Temperature, ^{o}F	250
Design Code	ASME III C

Low Level Waste Drain Filter

Number	1
Retention Size, Microns	5
Filter Element Material	Fibre
Capacity Normal, gpm	50
Capacity Maximum, gpm	75
Material	Stainless Steel Type 304
Design Pressure, psig	150
Design Temperature, ^{o}F	250
Design Code	ASME III C

TABLE 11.2.1 (Continued)

High Level Waste Drain Filter

Number	1
Retention Size, Microns	6
Filter Element Material	Fibre
Capacity Normal, gpm	50
Capacity Maximum, gpm	75
Material	Stainless Steel Type 304
Design Pressure, psig	150
Design Temperature, ^{o}F	250
Design Code	ASME III C

Contaminated Drains Filters

Number	2 (one required)
Filter Element Material	Fiber
Capacity Normal, gpm	50
Capacity Maximum, gpm	75
Material	Stainless Steel Type 304
Design Pressure, psig	150
Design Temperature, ^{o}F	120
Design Code	ASME VIII

11.2.3 LIQUID WASTE DISPOSAL SYSTEM

The Liquid Waste Disposal System, as shown in Figs. 11.2.3-1, -2 and -3, receives liquid wastes, either directly from various sources or from the Vent and Drain System discussed in Section 9.7. The Vent and Drain System classifies process liquids either for re-use or for disposal. The influent from the Vent and Drain System to the Liquid Waste Disposal System is a small fraction of the Vent and Drain System throughput.

System influents from the Vent and Drain System are discharged to either the high level waste drain tanks or the low level waste drain tanks, according to influent activity level.

Laundry waste, PCA shower drains, and PCA lavatory drains are discharged to the contaminated drain tanks.

Laboratory drains and various flush lines from the drumming of concentrated liquid wastes, including spent resin flush drains, are discharged to the high level waste drain tanks.

The contents of the high level waste drain tanks, which may have activity levels in the order of up to 10^{-1} uCi/cc are processed by evaporation. The evaporator subsystem is designed to produce liquid effluents of an activity level no greater than 10^{-4} to 10^{-8} uCi/cc, which are pumped to the waste disposal test tanks. The contents of the test tanks are sampled to determine the radioactivity level and the chemical composition. The evaporator effluent in the test tanks, if contaminated, may be purified by circulating the contents through a mixed bed demineralizer and filter or reprocessed. The test tank effluent is, after confirmation that activity and chemical concentrations are below the specified maximum level, discharged directly through the liquid waste radioactivity monitoring station and flow control station. Off gas from the evaporator is vented to the Gaseous Waste Disposal System. The concentrated bottoms in the evaporator is packaged for shipment offsite and ultimate disposal.

Provisions are made for the transfer of the high level waste drain tank contents to the low level waste drain tanks, by means of a line under administrative control, in the event that the high level waste drain tank contents do not require evaporation.

Provisions are also made for conveying the contents of the low level waste drain tanks and contaminated drain tanks to the high level tanks if the activity level of any of these liquids should exceed about 10^{-3} uCi/ml. The contents of the contaminated drain tanks are filtered by the contaminated drain filters and conveyed into the liquid waste effluent header, where the effluent is monitored by a radiation monitor and ultimately discharged into the circulating water discharge canal, via steam generator blowdown piping. If the activity level of the water exceeds the normal limits, it can be pumped to the high level waste drain tanks for reprocessing.

TABLE 11.2.1 (Continued)

Waste Gas Compressor

Number	2 (one required)
Type	Diaphragm
Motor Horsepower, hp	1
Capacity Each, scfm	1.5
Discharge Pressure at Rated Capacity, psig	120
Design Pressure, psig	220
Materials	
Cylinder	Carbon Steel
Piston Rod	Forged Steel
Piston	Nodular Iron
Diaphragms and Parts Contacting Waste Gas	304/316 SS

TABLE 11.2.1 (Continued)

Waste Disposal Evaporator Distillate Demineralizers

Number	1
Design Flow, gpm/ft^2	10
Demineralizer Resin	H-OH Mixed Bed
Active Volume, cu ft	17
Design DF	15
Design Pressure, psig	200
Design Temperature, ^oF	250
Material	Stainless Steel Type 316L
Design Code	ASME III C

Spent Resin Dewatering Filter

Number	1
Retention Size, Microns	5
Filter Element Material	Fibre
Capacity Normal, gpm	100
Capacity Maximum, gpm	150
Material	Stainless Steel Type 304
Design Pressure, psig	150
Design Temperature, ^oF	250
Design Code	ASME III C

The contents of the low level waste drain tanks are discharged through filters and a radiation monitoring station to the circulating water discharge canal. If the activity level of this effluent is not within the discharge tolerance then it is pumped to the high level waste drain tanks for subsequent evaporation.

All liquid waste discharges to the Circulating Water System and is monitored as described in Section 11.2.3 to provide radiation control of this discharge. Periodic sampling of the liquid waste effluent is conducted by station Health Physics personnel. The discharge rate is controlled by either of two parallel flow control valves; one handling low range, and the other handling high range flows. Excessive activities detected by the monitor overrides both valve controls and stops all discharge flow.

The discharge flow from the Liquid Waste Disposal System is combined and mixed with the circulating water such that the net activity of the combined effluent does not exceed 10CFR20 unrestricted area limits.

11.2.3.1 Components

High Level Waste Drain Tanks

Two high level waste drain tanks will be provided. Each tank has a capacity of 2,390 gal. Level indicators are provided. These are stainless steel tanks designed according to Section III C of the Boiler and Pressure Vessel Code.

Low Level Waste Drain Tanks

Two low level waste drain tanks are provided. Each tank has a capacity of 2,874 gal. Level indicators are provided. These are stainless steel tanks designed according to Section III C of the ASME Boiler and Pressure Vessel Code.

Waste Disposal Evaporator and Auxiliaries

One externally heated forced circulation evaporator with a feed and distillate capacity of 6 gpm is provided. The evaporator shell is fabricated from a high nickel alloy in accordance with Section III C of the ASME Boiler and Pressure Vessel Code. Internals are fabricated from an austenitic stainless steel not susceptible to stress cracking.

The external heat source is a shell and tube steam reboiler fabricated on the tube side from a high nickel alloy in accordance with Section III C of the ASME Boiler and Pressure Vessel Code, and TEMA Standards, and fabricated from carbon steel on the shell side in accordance with Section III C or the ASME Boiler and Pressure Vessel Code, and TEMA Standards. Distillate is condensed in a water cooled shell and tube condenser fabricated from austenitic stainless steel in accordance with Section III C of the ASME Boiler and Pressure Vessel Code and TEMA Standards.

The condensed distillate is held in the distillate accumulator. This tank is fabricated from austenitic stainless steel in accordance with Section III C of the ASME Boiler and Pressure Vessel Code.

A distillate cooler is provided to further cool the distillate. The tube side of the distillate cooler is fabricated from austenitic stainless steel and the shell side from carbon steel in accordance with Section III C of the ASME Boiler and Pressure Vessel Code.

Waste Disposal Evaporator Test Tanks

Two waste disposal evaporator test tanks, each of 548 gal capacity, with level indicators are provided. These tanks are stainless steel and designed according to Section III C of the ASME Boiler and Pressure Vessel Code.

Waste Disposal Evaporator Demineralizer

One waste disposal evaporator demineralizer is provided for evaporator distillate polishing. The demineralizer is fabricated from austenitic stainless steel not susceptible to stress cracking in accordance with Section III C of the ASME Unfired Boiler and Pressure Vessel Code.

Waste Disposal Filters

Liquid waste effluent filters and the distillate demineralizer filters will be cartridge type pressure filters. The vessels are fabricated from austenitic stainless steel in accordance Section III C of the ASME Boiler and Pressure Vessel Code. The filter elements are of the synthetic fibre disposal type. Filter cartridges are designed for removal as a single basket assembly. Contaminated drain tank filters are provided to remove lint and other laundry waste matter which could be radioactive. This filter is operated on a precoat-filter-backwash cycle.

Pumps

Centrifugal frame mounted pumps with single or double mechanical seals are provided. The waste disposal evaporator bottoms pump is a canned pump. One pump is provided for each tank with cross ties where appropriate, such as on high level waste drain tank pumps. External cooling and seal water is supplied to radioactive pump seals as required.

11.2.4 SOLID WASTE DISPOSAL SYSTEM

The Solid Waste Disposal System provides holdup, packaging and storage facilities for the eventual shipment offsite and ultimate disposal of radioactive waste material. Materials handled as solid waste include concentrated liquid wastes from the waste disposal evaporator, concentrated boric acid not to be reused in the system from the boron recovery bottoms tank, spent resin slurries, spent filter cartridges, and other miscellaneous solid materials resulting from station operation and maintenance. The operation of this system involves various unit operations described below.

11.2.4.1 Solid Waste Handling Operations

Drumming Operation

The drumming operation involves the mixing of the concentrated solution in a drum with an absorbant and solidifier such as cement, the mixing and setting up of the mixture, and the storage of the drum for ultimate offsite disposal.

The drumming of concentrated evaporator bottoms is a batch process using cement and/or diatamaceous earth. Concentrated waste is mixed with the cement and/or diatomaceous earth as it enters the drum. The amount of waste liquid per drum is predetermined by analysis of the waste. The liquid waste is cooled in a heat exchanger prior to drumming.

The drum is transported by overhead monorail to the drum rolling machine. The concentrated waste and absorbant is mixed by means of rotating the drum. After a predetermined time, the drum is transported to a storage area.

The radioactivity level on contact with the surface of the drum is measured, recorded, and attached to the drum in accordance with the applicable Federal Regulations. The drums are stored until such time as they are to be shipped offsite for ultimate disposal.

Baling Operation

Contaminated solid material resulting from station maintenance is stored in specified areas of the auxiliary building and decontamination building. The items are placed in polyethelene bags, if required, during storage with suitable labeling.

Materials which are compressible, such as absorbant paper, cloth, rubber, and plastic are placed in 55 gal. drums. The drum and its contents, including the plastic bags, are placed in position on the solid waste baler. The compression plate compacts the contents of the drum into a high density bale. Additional compressible material is added and the contents of the drum recompacted until the drum is filled. During the baling operation the area is closed and the ventilation air filtered to remove particulate matter. Contaminated metallic materials and other highly contaminated solid objects are placed inside a cylindrical concrete core at the center of a 55 gal. drum. The annular space between the core and drum and the bottom section of the drum is concreted prior to insertion of materials inside the core. After insertion of the contaminated materials, additional concrete is added to fill all remaining void spaces inside the drum.

Spent Resin Handling Operations

A spent catch tank properly shielded is provided to accumulate resin from ion exchangers. A spent resin transfer system permits spent resin to be flushed from the catch tank either to a disposable spent resin shipping cask or to a shielded tank truck. The disposable cask with its associated concrete shielding is normally in place and ready to receive high activity spent resin, while other casks may be in shipment to or from a licensed disposal site. To ready this equipment for receipt of spent resins, the shipping container and cask are placed in the spent resin cask pit at the decontamination building, with piping connections made up in advance. Provision is made to transfer spent resin directly to the disposable shipping drum from the ion exchanger. The spent resin from all high activity ion exchangers is handled in the same manner.

The resin in an ion exchanger is considered to be "spent" when the decontamination factor drops below a predetermined value or the dose rate on the outside of the ion exchanger approaches predetermined limits. The unit is then isolated and primary grade water used to flush the spent resin into the spent resin catch tank. The spent resin remains in the catch tank and the flush liquid passes through a filter element and discharges by way of the spent resin dewatering tank and the Vent and Drain System into one of the high level waste drain tanks. When all spent resin from a single source is transferred to the holdup tank, the flushing flow is stopped. A similar procedure transfers the resin to the spent resin shipping cask. After transferring the resin to the disposable drum, the disposable drum is dewatered in its cask and stored in the yard storage area until it can be transported for ultimate disposal.

Expended Filter Cartridge Handling Operations

Filters in radioactive liquid service are removed from the service when the pressure drop across the filters becomes excessive or when the radiation level exceeds a predetermined maximum. The filter cover is opened by personnel using appropriate tools, and protected by a filter removal shield, when required. High activity expended filter cartridges in their disposable basket are raised into the filter removal shield and placed in the filter shipping container shielded by a reinforced concrete shipping cask. Filter cartridges are packaged for ultimate disposal in 55 gal. drums to the extent possible. In each case, the packaging procedure is appropriate for the prevailing conditions.

Ultimate Disposal Operations

All packages containing radioactive non-fissionable material and the procedures used to prepare these for offsite shipment are in accordance with U. S. Department of Transportation regulations. All waste material is transferred either to a licensed disposal contractor or to a common carrier for delivery to a licensed disposal contractor.

11.2.4.2 Components

Spent Resin Catch Tank

One spent resin holdup tank is provided. The holdup tank is permanently installed in a concrete cubicle below grade. The tank capacity is designed to contain the total dewatered resin volume of approximately four demineralizers. The vessel is designed according to Section III C of the ASME Code for Boiler and Pressure Vessels.

Baler

One solid waste baler for contaminated, compressible materials is provided.

Drum Roller

Two 55 gal. drum rollers are provided for use at the bottoms drumming station.

Spent Filter Shipping Cask

Concrete shielded filter shipping casks, acceptable under the regulations, and related tools are provided.

Spent Resin Dewatering Tank

One spent resin dewatering tank is provided. The tank is designed in accordance with Section III C of the ASME Code for Boiler and Pressure Vessels.

Spent Resin Disposable Shipping Drum and Cask

A number of standard 55 gal. shipping drums and casks, acceptable under the regulations, are available.

FIG. 11.2.3-1
OCT. 15, 1970.

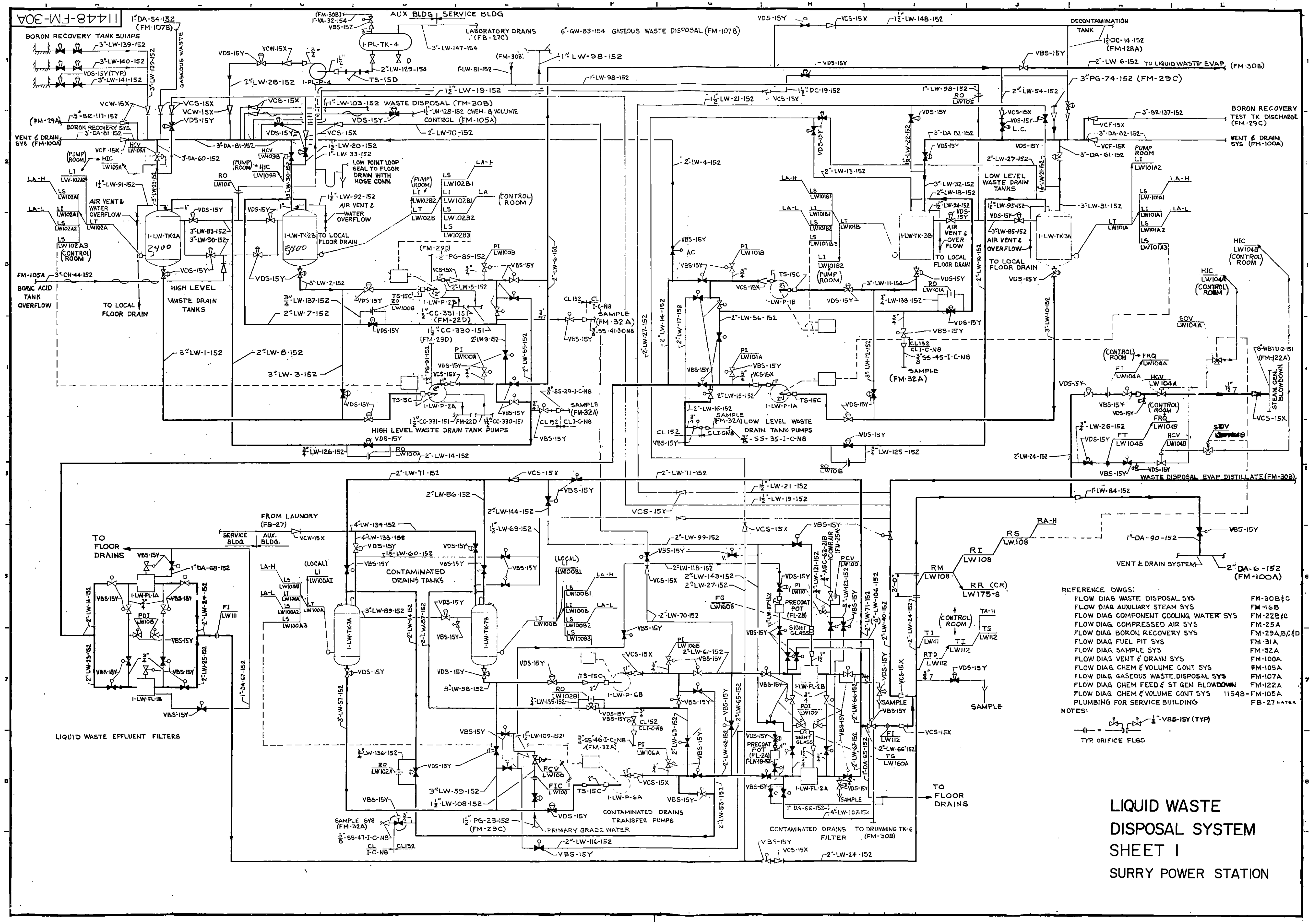
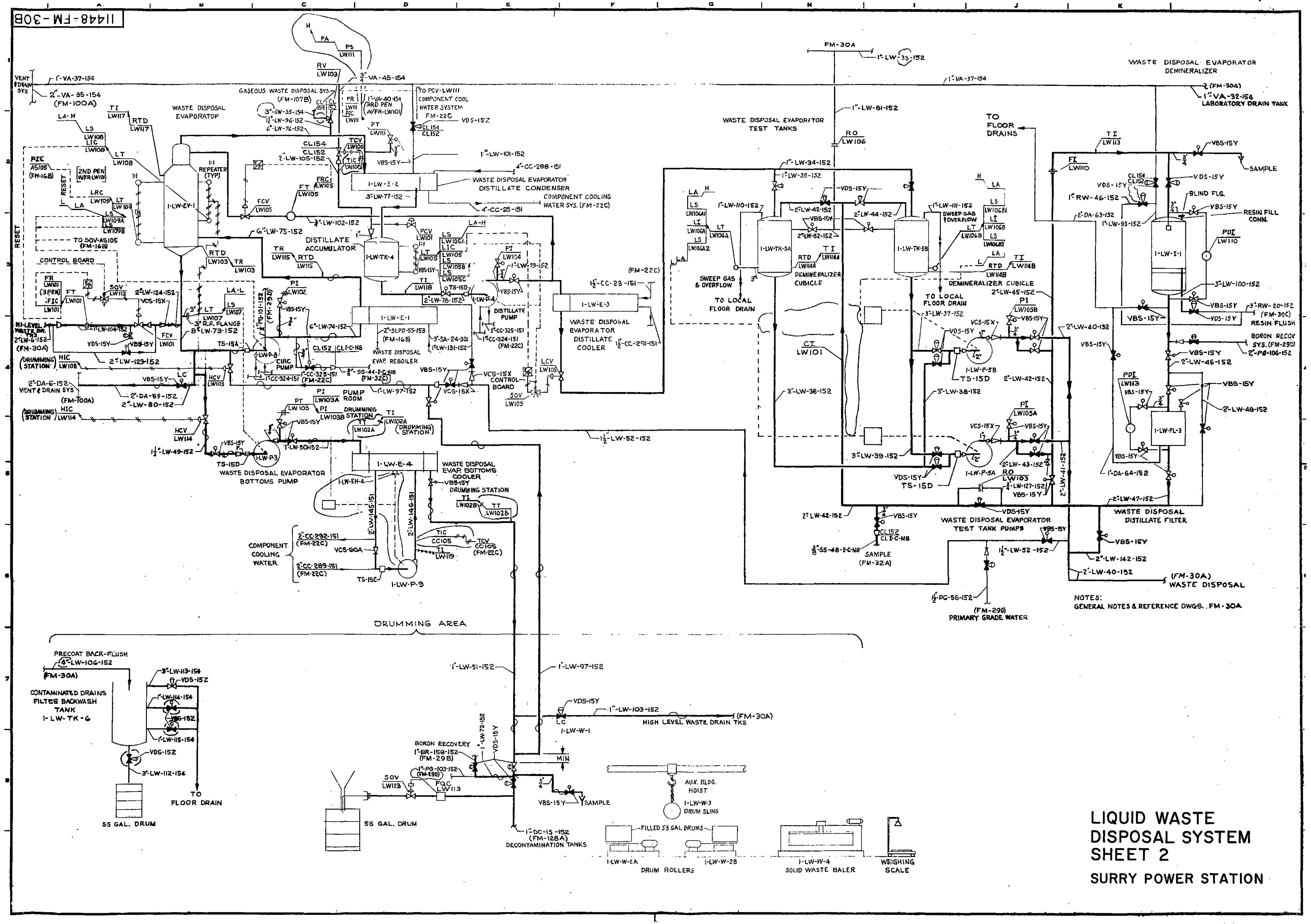
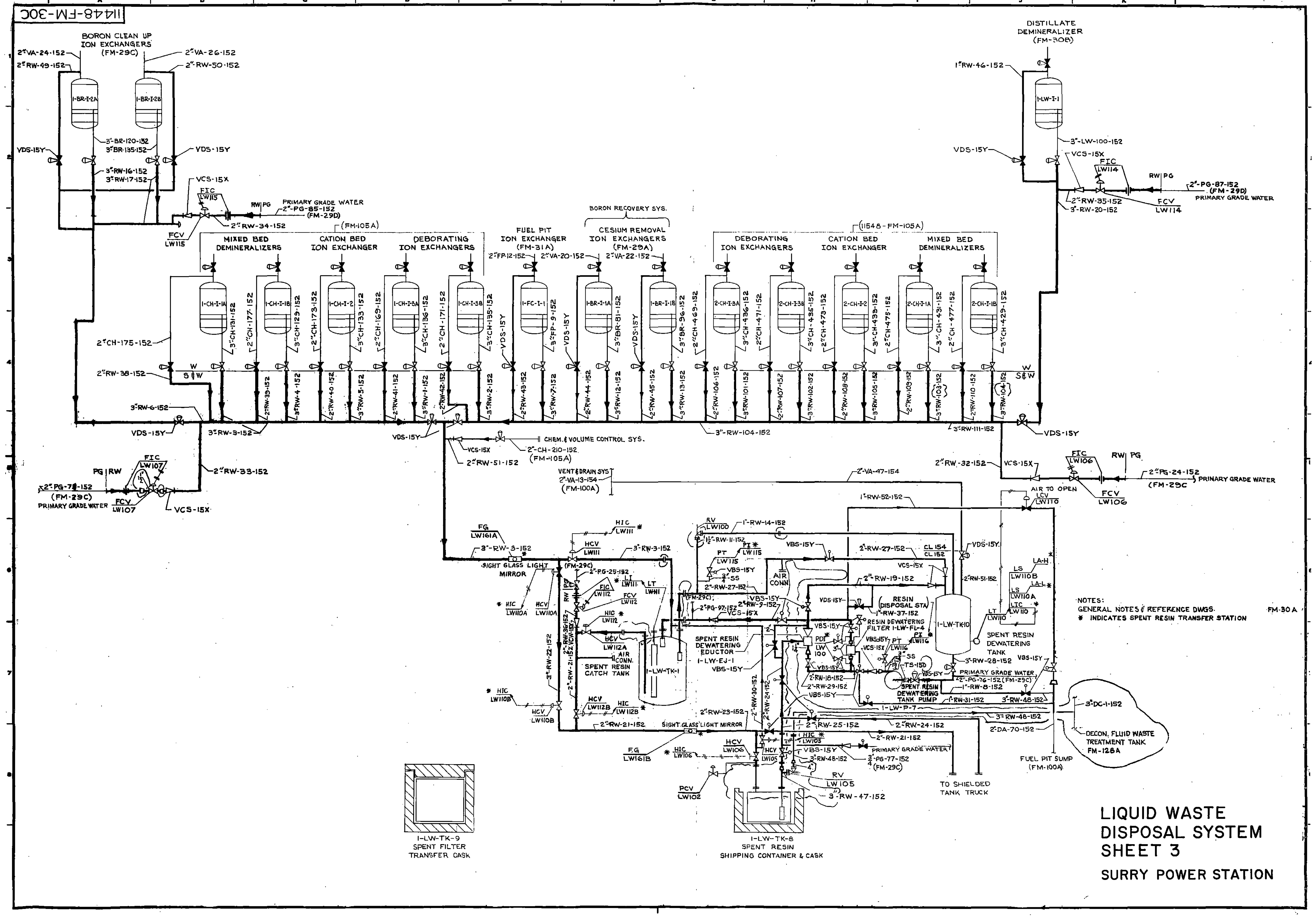


FIG. 11.2.3-2
OCT. 15, 1970





LIQUID WASTE DISPOSAL SYSTEM SHEET 3

SURRY POWER STATION

11.2.5 GASEOUS WASTE DISPOSAL SYSTEM

The Process Vent Subsystem regulates the discharge of potentially high activity waste gases to the atmosphere. The Ventilation Vent Subsystem described in Section 9.13 will regulate the discharge of potentially low activity air streams to the atmosphere. Radioactive waste discharges from both subsystems are monitored by particulate and gas monitors described in the Process Radiation Monitoring System described in Section 11.3.3.

11.2.5.1 Process Vent Subsystem

Gaseous wastes enter the Process Vent Subsystem from the Gaseous Waste Disposal System, from the stripper in the Boron Recovery System, the Vent and Drain System, various pressure relief valves and the Containment Vacuum System, as shown on Fig. 11.2.5-1, -2.

Waste gases, primarily hydrogen, nitrogen and minor amounts of fission product gases, such as xenon and krypton, are removed from reactor coolant letdown by the stripper in the Boron Recovery System. The stripped gases are processed in the Gaseous Waste Disposal System.

Before processing the stripped gases in the recombiner, nitrogen and oxygen are added as required and the entire mixture is preheated. The maximum hydrogen concentration is normally maintained at about 3 percent, which is below the lower hydrogen flammability limit of 4.4 percent. The gas mixture flows to the waste gas recombiner where about 99 percent of the hydrogen and oxygen is catalytically reacted to produce water vapor. The gas volume to the waste gas tanks from the recombiner is reduced by tenfold relative to the influent stripped gas entering from the boron recovery gas stripper.

The effluent gas from the recombiner is cooled in the recombiner after-cooler and then flows to the moisture separator. Condensed liquid from these two vessels is drained to the Liquid Waste Disposal System.

The waste gas stream from the moisture separator is recycled to the recombiners by the blowers.

The recombiners are maintained at a preset pressure by bleeding from the recycle line upstream of the blowers using pressure control. The bleed stream is pumped from a small surge tank to the buried gaseous waste decay tanks by diaphragm compressors. The redundant hydrogen analyzers on the effluent line from the moisture separator shut the valves in the bleed stream in event of a high hydrogen concentration in the effluent from the catalytic recombiner.

Duplicate oxygen analyzers on the moisture separator effluent reset the oxygen addition controller.

Redundant hydrogen analyzers on the recombiner feed stream shut the two valves in series on the stripped feed for hydrogen concentrations exceeding 4 percent. Simultaneously, redundant nitrogen purge lines open to dilute the hydrogen.

Two double-walled waste gas decay tanks are provided. Each tank is buried for tornado protection. The inner tank is fabricated from austenitic stainless steel in accordance with Section III C and the outer tank from carbon steel in accordance with Section VIII of the ASME Code for Boiler and Pressure Vessels. Sampling connections are provided for the tank contents, and for leakoff in the annular intercept space between the tanks. The decay tanks have piping connections for parallel operation with alternate feed and bleed.

Overpressure relief protection is provided at the waste gas decay tanks in accordance with Section III C of the ASME Code. The protective devices consist of bellows sealed pressure relief valves followed by rupture disc assemblies. The use of bellow seals and rupture discs preclude leakage of the waste gas to the environment during normal operation of the Gaseous Waste Disposal System. The piping downstream of the protective devices relieve to the process vent through the radiation monitor station.

Effluent from the waste gas decay tanks is mixed with dilution air, effluent from the Containment Vacuum System, and the aerated vents from the Vent and Drain System. The combined gaseous waste is filtered through charcoal filters prior to being released to the atmosphere. The process vent blowers maintain a small vacuum in the charcoal filters to prevent out-leakage from the filter assembly. The decay tank pressure relief valves discharge to the discharge side of the process vent blowers. The decay tank contents are sampled prior to any release to the process vent.

The entire discharge stream of radioactive letdown gas and dilution air is monitored for flow rate, pressure, temperature, and particulate and gaseous activity prior to release through the process vents. The total flow is regulated by a flow control valve on the dilution air. The ratio of dilution air to waste gas letdown flow is such that the mixed streams never enter the flammability region of the air, steam, hydrogen phase diagram.

The process vent and the process vent blowers are sized such that the minimum exit velocity is approximately 100 fps. This exit velocity prevents any significant downdrafting of the effluent with atmospheric winds as high as 35 mph. The process vent terminates at an elevation approximately 10 ft above the top of one of the containment structures.

The process vent monitors are set such that the effluent activity release rate results in concentrations less than those limits provided by 10CFR20 at the site boundary. In the event that the activity of the effluent stream exceeds the setting of the monitors, the process vent control station automatically terminates the release of waste effluents from the waste gas decay tanks and isolates the Containment Vacuum System from the Process Vent Subsystem. The monitor also alarms in the Main Control Room prior to valve closure if the activity approaches a preset value. Subsequent restart of the system is manual in accordance with administrative procedures. Discharge of gases from the waste gas decay tanks is initiated and controlled separately.

The Waste Gas Disposal System is designed to provide adequate radioactive decay storage time for the waste gases and, in addition, provide long term holdup of these gases when either high flow letdown is required or adverse meteorological conditions make it desirable to discontinue release of waste gas to the environment.

The combined volume of the two waste gas decay tanks is sized to process the gas stripped from the estimated annual average letdown flow of 17 gpm, based on simultaneous operation of two reactor units.

The average gas stripping rate is a function of the average letdown flow rate, and this flow rate is dependent on the assumed plan of operation as described in Section 9.2.

On this basis, the total annual letdown volume for two units is 8.22×10^6 gal and the average annual letdown flow rate for two units is 17 gpm.

The case when one unit is four weeks behind the other unit in the cycle and the first unit is in the last eight weeks of operation results in a two month average letdown rate of 17 gpm with daily load follow of the two units.

If the hydrogen volume is assumed to be 35 cc/kg and 90 percent of the total gas volume, then the hydrogen stripping rate at an annual average of 17 gpm letdown is .0792 scfm and the total gas stripping rate is .088 scfm.

The gas decay tanks are sized so that a 17 gpm letdown rate with the recombiner not operating gives an average holdup time equivalent to approximately 5 half-lives of Xe 133 (30 days).

Assuming 1 percent failed fuel, the estimated curies of each radionuclide released from the station via the Gaseous Waste Disposal System are listed in Tables 11.2.2-1 and 11.2.2-2. Table 11.2.2-1 is for a waste gas cycle with the recombiner not operating, which is the design basis for the system, and Table 11.2.2-2 is for a waste gas cycle with the recombiner operating. With the recombiner not operating the gas cycle is 30 days' feed - 20 days' decay - 10 days' bleed; most of the gas is hydrogen. With the recombiner operating, the feed portion of the waste gas cycle can vary between 30 days and approximately 330 days, the time to reach maximum design pressure in the tank. To be conservative, 330 days' feed was chosen as the basis for Table 11.2.2-2, and the waste gas cycle considered was 330 days' feed - 20 days' decay - 10 days' bleed.

In each case, it is assumed that all of the gases and 0.1 percent of the iodines are removed at the gas stripper and sent to the waste gas decay tanks. The system is operated so that one tank is on the feed portion of cycle while the other is on the decay and bleed portion.

The equilibrium reactor coolant activity is a function of the waste gas removal rate by the gas stripper. Using the parameters listed in Table 9.1-4 and a 17 gpm letdown rate to the gas stripper, the equilibrium coolant activity for each radionuclide was calculated. These are also listed in both Tables 11.2.2-1 and 11.2.2-2.

As can be seen from the Tables, the yearly dose at the site boundary is about 0.019 Rem for continuous operation with a 30 days' feed - 20 days' decay - 10 days' bleed cycle and about 0.007 Rem for the 330 days' feed - 20 days' decay - 10 days' bleed cycle. Both of these values are well below the unrestricted area dose of 0.5 Rem set forth in 10CFR20.

TABLE 11.2.2-1

ESTIMATED WASTE GAS RELEASE WITH RECOMBINER NOT OPERATING

Based upon gaseous activity from:

a. Waste gas cycle: 30 days' feed, 20 days' decay, 10 days' bleed
b. Two units, 2,546 MWt each
c. 1% failed fuel
d. 17 gpm total letdown flow rate to gas stripper from both units
e. 100% of gases and 0.1% of iodines removed in gas stripper
f. $X/Q = 7.5 \times 10^{-6}$ sec/m^3 for calculating dose at site boundary

Nuclide	Equilibrium Coolant Activity, μCi/cc	Curies in Tank at End of Feed Cycle	Curies in Tank after 20 Days Decay	Discharge Rate, Ci/Sec		Curies Discharge in One Cycle	Total Curies Discharged Per Year	Dose at Site Boundary, Rem Per Year
				Initial	Final			
Kr 85 m	1.22	3.63×10^{1}	2.05×10^{-35}	2.14×10^{-41}	1.61×10^{-59}	4.44×10^{-37}	5.40×10^{-36}	6.4×10^{-42}
Kr 85	3.25×10^{-1}	1.25×10^{3}	1.25×10^{3}	1.30×10^{-3}	1.30×10^{-3}	1.12×10^{3}	1.36×10^{4}	5.4×10^{-3}
Kr 87	8.21×10^{-1}	7.96	0	0	0	0	0	0
Kr 88	2.36	4.92×10^{1}	0	0	0	0	0	0
Xe 131 m	4.62×10^{-2}	1.95×10^{2}	6.16×10^{1}	6.43×10^{-5}	3.61×10^{-5}	4.22×10^{1}	5.13×10^{2}	1.6×10^{-4}
Xe 133 m	1.33	5.41×10^{2}	1.30	1.36×10^{-6}	6.66×10^{-8}	3.70×10^{-1}	4.50	1.8×10^{-6}
Xe 133	80.9	7.52×10^{4}	5.47×10^{3}	5.71×10^{-3}	1.54×10^{-3}	2.75×10^{3}	3.35×10^{4}	1.3×10^{-2}
Xe 135 m	7.13×10^{-1}	1.40	5.70×10^{-25}	5.95×10^{-31}	1.02×10^{-41}	2.07×10^{-26}	2.52×10^{-25}	3.0×10^{-31}
Xe 135	3.39	2.27×10^{2}	2.80×10^{-14}	2.92×10^{-20}	3.24×10^{-28}	1.38×10^{-15}	1.68×10^{-14}	1.9×10^{-20}
Xe 138	4.50×10^{-1}	9.59×10^{-1}	0	0	0	0	0	0
I 131	1.72	2.29×10^{-1}	4.11×10^{-2}	4.28×10^{-8}	1.81×10^{-8}	2.48×10^{-2}	3.01×10^{-1}	1.2×10^{-3}
I 132	5.85×10^{-1}	1.00×10^{-3}	0	0	0	0	0	0
I 133	2.65	4.15×10^{-2}	5.44×10^{-9}	5.68×10^{-15}	2.06×10^{-18}	6.19×10^{-10}	7.53×10^{-9}	7.9×10^{-12}
I 134	3.61×10^{-1}	2.35×10^{-4}	0	0	0	0	0	0
I 135	1.40	7.01×10^{-3}	2.03×10^{-24}	2.12×10^{-30}	3.60×10^{-41}	7.38×10^{-26}	8.98×10^{-25}	2.9×10^{-28}
Total		7.76×10^{4}	6.78×10^{3}	7.07×10^{-3}	2.87×10^{-3}	3.91×10^{3}	4.76×10^{4}	1.9×10^{-2}

TABLE 11.2.2-2

ESTIMATED WASTE GAS RELEASE WITH RECOMBINER OPERATING

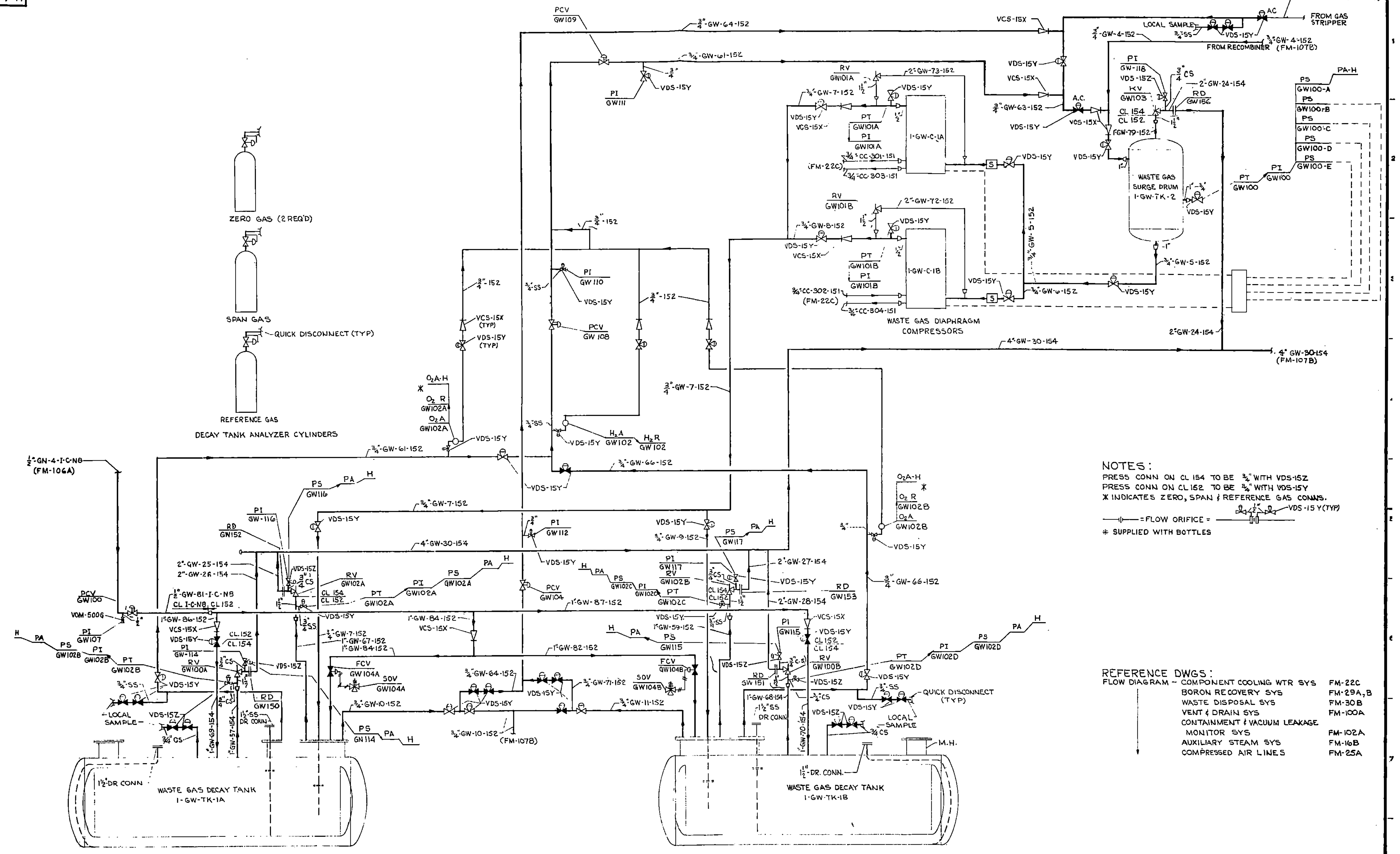
Based upon gaseous activity from:

a. Waste gas cycle: 300 days' feed, 20 days' decay, 10 days' bleed
b. Two units, 2,546 MWt each
c. 1% failed fuel
d. 17 gpm total letdown flow rate to gas stripper from both units
e. 100% of gases and 0.1% of iodines removed in gas stripper
f. $X/Q = 7.5 \times 10^{-6}$ sec/m^3 for calculating dose at site boundary

Nuclide	Equilibrium Coolant Activity, µCi/cc	Curies in Tank at End of Feed Cycle	Curies in Tank after 20 Days Decay	Discharge Rate, Ci/Sec Initial	Discharge Rate, Ci/Sec Final	Curies Discharge in One Cycle	Total Curies Discharged Per Year	Dose at Site Boundary, Rem Per Year
Kr 85 m	1.22	3.63×10^{1}	2.05×10^{-35}	2.14×10^{-4}	1.61×10^{-59}	4.44×10^{-37}	4.44×10^{-37}	5.3×10^{-43}
Kr 85	3.25×10^{-1}	1.63×10^{4}	1.63×10^{4}	1.70×10^{-2}	1.69×10^{-2}	1.46×10^{4}	1.46×10^{4}	5.8×10^{-3}
Kr 87	8.21×10^{-1}	7.96	0	0	0	0	0	0
Kr 88	2.36	4.92×10^{1}	0	0	0	0	0	0
Xe 131 m	4.62×10^{-2}	1.95×10^{2}	6.16×10^{1}	6.43×10^{-5}	3.61×10^{-5}	4.22×10^{1}	4.22×10^{1}	1.3×10^{-5}
Xe 133 m	1.33	5.42×10^{2}	1.30	1.36×10^{-6}	6.67×10^{-8}	3.71×10^{-1}	3.71×10^{-1}	1.5×10^{-7}
Xe 133	80.9	7.68×10^{4}	5.58×10^{3}	5.82×10^{-3}	1.57×10^{-3}	2.80×10^{3}	2.80×10^{3}	1.1×10^{-3}
Xe 135 m	7.13×10^{-1}	1.40	5.75×10^{-25}	6.00×10^{-31}	1.02×10^{-41}	2.09×10^{-26}	2.09×10^{-26}	2.5×10^{-32}
Xe 135	3.39	2.32×10^{2}	8.52×10^{-14}	2.98×10^{-20}	3.30×10^{-28}	1.40×10^{-15}	1.40×10^{-15}	1.7×10^{-21}
Xe 138	4.50×10^{-1}	9.59×10^{-1}	0	0	0	0	0	0
I 131	1.72	2.48×10^{-1}	4.44×10^{-1}	4.64×10^{-8}	1.96×10^{-8}	2.69×10^{-2}	2.69×10^{-2}	1.0×10^{-4}
I 132	5.85×10^{-1}	1.00×10^{-3}	0	0	0	0	0	0
I 133	2.65	4.23×10^{-2}	5.50×10^{-9}	5.79×10^{-15}	2.10×10^{-18}	6.31×10^{-10}	6.31×10^{-10}	6.6×10^{-13}
I 134	3.61×10^{-1}	2.35×10^{-4}	0	0	0	0	0	0
I 135	1.40	7.07×10^{-3}	2.05×10^{-24}	2.14×10^{-30}	3.63×10^{-41}	7.44×10^{-26}	7.44×10^{-26}	2.4×10^{-29}
Total		9.41×10^{4}	2.19×10^{4}	2.28×10^{-2}	1.85×10^{-2}	1.75×10^{4}	1.75×10^{4}	7.0×10^{-3}

OCT. 15, 1970

11448-FM-107A



NOTES:

PRESS CONN ON CL 154 TO BE ¾" WITH VDS-15Z

PRESS CONN ON CL 152 TO BE ¾" WITH VDS-15Y

✱ INDICATES ZERO, SPAN & REFERENCE GAS CONNS.

= FLOW ORIFICE = ½" VDS-15Y (TYP)

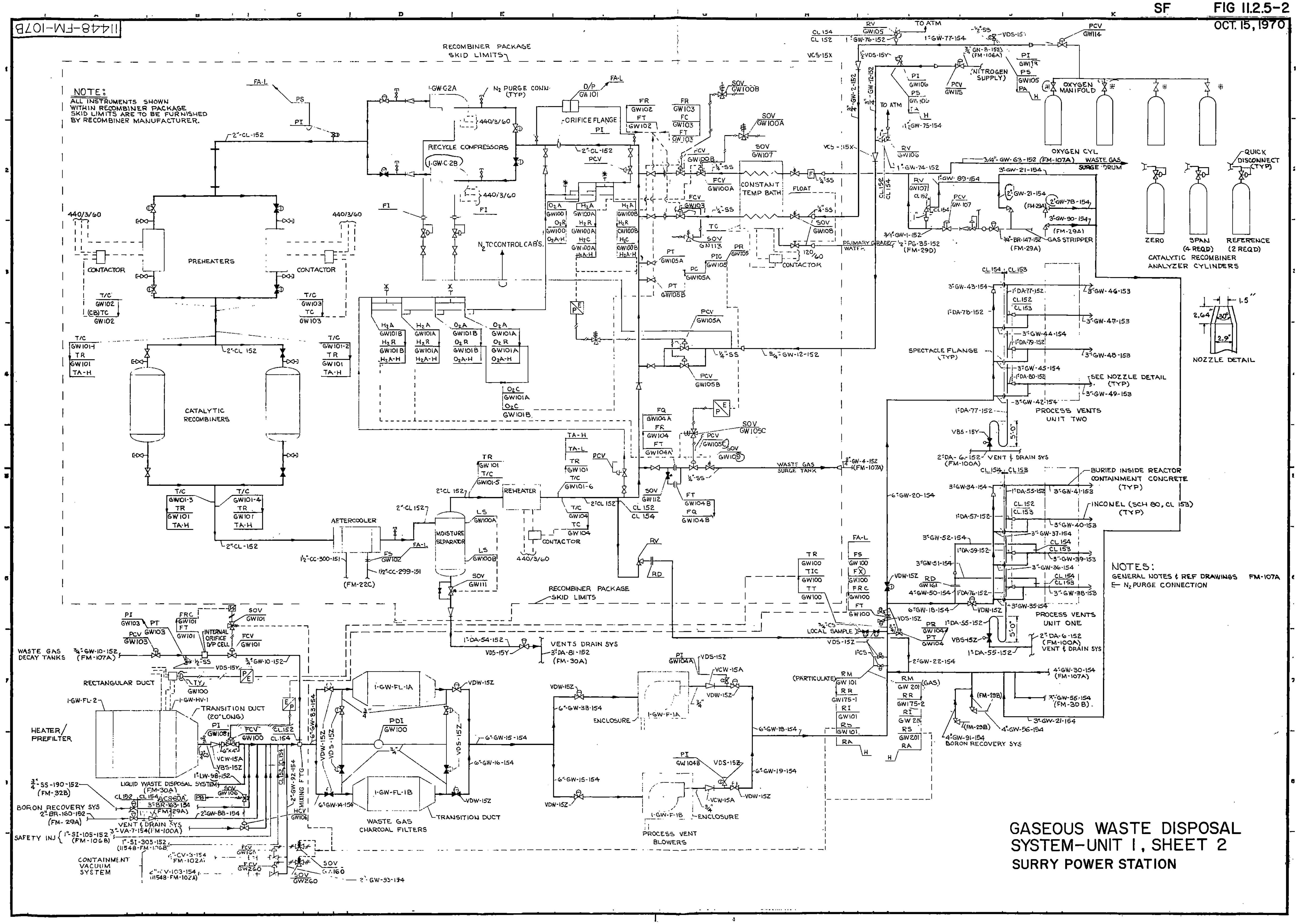
\# SUPPLIED WITH BOTTLES

REFERENCE DWGS:

FLOW DIAGRAM –	COMPONENT COOLING WTR SYS	FM-22C
	BORON RECOVERY SYS	FM-29A,B
	WASTE DISPOSAL SYS	FM-30B
	VENT & DRAIN SYS	FM-100A
	CONTAINMENT & VACUUM LEAKAGE MONITOR SYS	FM-102A
	AUXILIARY STEAM SYS	FM-16B
	COMPRESSED AIR LINES	FM-25A

GASEOUS WASTE DISPOSAL SYSTEM-UNIT 1, SHEET 1

SURRY POWER STATION



GASEOUS WASTE DISPOSAL SYSTEM-UNIT 1, SHEET 2
SURRY POWER STATION

11.2.5.2 Ventilation Vent Subsystem

The Ventilation Vent Subsystem is considered to be a portion of the Gaseous Waste Disposal System only for purposes of radiological surveillance, and it is designed on this basis. However, since it handles air streams of very low activity levels, and since the gases to be handled are predominantly of nonradioactive origin, this subsystem has been considered as an auxiliary system for the purpose of this report. A full description of this subsystem is included in Section 9.13.

11.2.5.3 Components

Catalytic Recombiner

One skid mounted catalytic recombiner system is provided. The system includes duplicate full capacity catalytic recombiners, duplicate electric preheaters, one aftercooler-condenser, one moisture separator, duplicate recycle blowers, duplicate hydrogen analyzers of the thermal conductivity type, for the recombiner influent and effluent, duplicate oxygen analyzers of the paramagnetic type on the recombiner effluent, a single oxygen analyzer on the recombiner influent, and one bleed stream cooler. The recombiner system operates at approximately 22 psia and has a feed capacity of 1.14 scfm. The diluent is nitrogen. The catalytic recombiner system is designed according to Section III-C of the ASME Code for Boiler and Pressure Vessels.

Waste Gas Surge Tank

One waste gas surge tank with a 15.7 ft^3 capacity is provided. This tank is operated at a pressure of approximately 10 to 20 psia. The tank is fabricated from austenitic stainless steel in accordance with Section III-C of the ASME Code for Unfired Boiler and Pressure Vessels.

Waste Gas Compressor

Two waste gas compressors of the diaphragm type are provided. Each has a capacity of 1.5 scfm and is capable of a discharge pressure up to 120 psig. The compressor heads are leak tested to insure that the leakage does not exceed a predetermined amount.

Waste Gas Decay Tank

Two buried waste decay tanks are provided. These tanks have double wall construction with feed and bleed lines, sample nitrogen purge, drain and relief valve lines to the inner tank, and sample, nitrogen purge, drain, and relief valve lines from the outer tank. An access opening is provided to the inner tank. In addition, adequate grounding and corrosion protection are provided. The inner tank is fabricated from stainless steel in accordance with Section III C and the outer tank from carbon steel in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code.

Process Vent Blowers

Two full capacity dilution air blowers of 300 cfm capacity at 2 psia are provided. The blowers are a centrifugal type located in a field fabricated box with the blower suction from the box's interior. Some inleakage is tolerated.

Charcoal Filters

Two charcoal filter beds are provided to service approximately **300 scfm** radioactive gas. The filters are maintained at a subatmospheric pressure.

11.2.6 TESTS AND INSPECTIONS

11.2.6.1 Construction and Fabrication

During the manufacturing period, the Applicant's inspectors inspected all equipment periodically, as required, to assure that all equipment had been provided in strict accordance with specifications. Shop hydrostatic and performance tests of principal equipment were witnessed by the Applicant's inspectors. Certified code inspection data sheets were provided by manufacturers for all equipment covered by ASME or other applicable codes.

During the construction period, all pressure systems were subjected to field hydrostatic or pneumatic tests to verify the integrity of welded connections and to assure that the system as a whole functioned as intended.

During the preliminary operation period, all equipment in the Waste Disposal System was tested to verify conformance with specification performance requirements. All control systems and interlocks were tested and operated to assure satisfactory functional performance and reliability.

11.2.6.2 Operation

The basic function of the Waste Disposal System is to release controlled amounts of radioactivity to the environment with no undue effects on the health and safety of the general public. This is accomplished by assuring that all releases from the station are less than the maximum levels of radioactivity set by applicable regulatory agencies, as given in Section 11.2.1.

The following is a list of the types and areas monitored to assure the proper functioning of the Waste Disposal System:

1. Continuous Process Monitoring: As described in Section 11.3.3, process radiation monitors continuously monitor certain key systems where radioactive material may exist. These monitors give an indication of the waste processing requirements of certain systems.

2. Batch Sample Process Monitoring: As described in Section 9.6, batch samples are obtained from certain subsystems which provide information on the effectiveness of ion exchangers, filters, and evaporators. This gives an indication of the effectiveness of the various waste processing subsystems. Monitoring of the gas in the waste gas holdup tanks avoids storage of excessive activity.

3. Continuous Monitoring of Discharge Effluents: As described in Section 11.3.3, radiation monitors continuously monitor discharges from the Process and Ventilation Vent Systems and the Liquid Waste Disposal and Service Water Systems. These monitors give an indication of liquid and gas radiation discharges to the environment, and provided with automatic valve closure when radiation levels exceed a preset level, thus terminating discharge.

4. Radiation Surveys on the Outside of Containers: Portable Geiger-Muller or scintillation detectors are used to survey spent resin casks and other containers for liquid and solid radioactive materials to assure acceptable contamination levels prior to handling or shipment.

5. Radiation Survey of Solid Waste Containers: Radiation surveys and smear samples are taken of all shipping casks, drums, etc. that contain solid waste to assure that such waste is properly contained and meets transportation regulations.

6. Environmental Monitoring: As described in Section 11.3.5, environmental samples are taken to indicate the effect of liquid and gas discharges on the environment and the compliance of these discharges with applicable regulations.

To assure that the performance of the Waste Disposal Systems is meeting design criteria, the following checks are made:

1. Standardized laboratory radiochemical analytical procedures are used to verify decontamination factors.

2. Radiation monitors are periodically checked with remotely operated check sources and, in addition, samples are withdrawn from the process streams being monitored for spot checks of the activity levels and monitor calibration. Those monitors which actuate control valves by a high radiation signal have check sources with sufficiently high count rates to actuate these valves for test purposes.

3. Portable Geiger-Muller and scintillation detectors are periodically calibrated with known radiation sources in accordance with station standard health physics procedures.

4. Radiation levels on the outside of components, pumps, valves, and piping in the Waste Disposal Systems are monitored on a periodic basis to avoid inadvertent discharge of activity which may accumulate with time.

11.3 RADIATION PROTECTION

11.3.1 DESIGN BASES

Radiation protection, including radiation shielding, is designed to ensure that the criteria specified in 10CFR20 are met during normal operation and that the guidelines suggested in 10CFR100 would be met in the event of the Design Basis Accident (Section 14.5.1).

Allowable dose rates are based on the expected frequency and duration of occupancy. Occupancy time and dose rates are such that no personnel shall receive in excess of that recommended in 10CFR20. All dose rate calculations are based upon 1 percent failed fuel elements. Allowable dose rates will not exceed the following:

Zone Description	Maximum Dose Rate, mrem/hr	Typical Locations
Full Power Operation		
Continuous access (Zone I)	0.75	Main Control Room, outside surface of containment, and all turbine plant and administration areas
Periodic access (Zone II)	2.5	Auxiliary and fuel building passageways in general, and inside reactor containment personnel lock
Limited access (Zone III)	15	Outside surface of shielded tank shields
Controlled access (Zone IV) General	15	Inside shielded equipment compartments
Access to incore instrumentation	100	Annulus between crane wall and containment wall

Zone Description	Maximum Dose Rate, mrem/hr	Typical Locations
Access to incore instrumentation	40	Vicinity of incore instrumentation transfer devices
Access to incore instrumentation	20	Vicinity of incore instrumentation motors
<u>Hot Shutdown</u> (after 15 min decay)		
Limited access (Zone III)	15	Reactor containment above charging floor and outside of crane wall
Controlled access (Zone IV)	>15	Inside shielded equipment compartments
<u>Cold Shutdown for Maintenance</u> (after 8 hr decay)		
Periodic access (Zone II)	2.5	Reactor containment above charging floor and outside of crane wall
Controlled access (Zone IV)	>15	Inside shielded equipment compartments
<u>Cold Shutdown for Refueling</u> (after 4 day decay)		
Periodic access (Zone II)	2.5	Reactor containment above charging floor, outside of crane wall, and adjacent to fuel transfer canal near incore instrumentation devices
Controlled access (Zone IV)	>15	Inside shielded equipment compartments
Surface of water over raised fuel elements	50	Fuel element above up-ender, and above other fuel elements in fuel building

<u>Design Basis Accident Conditions</u>

Location	Maximum Dose Criteria: Whole Body	Maximum Dose Criteria: Thyroid
Site boundary	25 rem in 2 hr	300 rem in 2 hr
Main Control Room	2.5 rem in 31 days	10 rem in 31 days

11.3.2 SHIELDING DESIGN AND EVALUATION

11.3.2.1 Primary Shielding

Primary shielding is provided to limit radiation emanating from the reactor vessel; the radiation consists of neutrons diffusing from the core, prompt fission gammas, fission product gammas, and gammas resulting from the slowing down and capture of neutrons.

The primary shielding is designed to:

1. Attenuate neutron flux to prevent excessive activation of unit components and structures.
2. Reduce the contribution of radiation from the reactor to obtain a reasonable division of the shielding function between primary and secondary shields.
3. Reduce residual radiation from the core to level which does not limit access to the region between the primary and secondary shields at a reasonable time after shutdown.

The primary shield consists of a water-filled neutron shield tank having a radial dimension of approximately 3 ft surrounded by 4 1/2 ft of reinforced concrete. The neutron shield tank is designed to prevent overheating and dehydration of the concrete primary shield wall and to prevent activation of the plant components within the reactor containment. A thermosiphon cooling system is provided for cooling the water in the shield tank.

A 17 ft high x 2 in. thick cylindrical lead shield is located beneath the neutron shield tank to protect station personnel servicing the neutron detectors during reactor shutdown.

In order to maintain the integrity of the primary shield, streaming shields fabricated from both masonite Benelex 401 and steel are provided in the annular gap between reactor vessel flange and the primary shield concrete. In addition, masonite and steel streaming shields located outside the primary concrete shield are provided around all of the reactor coolant pipe penetrations.

The primary shield arrangment is shown in Figures 11.3.2-1 and 11.3.2-2. The shield materials and thicknesses are listed in Table 11.3-2.

11.3.2.2 Secondary Shielding

Secondary shielding consists of reactor coolant loop shielding, reactor containment shielding, fuel handling shielding, auxiliary equipment shielding, and waste storage shielding.

Nitrogen-16 is the major source of radioactivity in the reactor coolant during normal operation and establishes the combined thickness of the crane and containment walls. Activiated corrosion and fission products in the Reactor Coolant System establish the shutdown radiation levels in the reactor coolant loop areas. Tables 9.1-5 and 11.3-2 list the activities which were used in designing the containment secondary shielding. Table 9.1-5 lists the fission product activities in the Reactor Coolant System with 1 percent failed fuel. Table 11.3-1 lists the activated corrosion product activities and the N-16 activity at the reactor vessel outlet nozzle.

Activated corrosion and fission products from the Reactor Coolant System are the radioactive sources for which shielding is required in the auxiliary and waste disposal systems.

11.3.2.3 Reactor Coolant Loop Shielding

Interior shield walls separate reactor coolant loop, pressurizer, incore instrumentation, and containment access sectors. This shielding allows access to the incore instrument sector during normal operation and facilitates maintenance in all sectors during shutdown. The crane support wall provides limited access protection in the annulus between the crane wall and the reactor containment wall and provides part of the exterior shielding required during power operation. Shield walls are provided around each steam generator above the charging floor to a height required for personnel protection. The shielding arrangement is shown in Figures 11.3.2-1 and 11.3.2-2. The shield materials and thicknesses are listed in Table 11.3-2.

11.3.2.4 Reactor Containment Shielding

The containment shielding consists of the steel-lined, steel-reinforced concrete cylinder and hemispherical dome as further described in Section 5. This shielding, together with the crane support wall, attenuates radiation during full power operation at the outside surface of the containment to less than 0.75 mrem per hr. In addition, it attenuates the dose rate from the Design Basis Accident to design levels.

11.3.2.5 Fuel Handling Shielding

Fuel handling shielding is designed to facilitate the removal and transfer of spent fuel assemblies from the reactor vessel to the spent fuel pit. It is designed to protect personnel against the radiation emitted from the spent fuel and control rod assemblies.

The refueling cavity above the reactor vessel is flooded to El.+45 ft-4 in. to provide a temporary water shield above the components being withdrawn from the reactor vessel. The water height is thus approximately 27 ft above the reactor vessel flange. This height assures a minimum of 107 in. of water above a withdrawn fuel assembly at its highest point of travel. Under these conditions, the dose rate is less than 50 mrem per hr at the water surface.

Upon removal of the fuel from the reactor vessel, it is moved to the spent fuel pit by the fuel transfer mechanism via the refueling canal.

The spent fuel pit in the fuel storage building is permanently flooded to provide a minimum of 96 in. of water above a fuel assembly when being withdrawn from the fuel assembly transfer basket. Water height above stored fuel assemblies is a minimum of 24 ft. The sides of the spent fuel pit, three of which also form part of the fuel storage building exterior walls, are 6 ft thick concrete to ensure a dose rate of no more than 2.5 mrem per hr outside the building.

Sixteen feet of earth shielding is provided above the fuel transfer tube between the reactor containment and the fuel storage pit wall.

11.3.2.6 Auxiliary Equipment Shielding

The auxiliary components exhibit varying degrees of radioactive contamination due to the handling of various fluids. The function of the auxiliary shielding is to protect personnel working near the various auxiliary system components, such as those in the Chemical and Volume Control System, the Boron Recovery System, the Waste Disposal System, and the Sampling System. Controlled access to the auxiliary building is allowed during reactor operation. Each equipment compartment is individually shielded so that compartments may be entered without having to shut down and, possibly, decontaminate the entire system. Ilmenite concrete is used in certain areas where substantial shielding is required and space is at a premium, such as the primary drain tank compartment and the mixed bed demineralizer compartments.

All ion exchangers and the most highly contaminated filters are located in the ion exchange structure along the north wall of the auxiliary building. Each ion exchanger or filter is enclosed in a separate, shielded compartment. The concrete thicknesses provided around the shielded compartments are sufficient to reduce the surrounding area dose rate to less than 2.5 mrem per hr and the dose rate to any adjacent cubicle to less than 100 mrem per hr. The shielding thicknesses around the mixed bed demineralizers are based upon a saturation activity which gives a contact radiation level of nearly 9,000 rem per hr.

In many areas, tornado missile protection in the form of thick concrete affords more shielding than that required for radiation protection.

11.3.2.7 Waste Storage Shielding

The waste storage and processing facilities in the auxiliary building and decontamination building and the waste storage tanks in the yard are shielded to provide protection of operating personnel in accordance with the radiation protection design bases set forth in Section 11.3.1.

Periodic surveys by Health Physics personnel using portable radiation detectors ensure that radiation levels outside the shield walls meet design specifications, and establish access limitations within the shielded cubicles. In addition, continuous surveillance is provided at the auxiliary building drumming area and control area by area radiation monitors.

Area and process monitoring also ensures that any accidental radioactivity release would be detected within a reasonable period of time. The largest accidental radioactivity release from the Waste Disposal System would be the rupturing of one of the waste gas decay tanks. An analysis of this accident is made in Section 14.4.2. Furthermore, periodic samples of the gas in the waste gas decay tanks are analyzed by Health Physics to ensure that the activity level in these tanks is never above the design level used in the accident analysis.

11.3.2.8 Accident Shielding

Accident shielding is provided by the reactor containment, which is a reinforced concrete structure lined with steel. For structural reasons, the thicknesses of the cylindrical walls and dome are 54 in. and 30 in., respectively. These thicknesses are more than adequate to meet the requirements of AEC Regulation 10CFR100 at the exclusion boundary.

Additional shielding is provided for the Main Control Room. This, together with the shielding afforded by its physical separation from the containment structure, ensures that an operator would be able to remain in the Main Control Room for 31 days after an accident and not receive an integrated whole body dose in excess of 2.5 rem. The calculational methods and radiation sources used in designing the Main Control Room shielding are discussed in Section 11.3.6, Control Areas.

In addition, the Main Control Room will serve as a fallout shelter with a protection factor of better than 500 as defined by the Office of Civil Defense.

TABLE 11.3-1

N-16 AND ACTIVITATED
CORROSION PRODUCT ACTIVITY

Isotope	Activity, µCi/cc @ 500 F
Mn 54	2.7×10^{-3}
Mn 56	5.7×10^{-2}
Fe 59	8.3×10^{-3}
Co 58	2.3×10^{-4}
Co 60	9.2×10^{-4}
N 16*	64.0

*At the reactor vessel outlet nozzle at 2,546 MWt

TABLE 11.3-2

CONTAINMENT SHIELDING SUMMARY

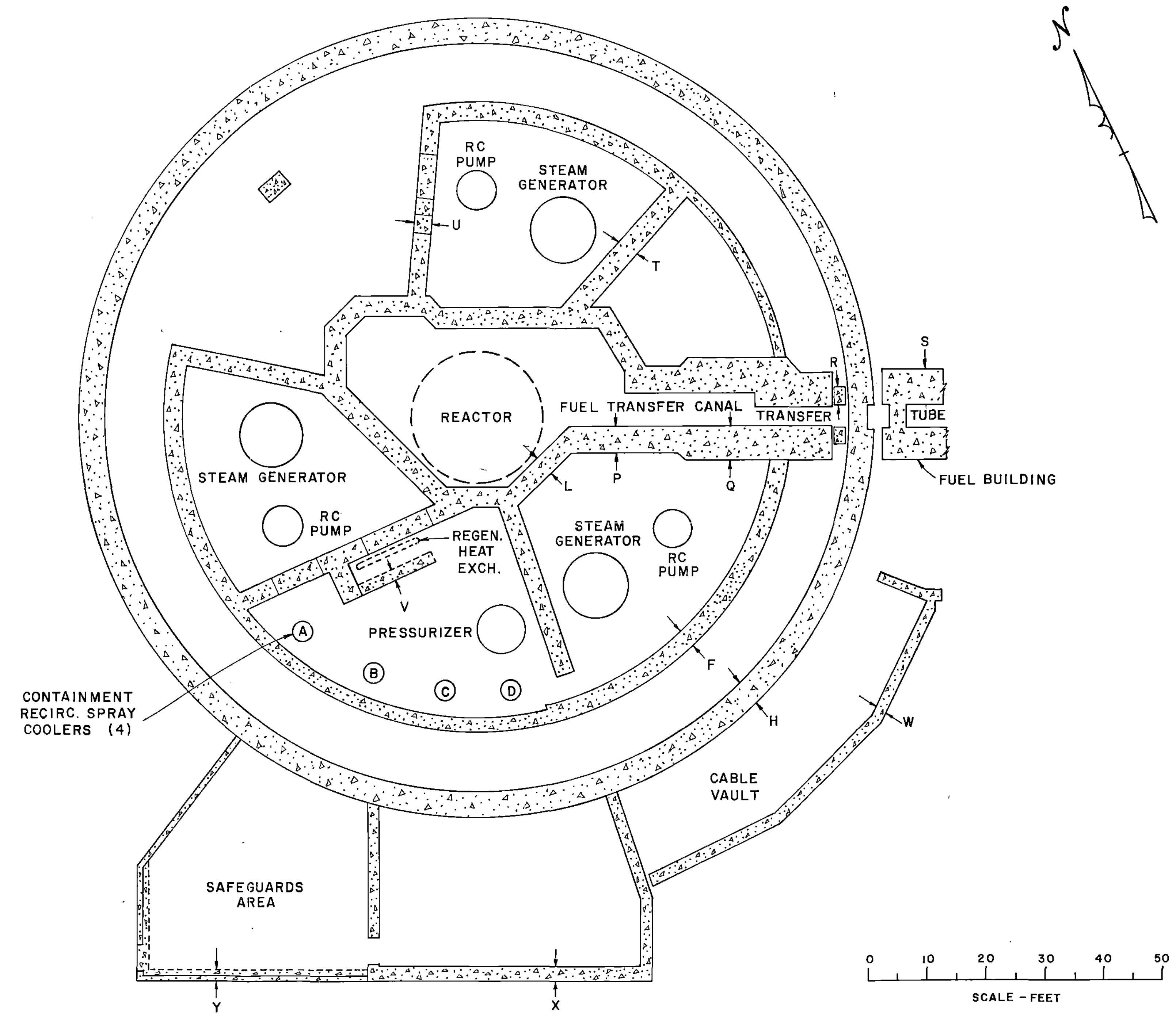
Symbol	Figure	Shield Description	Material	Thickness, In.
A	2	Neutron Shield Tank	Water Steel	34 3
B	2	Primary Shield	Concrete	54
C	2	Supplementary Neutron Shield	Benelex 70	14
D	2	Streaming Shield Ring	Benelex 401	5
E	2	Neutron Shield Tank Support	Steel Lead	1 1/2 2
F	1 & 2	Cubicle - Crane Support Wall	Concrete	33
G	2	Crane Support Wall	Concrete	24
H	1 & 2	Containment Wall	Concrete	54
I	2	Containment Dome	Concrete	30
J	2	Floor Elevation -3'-6"	Concrete	24
K	2	Charging Floor	Concrete	24
L	1 & 2	Refueling Cavity Wall	Concrete	36
M	2	Missile Shield	Concrete	24
N	2	Refueling Cavity Water	Water	-
O	2	Removable Block Wall	Concrete (Ilmenite)	12
P	1	Fuel Trans. Canal Wall	Concrete	54
Q	1	Fuel Trans. Canal Wall	Concrete	72
R	1	Fuel Trans. Tube Shielding	Concrete	36
S	1	Fuel Trans. Canal Wall	Concrete	72
T	1	Incore Inst. Cubicle Wall	Concrete	42

TABLE 11.3-2 (CONTINUED)

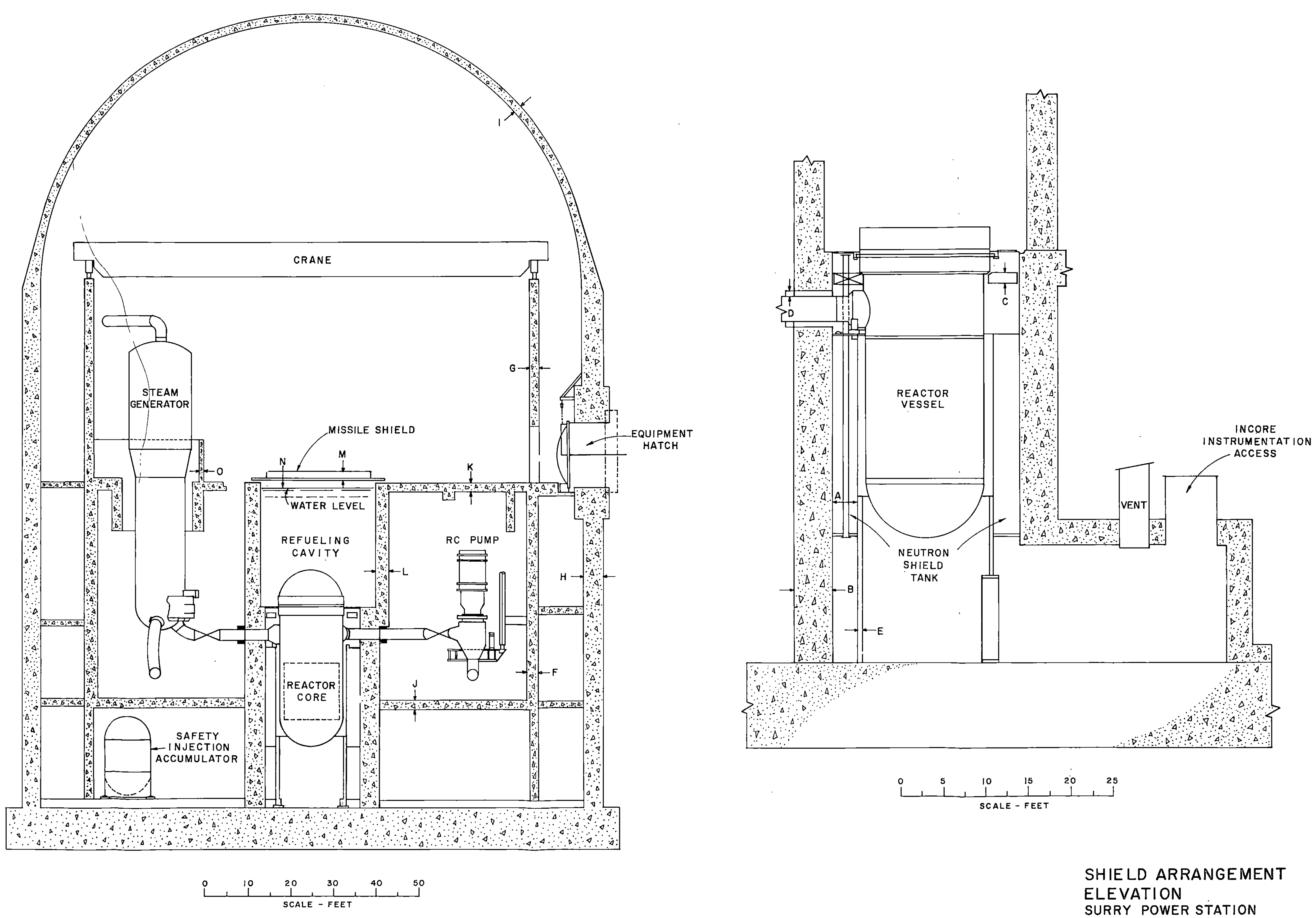
Symbol	Figure	Shield Description	Material	Thickness, In.
U	1	Cubicle Wall	Concrete	30
V	1	Regen. Heat Exchanger Wall	Concrete	24
W	1	Cable Vault Wall	Concrete	24
X	1	Auxiliary Feed Pump Cubicle Wall	Concrete	36
Y	1	Safeguards Area Wall	Concrete	24

Notes: 1) All concrete is reinforced with steel.

2) Figure 1 means "Figure 11.3.2-1," and Figure 2 means "Figure 11.3.2-2."



SHIELD ARRANGEMENT PLAN
SURRY POWER STATION



SHIELD ARRANGEMENT
ELEVATION
SURRY POWER STATION

11.3.3 PROCESS RADIATION MONITORING SYSTEM

The Process Radiation Monitoring System continuously monitors selected lines containing, or possibly containing, radioactive effluent. Lines through which waste liquids and gases are discharged to the environment are also monitored. Its function is to warn personnel of increasing radiation levels which could result in a radiation health hazard and to give early warning of a system malfunction. The Process Radiation Monitoring System serving both units is comprised of 28 channels (see Table 11.3.3-1) and four spare channels.

Each channel has a readout at the Main Control Room and selected ones, as indicated in Table 11.3.3-1, have a readout at the detector location. In addition, each channel has an audible and visual alarm for radiation levels in excess of pre-set values, as well as a visual alarm for detector malfunction. The output from all channels is recorded on strip chart recorders which produce a continuous record of radiation levels and radioactive discharges from the station. Each channel has its own power supply and check source, remotely operated from the Main Control Room, thus making it completely independent of any other channel. Adjustment of alarm set points, voltage, power, and other variables is made from the Main Control Room. The entire system is designed to be fail-safe, with emphasis on system reliability and availability. Certain channels, as indicated in the following text, actuate control valves on a high-activity alarm signal.

Channels monitoring Unit 1 are supplied from the emergency bus for Unit 1; channels monitoring Unit 2 are supplied from the emergency bus for Unit 2; and channels monitoring systems or areas common to both units can be supplied from the emergency bus for either Unit 1 or Unit 2.

The type of detector, sensitivity, range, background radiation, and other information for each channel are listed in Table 11.3.3-1. A description of each channel is included in the following text.

11.3.3.1 Process Vent Particulate Monitor

This channel continuously withdraws a 10 scfm sample from the process vent and passes the sample through a moving filter paper having a collection efficiency of 99 percent for particle sizes greater than 1.0 micron. The amount of deposited activity is continuously scanned by a lead-shielded beta scintillation detector. A high-activity alarm automatically initiates closure of the process vent discharge line valves. The sample system, which is common to both process vent monitors, includes a pump with 1.5 hp motor, a flowmeter, automatic pressure protective valves, a flow-regulating valve, and isolation valves. The sample system is controlled from the Main Control Room.

11.3.3.2 Process Vent Gas Monitor

This channel takes the continuous process vent sample, after it has passed through the particulate filter paper, and draws it through a sealed system to the process vent gas monitor assembly, which is a fixed lead-shielded sampler enclosing a beta scintillation detector. The sample activity is measured, and then it is returned to the process vent. A high-activity alarm automatically initiates closure of the process vent discharge line valves. A purge system is integral with the gas monitoring system for flushing the sampler with clean air for purposes of calibration.

11.3.3.3 Ventilation Vent Particulate Monitor and Ventilation Vent Gas Monitor

These two channels continuously sample the ventilation vent particulates and gas in the same way as the two channels monitor the process vent sample. All equipment is identical with that of the two process vent channels, except (1) pressure-protecting valves are not required, (2) an in-line, easily removable, charcoal filter is provided between the particulate and gas monitor, and (3) a multiprobe isokinetic sampler is provided to obtain a representative sample in the 80 in. diameter duct.

11.3.3.4 Component Cooling Water Monitors

These two channels continuously monitor the component cooling water by means of a gamma scintillation detector enclosed in 3 in. of lead shielding and mounted on the component cooling piping. The complex piping arrangement of this system dictates that two detectors are required to ensure that the system is properly monitored. Activity is indicative of a leak into the Component Cooling System (Section 9.4) from one of the radioactive systems which exchange heat to the Component Cooling System.

11.3.3.5 Component Cooling Heat Exchanger Service Water Monitor

This channel continuously monitors the service water effluent from the four component cooling water heat exchangers. A 1-2 gpm sample of the service water from each exchanger is drawn into a common header by a common pump powered by a 1/3 hp motor. The sample is passed through a liquid sampler, shielded by 4 in. of lead, enclosing a gamma scintillation detector. Upon indication of activity, a valving arrangement allows the heat exchangers to be individually sampled to determine the origin of the activity.

11.3.3.6 Liquid Waste Disposal System Monitor

This channel continuously monitors the Liquid Waste Disposal System (Section 11.2.3) effluent by means of a gamma scintillation detector mounted in a split sampler which is clamped around the 2 in. discharge pipe. The sampler is shielded by 8 in. of lead to ensure a sensitivity of 4×10^{-6} μCi/ml for Cs-137 in a background of 2.5 mrem/hr. The detector is located downstream of the last possible point of radioactive material addition. A high-activity alarm automatically initiates closure of a valve which terminates discharge from this system.

11.3.3.7 Condenser Air Ejector Monitors

Each of these identical channels (one channel per unit) continuously monitors the gaseous effluent from the condenser air ejectors by means of a G-M tube detector mounted in an in-line sampler surrounded by 2 in. of lead. Activity is indicative of a primary-to-secondary system leak. On a high-activity alarm, the flow is automatically diverted to the containment.

11.3.3.8 Steam Generator Blowdown Sample Monitors

Each of these channels (two channels per unit) monitors the liquid phase of the steam generators for radioactivity indicative of a primary-to-secondary system leak. The three steam generator blowdowns are combined and continuously monitored by one of the two detectors. Upon indication of radioactivity, a valving arrangement enables the steam generators to be individually sampled, in turn, to determine the source of the activity. Once it has been established which steam generator is leaking, one of the detectors monitors only the blowdown from that steam generator, while the other detector monitors the combined blowdown from the other two steam generators. After being monitored, the samples pass into the steam generator blowdown tank. The detectors are gamma scintillation detectors which are mounted in liquid samplers surrounded by 7 in. of lead.

11.3.3.9 Recirculation Spray Cooler Service Water Outlet Monitors

The recirculation spray coolers, as part of the Recirculation Spray System, (Section 6.3.1) operate only after the occurrence of a loss-of-coolant accident.

There are four recirculation spray coolers per unit, and each service water outlet line from the coolers is monitored, thus giving a total of eight channels. Each of these channels is identical. If the Recirculation Spray System is placed in service, a 5-10 gpm sample is drawn out of each service water outlet line by a small pump with a 1/3 hp motor and passed through an off-line liquid sampler, where it is monitored for activity indicative of a leak in the respective recirculation spray cooler. After passing through the liquid samples, which is located outside the containment behind heavy shielding, the sample is returned to the service water line. Each monitor consists of a gamma scintillation detector mounted in a standard off-line sampler surrounded by 3 in. of lead.

11.3.3.10 Reactor Coolant Letdown Gross Activity Monitors

Each of the units has its reactor coolant continuously monitored by means of a sample taken from the letdown line to the Chemical and Volume Control System (Section 9.1). In this system, large variations in activity level are possible in the event of fuel assembly failure. This is a two-stage monitoring system consisting of a low-range channel and a high-range channel.There is one such system for each unit. After being withdrawn from the letdown line, the sample is passed through a delay line to allow N-16 to decay and then enters a sampler consisting of 2 gamma scintillation detectors surrounded by 7 to 8 in. of lead, and is then discharged to the volume control tank. Both detectors sit on a 1/2 in. removable stainless steel tube providing flow through the sampler. Shielded lead plugs are used to convert the two detectors into either high or low range letdown monitors. Normally, the low range detector will be sitting on the 1/2 in. tubing and the high range will be sitting on the shielded lead plug. In the event of a fuel element failure, the activity released could be sufficient to raise the coolant activity level above 1.0 μCi/cc gross fission products. This causes the high-range monitor to begin to indicate activity level at 10^{-1} μCi/cc, providing a one-decade overlap. At this point, the high-range channel provides the activity data, and the low-range monitor can be converted into a high range monitor by inserting a shielded lead plug.

11.3.3.11 Circulating Water Discharge Tunnel Monitors

Each of these identical channels (one per unit) monitors the effluent (service water, condenser circulating water, and liquid waste) in the circulating water discharge tunnel beyond the last point of possible radioactive material addition. A gamma scintillation detector slides into a capped pipe which is then inserted directly into the discharge tunnel and acts as a well. At the top of the pipe is a waterproof support assembly which encloses a check source. The entire device is waterproof.

11.3.3.12 Ventilation Vent Sample Particulate Monitor and Ventilation Vent Sample Gas Monitor

These two channels continuously sample one of seven areas located in the auxiliary, fuel or decontamination buildings for particulates and gas in the same way as the two channels monitor the process vent sample. All equipment is identical with that of the two process vent channels, except (1) pressure protecting valves are not required, (2) an in-line, easily removable, charcoal filter is provided between the particulate and gas monitor, (3) seven isokinetic samplers are provided, (4) an eight valve manifold (one spare valve) and control valves are provided, and (5) a control panel that will automatically switch the valves on a periodic basis to permit sampling from eight different areas.

A manual positioner is also provided to permit locking in on any chosen area. The sequencing of the valves will be synchronized to the step advance programmer and the operator will be able to select a sampling time between 30 minutes and 24 hours for each of the eight areas.

TABLE 11.3.3-1

PROCESS RADIATION MONITORING SYSTEM

Monitor	No.	Type of Detector	Medium	Limiting Isotopes	Sensitivity, µCi/cc	Range, Decades	Maximum Background, mr/hr
Process Vent Particulate	1	Beta - Scint.	Gas, air	I 131	1×10^{-9} (I 131)	3	0.75
Process Vent Gas	1	Beta - Scint.	Gas, air	I 131, Xe 133, Kr 85	5×10^{-6} (Kr 85)	4	0.75
Ventilation Vent Particulate	1	Beta - Scint.	Air	I 131	1×10^{-9} (I 131)	3	0.75
Ventilation Vent Gas	1	Beta - Scint.	Air	I 131,Xe 133, Kr 85	5×10^{-6} (Kr 85)	3	0.75
Component Cooling Water	2	Gamma - Scint.	Water	Co 60, Cs 137	1×10^{-5} (Cs 137)	3	0.75
Component Cooling HX Service Water	1	Gamma - Scint.	Water	Co 60, Cs 137	1×10^{-5} (Cs 137)	3	0.75
Liquid Waste Disposal *	1	Gamma - Scint.	Water	Co 60, Cs 137	4×10^{-6} (Cs 137)	4	2.5
Condenser Air Ejector	2	G-M Tube	Vapor	I 131, Xe 133, Kr 85	4×10^{-3} (Kr 85)	3	0.75
Steam Generator Blowdown*	4	Gamma - Scint.	Water	Co 60, Cs 137	4×10^{-6} (Cs 137)	3	2.5
Recirc. Spray Cooler	8	Gamma - Scint.	Water	Co 60, Cs 137	3×10^{-4} (Cs 137)	2	5.0
R.C. Letdown High Range*	2	Gamma - Scint.	Water	Co 60, Mixed Fission Products	1×10^{-1} (Co 60)	4	2.5
R.C. Letdown Low Range*	2	Gamma - Scint.	Water	Co 60, Mixed Fission Products	1×10^{-4} (Co 60)	4	2.5
C. W. Discharge Tunnel	2	Gamma - Scint.	Water	Co 60, Cs 137	1×10^{-6} (Cs 137)	3	0.05
Vent. Vent Sample - Part.	1	Beta - Scint.	Air	I 131	1×10^{-9} (I 131)	3	0.75
Vent. Vent Sample - Gas	1	Beta - Scint.	Air	I 131, Xe 133, Kr 85	5×10^{-6} (Kr 85)	3	0.75
Spares	4						

* These channels also have local readout and alarm

TABLE 11.3.3-2

PROCESS RADIATION MONITORING SYSTEM

COUNTING RATES OF LIMITING ISOTOPES (cpm/μCi/cc)

Monitor	I 131	Xe 133	Kr 85	Cs 137	Co 60
Process Vent Particulate	8.7×10^{11}	-	-	-	-
Process Vent Gas	8.5×10^{6}	3.96×10^{6}	1.04×10^{7}	-	-
Ventilation Vent Particulate	8.7×10^{11}	-	-	-	-
Ventilation Vent Gas	8.5×10^{7}	3.96×10^{7}	1.04×10^{8}	-	-
Ventilation Vent Sample Particulate	8.7×10^{11}	-	-	-	-
Ventilation Vent Sample Gas	8.5×10^{7}	3.96×10^{7}	1.04×10^{8}	-	-
Component Cooling Water	-	-	-	4.89×10^{7}	8.0×10^{7}
Component Cooling HX Service Water	-	-	-	7.16×10^{7}	1.52×10^{8}
Liquid Waste Disposal	-	-	-	1.35×10^{7}	2.65×10^{7}
Condenser Air Ejector	2.08×10^{5}	8.25×10^{4}	4.85×10^{3}	-	-
Steam Generator Blowdown	-	-	-	7.16×10^{7}	1.52×10^{8}
Recirculating Spray Cooler	-	-	-	7.16×10^{7}	1.52×10^{8}
R.C. Letdown High Range	-	-	-	-	9.92×10^{2}
R. C. Letdown Low Range	-	-	-	-	9.65×10^{5}
C. W. Discharge Tunnel	-	-	-	1.99×10^{8}	5.52×10^{8}

11.3.4 AREA RADIATION MONITORING SYSTEM

11.3.4.1 General

The Area Radiation Monitoring System reads out and records the radiation levels in selected areas throughout the station and alarms (audible and visual) if these levels exceed a pre-set value or if the detector malfunctions. With the exception of the containment gas and particulate monitors, each detector reads out and alarms both in the Main Control Room and at its station location. Each channel is equipped with a check source remotely operated from the Main Control Room. Strip charts produce a continuous permanent record of radiation levels while the detectors are functioning. Detectors monitoring Unit 1 are supplied with power from the emergency bus for Unit 1; detectors monitoring Unit 2 are supplied with power from the emergency bus for Unit 2; and detectors monitoring areas common to both units have the capability of being supplied with power from the emergency bus for either Unit 1 or Unit 2.

The alarm set point of each area monitor is variable, and is set at a level slightly above the normal background radiation level in the respective area.

The Area Radiation Monitoring System consists of the detectors listed in Table 11.3.4-1, plus the containment gas and particulate monitors for each unit described in the following text.

The high range reactor containment area monitor is located outside the containment structure. The detector is permanently mounted and aimed at the personnel hatch. The monitor has a range of 0.1 to 10^7 mr/hr to measure the expected high gamma dose rate in the containment following a loss-of-coolant accident.

11.3.4.2 Containment Particulate Monitors

This channel continuously withdraws a sample from the containment atmosphere into a closed, shielded system exterior to the containment. The sample is passed through a moving filter paper having a collection efficiency of 99 percent for particles greater than 1.0 micron. The amount of deposited activity is continuously scanned by a lead shielded beta-scintillation detector having a sensitivity of $1x10^{-9}$ μCi/cc for I 131 in a background of 0.75 mr/hr. The sample system, which is common to both the particulate and gas monitors, includes a pump with a 1.5 hp motor, a flowmeter, automatic pressure protecting valves, a flow regulating valve, and isolation valves. The pump and motor are located inside the containment. A sample point is available for taking a sample of the containment atmosphere after an incident for spectrum analysis in the laboratory. During refueling, if the I 131 concentration exceeds 9×10^{-9} μCi/cc, a high activity alarm automatically trips the containment purge air supply and exhaust fans and closes the purge system butterfly valves, thus isolating the purge system. The counting rate of the limiting isotope, I 131, is 8.7×10^{11} cpm/μCi/cc.

11.3.4.3 Containment Gas Monitors

This channel takes the continuous containment atmosphere sample, after it has passed through the particulate filter paper, and draws it through an in-line, easily removable, charcoal cartridge arrangement to the containment gas monitor assembly, which is a fixed volume, lead shielded sampler enclosing a beta-scintillation detector. The sensitivity of this detector is 5×10^{-6} μCi/cc for Kr 85 in a background of 0.75 mr/hr. The sample activity is measured, and then the sample is returned to the containment.

During refueling, if the Kr 85 concentration exceeds 1×10^{-5} μCi/cc, a high activity alarm automatically trips the containment purge air supply and exhaust fans and closes the purge system butterfly valves, thus isolating the purge system.

A purge valve arrangement blocks the normal sample flow to permit purging the detector with a "clean" sample for calibration. Purged gases are discharged to the containment. Protection and isolation are provided by the equipment described under "Containment Particulate Monitors" above.

The counting rates of the limiting isotopes, I 131, Xe 133, and Kr 85 are 8.5×10^7, 3.96×10^7, and 1.04×10^8 cpm/μCi/cc, respectively.

11.3.4.4 Other Area Radiation Monitoring Equipment

This equipment consists of fixed-position, ion-chamber type gamma detectors and associated electronic equipment. These channels warn personnel of any increase in radiation level at locations where personnel may be expected to remain for extended periods of time. All monitors have an eight-decade display (0.1 to 10^7 mr/hr), and for better resolution can also be set on any three consecutive decades within this range. The channels and their ranges are listed in Table 11.3.4-1.

In addition, if the dose rate at the manipulator crane area monitor exceeds 50 mr/hr during refueling, the alarm automatically trips the containment's purge air supply and exhaust fans and closes the purge system butterfly valves, thus isolating the containment from the environment.

TABLE 11.3.4-1

AREA RADIATION MONITORING LOCATIONS, NUMBER, AND RANGES

Channel Location (Number)	Range, mr/hr
Containment High Range Gamma (2)	$0.1-10^7$
Manipulator Crane (2)	$0.1-10^7$
Reactor Containment Area (2)	$0.1-10^7$
In-Core Instrument Transfer Area (2)	$0.1-10^7$
New Fuel Storage Area (1)	$0.1-10^7$
Fuel Pit Bridge (1)	$0.1-10^7$
Auxiliary Building Control Area (1)	$0.1-10^7$
Solid Waste Drum Storage and Handling Area (1)	$0.1-10^7$
Sample Room (1)	$0.1-10^7$
Main Control Room (1)	$0.1-10^7$
Laboratory (1)	$0.1-10^7$
Spare (2)	$0.1-10^7$
Decontamination Area (1)	$0.1-10^7$

11.3.5 ENVIRONMENTAL SURVEY PROGRAM

A post-operational radiation surveillance program has been developed using the knowledge and information obtained from the pre-operational surveillance program. The latter will have been in effect over a two year period and served to train plant personnel in sampling and analytical techniques, aid in identifying those "indicator samples" which may be an indication of a slow build-up of radioactivity in the environment, establish the degree of variability between measurements resulting from seasonal changes in the weather since fluctuations do occur and are expected, generate meaningful environmental data based primarily on scientific and technical requirements, and establish correlation of data between the consulting service and the station's laboratory group.

The ultimate objective of the post-operational surveillance program is verification of the adequacy of radiation source control. Therefore, analytical efforts are being directed toward those samples which have the ability to concentrate the radioelements of concern and afford an integrated and sensitive sampling mechanism. Milk, shellfish (oysters and clams), and silt are considered "indicator samples" and indicative of radioactivity levels in the environment. Samples are collected from the environment surrounding the station on a quarterly basis. Radioanalysis of these samples indicates conditions both in time and space, and thus a slow build-up of radioactivity can be determined. In addition, comparison in the trends of radioactivity levels are more meaningful since the biological variation caused by using many different mediums has been eliminated.

Air Monitoring

Normally the gaseous wastes discharged from the station consist almost entirely of the noble gases, xenon and krypton. The radiation hazard from these gases, due to their inertness, is external radiation exposure. Therefore, radiation surveillance can be maintained by using devices to measure total external body radiation levels in the station environs. Thus, thermoluminesent dosimeters are placed at each of the nine air monitoring stations and read on a monthly basis.

In order to meet the surveillance objective previously stated, it is desirable for the surveillance methods to indicate changes in radioactivity above background levels. Therefore, continuous duty air particulate samplers are also used to ensure that the station presents no hazard to the public. The nine air sampling stations are located at Hog Island Point, Bacon's Castle, Alliance, Colonial Parkway, Lee Hall, Fort Eustis, Newport News, Richmond, and the station site. Establishment of the air particulate network takes into account three general considerations:

1. The average meteorological conditions in the vicinity of the site.
2. The current and projected population densities near the station.
3. The proximity of other nuclear facilities.

Air particulates are analyzed weekly for gross beta radioactivity. Gamma spectrometry is also used to identify the gamma emitters if the sample activity is high enough to warrant such an investigation. Results to date indicate less than 1 pCi per cubic meter of gross beta activity at all stations.

Milk

It has been estimated that since a cow grazes over an area of 160 M^2 per day*, cow's milk affords a good integrated sample. Since milk is one of the best and most direct biosamplers for determining the radiocesium, radiostrontium, and radioiodine levels in the environment, samples are collected from local dairy farms in the vicinity of the station. The Sr-90 and Cs-137 concentrations to date have been less than 20 pCi per liter. Iodine-131 has not been detected in any sample.

Shellfish

Shellfish have the ability to concentrate certain stable elements and radionuclides far above the normal concentrations found in their saline water environment. Oysters and clams, Crasostrea virginica and Mercenaria mercenaria respectively, found in the James River are thus one of the more sensitive mechanisms for the determination of radioactivity released from the station. Samples are collected from upstream of the station and downstream to Newport News. The net beta radioactivity content (gross beta minus the K-40 contribution) is determined and gamma spectrometry used to identify and

---

*Wortley, G., "Contamination of Milk with Radionuclides Dispersed into the Biosphere," A report submitted to the Joint FAO/WHO Expert Committee on Milk Hygiene, WHO Headquarters, Geneva, Switzerland, (April, 1969).

quantify any gamma emitters. Net beta results to date indicate less than 0.35 pCi per gm. wet weight. The only gamma emitter detected has been the naturally occuring radionuclide, K-40.

Additional aquatic vectors useful as integrating samples have been investigated but none have proven as sensitive or indicative as shellfish.

Silt and Sediment

Since shellfish do not concentrate all radionuclides, river bottom sediment samples are collected upstream and opposite the station's intake and discharge canal. Due to the interaction of a number of mechanisms, radionuclides accumulate in silt and bottom sediments. These samples thus afford an integrated sample indicative of average water concentrations. The net beta activity is determined and gamma spectrometry used to identify and quantify any gamma emitters. Results to date indicate less than 20 pCi per gm. for the net beta activity with the only gamma emitter being K-40.

The collection and analysis of the above mentioned samples is yielding meaningful data that meets the primary objective. Should environmental radiation levels increase, the program will be expanded to include other samples that may be considered as vectors for transporting radioactivity from source to man.

This program is periodically reviewed and evaluated to ensure maximum effectiveness.

Equipment

Analytical equipment used at the station for the surveillance programs include:

1. Nuclear Chicago Model 4336, Fifty Sample Automatic Low Background Gas Flow Planchet Counting System with a scintillation "guard" detector, dual scaler, and calculator.
2. Nuclear Measurements Corporation, Model PC-3T Proportional Counter.
3. Nuclear Data Incorporated, Model 2200 Multichannel Pulse Height Analyzer System, 512 Channels, with fast speed printout, oscilloscope, X-Y plotter, and 3" x 3" Harshaw NaI(Tl) Detector.

11.3.6 CONTROL AREAS

The Control Area is described in Section 7.7

The design basis for the walls bounding the Main Control Room is that the radiation dose to personnel inside the Main Control Room be less than 2.5 rem from the Design Basis Accident. This dose includes the 31-day direct radiation dose from activity inside the containment, assuming no cleanup, and the external radiation contribution from the postulated radioactive plume leaking from the containment (as discussed in Section 14.5.6) until Engineered Safeguards return the containment to subatmospheric pressure and terminate the leakage. The radiation sources in the containment during the Design Basis Accident were calculated by the TID-14844 method and are plotted on Figure 14.5.6-4. These sources were assumed to be evenly distributed throughout the containment. The containment was then treated as a volume source and the 31-day direct radiation dose inside the 24 in. thick concrete wall of the Main Control Room was calculated using typical shielding computational techniques. The integrated whole body dose from this source is 0.029 rem.

The radioactive plume was assumed to leak from the containment at a leak rate of 0.1 percent of the contained volume per day until containment leakage was terminated after 40 min. Using Pasquill "F" meteorology conditions and a 1 m per sec wind speed, the plume was converted into a semi-infinite volume source surrounding the Main Control Room. The dose rate from the plume was then calculated using a computer program developed specifically for this case. The integrated whole body dose from this source is 0.013 rem.

Therefore, the integrated whole body dose from both sources is 0.042 rem in 31 days. This dose is well below the criterion dose of 2.5 rem. Thus, the Main Control Room walls, which must be a minimum of 24 in. thick for tornado missile protection, provide more than adequate shielding from radiation.

Special consideration has been given to the design of penetrations and structural details of the Main Control Room so as to establish an acceptable condition of leaktightness.

The air-conditioning systems are installed within the spaces served and designed to provide uninterrupted service under accident conditions. Upon an emergency signal, the normal replenishment air and exhaust systems are isolated manually from the Main Control Room by tight closures in the ductwork. Breathing-quality compressed air is supplied from high pressure storage bottles to maintain a small positive outflow from the Main Control Room for a period exceeding the containment leakage period. This outflow can be verified by means of a pressure gage reading the inside and outside pressure difference in inches of water. The Main Control and Relay Rooms are also provided with an emergency ventilation system fitted with particulate and impregnated charcoal filters to introduce cleaned outside air into the protected spaces upon depletion of the high pressure air. This can continue indefinitely to hold the area pressure above atmospheric to assure outflow leakage.

The radiation level in the Main Control and Relay Rooms is measured by gamma monitors to verify safe operating conditions.

As a secondary precaution, personnel air-packs are available in the control area.

11A.1 ESTIMATED CONCENTRATIONS IN WASTE DISPOSAL SYSTEM WITH 1 PERCENT FAILED FUEL

A flow diagram of the Liquid Waste Disposal System is shown in Fig. 11.2.3.1, and Fig. 11A-1 is a flow chart, listing estimated quantities and activities of waste liquid. Also shown in Fig. 11A-1 are the sources of radionuclides which require release from the station and the waste treatment which each undergoes. Tables 11A-1 through 11A-6 expand this information on a nuclide-by-nuclide basis, concluded by Tables 11A-7 and 11A-8 which list the estimated activity and discharge rate of each radionuclide from the station and the activity in the discharge canal. Tables 11A-9, 10 and 11 show how steam generator tube leakage affects station discharge rate and activity in the discharge canal.

These estimated activities are based upon the following conservative assumptions:

1. Two units, 2,546 MWt each

2. One percent failed fuel and equilibrium corrosion products in each unit

3. DF of mixed-bed demineralizer equal to 10 for all radionuclides, except Y90, Mo99, Cs134, Cs136, and Cs137, for which DF equals 1

4. DF of waste disposal evaporator equal to 10^4 for all radionuclides

5. Decay time of 10 hr for all radionuclides which pass through the waste disposal evaporator

6. No decay time for radionuclides which do not pass through the waste disposal evaporator

7. Steam generator tube leakage of 1 liter/hour per unit

In addition to those radionuclides listed in the tables, a significant amount of tritium may be accumulated in the Reactor Coolant System. Tritium is considered separately because it is relatively insensitive to waste treatment. It is usually in the form of tritiated water, which behaves in the liquid waste disposal system essentially the same as ordinary water. Assuming one unit on its equilibrium cycle and one unit on its initial cycle, it is conservatively estimated in Table 9.1-7 that 5,495 curies per year of tritium may be available for release from the station. The fraction of this amount which is actually discharged from the station depends upon the primary grade water management at the station. However, for this analysis the conservative assumption that all of the tritium produced is discharged.

Based upon the above considerations, the fraction of each radionuclide in the mixture at the point of discharge from the station was determined. Assuming that these fractions remain constant and assuming that the discharge rate of tritium remains constant, the maximum permissible release rate of each radionuclide from the Waste Disposal System was calculated, as follows:

Let X = Gross activity (excluding tritium) discharge rate from the Waste Disposal System (μCi/sec)

f_i = Fraction of nuclide i in the nontritium mixture

^3H = Tritium

Then Xf_i = Discharge rate of nuclide i in a nontritium mixture of known concentration.

$\frac{Xf_i}{Q_{DC}}$ = Concentration in the discharge canal of nuclide i (μCi/cc)

Where Q_{DC} = Discharge canal flow rate (cc/sec)

10CFR20 requires that the concentrations of nuclides in the discharge canal be such that

1. $$\frac{C_a}{MPC_a} + \frac{C_b}{MPC_b} + \frac{C_c}{MPC_c} + \ldots\ldots \leq 1$$

Therefore

2. $$\frac{C_{^3H}}{MPC_{^3H}} + \sum_{i=1}^{m} \frac{Xf_i}{Q_{DC}MPC_I} \leq 1$$

and

$$X \leq \frac{\left[1 - \frac{C_{^3H}}{MPC_{^3H}}\right]}{\sum_{i=1}^{m} \frac{f_i}{Q_{DC}MPC_i}}$$

As can be seen by examining Tables 11A-7 and 11A-8, even with 1 percent failed fuel, the total estimated activity level of all nuclides in the discharge canal excluding tritium will be about 2,200 times below that allowed by 10CFR20. In the actual operation of the plant, it is anticipated that much less than 1 percent failed fuel will exist and that much less than 30 percent (assumed diffusion into reactor coolant) of the tritium produced in the core will be released. Accordingly, much less activity should be discharged than assumed here.

TABLE 11A-1

ACTIVITY FROM LAUNDRY DRAINS

MIXED BED DEMIN.OPTION=	0
DF OF WASTE DISPOSAL SYSTEM FOR THIS SOURCE=	1.00E 00
DECAY TIME IN WASTE DISP SYS (HOURS)=	0.0
FLOW RATE (GAL/YR)=	2.50E 05

IF MIXED BED DEMIN.OPTION= 1,INITIAL ACTIVITY PASSES THROUGH MIXED BED DEMIN WITH DF OF 10 FOR ALL RADIONUCLIDES, EXCEPT Y90,MO99,CS134,CS136,AND CS137

NUCLIDE	INITIAL ACTIVITY (UC/CC)	ACTIVITY AFTER TREATMENT (UC/CC)	DISCHARGE RATE FROM W.D.SYSTEM (CI/YR)
MN54	4.620E-09	4.620E-09	4.372E-06
MN56	9.753E-08	9.753E-08	9.229E-05
CO58	3.935E-10	3.935E-10	3.724E-07
FE59	1.420E-08	1.420E-08	1.344E-05
CO60	1.574E-09	1.574E-09	1.490E-06
SR89	4.791E-09	4.791E-09	4.534E-06
SR90	1.454E-10	1.454E-10	1.376E-07
SR91	2.224E-09	2.224E-09	2.105E-06
Y90	1.711E-10	1.711E-10	1.619E-07
Y91	8.384E-10	8.384E-10	7.934E-07
Y92	9.068E-10	9.068E-10	8.582E-07
ZR95	9.239E-10	9.239E-10	8.744E-07
NB95	9.239E-10	9.239E-10	8.744E-07
MO99	3.816E-06	3.816E-06	3.611E-03
I131	2.874E-06	2.874E-06	2.720E-03
I132	1.069E-06	1.069E-06	1.012E-03
I133	4.671E-06	4.671E-06	4.420E-03
I134	6.502E-07	6.502E-07	6.153E-04
I135	2.447E-06	2.447E-06	2.315E-03
TE132	3.182E-07	3.182E-07	3.012E-04
CS134	3.011E-07	3.011E-07	2.850E-04
CS136	4.449E-08	4.449E-08	4.210E-05
CS137	1.668E-06	1.668E-06	1.579E-03
BA140	2.738E-10	2.738E-10	2.591E-07
LA140	1.061E-09	1.061E-09	1.004E-06
CE144	3.593E-09	3.593E-09	3.400E-06
TOTAL	1.799E-05	1.799E-05	1.703E-02

TABLE IIA-2

ACTIVITY FROM SAMPLING SINKS

MIXED BED DEMIN.OPTION=	0
DF OF WASTE DISPOSAL SYSTEM FOR THIS SOURCE=	1.00E 04
DECAY TIME IN WASTE DISP SYS (HOURS)=	1.00E 01
FLOW RATE (GAL/YR)=	9.20E 04

IF MIXED BED DEMIN.OPTION= 1,INITIAL ACTIVITY PASSES THROUGH MIXED BED DEMIN WITH DF OF 10 FOR ALL RADIONUCLIDES, EXCEPT Y90,MO99,CS134,CS136,AND CS137

NUCLIDE	INITIAL ACTIVITY (UC/CC)	ACTIVITY AFTER TREATMENT (UC/CC)	DISCHARGE RATE FROM W.D.SYSTEM (CI/YR)
MN54	2.700E-03	2.698E-07	9.394E-05
MN56	5.700E-02	3.886E-07	1.353E-04
CO58	2.300E-04	2.291E-08	7.978E-06
FE59	8.300E-03	8.247E-07	2.872E-04
CO60	9.200E-04	9.199E-08	3.203E-05
SR89	2.800E-03	2.784E-07	9.696E-05
SR90	8.500E-05	8.500E-09	2.960E-06
SR91	1.300E-03	6.374E-08	2.220E-05
Y90	1.000E-04	8.973E-09	3.125E-06
Y91	4.900E-04	4.876E-08	1.698E-05
Y92	5.300E-04	7.724E-09	2.690E-06
ZR95	5.400E-04	5.376E-08	1.872E-05
NB95	5.400E-04	5.356E-08	1.865E-05
MO99	2.230E 00	2.011E-04	7.004E-02
I131	1.680E 00	1.621E-04	5.645E-02
I132	6.250E-01	3.071E-06	1.069E-03
I133	2.730E 00	1.962E-04	6.834E-02
I134	3.800E-01	1.381E-08	4.809E-06
I135	1.430E 00	5.089E-05	1.772E-02
TE132	1.860E-01	1.702E-05	5.926E-03
CS134	1.760E-01	1.759E-05	6.127E-03
CS136	2.600E-02	2.082E-06	7.251E-04
CS137	9.750E-01	9.750E-05	3.395E-02
BA140	1.600E-04	1.564E-08	5.448E-06
LA140	6.200E-04	5.218E-08	1.817E-05
CE144	2.100E-03	2.098E-07	7.306E-05
TOTAL	1.052E 01	7.500E-04	2.612E-01

TABLE II A-3

ACTIVITY FROM BORON RECOVERY LETDOWN

MIXED BED DEMIN.OPTION= 1
DF OF WASTE DISPOSAL SYSTEM FOR THIS SOURCE= 1.00E 04
DECAY TIME IN WASTE DISP SYS (HOURS)= 1.00E 01
FLOW RATE (GAL/YR)= 1.23E 06

IF MIXED BED DEMIN.OPTION= 1,INITIAL ACTIVITY PASSES THROUGH MIXED BED DEMIN WITH DF OF 10 FOR ALL RADIONUCLIDES, EXCEPT Y90,MO99,CS134,CS136,AND CS137

NUCLIDE	INITIAL ACTIVITY (UC/CC)	ACTIVITY AFTER TREATMENT (UC/CC)	DISCHARGE RATE FROM W.D.SYSTEM (CI/YR)
MN54	2.700E-03	2.698E-08	1.256E-04
MN56	5.700E-02	3.886E-08	1.809E-04
CO58	2.300E-04	2.291E-09	1.067E-05
FE59	8.300E-03	8.247E-08	3.840E-04
CO60	9.200E-04	9.199E-09	4.283E-05
SR89	2.800E-03	2.784E-08	1.296E-04
SR90	8.500E-05	8.500E-10	3.958E-06
SR91	1.300E-03	6.374E-09	2.968E-05
Y90	1.000E-04	8.973E-09	4.178E-05
Y91	4.900E-04	4.876E-09	2.270E-05
Y92	5.300E-04	7.724E-10	3.596E-06
ZR95	5.400E-04	5.376E-09	2.503E-05
NB95	5.400E-04	5.356E-09	2.494E-05
MO99	2.230E 00	2.011E-04	9.364E-01
I131	1.680E 00	1.621E-05	7.547E-02
I132	6.250E-01	3.071E-07	1.430E-03
I133	2.730E 00	1.962E-05	9.137E-02
I134	3.800E-01	1.381E-09	6.430E-06
I135	1.430E 00	5.089E-06	2.369E-02
TE132	1.860E-01	1.702E-06	7.923E-03
CS134	1.760E-01	1.759E-05	8.191E-02
CS136	2.600E-02	2.082E-06	9.694E-03
CS137	9.750E-01	9.750E-05	4.540E-01
BA140	1.600E-04	1.564E-09	7.283E-06
LA140	6.200E-04	5.218E-09	2.430E-05
CE144	2.100E-03	2.098E-08	9.768E-05
TOTAL	1.052E 01	3.615E-04	1.683E 00

TABLE 11A-4

ACTIVITY FROM SPENT RESIN FLUSH

MIXED BED DEMIN.OPTION=	0
DF OF WASTE DISPOSAL SYSTEM FOR THIS SOURCE=	1.00E 04
DECAY TIME IN WASTE DISP SYS (HOURS)=	1.00E 01
FLOW RATE (GAL/YR)=	3.16E 04

IF MIXED BED DEMIN.OPTION= 1,INITIAL ACTIVITY PASSES THROUGH MIXED BED DEMIN WITH DF OF 10 FOR ALL RADIONUCLIDES, EXCEPT Y90,MO99,CS134,CS136,AND CS137

NUCLIDE	INITIAL ACTIVITY (UC/CC)	ACTIVITY AFTER TREATMENT (UC/CC)	DISCHARGE RATE FROM W.D.SYSTEM (CI/YR)
MN54	3.915E-02	3.911E-06	4.679E-04
MN56	8.265E-01	5.635E-06	6.741E-04
CO58	3.335E-03	3.322E-07	3.973E-05
FE59	1.203E-01	1.196E-05	1.430E-03
CO60	1.334E-02	1.334E-06	1.595E-04
SR89	0.0	0.0	0.0
SR90	0.0	0.0	0.0
SR91	0.0	0.0	0.0
Y90	0.0	0.0	0.0
Y91	0.0	0.0	0.0
Y92	0.0	0.0	0.0
ZR95	0.0	0.0	0.0
NB95	0.0	0.0	0.0
MO99	0.0	0.0	0.0
I131	0.0	0.0	0.0
I132	0.0	0.0	0.0
I133	0.0	0.0	0.0
I134	0.0	0.0	0.0
I135	0.0	0.0	0.0
TE132	0.0	0.0	0.0
CS134	0.0	0.0	0.0
CS136	0.0	0.0	0.0
CS137	0.0	0.0	0.0
BA140	0.0	0.0	0.0
LA140	0.0	0.0	0.0
CE144	0.0	0.0	0.0
TOTAL	1.003E 00	2.317E-05	2.772E-03

TABLE IIA-5

ACTIVITY FROM LABORATORY WASTES

MIXED BED DEMIN.OPTION=	0
DF OF WASTE DISPOSAL SYSTEM FOR THIS SOURCE=	1.00E 04
DECAY TIME IN WASTE DISP SYS (HOURS)=	1.00E 01
FLOW RATE (GAL/YR)=	4.03E 04

IF MIXED BED DEMIN.OPTION= 1,INITIAL ACTIVITY PASSES THROUGH MIXED BED DEMIN WITH DF OF 10 FOR ALL RADIONUCLIDES, EXCEPT Y90,MO99,CS134,CS136,AND CS137

NUCLIDE	INITIAL ACTIVITY (UC/CC)	ACTIVITY AFTER TREATMENT (UC/CC)	DISCHARGE RATE FROM W.D.SYSTEM (CI/YR)
MN54	7.700E-06	7.693E-10	1.174E-07
MN56	1.626E-04	1.108E-09	1.691E-07
CO58	6.560E-07	6.533E-11	9.966E-09
FE59	2.367E-05	2.352E-09	3.588E-07
CO60	2.624E-06	2.623E-10	4.002E-08
SR89	7.986E-06	7.941E-10	1.211E-07
SR90	2.424E-07	2.424E-11	3.698E-09
SR91	3.708E-06	1.818E-10	2.773E-08
Y90	2.852E-07	2.559E-11	3.904E-09
Y91	1.397E-06	1.391E-10	2.121E-08
Y92	1.512E-06	2.203E-11	3.360E-09
ZR95	1.540E-06	1.533E-10	2.339E-08
NB95	1.540E-06	1.527E-10	2.330E-08
MO99	6.360E-03	5.736E-07	8.750E-05
I131	4.791E-03	4.623E-07	7.052E-05
I132	1.782E-03	8.758E-09	1.336E-06
I133	7.786E-03	5.597E-07	8.538E-05
I134	1.084E-03	3.938E-11	6.008E-09
I135	4.078E-03	1.451E-07	2.214E-05
TE132	5.305E-04	4.853E-08	7.404E-06
CS134	5.020E-04	5.018E-08	7.654E-06
CS136	7.415E-05	5.938E-09	9.059E-07
CS137	2.781E-03	2.781E-07	4.242E-05
BA140	4.563E-07	4.461E-11	6.806E-09
LA140	1.768E-06	1.488E-10	2.270E-08
CE144	5.989E-06	5.983E-10	9.127E-08
TOTAL	2.999E-02	2.139E-06	3.263E-04

TABLE 11A-6

ACTIVITY FROM PRIMARY COOLANT SYSTEM LEAKAGE

MIXED BED DEMIN.OPTION=	0
DF OF WASTE DISPOSAL SYSTEM FOR THIS SOURCE=	1.00E 04
DECAY TIME IN WASTE DISP SYS (HOURS)=	1.00E 01
FLOW RATE (GAL/YR)=	3.23E 03

IF MIXED BED DEMIN.OPTION= 1,INITIAL ACTIVITY PASSES THROUGH MIXED BED DEMIN WITH DF OF 10 FOR ALL RADIONUCLIDES, EXCEPT Y90,MO99,CS134,CS136,AND CS137

NUCLIDE	INITIAL ACTIVITY (UC/CC)	ACTIVITY AFTER TREATMENT (UC/CC)	DISCHARGE RATE FROM W.D.SYSTEM (CI/YR)
MN54	2.700E-03	2.698E-07	3.298E-06
MN56	5.700E-02	3.886E-07	4.752E-06
CO58	2.300E-04	2.291E-08	2.801E-07
FE59	8.300E-03	8.247E-07	1.008E-05
CO60	9.200E-04	9.199E-08	1.125E-06
SR89	2.800E-03	2.784E-07	3.404E-06
SR90	8.500E-05	8.500E-09	1.039E-07
SR91	1.300E-03	6.374E-08	7.793E-07
Y90	1.000E-04	8.973E-09	1.097E-07
Y91	4.900E-04	4.876E-08	5.962E-07
Y92	5.300E-04	7.724E-09	9.444E-08
ZR95	5.400E-04	5.376E-08	6.573E-07
NB95	5.400E-04	5.356E-08	6.548E-07
MO99	2.230E 00	2.011E-04	2.459E-03
I131	1.680E 00	1.621E-04	1.982E-03
I132	6.250E-01	3.071E-06	3.755E-05
I133	2.730E 00	1.962E-04	2.399E-03
I134	3.800E-01	1.381E-08	1.688E-07
I135	1.430E 00	5.089E-05	6.222E-04
TE132	1.860E-01	1.702E-05	2.081E-04
CS134	1.760E-01	1.759E-05	2.151E-04
CS136	2.600E-02	2.082E-06	2.546E-05
CS137	9.750E-01	9.750E-05	1.192E-03
BA140	1.600E-04	1.564E-08	1.913E-07
LA140	6.200E-04	5.218E-08	6.380E-07
CE144	2.100E-03	2.098E-07	2.565E-06
TOTAL	1.052E 01	7.500E-04	9.170E-03

TABLE 11A-7
2-15-71

ACTIVITY IN WASTE DISPOSAL SYSTEM WITH NO STEAM GEN LEAKAGE

CALCULATED MAXIMUM ALLOWABLE DISCHARGE RATE (CI/YR)= 9.86E 03
CALCULATED TOTAL DISCHARGE FLOW RATE (GPM)= 3.13

NUCLIDE	I.D.NUMBER	ACTUAL ACTIVITY (UC/CC)	ACTUAL DISCH. RATE (CI/YR)	ALLOWABLE DISCHARGE RATE MIXTURE (CI/YR)	ALLOWABLE DISCHARGE RATE SINGLE (CI/YR)
H3	1	8.813E-01	5.495E 03	5.495E 03	4.617E 06
MN54	2	1.115E-07	6.952E-04	1.539E 00	1.539E 05
MN56	3	1.744E-07	1.088E-03	2.407E 00	1.539E 05
CO58	4	9.469E-09	5.904E-05	1.307E-01	1.385E 05
FE59	5	3.409E-07	2.125E-03	4.704E 00	7.695E 04
CO60	6	3.802E-08	2.371E-04	5.246E-01	4.617E 04
SR89	7	3.764E-08	2.347E-04	5.193E-01	4.617E 03
SR90	8	1.149E-09	7.163E-06	1.585E-02	4.617E 02
SR91	9	8.786E-09	5.478E-05	1.212E-01	7.695E 04
Y90	10	7.246E-09	4.518E-05	9.999E-02	3.078E 04
Y91	11	6.591E-09	4.110E-05	9.095E-02	4.617E 04
Y92	12	1.161E-09	7.242E-06	1.603E-02	9.234E 04
ZR95	13	7.267E-09	4.531E-05	1.003E-01	9.234E 04
NB95	14	7.240E-09	4.514E-05	9.990E-02	1.539E 05
MO99	15	1.624E-04	1.013E 00	2.241E 03	6.156E 04
I131	16	2.192E-05	1.367E-01	3.025E 02	4.617E 02
I132	17	5.694E-07	3.550E-03	7.857E 00	1.231E 04
I133	18	2.672E-05	1.666E-01	3.687E 02	1.539E 03
I134	19	1.005E-07	6.267E-04	1.387E 00	3.078E 04
I135	20	7.117E-06	4.438E-02	9.821E 01	6.156E 03
TE132	21	2.304E-06	1.437E-02	3.179E 01	3.078E 04
CS134	22	1.420E-05	8.855E-02	1.960E 02	1.385E 04
CS136	23	1.682E-06	1.049E-02	2.321E 01	9.234E 04
CS137	24	7.870E-05	4.907E-01	1.086E 03	3.078E 04
BA140	25	2.115E-09	1.319E-05	2.919E-02	3.078E 04
LA140	26	7.078E-09	4.413E-05	9.767E-02	3.078E 04
CE144	27	2.836E-08	1.768E-04	3.913E-01	1.539E 04
TOTAL		8.816E-01	5.497E 03	9.862E 03	
TOTAL (NON-TRITIUM)		3.165E-04	1.973E 00	4.367E 03	

RATIO OF ALLOWABLE TO ACTUAL DISCHARGE RATE OF MIXTURE,EXCLUDING H3 = 2.213E 03

ACTIVITY IN DISCHARGE CANAL

TABLE IIA-8

2-15-71

DISCHARGE CANAL FLOW RATE (GPM)= 7.73E 05
TRITIUM DISCHARGE RATE (CURIES/YEAR)= 5.50E 03

NUCLIDE	I.D.NUMBER	MPC (UC/CC)	HALF LIFE (DAYS)	ACTUAL ACTIVITY (UC/CC)	RATIO (UC/CC) /MPC	ALLOWABLE ACTIVITY (UC/CC)	RATIO (UC/CC) /MPC
H3	1	3.000E-03	4.481E 03	3.570E-06	1.190E-03	3.570E-06	1.190E-03
MN54	2	1.000E-04	3.121E 02	4.517E-13	4.517E-09	9.997E-10	9.997E-06
MN56	3	1.000E-04	1.075E-01	7.067E-13	7.067E-09	1.564E-09	1.564E-05
CO58	4	9.000E-05	7.161E 01	3.836E-14	4.262E-10	8.489E-11	9.433E-07
FE59	5	5.000E-05	4.506E 01	1.381E-12	2.762E-08	3.056E-09	6.113E-05
CO60	6	3.000E-05	1.919E 03	1.540E-13	5.134E-09	3.409E-10	1.136E-05
SR89	7	3.000E-06	5.109E 01	1.525E-13	5.082E-08	3.374E-10	1.125E-04
SR90	8	3.000E-07	1.044E 04	4.654E-15	1.551E-08	1.030E-11	3.433E-05
SR91	9	5.000E-05	4.051E-01	3.560E-14	7.119E-10	7.878E-11	1.576E-06
Y90	10	2.000E-05	2.665E 00	2.936E-14	1.468E-09	6.497E-11	3.248E-06
Y91	11	3.000E-05	5.898E 01	2.670E-14	8.901E-10	5.909E-11	1.970E-06
Y92	12	6.000E-05	1.499E-01	4.706E-15	7.843E-11	1.041E-11	1.736E-07
ZR95	13	6.000E-05	6.521E 01	2.944E-14	4.907E-10	6.515E-11	1.086E-06
NB95	14	1.000E-04	3.503E 01	2.933E-14	2.933E-10	6.491E-11	6.491E-07
MO99	15	4.000E-05	2.795E 00	6.579E-10	1.645E-05	1.456E-06	3.640E-02
I131	16	3.000E-07	8.061E 00	8.881E-11	2.960E-04	1.966E-07	6.552E-01
I132	17	8.000E-06	9.583E-02	2.307E-12	2.883E-07	5.105E-09	6.381E-04
I133	18	1.000E-06	8.747E-01	1.083E-10	1.083E-04	2.396E-07	2.396E-01
I134	19	2.000E-05	3.646E-02	4.072E-13	2.036E-08	9.012E-10	4.506E-05
I135	20	4.000E-06	2.795E-01	2.883E-11	7.208E-06	6.381E-08	1.595E-02
TE132	21	2.000E-05	3.247E 00	9.335E-12	4.667E-07	2.066E-08	1.033E-03
CS134	22	9.000E-06	7.567E 02	5.754E-11	6.393E-06	1.273E-07	1.415E-02
CS136	23	6.000E-05	1.300E 00	6.815E-12	1.136E-07	1.508E-08	2.514E-04
CS137	24	2.000E-05	1.088E 04	3.188E-10	1.594E-05	7.056E-07	3.528E-02
BA140	25	2.000E-05	1.279E 01	8.569E-15	4.285E-10	1.896E-11	9.482E-07
LA140	26	2.000E-05	1.674E 00	2.867E-14	1.434E-09	6.346E-11	3.173E-06
CE144	27	1.000E-05	2.854E 02	1.149E-13	1.149E-08	2.542E-10	2.542E-05
TOTAL				3.572E-06	1.641E-03	6.408E-06	1.000E 00

TABLE 11A-9

ACTIVITY FROM STM.GEN.BLOWDOWN

MIXED BED DEMIN.OPTION=	0
DF OF WASTE DISPOSAL SYSTEM FOR THIS SOURCE=	1.00E 00
DECAY TIME IN WASTE DISP SYS (HOURS)=	0.0
FLOW RATE (GAL/YR)=	1.14E 07

IF MIXED BED DEMIN.OPTION= 1,INITIAL ACTIVITY PASSES THROUGH MIXED BED DEMIN WITH DF OF 10 FOR ALL RADIONUCLIDES, EXCEPT Y90,MO99,CS134,CS136,AND CS137

NUCLIDE	INITIAL ACTIVITY (UC/CC)	ACTIVITY AFTER TREATMENT (UC/CC)	DISCHARGE RATE FROM W.D.SYSTEM (CI/YR)
MN54	5.750E-07	5.750E-07	2.490E-02
MN56	9.550E-07	9.550E-07	4.136E-02
CO58	4.820E-08	4.820E-08	2.087E-03
FE59	1.740E-06	1.740E-06	7.535E-02
CO60	1.970E-07	1.970E-07	8.531E-03
SR89	5.900E-07	5.900E-07	2.555E-02
SR90	2.960E-08	2.960E-08	1.282E-03
SR91	6.600E-08	6.600E-08	2.858E-03
Y90	5.200E-08	5.200E-08	2.252E-03
Y91	1.170E-07	1.170E-07	5.067E-03
Y92	2.030E-08	2.030E-08	8.791E-04
ZR95	1.240E-07	1.240E-07	5.370E-03
NB95	1.270E-07	1.270E-07	5.500E-03
MO99	5.180E-04	5.180E-04	2.243E 01
I131	3.120E-04	3.120E-04	1.351E 01
I132	3.520E-05	3.520E-05	1.524E 00
I133	2.350E-04	2.350E-04	1.018E 01
I134	2.320E-06	2.320E-06	1.005E-01
I135	5.750E-05	5.750E-05	2.490E 00
TE132	2.820E-05	2.820E-05	1.221E 00
CS134	4.700E-05	4.700E-05	2.035E 00
CS136	3.670E-06	3.670E-06	1.589E-01
CS137	3.710E-05	3.710E-05	1.607E 00
BA140	2.940E-08	2.940E-08	1.273E-03
LA140	8.700E-08	8.700E-08	3.768E-03
CE144	4.460E-07	4.460E-07	1.931E-02
TOTAL	1.281E-03	1.281E-03	5.548E 01

TABLE IIA-10 2-15-71

ACTIVITY IN WASTE DISPOSAL SYSTEM WITH STEAM GEN LEAKAGE

CALCULATED MAXIMUM ALLOWABLE DISCHARGE RATE (CI/YR)= 7.03E 03
CALCULATED TOTAL DISCHARGE FLOW RATE (GPM)= 24.88

NUCLIDE	I.D.NUMBER	ACTUAL ACTIVITY (UC/CC)	ACTUAL DISCH. RATE (CI/YR)	ALLOWABLE DISCHARGE RATE MIXTURE (CI/YR)	SINGLE (CI/YR)
H3	1	1.109E-01	5.495E 03	5.495E 03	4.617E 06
MN54	2	5.167E-07	2.560E-02	6.822E-01	1.539E 05
MN56	3	8.568E-07	4.244E-02	1.131E 00	1.539E 05
CO58	4	4.333E-08	2.146E-03	5.721E-02	1.385E 05
FE59	5	1.564E-06	7.748E-02	2.065E 00	7.695E 04
CO60	6	1.770E-07	8.768E-03	2.337E-01	4.617E 04
SR89	7	5.205E-07	2.578E-02	6.872E-01	4.617E 03
SR90	8	2.602E-08	1.289E-03	3.436E-02	4.617E 02
SR91	9	5.880E-08	2.913E-03	7.764E-02	7.695E 04
Y90	10	4.637E-08	2.297E-03	6.122E-02	3.078E 04
Y91	11	1.031E-07	5.108E-03	1.361E-01	4.617E 04
Y92	12	1.789E-08	8.863E-04	2.362E-02	9.234E 04
ZR95	13	1.093E-07	5.415E-03	1.443E-01	9.234E 04
NB95	14	1.119E-07	5.545E-03	1.478E-01	1.539E 05
MO99	15	4.732E-04	2.344E 01	6.249E 02	6.156E 04
I131	16	2.755E-04	1.365E 01	3.638E 02	4.617E 02
I132	17	3.084E-05	1.528E 00	4.072E 01	1.231E 04
I133	18	2.088E-04	1.034E 01	2.757E 02	1.539E 03
I134	19	2.041E-06	1.011E-01	2.694E 00	3.078E 04
I135	20	5.116E-05	2.534E 00	6.755E 01	6.156E 03
TE132	21	2.494E-05	1.236E 00	3.293E 01	3.078E 04
CS134	22	4.287E-05	2.124E 00	5.661E 01	1.385E 04
CS136	23	3.420E-06	1.694E-01	4.515E 00	9.234E 04
CS137	24	4.234E-05	2.097E 00	5.590E 01	3.078E 04
BA140	25	2.597E-08	1.286E-03	3.428E-02	3.078E 04
LA140	26	7.694E-08	3.812E-03	1.016E-01	3.078E 04
CE144	27	3.934E-07	1.949E-02	5.195E-01	1.539E 04
TOTAL		1.121E-01	5.552E 03	7.026E 03	
TOTAL (NON-TRITIUM)		1.160E-03	5.746E 01	1.531E 03	

RATIO OF ALLOWABLE TO ACTUAL DISCHARGE RATE OF MIXTURE,EXCLUDING H3 = 2.665E 01

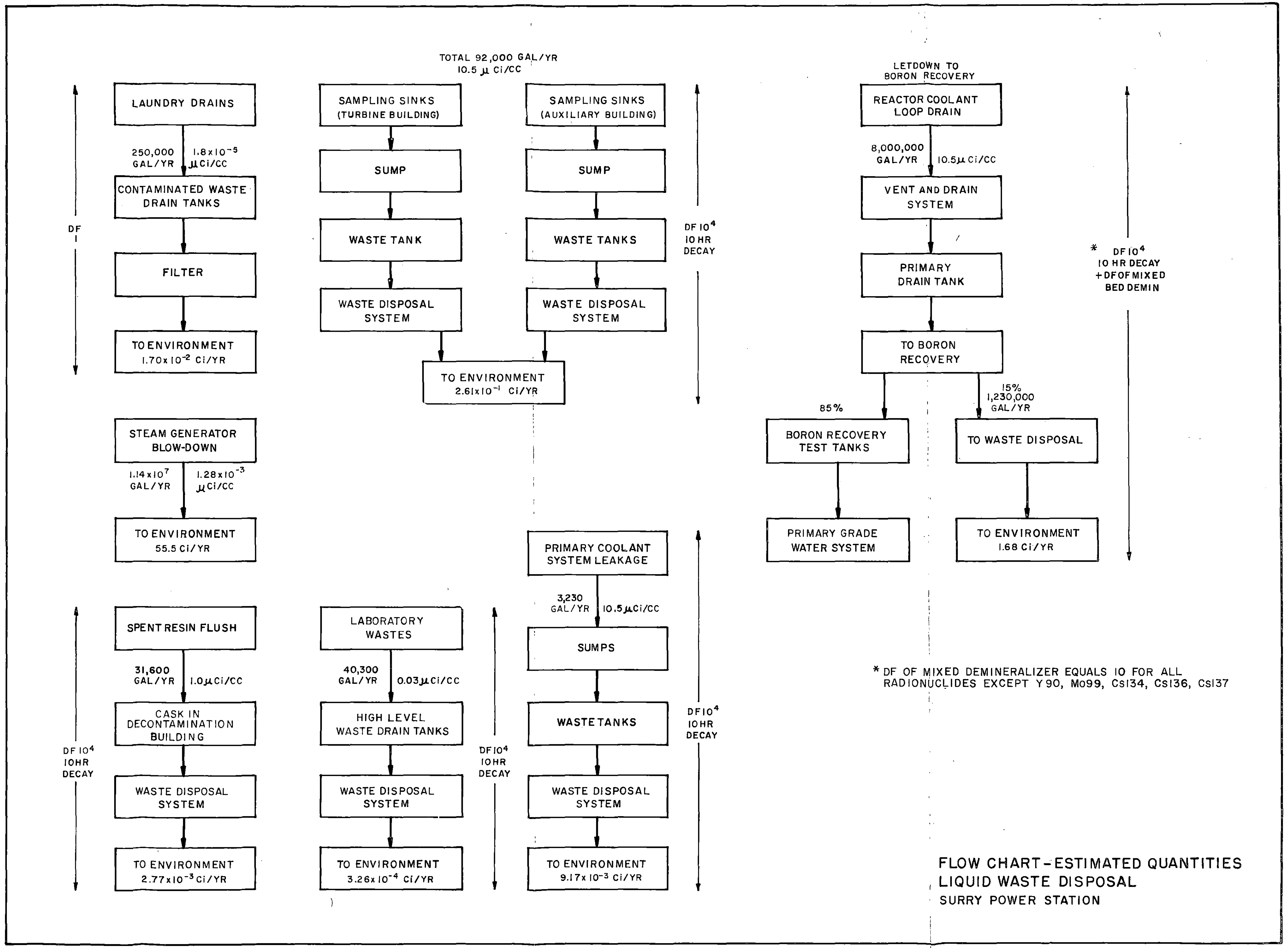
ACTIVITY IN DISCHARGE CANAL

TABLE IIA-11 2-15-71

DISCHARGE CANAL FLOW RATE (GPM)= 7.73E 05
TRITIUM DISCHARGE RATE (CURIES/YEAR)= 5.50E 03

NUCLIDE	I.D.NUMBER	MPC (UC/CC)	HALF LIFE (DAYS)	ACTUAL ACTIVITY (UC/CC)	RATIO (UC/CC) /MPC	ALLOWABLE ACTIVITY (UC/CC)	RATIO (UC/CC) /MPC
H3	1	3.000E-03	4.481E 03	3.570E-06	1.190E-03	3.570E-06	1.190E-03
MN54	2	1.000E-04	3.121E 02	1.663E-11	1.663E-07	4.433E-10	4.433E-06
MN56	3	1.000E-04	1.075E-01	2.758E-11	2.758E-07	7.350E-10	7.350E-06
CO58	4	9.000E-05	7.161E 01	1.395E-12	1.550E-08	3.717E-11	4.130E-07
FE59	5	5.000E-05	4.506E 01	5.034E-11	1.007E-06	1.342E-09	2.683E-05
CO60	6	3.000E-05	1.919E 03	5.697E-12	1.899E-07	1.518E-10	5.061E-06
SR89	7	3.000E-06	5.109E 01	1.675E-11	5.585E-06	4.465E-10	1.488E-04
SR90	8	3.000E-07	1.044E 04	8.375E-13	2.792E-06	2.232E-11	7.441E-05
SR91	9	5.000E-05	4.051E-01	1.893E-12	3.785E-08	5.045E-11	1.009E-06
Y90	10	2.000E-05	2.665E 00	1.493E-12	7.463E-08	3.978E-11	1.989E-06
Y91	11	3.000E-05	5.898E 01	3.319E-12	1.106E-07	8.846E-11	2.949E-06
Y92	12	6.000E-05	1.499E-01	5.759E-13	9.598E-09	1.535E-11	2.558E-07
ZR95	13	6.000E-05	6.521E 01	3.518E-12	5.864E-08	9.378E-11	1.563E-06
NB95	14	1.000E-04	3.503E 01	3.603E-12	3.603E-08	9.602E-11	9.602E-07
MO99	15	4.000E-05	2.795E 00	1.523E-08	3.808E-04	4.060E-07	1.015E-02
I131	16	3.000E-07	8.061E 00	8.868E-09	2.956E-02	2.364E-07	7.878E-01
I132	17	8.000E-06	9.583E-02	9.927E-10	1.241E-04	2.646E-08	3.307E-03
I133	18	1.000E-06	8.747E-01	6.721E-09	6.721E-03	1.791E-07	1.791E-01
I134	19	2.000E-05	3.646E-02	6.569E-11	3.284E-06	1.751E-09	8.754E-05
I135	20	4.000E-06	2.795E-01	1.647E-09	4.117E-04	4.389E-08	1.097E-02
TE132	21	2.000E-05	3.247E 00	8.028E-10	4.014E-05	2.140E-08	1.070E-03
CS134	22	9.000E-06	7.567E 02	1.380E-09	1.533E-04	3.678E-08	4.087E-03
CS136	23	6.000E-05	1.300E 00	1.101E-10	1.835E-06	2.934E-09	4.890E-05
CS137	24	2.000E-05	1.088E 04	1.363E-09	6.814E-05	3.632E-08	1.816E-03
BA140	25	2.000E-05	1.279E 01	8.358E-13	4.179E-08	2.228E-11	1.114E-06
LA140	26	2.000E-05	1.674E 00	2.477E-12	1.238E-07	6.601E-11	3.300E-06
CE144	27	1.000E-05	2.854E 02	1.266E-11	1.266E-06	3.375E-10	3.375E-05
TOTAL				3.608E-06	3.866E-02	4.565E-06	1.000E 00

FIG. 11A-1
DEC. 1, 1969



FLOW CHART - ESTIMATED QUANTITIES
LIQUID WASTE DISPOSAL
SURRY POWER STATION

TABLE OF CONTENTS

12 CONDUCT OF OPERATIONS

12.1 GENERAL

Vepco has a long history of building and operating safe electric generating stations, both coal-fired and hydro, and safe transmission systems. Through thorough planning and careful implementation, Vepco has pioneered such industry milestones as a 500 kilovolt transmission system and new innovations in the Mine Mouth Station concept. In the construction of its first nuclear station, Vepco has applied the same managerial techniques used in the past to enforce high standards of design and construction.

For the past fourteen years in the preparation for the eventual use of nuclear energy in the generation of electricity, Vepco has followed a policy of educating and training its officers, managers, and employees, both supervisory and non-supervisory, in the rapidly expanding nuclear field. Vepco management is consequently prepared for the added responsibility of operating its first nuclear generating station. The station has been constructed and will be operated in such a manner that the safety of the public and operating personnel will be assured.

12.2 ORGANIZATION

12.2.1 NUCLEAR PARTICIPATION BY VIRGINIA ELECTRIC & POWER COMPANY

Vepco has participated in nuclear power activities since the passage of the Atomic Energy Act of 1954. In 1954, Vepco participated in a series of studies with Stone & Webster Engineering Corporation. In 1955, the Company commenced further studies with Carolina Power and Light Company, Duke Power Company, and South Carolina Electric & Gas Company. In 1956, they formed Carolinas Virginia Nuclear Power Associates, Inc. (CVNPA), a nonprofit, membership organization. Subsequently, under the third round invitation of the Reactor Demonstration Program, CVNPA built and operated the CVTR, a 65 MWt heavy water moderated and cooled, pressure tube reactor located at Parr, South Carolina. The CVTR achieved criticality for the first time in March, 1963, and produced its first electric power in December, 1963. From the early summer of 1964 to 1967, the CVTR produced electric power on a reliable basis. CVNPA, and Westinghouse as its subcontractor, carried out an extensive research and development program for the Commission, both before and after construction of the CVTR. The plant was decommissioned in 1967 after fulfilling the objectives of the program.

Vepco was a significant participant in the work of CVNPA since its incorporation. Employees of Vepco served on the CVNPA Board of Directors and on several of the management committees, including the Steering Committee, the Technical Advisory Committee, and the Manpower Committee. Four Vepco employees were associated with CVNPA on a resident basis and had an integrated total of 22 man-years of project experience in responsible positions relating to design, engineering, construction, operation, maintenance, health physics, and chemistry. Individual periods of resident service with CVTR ranged from two to nine years.

12.2.2 VEPCO ORGANIZATION

The execution of the Surry Power Station project was solely the responsibility of Vepco. In this connection, Vepco engaged Stone & Webster Engineering Corporation as its agent for engineering and construction and contracted with Westinghouse Electric Corporation for furnishing the Nuclear Steam Supply Systems, the nuclear fuel and the turbine generators. Dames & Moore, Inc. was retained to perform Site Geology, Hydrology and Seismology studies, and NUS Corporation was retained for Site Meteorology, Climatology, and general nuclear consultation.

Station Organization

The station's organization chart is shown in Figures 12.2-1A and 12.2-1B. The numbers shown in these figures for the Shift Supervisor, the Control Room Operator, the Assistant Control Room Operator, and the Auxiliary Operator are for one shift with one unit and two units respectively in operation. (Section 12.4)

The minimum qualifications of the key supervisory positions are specified below:

Manager

Degree in Engineering or equivalent in technical education and experience. Sufficient station and supervisory experience as to be well versed in the design, operation, and maintenance requirements of power generating facilities. Specialized training in nuclear technology.

VIRGINIA ELECTRIC AND POWER COMPANY

ORGANIZATION CHART

SURRY POWER STATION

1-UNIT OPERATION

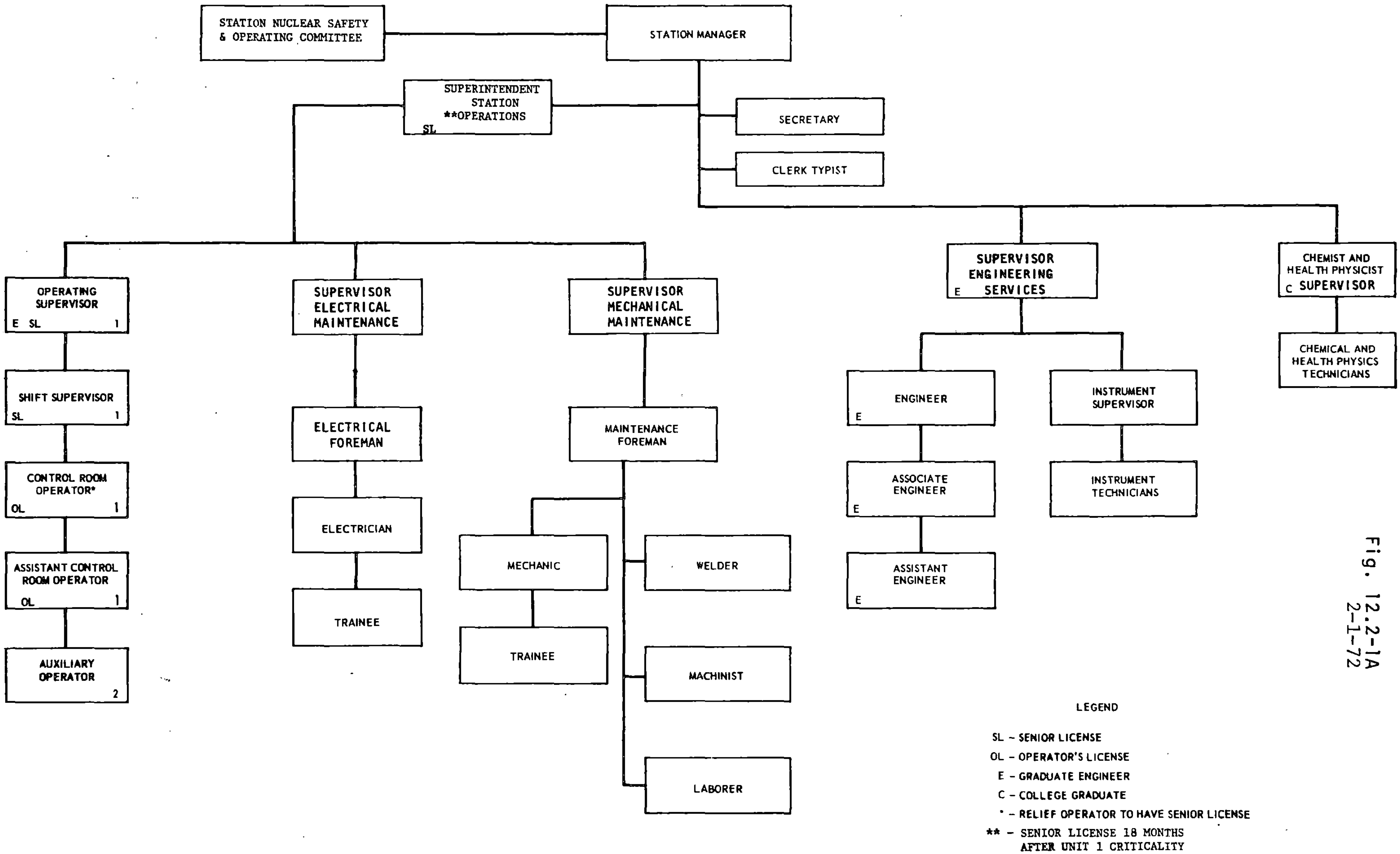


Fig. 12.2-1A
2-1-72

VIRGINIA ELECTRIC AND POWER COMPANY

ORGANIZATION CHART

SURRY POWER STATION

2-UNIT OPERATION

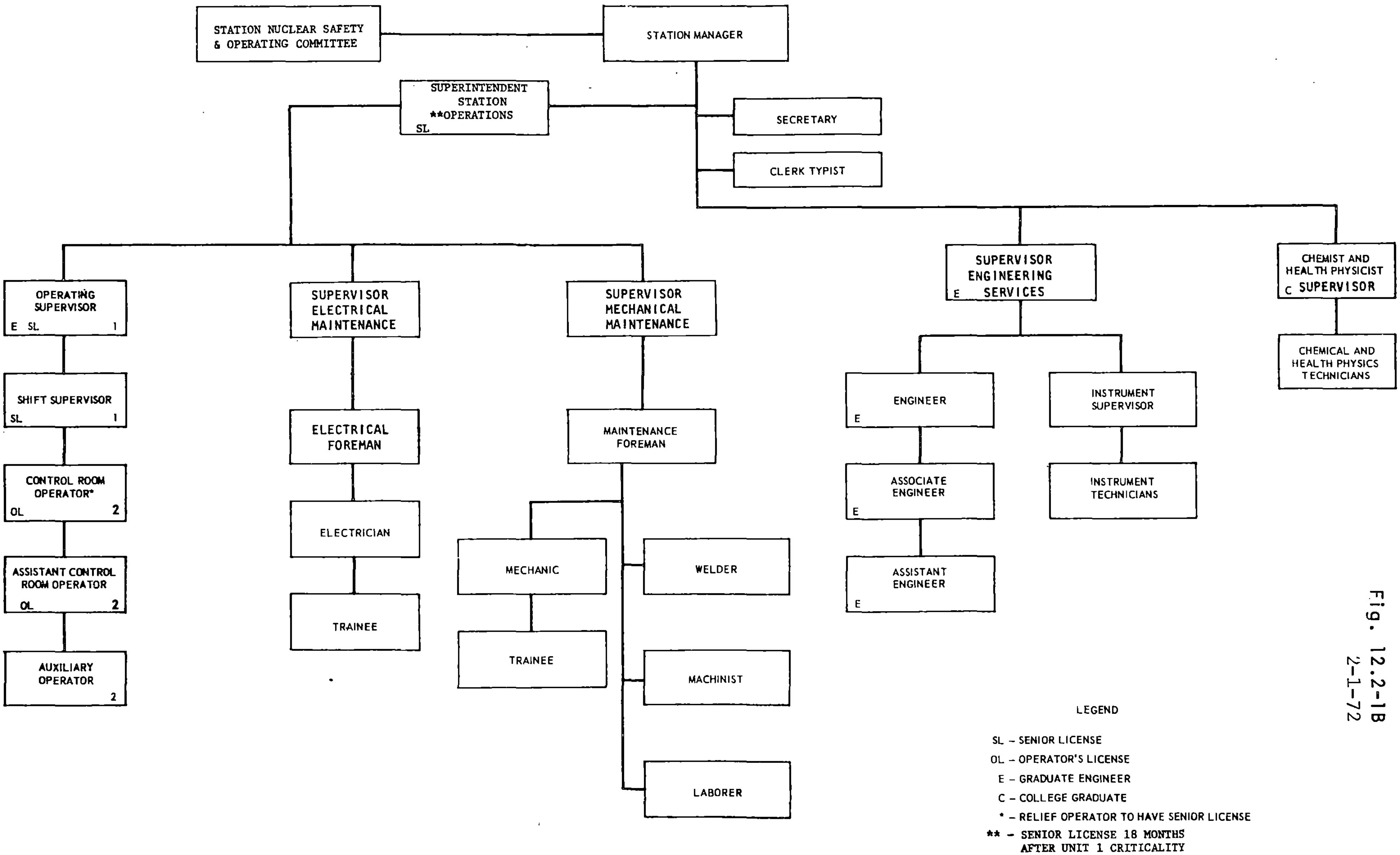


Fig. 12.2-1B
2-1-72

Superintendent-Station Operations

Degree in Engineering or equivalent in technical education and experience. Supervisory experience in an operating nuclear facility and possession of a senior reactor operator license within 18 months after initial criticality of Unit No. 1.

Operating Supervisor

Degree in Engineering or equivalent in technical education and experience. Supervisory experience in an operating nuclear facility and possession of a senior reactor operator license.

Supervisor-Electrical Maintenance

Appropriate education and technical experience. Supervisory experience in the installation and maintenance of electric systems and equipment normally found in a power generating facility.

Supervisor-Mechanical Maintenance

Appropriate education and technical experience. Supervisory experience in the installation and maintenance of mechanical equipment normally found in a power generating facility.

Supervisor-Engineering Services

Degree of Engineering or equivalent in technical education and experience. The ability to supervise and evaluate the performance of pressurized water reactors and reactor associated systems.

Chemistry and Health Physics Supervisor

Degree in Chemistry or Health Physics or equivalent in technical education and experience. Supervisory experience in these areas in an operating nuclear facility.

The relationship of the Manager to other levels of company management is shown in Figure 12.2-2.

Operating Lines of Authority

Station Manager

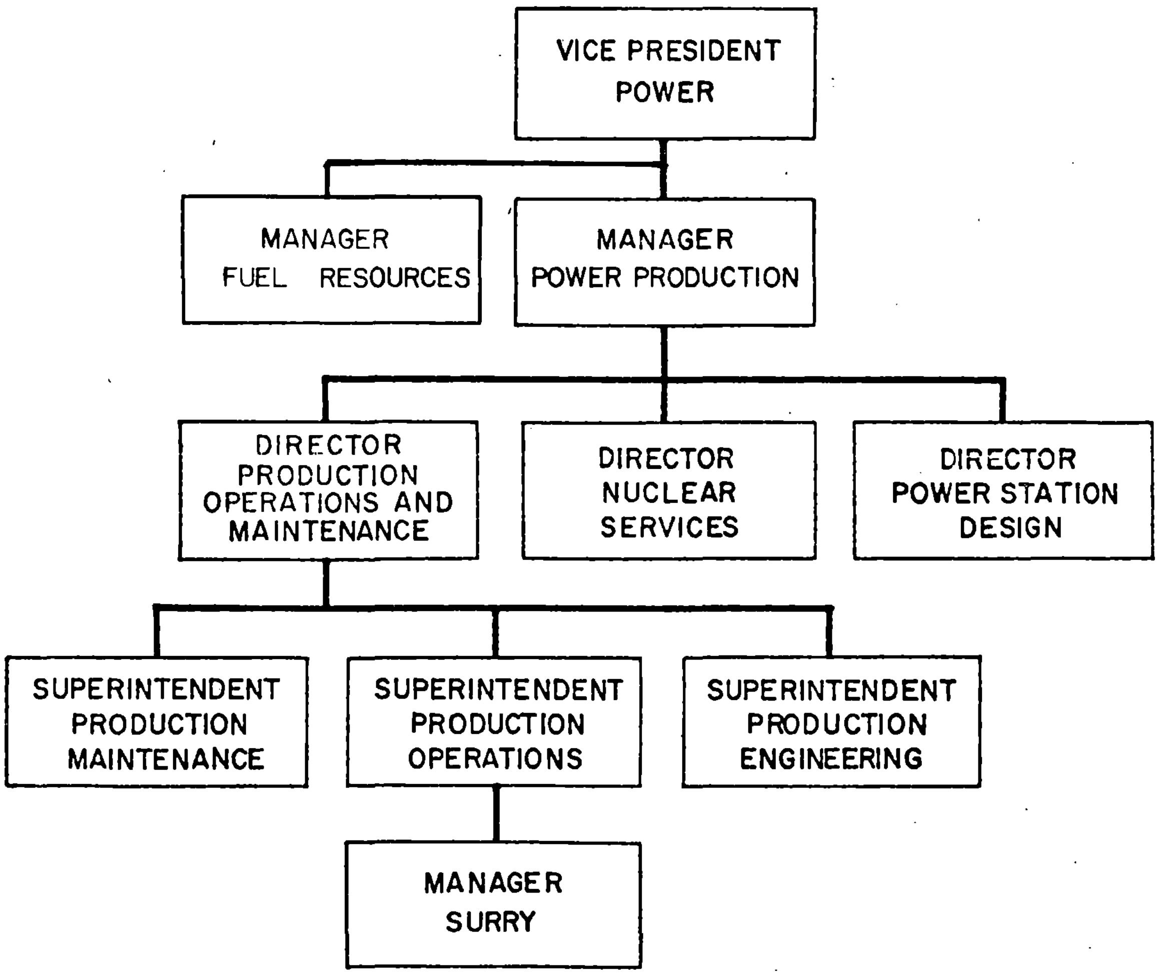
The Station Manager has supervisory control over all Vepco personnel assigned to the Surry Power Station and administrative control over all other personnel or persons within the site boundaries.

He is responsible for maintaining the station as a functional part of the Vepco power generating system in a safe manner by insuring compliance with the Technical Specifications.

Fig. 12.2-2
6-30-71

POWER PRODUCTION DEPARTMENT ORGANIZATION CHART

VIRGINIA ELECTRIC AND POWER COMPANY



He is a member of the System Nuclear Safety and Operating Committee for Surry Power Station and the Chairman of the Station Nuclear Safety and Operating Committee.

The Manager has the Superintendent - Station Operations, the Supervisor - Engineering Services, and the Station Chemistry and Health Physics Supervisor reporting directly to him and is responsible for coordinating their functions.

In general, the Manager is the representative of company management. He is empowered to implement all company policy with regard to the operations of the facility and is responsible for the coordination of tactical station functions with outside agencies and services.

In the absence of the Manager the Superintendent - Station Operations assumes the duties of this position.

Superintendent - Station Operations

The Superintendent - Station Operations has the responsibility of coordinating the work of the station operating and maintenance groups. He is responsible for the safe operation of the reactor plant and secondary plant and overall station maintenance and refueling operations.

The Operating Supervisor, Supervisor-Electrical Maintenance, and Supervisor-Mechanical Maintenance report to him, and any functions that require coordination with Engineering and/or Chemistry and Health Physics groups are also his responsibility.

He is the Vice Chairman of the Station Nuclear Safety and Operating Committee.

The Superintendent reports to the Manager any unusual occurrences in connection with station operation.

He is responsible for the maintenance of complete operating data and approval of any changes in operating procedures. In addition, he is responsible for maintaining personnel AEC operating licenses on a current basis.

In the absence of the Superintendent-Station Operations the Operating Supervisor assumes the duties of this position.

Operating Supervisor

The Operating Supervisor is responsible for the safe and productive operation of the station through direct supervision of the Shift Supervisors and thereby operation of the reactors and secondary plants.

He is a member of the Station Nuclear Safety and Operating Committee.

He is responsible for implementation of Operator Training policies.

He endorses and/or implements any proposed changes to operating procedures and reports any unusual occurrences, deviations from the Technical Specifications, or unusual events to the Superintendent of Station Operations.

In the absence of the Operating Supervisor a qualified engineer or Shift Supervisor, assigned by either the Station Manager or Superintendent - Station Operations, will assume his duties.

He ensures that all required operating data is properly logged or recorded and that complete records are maintained in regard to station operations. He ensures that each shift is adequately staffed and that reserve personnel are available.

Supervisor - Engineering Services

This Supervisor has the responsibility of a continuing technical evaluation of the performance of the station and has administrative control of the instrumentation group at the station. The non-supervisory engineering group and the Instrument Supervisor report directly to the Supervisor - Engineering Services and he in turn reports to the Station Manager. When necessary he coordinates his activities with the Superintendent-Station Operations, and the Chemistry and Health Physics Supervisor.

He is responsible for originating any procedures involving experiments or tests on the reactors or secondary plants.

He is responsible for periodic determinations of fuel composition and burnup.

He is responsible for the determination of fuel loading sequences and is available for technical support during any refueling or fuel loading of the reactor.

He has the responsibility of obtaining performance data on overall station and equipment operation.

He is responsible for evaluating and determining the control rod worths and operating sequences.

He is responsible for resolving, by accepted engineering practices, any design modifications or engineering projects as the need arises at the station.

His responsibility in nuclear and conventional station instrumentation is to ensure that all systems are properly functional, calibrated and checked on a specified basis.

This supervisor shall be a member of the Station Nuclear Safety and Operating Committee. His group also has the responsibility for originating all written technical reports for company and regulatory purposes.

Chemistry & Health Physics Supervisor

This supervisor has the responsibility of insuring that all routine chemical analysis and evaluation is properly performed during all phases of station operation. He also directs the activities of the Health Physics group at the station, insuring that all data is correctly interpreted and acted upon when necessary. He is directly responsible to the Station Manager and coordinates his efforts with the Superintendent-Station Operations and the Supervisor-Engineering Services. He is also a member of the Station Nuclear Safety and Operating Committee. He is responsible for keeping records of radiological exposure to all persons working or visiting within the station's restricted areas as designated. This includes the organization of all such written reports required for company or regulatory purposes.

He conducts regular surveys of the station and environs to ascertain the levels of natural and induced radioactivity.

He has the responsibility of determining the radiation levels of all areas where work is to be conducted when it is anticipated such radiation may exist and has the responsibility of establishing, recording, checking, and suitable posting all areas where radiation is a known problem.

He has the responsibility for keeping records and checking on all radioactive material release and shipments from the station.

He has the responsibility of writing and amending the Health Physics Manual of Procedures and Safety Protection and insuring that all station personnel receive instruction in the implementation of such procedures.

In general, the Chemistry and Health Physics Supervisor directs the activities of his group in such a manner as to prevent the exposure of station personnel to excessive accumulative doses of radiation and to prevent the spread of radioactive contamination. He further ensures that the chemical treatment of all liquid systems in the station are such that corrosion products and carryover are minimized. All activities in these areas are coordinated with other station groups to ensure full awareness of problems when and if they arise.

Supervisor-Mechanical & Electrical Maintenance

The Supervisor-Mechanical Maintenance and the Supervisor-Electrical Maintenance have the overall responsibility of administering mechanical and electrical maintenance. They report directly to the Superintenden-Station Operations. The group working for these supervisors also includes the station labor force.

They are responsible for the maintenance necessary to assure safe and prolonged operation of all station components not specifically assigned to other groups, and when required they direct approved station system modifications. They are responsible for the implementation of safe working practice within their group and coordinate their groups' activity with both Operations and Health Physics.

They have the responsibility of writing, and amending when necessary, maintenance procedures where such documentation is necessary and implementing training programs within their group.

They are responsible for keeping proper maintenance records and files.

They are members of the Station Nuclear Safety and Operating Committee.

Instrument Supervisor

The Instrument Supervisor reports directly to the Supervisor of Engineering Services and coordinates the efforts of his group with the operating and maintenance groups.

He is responsible for maintaining adequate and accurate instrumentation status on all station systems, and recalibrating instrumentation and controls.

This supervisor is responsible for the training of his group and writes, and when necessary, amends the instrumentation procedures. He is further responsible for maintaining calibrated test instrumentation for test and experimental work.

Shift Supervisors

This supervisor has the responsibility of directing the actions of the station operators to ensure safe and prudent operation of the facility. He reports all abnormal occurrences to the Operating Supervisor. He also is responsible for the entire facility during the absence of his supervisors.

He is responsible for ensuring that the operating procedures are kept on a current basis and when necessary initiates changes in operating procedures. He ensures that his shift is adequately manned and that other groups are called in to work when necessary.

He keeps a shift log noting non-routine operations.

In general, the Shift Supervisor is informed of all actions by other groups that affect the operation of the station. He is responsible for station security.

The Shift Supervisor is responsible for the training of personnel assigned to his shift in coordination with the overall station training, and re-training programs.

12.3 TRAINING

The key staff complement for the Surry Power Station consists of selected individuals with nuclear and/or conventional power station experience who have attained a high level of development in the Westinghouse Nuclear Training Program. A number in this group have received additional training at the graduate level in Nuclear Engineering and Health Physics at the following universities:

University of North Carolina at Chapel Hill
North Carolina State University
University of Virginia

Under a contractual agreement with Westinghouse, a comprehensive training program involving reactor physics, actual pressurized water reactor operation, and the study of station component systems has been administered to nineteen members of the Surry nuclear engineering and operating staff.

Many of the above group are available, and were available, for on site training duties. Training was given to all station personnel, varying according to need. As new employees join the station staff they will be entered in formal training programs. Training programs are also available to upgrade or to improve the skills of present employees at the station.

Personnel to staff the Surry Power Station have been selected to insure that each individual possesses the education, training, and experience necessary to satisfactorily perform his assigned function.

To augment the formal education, training, and experience received by station personnel prior to being stationed at Surry, numerous training programs have been instituted to specifically familiarize employees with the Surry facility.

Tables 12.3-1 and 12.3-2 summarizes by job classification the training received.

TABLE 12.3-2 (CON'D)

COURSE NO.	DESCRIPTION
15	Basic radiological protection sponsored by U. S. Public Health Service.

TABLE 12.3-1
TRAINING PROGRAMS
SURRY POWER STATION

POSITION	COURSE**														
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
Station Manager	X	X	X	X											
Supt.- Station Oper	X	X	X	X		X									
Supv.- Engr. Services	X	X	X	X		X				X					
Engineer	X		X	X		X*			X*	X*					
Operating Supv.	X	X					X								
Shift Supervisor	X	X	X*	X*	X*	X	X								
Control Rm. Oper	X	X	X*	X*	X*	X*	X								
Asst. Control Rm. Oper	X				X*										
Aux. Operator	X														
Supv. - Chem. & H.P.	X	X													
Health Physcists	X	X													
Chem. & H.P. Tech	X	X													X*
Instrument Supv.	X	X													
Instrument Tech	X					X*		X*		X*	X*	X*	X*	X*	
Electrical Supv.	X	X													
Electricians	X														
Maintenance Supv.	X	X													
Maintenance Foreman	X	X													
Mechanic	X														
Storekeeper	X														

* All persons in classification did not receive training
** See Table 12.3-2 for course description

TABLE 12.3-2
TRAINING PROGRAMS
SURRY POWER STATION

COURSE NO.	DESCRIPTION
1	Health Physics I - A review of basic health physics to familiarize personnel with radiation hazards in the station and procedures to be followed to insure safe operation (40 hours)
2	Health Physics II - A sequel to HP-I to acquaint supervisory personnel, health physics technicians, and operators with health physic procedures at the station. (40 hours)
3	Westinghouse Reactor Operator Training Program - Training which prepared supervisory and operating personnel for cold licensing through classroom and on-the-job-training at an operating reactor facility. (10 months)
4	Westinghouse Atomic Power Division Training Program - Introduction to reactor systems. (5 weeks)
5	A Series - On-site training consisting of mathematics and theory of nuclear physics (6 weeks)
6	B Series - On-site training consisting of a review of station systems and controls (18 weeks)
7	Cold License Training - Accelerated review of systems and controls. (14 weeks)
8	Hagan Control Training - A review of control instrumentation, operation, calibration, and maintenance. (2 weeks)
9	Hagan Protection Training - A review of protection instrumentation operation, calibration, testing, and maintenance. (7 weeks)
10	Reactor Protection and Engineered Safeguards Training - A detailed study of reactor protection and safeguards circuits (80 hours)
11	Computer Training - Comprehensive training on P-250 computer hardware. (14 weeks)
12	Catalytic Recombiner Training - Review of operation and maintenance of recombiner system. (3 days)
13	EHC System Training - A review of the operation and maintenance of the electrohydraulic control system (3 weeks)
14	Radiation Monitoring - A review of the use, calibration, and maintenance of radiation monitoring equipment.

12.4 SHIFT PERSONNEL

The station is normally staffed to provide adequate coverage for the operation of both units, however, there are periods when only one unit will be operating. Listed below are the complements of both of these shift conditions.

1. One Unit Operation

   1 Shift Supervisor

   1 Control Room Operator

   1 Asst. Control Room Operator

   2 Auxiliary Operators

2. Two Unit Operation

   1 Shift Supervisor

   2 Control Room Operators

   2 Asst. Control Room Operators

   2 Auxiliary Operators

Health Physics and Chemistry coverage will be available for all shift operations.

Normally, four shifts report for work on a rotating basis, one shift always being off.

12.5 HEALTH PHYSICS

Established administrative controls assure that all procedures and requirements related to radiation protection are followed by all station personnel. These procedures include a Radiation Work Permit System. All work on systems or locations where exposure to radiation or radioactive materials is involved requires an appropriate Radiation Work Permit before work can begin. The radiological hazards associated with the job are determined and evaluated prior to issuing the permit.

12.5.1 PERSONNEL MONITORING SYSTEMS

Personnel monitoring equipment consisting of film badges or thermoluminescent dosimeters are assigned to and worn by personnel within the restricted area of the Surry Power Station as required by 10CFR20. In addition, those persons who work in Radiation Control Areas or whose job requires access to these areas are required to use one or more of the following types of monitors; pocket chambers, self-reading dosimeters, pocket high radiation alarms, wrist badges, and finger tabs in addition to the film badge. Records of radiation exposure history and current occupational exposure are maintained by the Health Physics group for each individual for whom personnel monitoring is required and is reported to the AEC as required by 10CFR20.

12.5.2 PERSONNEL PROTECTIVE EQUIPMENT

Special protective clothing is furnished and worn as necessary to protect personnel against contact with radioactive contamination. A change room is conveniently located for use of this protective clothing. Respiratory protective equipment is also available for the protection of personnel against airborne radioactive contamination.

12.5.3 CHANGE ROOM AREA

A change room has been provided so that personnel may obtain clean protective clothing required for station work. This area is the main access dividing point between the "clean" and "Radiation Control" areas of the facility. The "Radiation Control" section of the change room is used for the removal and handling of potentially contaminated protective clothing after use. Showers, sinks, and necessary monitoring equipment also is provided in both sections of the change area.

12.5.4 DECONTAMINATION FACILITY

A Decontamination Facility is provided at the station for decontamination of large and small items of plant equipment and components as discussed in Section 9.14.

12.5.5 ACCESS CONTROL

In order to protect personnel from access to high radiation areas, warning signs, audible and visual indicators, barricades and locked doors are used as necessary. Administrative procedures have been established to control access to radiation and high radiation areas.

12.5.6 HEALTH PHYSICS LABORATORY FACILITIES

Surry Power Station includes a Health Physics Laboratory with adequate equipment for detecting, separating, analyzing, and measuring the types of radiation of concern, and for evaluating any radiological problem which may be anticipated. Counting equipment such as G-M, scintillation and proportional counters, and scalers is provided in an appropriately designed counting room for detecting and measuring radiation as well as equipment for the identification of specific radionuclides. Equipment and facilities for analyzing environmental survey bioassay samples are also included in the Health Physics Laboratory.

12.5.7 HEALTH PHYSICS INSTRUMENTATION

Portable radiation survey instruments are provided for use by Health Physics, operating, and maintenance personnel. A sufficient number are available to allow for use, calibration, maintenance, and repair. The types of instruments used include those for detecting and measuring alpha, beta, gamma, and neutron radiation. Fixed hand and foot monitoring instruments or G-M survey and count rate meters are located at exits from Radiation Control Areas. These instruments assist operating personnel in detecting and preventing the spreading of contamination.

Portable survey instruments are also available at various locations within the Radiation Control Area as conditions warrant.

Permanently installed area radiation monitoring equipment is included in the Surry facility. These detectors continuously monitor radiation levels within certain areas as discussed in Section 11.3.4. These monitors will have an indicator readout in the Main Control Room and present an audible alarm both in the area of concern and in the Main Control Room.

12.5.8 BIOASSAY AND MEDICAL PROGRAMS

Appropriate bioassay samples are collected and evaluated as necessary from personnel who work in Radiation Control Areas as an aid in evaluating internal exposures.

A comprehensive medical examination program, appropriate for radiation exposed workers is conducted to determine whether all workers are physically suited for radiation work.

A more complete discussion of the emergency medical program is included in the Emergency Operating Procedures.

12.6 OPERATIONS PROCEDURES

Detailed written procedures for all normal nuclear operations and for abnormal and emergency nuclear situations have been prepared. Included in the procedures is a plan to provide the necessary arrangement and organization of personnel to deal effectively with any foreseeable station emergency. The station design is such that none of the credible nuclear accidents will create an undue hazard to the public; however, organization for action in various types of emergencies has been prearranged. All station personnel are thoroughly familiar with the emergency plan, and practice drills are held as necessary for training.

12.7 RECORDS

Records documenting the nuclear operation and maintenance of, modifications to, the station are maintained on file at the station site. Operating records include appropriate log books, log sheets, data logger output, and recorder charts covering all aspects of operation. Detailed records of routine operational testing of nuclear safeguards systems and components are also maintained. Maintenance and modifications records are kept for documenting inspections, preventative maintenance, alterations, and repairs to mechanical, electrical, and instrumentation systems. Copies of administrative records such as personnel radiation exposure records, station licenses, and permits are maintained on file at the station.

12.8 REVIEW AND AUDIT OF OPERATIONS

A review procedure has been established to review the following:

1. All operating procedures
2. All proposed tests and experiments
3. Changes in design
4. Proposed changes to the Technical Specifications
5. Unusual occurrences
6. Violations of the Technical Specifications

In addition, occasional inspection of station operations and other special reviews are performed as necessary. Two committees are planned for this purpose; a Station Nuclear Safety and Operating Committee, and a System Nuclear Safety and Operating Committee. The members of these committees and their responsibilities have been noted in the administrative controls section of the station Technical Specifications.

12.9 INSERVICE INSPECTION

The inservice inspection program for the Surry Power Station is designed to verify that the structual integrity of the reactor coolant pressure boundary is maintained throughout the life of the station. The program has been developed by adopting, insofar as practicable, the principles and intent embodied in the ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems. However due to the advanced stage of design and construction and the date of publication of the code, consideration has been given to the "Inservice Inspection Requirement for Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspection", dated January 31, 1969.

The following components and areas are available for visual and/or non-destructive inspection:

1. Reactor Vessel - The entire inside surface.
2. Reactor Vessel Nozzles - The entire inside surface.
3. Reactor Vessel Closure Head - The entire inside and outside surface.
4. Reactor Vessel Studs, Nuts, and Washers.
5. Field Welds between the Main Coolant Piping, Reactor Vessel, Steam Generators, Reactor Coolant Pumps, the Chemical and Volume Control Piping, and Safety Injection System Piping.
6. Reactor Internals.
7. Reactor Vessel Flange Seal Surface.

8. Control Rod Drive Shafts.
9. Control Rod Drive Mechanism Assemblies.
10. Selected Areas of Reactor Coolant Pipe External Surfaces (except for the five foot penetration of the primary shield).
11. Steam Generator - The external surface, the internal surfaces of the steam drum, and channel head.
12. Pressurizer - The internal and external surfaces.
13. Reactor Coolant Pump - The external surfaces, motor and impeller.
14. Loop Stop Valves.
15. Regenerative Heat Exchanger - The external surface and nozzle welds.

The considerations which are incorporated into the Reactor Coolant System design to permit the above inspections are as follows:

1. All reactor internals are completely removable. The storage space required to permit these inspections is provided.
2. The closure head is stored dry in the containment structure during refueling to facilitate direct visual inspection.
3. All reactor vessel studs, nuts, and washers are removed to dry storage during refueling.
4. Provision is made to remove portions of the supplementary neutron shield of the coolant nozzles, and the insulation covering the nozzle welds may be removed.
5. Access holes are provided in the lower internals barrel flange to allow remote access to the reactor vessel internal surfaces between the flange and the nozzles without removal of the internals.

6. A removable plug is provided in the lower core support plate to allow access for inspection of the bottom head without removal of the lower internals.

7. The storage stands provided for storage of the upper internals package allows for inspection access to both the inside and outside of the structure. No permanent storage stand is provided for the lower internals package. However, it can be removed from the reactor vessel and temporarily stored, if inservice inspection is required.

8. The control rod drive mechanism is designed to allow removal of the mechanism assembly from the reactor vessel head by cutting of seal welds.

9. Manways are provided in the steam generator steam drum and channel head to allow access for internal inspection.

10. A manway is provided in the pressurizer top head to allow access for internal inspection.

11. Insulation on primary system components (except the reactor vessel) and selected portions of the piping (except for the penetration in the primary shield) may be removed.

Conventional non-destructive test techniques can be used for the inspection of primary loop components other than the reactor vessel. The reactor vessel presents special problems because of radiation levels and underwater accessibility which severely restrict available test techniques. Several steps are incorporated in the design and manufacturing procedures to prepare the vessel for non-destructive test techniques which may be available in the future.

These are:

1. Ultrasonic examinations and mapping on all internally clad surfaces of the vessel to provide a reference for future ultrasonic testing.
2. The internal reactor vessel shell, in the core area, presents a clean, uncluttered cylindrical surface which should permit positioning of future test equipment without obstruction.
3. During the manufacturing stage, additional areas of the reactor vessel were ultrasonic tested and mapped to provide a reference point for possible inspections in the future.

   The areas selected for ultrasonic testing and mapping include:

   a. Vessel flange radius, including the vessel flange to upper shell weld.
   b. Middle shell course.
   c. Lower shell course above the radial core supports.
   d. Exterior surface of the closure head from the flange knuckle to the cooling shroud.
   e. Nozzle to upper shell weld.
   f. Middle shell to lower shell weld.
   g. Upper shell to middle shell weld.

The pre-operational ultrasonic testing of these areas is performed after shop hydrostatic test.

The internal surface of the reactor vessel is inspected periodically using optical devices over the accessible areas. During refueling, the vessel cladding can be inspected in certain areas between the closure flange and the primary coolant inlet nozzles, and, if deemed necessary by this inspection, the core barrel could be removed making the entire inside vessel surface accessible. If more advanced inspection methods (such as ultrasonic testing), are successfully developed for the vessel, these methods will be employed as appropriate.

Externally, the control rod drive mechanism nozzles on the closure head, the instrument nozzles on the bottom of the vessel, and the extension spool pieces on the primary coolant outlet nozzles are accessible for inspection during refuelings.

The closure head is examined visually during each refueling. Optical devices permit visual inspection of the cladding, control rod drive mechanism nozzles, and the gasket seating surface. The knuckle transition piece, which is the area of highest stress of the closure head, also is accessible on the outer surface for inspection.

The closure studs are inspected periodically, and it is possible to perform strain tests during the tensioning which assists in verifying the material properties.

A complete program dealing with the frequency of inspection and the methods of such inspections is defined in the Technical Specifications.

The program was developed with an attempt to conform, where practical with the Code for Inservice Inspection of Nuclear Reactor Coolant Systems, dated January, 1970. Due to the advanced stage of design and construction and the date of publication of the ASME Code, consideration has also been given to the "Inservice Inspection Requirements for Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspection", dated January 31, 1969. However, since proven practical methods of performing all the inspections as proposed in the ASME Code have not been fully developed, available methods that represent a logical approach to the intent of the code will be employed.

In consideration of the Surry Power Station schedule of design and construction and the date of publication of the code, Technical Specifications describe the general degree of conformance to the USASI-N45 code.

TABLE OF CONTENTS

13 INITIAL TESTS AND OPERATION

13.1 TESTS PRIOR TO REACTOR INITIAL FUELING

The comprehensive testing program ensured that equipment and systems performed in accordance with design criteria prior to fuel loading. As the installation of individual components and systems was completed, they were tested and evaluated according to predetermined and approved written testing techniques, procedures, or check-off lists. Field and Engineering analyses of test results were made to verify that systems and components were performing satisfactorily and to recommend corrective action, if necessary.

The program included tests, adjustments, calibrations, and system operations necessary to assure that initial fuel loading and subsequent power operation could be safely undertaken. In general, the types of tests are classified as hydrostatic, functional, electrical, and operational. Functional tests verified that the system or equipment was capable of performing the function for which it was designed. Operational tests involved actual operation of the system and equipment under design or simulated design conditions.

Whenever possible, these tests were performed under the same conditions as experienced under subsequent station operations. During system tests for which unit parameters were not available and could not be simulated, the systems were operational tested as far as possible without these parameters.

The remainder of the tests were performed when the parameters were available. Abnormal unit conditions were simulated during testing when such conditions did not endanger personnel or requipment, or contaminate clean systems. The detailed procedure took into account the predicted emergency or abnormal conditions involved in the tests program, and appropriate measures were included in the procedure.

During the preoperational tests, piping systems were checked to insure correct and satisfactory performance under normal operating conditions, including expected routine transients. Any abnormal conditions, such as water hammer, excessive vibration or displacement were noted and referred to the start-up engineer for investigation. If the conditions warranted, remedial steps were implemented to correct the condition to insure the operational integrity of the system. If no abnormal conditions were observed, the system was deemed to be satisfactory and no other action taken.

For purpose of illustration, a listing of representative tests required prior to initial reactor fueling is contained in Table 13.1-1. Additional information of pre-operational testing of specific components and systems is contained in the Inspection and Tests subsections of Sections 3, 4, 5, 6, 7, 8, 9, 10 and 11. The Quality Assurance Section (15.4.6) contains supplemental information concerning procedural and organizational matters.

All systems have System Descriptions in which individual equipment tests are listed.

TABLE 13.1-1

OBJECTIVES OF SYSTEM TEST PRIOR TO INITIAL REACTOR FUELING

SYSTEM TESTS	TEST OBJECTIVE
1. Electrical System	To ensure continuity, circuit integrity, and the correct and reliable functioning of electrical apparatus. Electrical tests were performed on transformers, switchgear, turbine-generators, motors, cables, control circuits, excitation switchgear, d-c systems, annunciator systems, lighting distribution switchboards, communication systems and miscellaneous equipment. Special attention was directed to the following tests: (a) High voltage switchgear breaker interlock test. (b) Station loss of voltage auto-transfer test. (c) Emergency power transfer test. (d) Tests of protective devices. (e) Equipment automatic start tests. (f) Exciter check for proper voltage build up. (g) Insulation tests.

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
2. Voice Communication System	To verify proper communication between all local stations, for interconnection to commercial phone service, and to balance and adjust amplifiers and speakers.
3. Service Water System	To verify, prior to critical operations, the design head-capacity characteristics of the service water system, that the system would supply design flow rate through all heat exchangers, and would meet the specified requirements when operated in the safeguards mode.
4. Fire Protection System	To verify proper operation of the system by ensuring the design intent was met for the fire pumps, to verify that automatic start functions operated as designed, and to verify that level and pressure controls met specifications.
5. Compressed Air System	To verify leak tightness of the system, proper operation of all compressors, the manual and automatic operation of controls at design set-points, design air-dryer cycle

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
	time and moisture content of discharge air, and proper air pressure to each controller served by the system.
6. Reactor Coolant System Cleaning	To flush and clean the reactor coolant and related primary systems to obtain the degree of cleanliness required for the intended service. Provisions to maintain cleanliness and protection from contaminated sources were made after system cleaning and acceptance. After systems were flushed clean of particulate matter, the cleanliness of the system was maintained. Coolant was analyzed for chloride content, solids, pH, and conductivity. Oxygen content was analyzed and brought to specifications prior to exceeding 200^{o}F.
7. Ventilation Systems	To verify proper operability of fans, controls, and other components of the Containment Ventilation System and Auxiliary Ventilation System.

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
8. Condensate and Feedwater System	To verify value and control operability and set-points, an inspection for completeness and integrity was made. Functional testing was performed when the Main Steam System was available. Flushing and hydrostatic test were performed where applicable.
9. Auxiliary Coolant Systems	To verify component cooling flow to all components, and tc verify proper operation of instrumentation, controllers, and alarms. Specifically, each of the three systems; i.e., Component Cooling System (including the Charging Pump Cooling System), Residual Heat Removal System and Fuel Pit Cooling System, was tested to ensure: (a) All manual and remotely operated valves were operable manually and/or remotely. (b) All pumps performed their design functions satisfactorily.

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
	(c) All temperature, flow, level, and pressure controllers functioned to control at the required set-point when supplied with appropriate signals.
	(d) All temperature, flow level, and pressure alarms provided alarms at the required locations when the alarm set-point was reached and cleared when the reset point was reached.
	(e) Design flow rates were established through the principal heat exchangers.
10. Boron Recovery System	To verify valve and control operability and set-points, flushing and hydrostatic testing as applicable, inspection for completeness and integrity. Functional testing was performed when a steam supply was available.
11. Chemical and Volume Control System	To verify, prior to critical operation, that the Chemical and Volume Control System functioned as specified in the system

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
	description and appropriate manufacturers' technical manuals. More Specifically that: (a) All manual and remotely operated valves were operable manually and/or remotely. (b) All pumps performed to specifications. (c) All temperature, flow, level and pressure controllers functioned to control at the required set-point when supplied with appropriate signal(s). (d) All temperature, flow, level, and pressure alarms provided alarms at the required locations when the alarm set-point was reached and cleared when the reset point was reached. (e) The reactor makeup control regulated blending, dilution, and boration as designed. (f) The design seal water flow rates were attainable at each reactor coolant pump.

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
	(g) Chemical Addition Subsystem functioned as specified.
12. Safety Injection System	To verify prior to critical operation, response to control signals and sequencing of the pumps, valves, and controllers of this system as specified in the system description and the manufacturer's technical manuals; and to check the time required to actuate the system after a safety injection signal was received. More Specifically that: (a) All manual and remotely operated valves were operable manually and/or remotely. (b) For each pair of valves installed for redundant flow paths, disabling one of the valves did not impair operation of the other. (c) All pumps performed their design functions satisfactorily.

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
	(d) The proper sequencing of valves and pumps occurred on initiation of a safety injection signal. (e) The fail position on loss of power for each remotely operated valve was as specified. (f) Valves requiring signals or high containment pressure to operate did so when supplied with these signals. (g) All level and pressure instruments were set at the specified points and provided alarm and reset at the required location(s). (h) The time required to actuate the system was within the design specifications.
13. Containment Spray System	To verify, prior to critical operation, response to control signals and sequencing of the pumps, valves, and controllers as specified in the system description and the manufacturers' technical manuals; and to

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
	check the time required to actuate the system after a containment high-high pressure signal was received. More specifically, see the test objective listing for Safety Injection System above.
14. Fuel Handling System*	To show that the system design is capable of providing a safe and effective means of transporting and handling fuel from the time it reaches the station until it leaves the station. In particular, the tests were designed to verify that: (a) The major structures required for refueling, such as the reactor cavity, refueling canal, new fuel and spent fuel storage, and decontamination facilities, were in accordance with the design intent. (b) The major equipment required for refueling such as the manipulator crane, fuel handling tools, spent

* Tests were conducted with a dummy fuel element

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
	fuel transfer system, operated in accordance with the design specifications. (c) All auxiliary equipment and instrumentation functioned properly.
15. Radiation Monitoring Systems	To verify the calibration, operability, and alarm set-points of all area radiation monitors, air particulate monitors, gas monitors and liquid monitors which are included in the process Radiation Monitor System and the Area Radiation Monitor System.
16. Reactor Control and Protection System	To verify calibration, operability, and alarm settings of the Reactor Control and Protection System; to test its operability in conjunction with other systems. As an example, the Nuclear Instrumentation System tests are detailed below.

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
17. Nuclear Instrumentation System	To ensure that the instrumentation system was capable of monitoring the reactor leakage neutron flux from source range through 120 percent of full power and that protective functions were operating properly. In particular the tests were designed to verify that: (a) All system equipment, cabling, and interconnections were properly installed. (b) The source range detector and associated instrumentation responded to neutron level changes, and that the source range protection (high flux level reactor trip) as well as alarm features and audible count rate operated properly. (c) The intermediate range instrumentation operated properly, the reactor protective and control features such as high level reactor trip and high level rod stop signals operated properly, and

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
	the permissive signals for blocking source range trip and source range high voltage off operated properly.
	(d) The power range instrumentation operated properly; the protective features such as the overpower trips, permissive and dropped-rod functions operated with the required redundancy and separation through the associated logic matrices; and the nuclear power signals to other systems were available and operating properly.
	(e) All auxiliary equipment such as the startup rate channel, recorders, and indicators operated properly.
	(f) All instruments were properly calibrated and all set-points and alarms were properly adjusted.
18. Radioactive Waste System	To verify satisfactory flow characteristics through the equipment, to demonstrate satisfactory performance of pumps and instruments,

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
	to check for leaktightness of piping and equipment, and to verify proper operation of monitors, alarms and controls. More specifically that: (a) All manual and automatic valves were operable. (b) All instrument controllers operated to control system at required values. (c) All alarms were operable at required locations. (d) All pumps performed their design functions satisfactorily. (e) All pump indicators and controls were operable at required locations. (f) The waste gas compressors and blowers operated as specified. (g) The gas analyzers and recombiners operated as specified. (h) The waste evaporator operated as specified.

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
19. Sampling System	To verify that a quantity of representative fluid could be obtained safely from each sampling point. In particular the tests were designed to verify that: (a) All system piping and components were properly installed. (b) All remotely and manually operated valving operated in accordance with the design specifications. (c) All sample containers and quick-disconnect couplings functioned properly.
20. Emergency Power System	To demonstrate that the system was capable of providing power for operation of vital equipment under power failure conditions. In particular the tests were designed to verify that: (a) All system components were properly installed. (b) Each emergency diesel functioned according to the design intent under emergency conditions.

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
	(c) The emergency units were capable of supplying the power to vital equipment as required under Emergency Conditions.
	(d) All redundant features of the system functioned according to the design intent.
21. Hot Functional Tests	Using pump heat, the Reactor Coolant System was tested to check heatup and cooldown procedures to demonstrate satisfactory performance of components that are exposed to the reactor coolant temperature; to verify proper operation of instrumentation, controllers and alarms, and to provide design operating conditions for checkout of auxiliary systems. The Chemical and Volume Control System was tested to determine that water could be charged at rated flow against normal Reactor Coolant System pressure; to check letdown flow against design rate for each pressure reduction station; to

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
	determine the response of the system to changes in pressurizer level; to check procedures and components used in boric acid batching and transfer operations; to check operation of the reactor makeup control; to check operation of the excess letdown and seal water flowpath; and to verify proper operation of instrumentation controls and alarms.
	The Sampling System was tested to determine that a specified quantity of representative fluid could be obtained safely and at design conditions from each sampling point.
	The Component Cooling System was tested to evaluate its ability to remove heat from systems containing radioactive fluid and other special equipment, under varied service water conditions; to verify component cooling flow to all components; to verify that the charging pumps cooling water subsystem functioned as design and to verify proper operation of instrumentation, controllers and alarms.

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
	Following this hot functional the reactor internals were examined for evidence of vibration.
22. Pressurizer Level Control System	To ensure that the system was capable of monitoring the full range of pressurizer level and to verify alarms and set-points. Also to verify that the system in conjunction with the Chemical and Volume Control System controls pressurizer level.
23. Rod Position Indication System	To check the systems response to test signals and to verify correct indicating and control functions. After fuel loading and after the position indication coils were installed, a calibration and complete operational check was performed by operating individual control rod drive mechanisms.
24. Reactor Thermocouple Instrumentation	To check and calibrate the system and compare thermocouple readings with other temperature instrumentation indications up to the maximum allowable temperature.
25. Auxiliary Steam Generator Feedwater Pumps	To verify that all pumps performed their design functions satisfactorily.

TABLE 13.1-1 (Continued)

SYSTEM TESTS	TEST OBJECTIVE
26. Primary System Safety and Relief Valves	To verify correct relief and lift pressures as necessary.
27. Cold Hydrostatic Tests	To verify the integrity and leaktightness of the Reactor Coolant System and auxiliary primary systems with the performance of a hydrostatic test at the specified test pressure.
28. Main Steam Trip Valves	To verify that they will terminate steam flow to the turbine by testing at steam temperature and pressure associated with hot functional conditions.

13.2 FINAL STATION PREPARATION

Fuel loading began when all prerequisite system tests and operations were satisfactorily completed and the facility operating license was obtained. Upon completion of fuel loading, the reactor upper internals and pressure vessel head were installed and additional mechanical and electrical tests were performed. The purpose of this phase of activities was to prepare the system for nuclear operation and to establish that all design requirements necessary for operation had been achieved. The core loading and post loading tests are described below.

13.2.1 CORE LOADING

The overall responsibility and direction for initial core loading was exercised by the Station Manager assisted by the Superintendent of Station Operations. The overall process of initial core loading was normally directed from the charging floor of the containment structure. Standard procedures for the control of personnel and the maintenance of containment security was established prior to fuel loading. Westinghouse provided technical advisors to assist during the initial core loading operation.

The as-loaded core configuration was specified as part of the core design studies conducted in advance of station startup and as such was not subject to change at startup. If a fuel assembly had sustained mechanical damage during core loading operations, a replacement fuel assembly of the same type would have been used, if available, and the core loading would have continued according to the prescribed plan. If such an assembly had not been available, an alternate core loading scheme would have been devised which would have insured operation within technical specifications, and this scheme would have been invoked.

The core was assembled in the reactor vessel, submerged in water containing enough dissolved boric acid (at least 1500 ppm boron) to maintain a core effective multiplication constant of 0.90 or lower. The refueling cavity was dry during initial core loading. Core moderator chemistry conditions (particularly, boron concentration) were prescribed in the core loading procedure document and were verified periodically by chemical analysis of moderator samples taken prior to and during core loading operation.

Core loading instrumentation consisted of two permanently installed source range (pulse type) nuclear channels and two temporary incore source range channels plus a third temporary channel which could be used as a spare. The permanent channels were monitored in the Main Control Room by licensed station operators; the temporary channels were installed in the containment structure and monitored by reactor engineering personnel. At least one channel and one temporary channel was equipped with audible count range indicators. Both channels and both regular temporary channels displayed neutron count rate on count rate meters and strip chart recorders. Minimum count rates of two counts per second, attributable to core neutrons, were required on at least two of the four available nuclear channels at all times during core loading operations.

At least two artifical neutron sources were introduced into the core at appropriate specified points in the core loading program to ensure a neutron population large enough for adequate monitoring of the core.

Fuel assemblies together with inserted components (control rod assemblies, burnable poison inserts, source spider, or thimble plugging devices) were placed in the reactor vessel one at a time according to a previously established and approved sequence which was developed to provide reliable core monitoring with minimum possibility of core mechanical damage. The core loading procedure documents included a detailed tabular check sheet which prescribed and verified the successive movements of each fuel assembly and its specified inserts from its initial position in the storage racks to its final positions in the core. Multiple checks were made of component serial numbers and types at successive transfer points to guard against possible inadvertent exchanges or substitutions of components.

An initial nucleus of eight fuel assemblies, the first of which contained an activated neutron source, was the minimum source-fuel nucleus which would permit subsequent meaningful inverse count rate monitoring. This initial nucleus was determined by calculation and previous experience to be markedly subcritical (k_{eff}= 0.90) under the required conditions of loading.

Each subsequent fuel addition was accomplanied by detailed neutron count rate monitoring to determine that the just loaded fuel assembly had not excessively increased the count rate and that the extrapolated inverse count rate ratio was not decreasing for unexplained reasons. The results of each loading step was evaluated by the Applicant and technical advisors before the next prescribed step was started.

Criteria for safe loading required that loading operations stop immediately if:

a. The neutron count rates on all responding nuclear channels doubled during any single loading step after the initial nucleus of eight fuel assemblies had been loaded.
b. The neutron count rate on any individual nuclear channel increased by a factor of five during any single loading step.

An alarm in the containment and Main Control Room was coupled to the source range channels with a set-point at five times the current count rate. This alarm would have automatically alerted the loading operation to an indication of high count rate and would have required an immediate stop of all operations until the incident was evaluated by the Applicant and technical advisors.

In the event that the licensed station operator in the Main Control Room determined that an unacceptable increase in count rate was being observed in any or all responding nuclear channels, he executed one of, or combinations of, the prepared special procedures which involved withdrawing fuel from the core, manually actuating the containment evacuation alarm, or charging of concentrated boric acid into the moderator.

Core loading procedures specified alignment of fluid systems to prevent inadvertent dilution of the reactor coolant, restricted the movement of fuel to preclude the possibility of mechanical damage, prescribed the conditions under which loading could proceed, identified chains of responsibility and authority and provided for continuous and complete fuel and core component accountability.

13.2.2 POSTLOADING TESTS

Upon completion of core loading, the reactor upper internals and the pressure vessel head were installed and additional mechanical and electrical tests were performed prior to initial criticality. The final hydrostatic tests were conducted after filling and venting was completed.

Mechanical and electrical tests were performed on the control rod drive mechanisms at both the cold and hot conditions. These tests included a complete operational checkout of the mechanisms. Checks were made to ensure that the control rod assembly position indicator coil stacks were connected to their position indicators. Similar checks were made on control rod drive mechanism coils.

Tests were performed on the reactor trip circuits to test manual trip operation and actual control rod assembly drop times were measured for each control rod assembly. By use of dummy signals, the Reactor Control and Protection System was made to produce trip signals for the various unit abnormalities that required tripping.

At all times that the control rod drive mechanisms were being tested, the boron concentration in the coolant-moderator was large enough (approximately 1500 ppm boron) that criticality could not be achieved with all control rod assemblies out. Furthermore, the number of control rod assemblies operated at any one time was restricted to no more than approximately half the total number of assemblies.

A complete functional electrical and mechanical check was made of the incore nuclear flux mapping system at the operating temperature and pressure.

13.3 INITIAL TESTING IN THE OPERATING REACTOR

After satisfactory completion of fuel loading and final station tests, nuclear operation of the reactor was begun. This final phase of startup and testing included Initial Criticality, Initial Unit Verification Test, Zero Power Testing and Power Level Escalation. The purpose of these tests was to establish the operational characteristics of the unit and core, to verify design prediction, to demonstrate that license requirements had been met, and to ensure that the next prescribed step in the test sequence could be safely undertaken. A brief description of the testing is presented in the following sections. Table 13.3-1 summarizes the tests which were performed from initial core loading to rated power.

13.3.1 INITIAL CRITICALITY

Initial criticality was established by sequentially withdrawing the shutdown and control groups of control rod assemblies from the core, leaving the last withdrawn control group inserted far enough in the core to provide effective control when criticality was achieved, and then slowly and continuously diluting the heavily borated reactor coolant until the chain reaction was self-sustaining.

Successive stages of control rod assembly group withdrawal and of boron concentration reduction was monitored by observing changes in neutron count rate as indicated by the regular source range nuclear instrumentation as functions of control rod assembly group position and, subsequently, of primary water addition to the Reactor Coolant System during dilution.

Primary safety reliance was based on inverse count rate ratio monitoring as an indication of the nearness and rate of approach to criticality of the core during control rod assembly group withdrawal and during reactor coolant boron dilution. The rate of approach was reduced as the reactor approached extrapolated criticality to ensure that effective control was maintained at all times.

Written procedures specified alignment of fluid systems to allow controlled start and stop and adjustment of the rate at which the approach to criticality could proceed, indicated values of core conditions under which criticality was expected, specified allowed deviations in expected values, and identified chains of responsibility and authority during reactor operations.

13.3.2 INITIAL UNIT VERIFICATION TESTS

Upon establishment of criticality, a series of tests was initiated to determine the overall unit behavior and to checkout the system under operating conditions. The initial tests consisted of selected zero power physics measurements and power escalation tests to ensure safe reactor operation while performing the overall unit checkout.

The selected zero power measurements were made at or near operating temperature and pressure and were planned to consist of the last withdrawn control rod assembly group reactivity worth, boron concentration reactivity worth determined from data taken during the control rod assembly reactivity worth measurement, an isothermal temperature coefficient, and the control rod assemblies out critical boron concentration and power distribution. Concurrent tests were conducted on the unit instrumentation including the source and intermediate range nuclear channels. Control rod assembly operation and the behavior of the associated control and indicating circuits were demonstrated under zero power operating conditions. The results of these tests and measurements were compared to the expected design behavior and a decision made by the Station Manager and technical advisors whether to continue with the preliminary testing or to do the complete zero power testing to better verify design values. The remainder of the initial station verification tests were performed during power escalation to no more than 40% of full power.

The purpose of the above nuclear tests was to survey overall station performance and to determine the adequacy of the design and the integrity of the systems used.

Detailed procedures specified the sequence of tests and measurements conducted and the conditions under which each was performed. If deviations from design predictions existed, unacceptable behavior revealed, or apparent anomalies developed, the testing was suspended and the situation reviewed by the Applicant to determine whether a question of safety was involved, prior to resumption of testing.

13.3.3 ZERO POWER TESTING

A prescribed program of reactor physics measurements was undertaken to verify that the basic static and kinetic characteristics of the core were as expected and that the values of the kinetic coefficients assumed in the safeguards analysis were indeed conservative.

The measurements were made at zero power and primarily at or near operating temperature and pressure. Measurements, to include verification of calculated values of control rod assembly group and unit reactivity worths, of isothermal temperature coefficient under various core conditions, of differential boron concentration reactivity worth and of critical boron concentrations as functions of control rod assembly group configuration were made. Relative power distribution checks were made in normal and abnormal control rod assembly configurations.

Detailed procedures were prepared to specify the sequence of tests and measurements to be conducted and the conditions under which each was to be performed to ensure both safety of operation and the relevancy and consistency of the results obtained.

13.3.4 POWER LEVEL ESCALATION

When the operating characteristics of the reactor and unit were verified by the preliminary zero power tests, a program of power level escalation in successive stages brought the unit to its full rated power level. Both reactor and unit operational characteristics were closely examined at each stage and the relevance of the safeguards analysis verified before escalation to the next programmed level was effected.

Reactor physics measurements were made to determine the magnitudes of reactivity effects, of control rod assembly group differential reactivity effects, of control rod assembly group differential reactivity worth and of relative power distribution in the core as functions of power level and control rod assembly group position.

Concurrent determinations of primary and secondary heat balances ensured that the several indications of power level were consistent and provided bases for calibration of the power range nuclear channels. The ability of the Reactor Control and Protection System to respond effectively to signals from primary and secondary instrumentation under a variety of conditions encountered in normal operations was verified.

At prescribed power levels the response characteristics of the reactor coolant and steam systems to dynamic stimuli was evaluated. The responses of system components was measured for 10% loss of load and recovery, 50%

loss of load and recovery, turbine trip, and trip of a single control rod assembly.

After rated power level was achieved, a series of load follow tests was performed at selected power level escalation steps. The results of these tests gave actual reactor and unit behavior under operating conditions and was used to verify predicted load follow capabilities.

Adequacy of radiation shielding was verified by gamma and neutron radiation surveys inside the containment and throughout the station site.

The sequence of tests, measurements and intervening operations was prescribed in the power escalation procedures together with specific details relating to the conduct of the several tests and measurements. The measurement and test operations during power escalation was similar to normal operations.

13.3.5 POST STARTUP SURVEILLANCE AND TESTING REQUIREMENTS

Post startup surveillance and testing requirements are designed to provide assurance that essential systems, which include equipment components and instrument channels, are always capable of functioning in accordance with their original design criteria. These requirements can be separated into two categories:

a) The system must be capable of performing its function, i.e., pumps deliver at design flow and heat, and instrument channels respond to initiating signals within design calibration and time responses.

b) Reliability is maintained at levels comparable to those established in the design criteria and during early station life.

The testing requirements, as described in the Technical Specifications, establish this reliability and, in addition, provide the means by which this reliability is continually reconfirmed. Verification of operation of complete systems is checked at refueling intervals. Individual checks of components and instrumentation are made at more frequent intervals as outlined in the Technical Specifications.

The techniques used for the testing of instrument channels included a pre-operational calibration which confirmed values obtained during factory test programs. These reconfirmed calibration values become the reference for recalibration maintenance at refueling intervals during station life. Periodic testing, as defined in the Technical Specifications, includes the insertion of a predetermined signal that will trip the channel bistable. Indication of the operation is confirmed and recorded.

Testing of components is initiated through manual actuation. If response times are important, they are measured and recorded. The capability to deliver design output is checked by instrumentation and compared against design data. Allowable discrepancies are established in the Technical Specifications. The component is operated sufficiently long to allow equalization of operating temperatures in bearings, seals, and motors. Checks are made on these parameters. The component is surveyed for excessive vibration. Readings are recorded.

The Applicant believes that testing in accordance with the above described program provides a realistic basis for determining maintenance requirements and, as such, ensures continued system capabilities including reliability equal to that established in the original criteria.

TABLE 13.3-1

INITIAL TESTING SUMMARY

Test	Conditions	Objectives	Acceptance Criteria
Control Rod Assembly Drop Tests	a) Cold, Shutdown b) Hot, Shutdown	To measure the droptime of control rod assemblies under full flow and no flow conditions	Droptime less than value assumed in Section 3, Safety Analysis Report
Thermocouple/RTD Intercalibration	Various temperatures during system heatup at zero power	To determine inplace isothermal correction constants for all core exit thermocouples and reactor coolant RTD's	Sensors showing excessive deviations from average were removed from service or replaced
Nuclear Design Check Tests	All two dimensional control rod assembly group configurations at hot, zero power	To verify that nuclear design predictions for endpoint boron concentrations, isothermal temperature coefficients and power distributions were valid	Within limits established in FSAR $\partial\rho/\Delta T$ and $F_{\Delta H}$
Control Rod Assembly Group Worth	All control rod assembly groups at hot, zero power	To verify that nuclear design predictions for control rod assembly group differential worths with and without partial length control rod assemblies were valid	Within limits established in FSAR for $\Delta\rho/\Delta h$, Δ_ρ/h, $\partial_\rho/\partial h$ and Δ_ρ/h
Power Coefficient Measurement	0 percent to 100 percent of rated power	To verify that nuclear design predictions for differential power coefficients were valid	Technical Specification limiting values
Automatic Control System Checkout	Approximately 30 percent of rated power	To verify control system response characteristics for the: a) Steam generator level control system	Safety Analysis Report criteria applicable

TABLE 13.3-1 (Continued)

Test	Conditions	Objectives	Acceptance Criteria
		b) Control rod assembly automatic control system c) turbine control system	
Power Range Instrumentation Calibration	During static and/or transient conditions at the following percentages of rated power: 30 percent 50 percent 75 percent 90 percent 100 percent	To verify all power range instrumentation consisting of power range nuclear channels, incore exit thermocouple system, and reactor coolant RTD's were responsive to changes in reactor power level and to intercalibrate the several systems	Verify that allowable errors sited in Technical Specifications are met
Load Swing Test	10 percent steps at the following percentages of rated power: 50 percent 75 percent 100 percent	To verify reactor control performance	Unit performance criteria applicable
Station Trip	Full Load Rejection from the following percentages of rated power: 30 percent 100 percent	To verify reactor control performance	Unit performance criteria applicable
Pressurizer Effectiveness Test	Hot, shutdown	To verify that pressurizer pressure was reduced at the required rate by pressurizer spray actuation	Unit performance criteria applicable

TABLE 13.3-1 (Continued)

Test	Conditions	Objectives	Acceptance Criteria
Minimum Shutdown Verification	Hot, zero power	To verify the nuclear design prediction of the minimum shutdown boron concentration with one "stuck" control rod assembly	Verify stuck control rod assembly shutdown criteria
Power Redistribution Follow	75 percent of rated power	To verify that out-of-core nuclear instrumentation adequately monitors changes in core power distribution under transient xenon conditions	Technical Specification limiting values
RCC Out-of-Position Test	50% percent of rated power	To verify that a single control rod assembly inserted fully or part way below the control bank is detected by out-of-core nuclear instrumentation, core exit, thermocouples under typical operating conditions and to provide bases for adjustment of protection system set-points	Inserted control rod assembly detectable with station instrumentation
Step Load Reduction Test	Reduction from 75 percent to 25 percent of rated power 50 percent reduction from 100 percent of rated power	To verify reactor control system	Unit performance report criteria applicable

TABLE 13.3-1 (Continued)

Test	Conditions	Objectives	Acceptance Criteria
Part Length Group Operational Maneuvering	90 percent of rated power	To verify that the part length control rod assembly maneuvering scheme is effective in containing and suppressing spatial xenon transients	Technical Specification limiting values
Load Cycle Test	40 percent to 85 percent of rated power	To verify that all station systems are capable of sustaining load follow operations without encountering unacceptable operational limits through a typical weekly cycle	Technical Specification limiting values for F_Z, power distribution shutdown margin
Dynamic Control Rod Assembly Drop Test	75 percent of rated power	To verify automatic detection of dropped control rod assembly, and subsequent automatic rod stop and turbine cutback	Required power reduction and control rod assembly withdrawal block accomplishment
Turbine Generator Startup Tests	Pre- and Post-Synchronization	To verify that the turbine generator unit and associated controls and trips were in good working order and ready for service	Successful completion of all mechanical, electrical and control functional checks
Turbine Generator	30 percent of rated power	To verify normal trouble free performance of the turbine generator at low power	Performance within manufacturers limitations
Control Valve Tests	75 percent of rated power	To verify capability of exercising control valves at significant load and evaluate function of valves and controls	Normal trouble free operation
Acceptance Run	100 hours at rated power	To verify reliable steady state full power capability	100 hours reliable equilibrium operation at full power

13.4 OPERATING RESTRICTIONS

13.4.1 SAFETY PRECAUTIONS

The measurements and test operations during zero power and power escalation were similar to normal station operations at power such that normal safety precautions were adequate.

13.4.2 INITIAL OPERATION RESPONSIBILITIES

The Applicant had overall responsibility for supervising and directing all phases of testing. Technical responsibility for each individual phase of actual startup resides with the functional group most directly concerned with the results of the phase. Stone & Webster and Westinghouse had onsite representatives of supporting functional groups to provide technical advice, recommendations and assistance in planning and executing the respective phases of unit startup. Specific responsibilities during each phase of testing are discussed in preceeding respective sections.

All system operations in the testing program were performed by station operators in accordance with the approved written procedures. These procedures included such items as delineation of administrative procedures and test responsibilities, equipment clearance procedures, test purpose, conditions, precautions, limitations, and sequence of operations. Procedural changes were made only in accordance with an approved standard operating procedure that required review and approval of the changes by experienced supervisory personnel.

Test procedures stating the test purpose, conditions, precautions, limitations and criteria for acceptance were prepared for each test by station personnel with assistance from Westinghouse and Stone and Webster technical advisors. All such procedures were reviewed and approved by the Applicants senior personnel in accordance with approved standard operating procedures prior to implementation.

As part of the precautions, all licensed Senior Reactor Operators and manufacturer's representatives whose equipment was being tested were instructed to stop a test or a portion of a test if the test was not being performed safely or in accordance with the written test procedures. The test procedure was required and was approved by the Station Manager or his representative. If substantial revision was required, however, the Station Manager reviewed the change with the same approach as a new test procedure before approving continuation.

The Applicant had overall responsibility during plant startup, including pre-criticality tests, approach to criticality, and post-criticality operation. The station staff was assisted by the nuclear steam supply system supplier, Westinghouse Electric Corporation. Experienced Westinghouse reactor engineers were assigned to the station from fuel loading, through power ascension until completion of the 100 hour full load test. These reactor engineers had previously participated in reactor startups of similar units and were qualified and knowledgeable in reactor operations. At least one reactor engineer was on site during all shifts when the reactor was operating. The responsible shift reactor engineer reported directly to the Shift Supervisor and received instructions from him. The reactor engineer acted in an advisory capacity only - the Applicant retained responsibility and control of the unit. Reactor specialists, e.g. control engineers, etc., were available and utilized as required.

TABLE OF CONTENTS

14 SAFETY ANALYSIS

14.1 GENERAL

This section evaluates the safety aspects of the station and demonstrates that the station can be operated safely and that exposures from credible accidents do not exceed the guide lines of 10CFR100.

This section is divided into subsections, each dealing with a different behavior category. These subsections are:

1. CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS, SECTION 14. 2

The incidents presented in Section 14.2 are associated with an individual unit within the station, and have no off-site radiation consequences.

2. OTHER SINGLE UNIT ACCIDENTS, SECTION 14.3

The accidents presented in Section 14.3 are the rupture of a steam generator tube, break of a steam line, and control rod ejection.

3. GENERAL STATION ACCIDENT ANALYSIS, SECTION 14.4

The accidents presented in Section 14.4 are associated with shared systems and facilities which may cause release of radioactive material to the environment.

4. LOSS OF COOLANT ACCIDENT, INCLUDING THE DESIGN BASIS ACCIDENT SECTION 14.5

The loss-of-coolant accident, or the rupture of a reactor coolant pipe, is the worst accident case and is the primary basis for the design basis accident; the unit design requirement.

It is shown that even this accident meets the guide lines of 10CFR100 by a wide margin, assuming that the core has been operating at its maximum expected reactor thermal power rating of 2546 MWt.

14.2.1 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL FROM A SUBCRITICAL CONDITION

A control rod assembly withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of control rod assemblies resulting in a power excursion. While the probability of a transient of this type is extremely low, such a transient could be caused by a malfunction of the Reactor Control or Control Rod Drive Systems. This could occur with the reactor either subcritical or at power. The "at power" case is discussed in Section 14.2.2.

Reactivity is added at a prescribed and controlled rate in bringing the reactor from a shutdown condition to a low power level during startup by control rod withdrawal. Although the initial startup procedure uses the method of boron dilution, the normal startup is with control rod assembly withdrawal. Control rod assembly motion can cause much faster changes in reactivity than can be made by changing boron concentration.

The control rod drive mechanisms are wired into preselected banks, and these bank configurations are not altered during core life. The assemblies are therefore physically prevented from being withdrawn in other than their respective banks. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. The control 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 analysis assuming the simultaneous withdrawal of the combination of the two rod banks of the maximum combined worth at maximum speed.

Should a continuous control rod assembly withdrawal be initiated, the transient will be terminated by the following automatic Safety Features:

1. 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.

2. Intermediate range control rod stop - actuated when either of two independent intermediate range channels indicates a flux level above a preselected, manually adjustable value. This control rod stop may be manually bypassed when two out of the four power range channels indicate a power level above approximately 10 percent of full power. It is automatically reinstated when three of the four power range channels are below this value.

3. Intermediate range flux level trip - actuated when either of two independent intermediate range channels indicate a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed when two of the four power range channels are reading above approximately 10 percent of full power and is automatically reinstated when three of the four channels indicate a power level below this value.

4. Power range flux level trip (low setting) - actuated when two out of the four power range channels indicate a power level above approximately 25 percent of full power. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately 10 percent of full power and is automatically reinstated when three of the four channels indicate a power level below this value.

5. Power range control rod stop - actuated when one out of the four power range channels indicates a power level above a preset setpoint. This function is always active.

6. 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 fast negative fuel temperature feedback (Doppler effect) and is of prime importance during a startup incident 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 incident is not terminated by a reactor trip, the nuclear power increases again, but at a much slower rate.

Termination of the startup incident by the above protection channels prevents core damage. In addition, the reactor trip from high reactor pressure serves as a backup to terminate the incident before an overpressure condition could occur.

Method of Analysis

Analysis of this transient is performed by digital computation incorporating the neutron kinetics, including six delayed groups, and the core thermal and hydraulic equations. In addition to the nuclear flux response, the average fuel, clad, and water temperature, and also the heat flux response, are computed.

In order to give conservative results for a startup incident, the following additional assumptions are made concerning the initial reactor conditions:

1. Since the magnitude of the nuclear power peak reached during the initial part of the transient, for any given rate of reactivity insertion, is strongly dependent on the Doppler power reactivity coefficient, a conservatively low value is used for the startup incident (See Figure 14.2.1-1).

2. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the nuclear flux response time constant. However, after the initial

nuclear flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. Although during normal operation the moderator coefficient will not be positive at any time in core life, a conservative value of $+1.00 \times 10^{-4}$ $\Delta k/°F$ has been used in the analysis since the positive value yields the maximum peak core heat flux.

3. The reactor is assumed to be at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel to water thermal conductivity, a larger fuel thermal capacity, and a less negative (smaller absolute magnitude) Doppler coefficient. The less negative Doppler coefficient reduces the Doppler feedback effect thereby increasing the nuclear flux peak. The high nuclear flux peak combined with a high fuel thermal capacity and large thermal conductivity yields a larger peak heat flux. Initial multiplication (k_o) is assumed to be 1.0 since this results in the maximum nuclear power peak.

4. The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and control rod assembly release, are taken into account. A 10% increase has been assumed for the power range flux trip setpoint raising it from the nominal value of 25% to 35%. Reference to Figure 14.2.1-3, however, shows that the rise in nuclear flux is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods are released in negligible. In

addition to the above, the rate of negative reactivity insertion corresponding to the trip action is based on the assumption that the highest worth control rod assembly is stuck in its fully withdrawn position.

Results

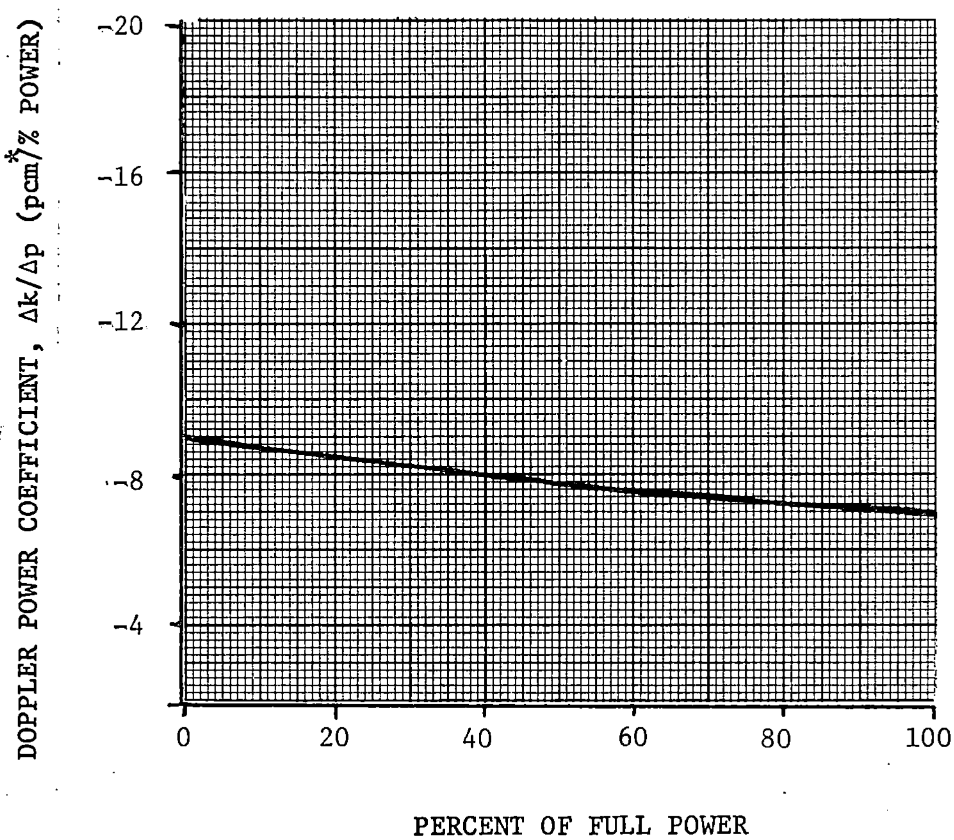
Figure 14.2.1-2 shows the effect of initial power level on peak heat flux for various reactivity insertion rates. It shows that peak heat flux initially decreases with increasing initial power level and then, depending on the rate, it increases again and approaches 35 percent of full power (reactor trip is assumed to be initiated at this value). It can also be seen that for the faster insertion rates, which result in the greatest energy addition, the flux peak is greatest for the lowest intial power level.

Figures 14.2.1-3 through 14.2.1-5 show the transient behavior for a reactivity insertion rate of 6×10^{-4} Δk/sec with the incident terminated by reactor trip at 35 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. It is also greater (by more than a factor of 10) than the maximum insertion rate of the part length control rod assemblies which is 5.25×10^{-5} Δk/sec. Figure 14.2.1-3 shows the nuclear power increase. The nuclear power is seen to increase to the trip point in 12.5 seconds.

The nuclear power overshoots to approximately 315 percent, but this occurs for only a very short time period. Hence, the energy release and the fuel temperature increases are small. The thermal flux response, of interest for DNB considerations, is shown on Figure 14.2.1-4. The beneficial effect of the inherent thermal lag of the fuel is evidenced by a peak heat flux of only 67 percent of the nominal value. There is a larger margin to DNB during the transient since the rod surface heat flux remains below the design value, and there is a high degree of subcooling at all times in the core. Figure 14.2.1-5 shows the response of the average fuel, cladding, and coolant temperature. The average fuel temperature increases to 852°F which is lower than the nominal full power value of about 1475°F. The average coolant temperature increases to only 560°F.

Conclusion

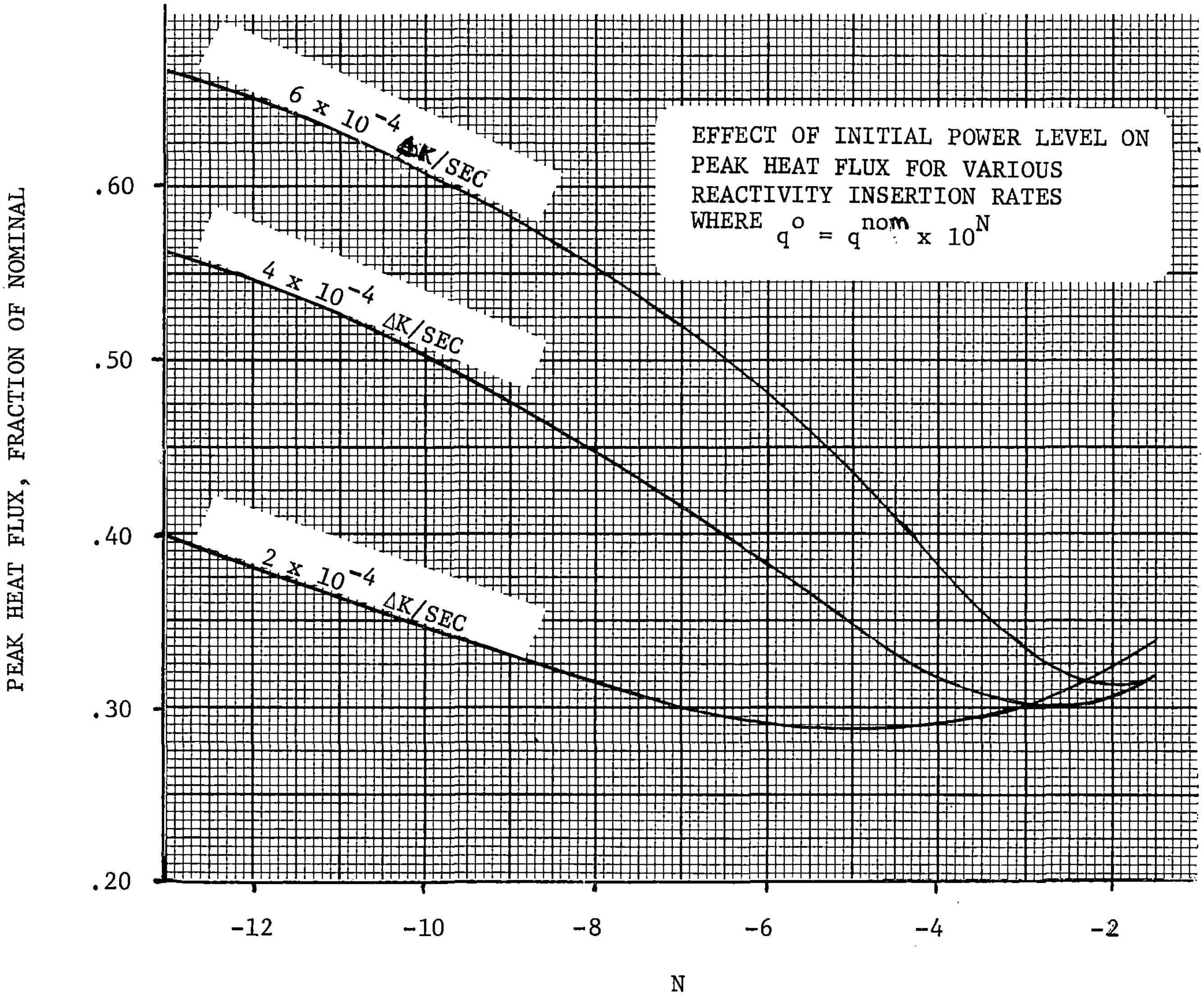
Taking into account the conservative assumptions with which the incident has been analyzed, it is concluded that in the unlikely event of a control rod assembly withdrawal incident, the core and Reactor Coolant System are not adversely affected since the thermal power reached is only 67 percent of the nominal value and the core water temperature reached is 560°F compared to 577°F for the nominal conditions. This combination of thermal power and core water temperature results in a DNBR well above the limiting value of 1.30. The peak average clad temperature (≮600°F) is less than the nominal full power value of 633°F and thus there is no clad damage.



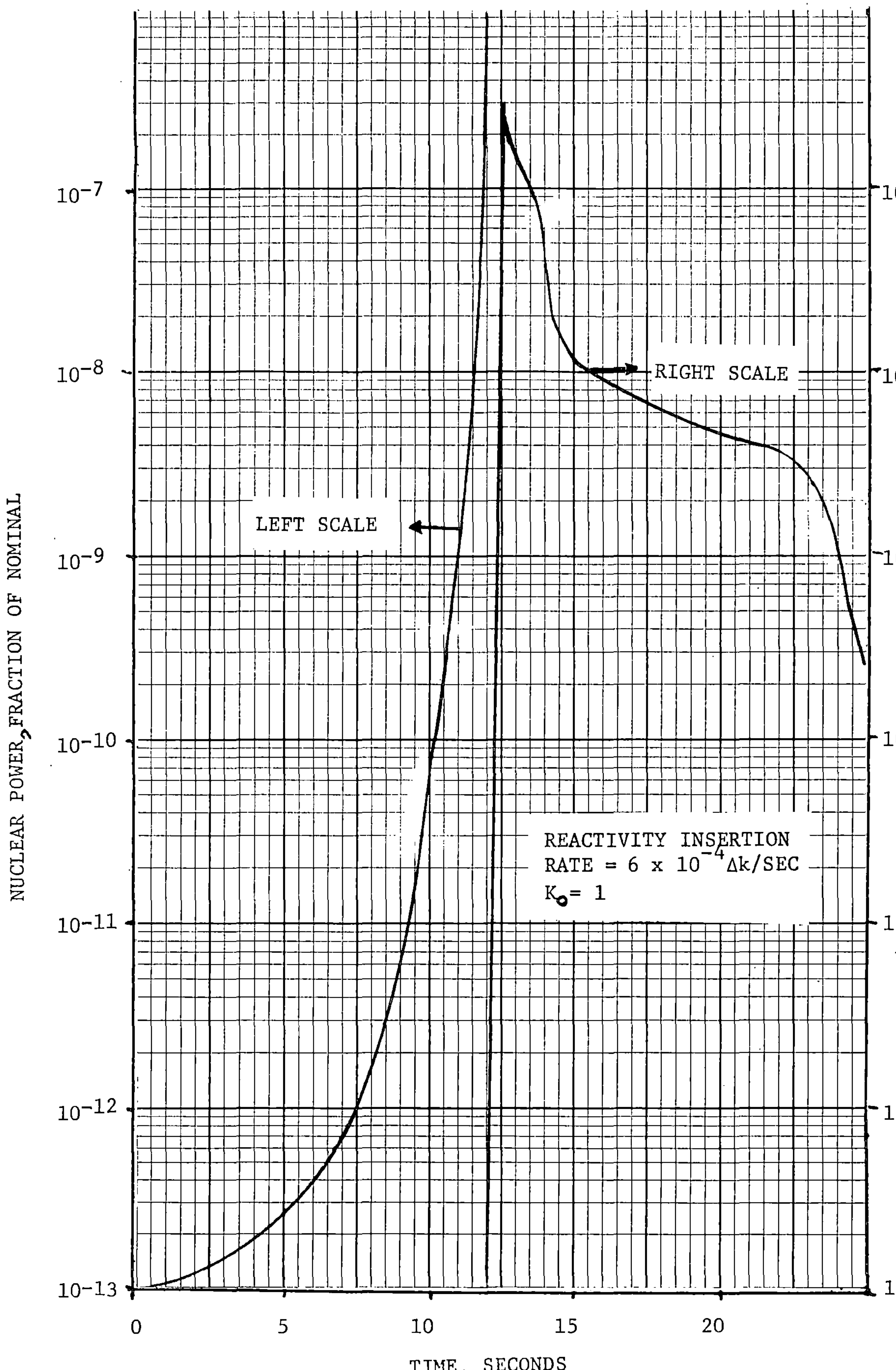
POWER COEFFICIENT (DOPPLER CONTRIBUTION ONLY) VS. POWER LEVEL FOR AN UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION

* 1 pcm = $10^{-5}\Delta K$

FIGURE 14.2.1-2
12-1-69

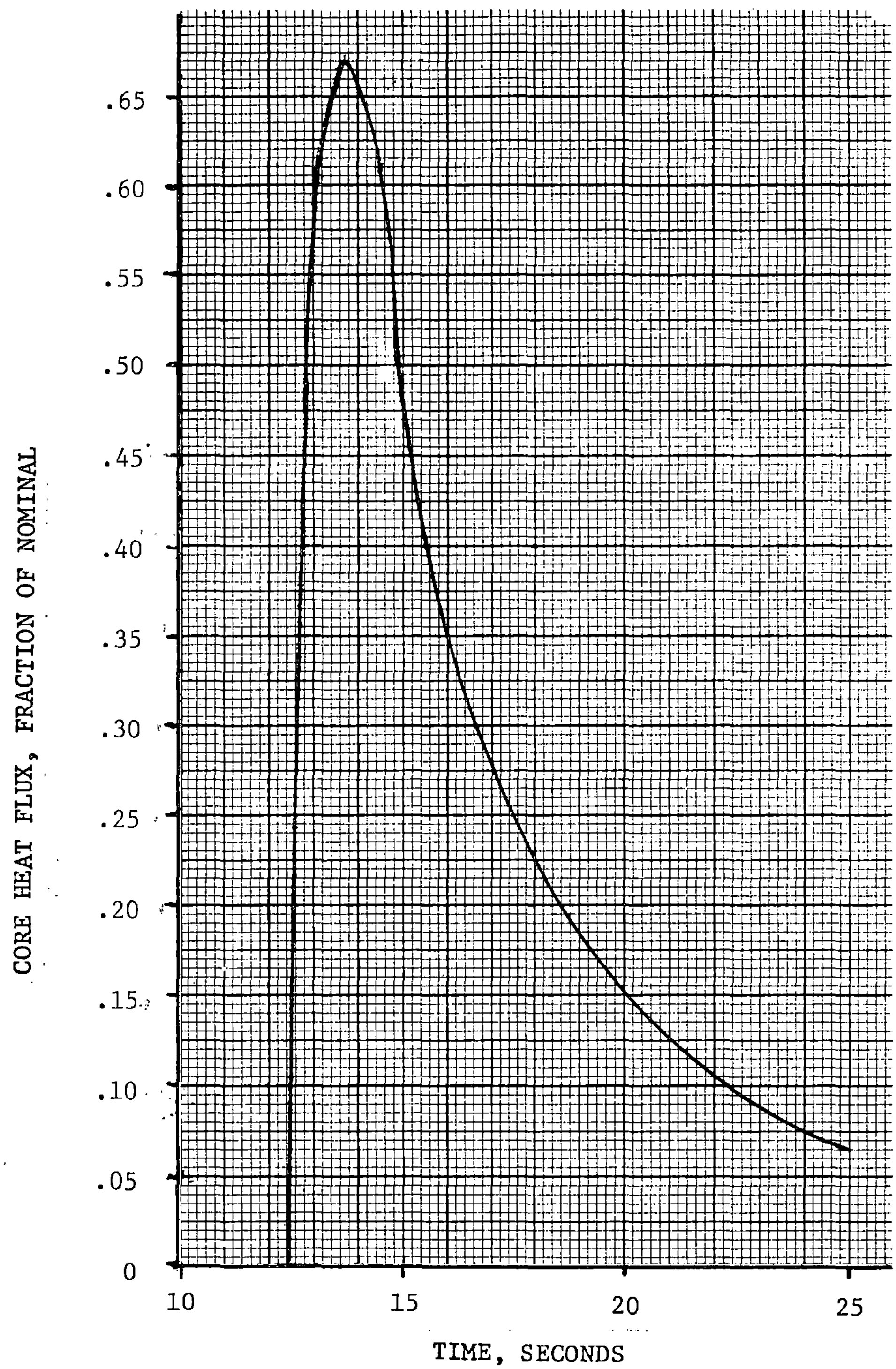


UNCONTROLLED ROD WITHDRAWAL FROM
A SUBCRITICAL CONDITION



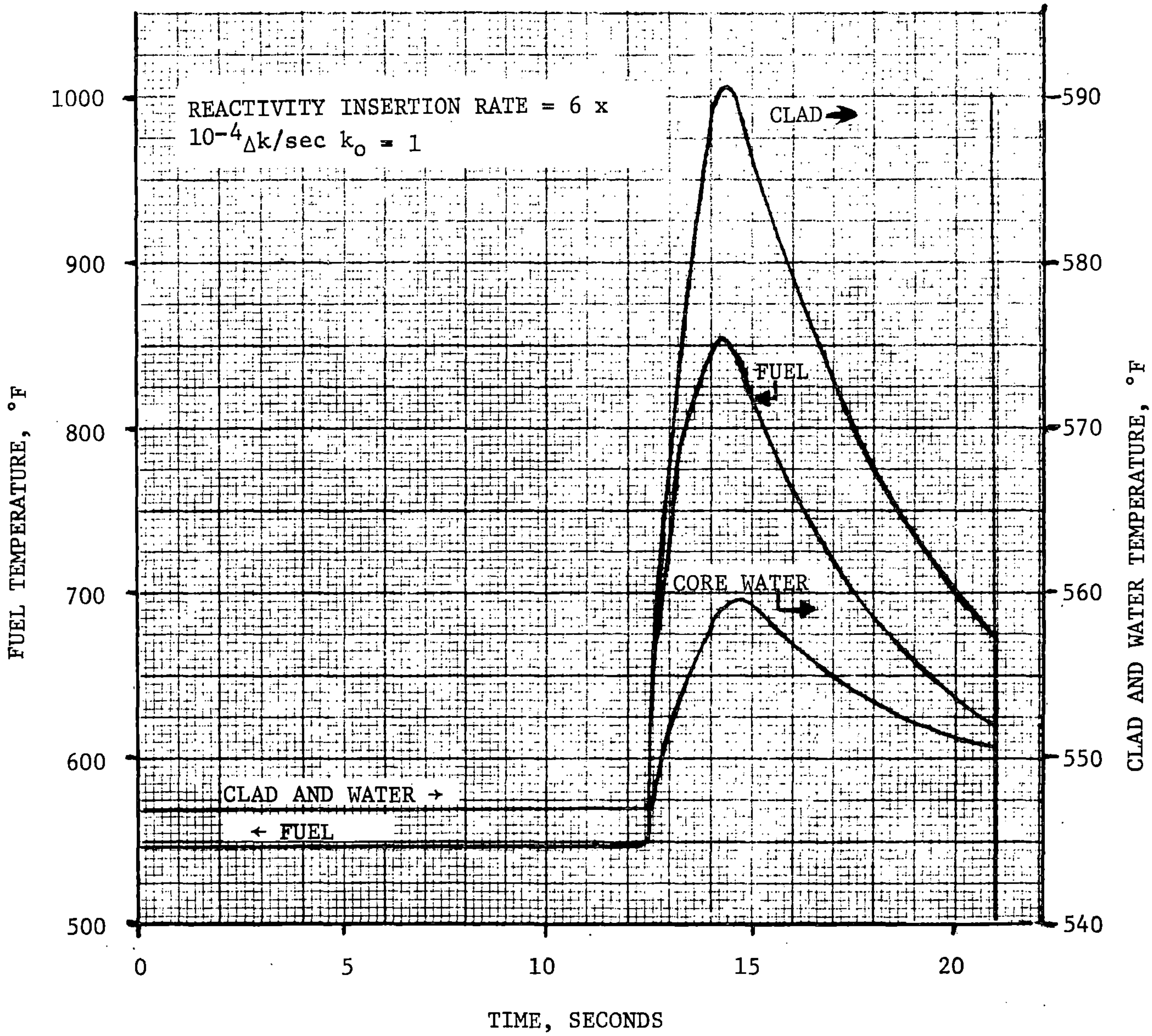
UNCONTROLLED ROD WITHDRAWAL
FROM A SUBCRITICAL CONDITION

FIGURE 14,2.1-4
12-1-69



UNCONTROLLED ROD WITHDRAWAL
FROM A SUBCRITICAL CONDITION,
CORE HEAT FLUX VS. TIME

FIGURE 14.2.1-5
12-1-69



UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION FUEL TEMPERATURE AND TIME

14.2.2 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER

An uncontrolled control rod assembly withdrawal at power results in an increase in core heat flux. Since the heat extraction from the steam generator remains constant until the steam generator pressure reaches the relief or safety valve set point, there is a net increase in reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, to prevent the possibility of damage to the cladding, the Reactor Protection System is designed to terminate any such transient before the DNBR falls below 1.3.

The automatic features of the Reactor Protection System which prevent core damage in a control rod assembly withdrawal incident at power include the following:

1. Nuclear power range instrumentation actuates a reactor trip if two out of the four channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two out of three ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power distribution to ensure that the allowable fuel power rating is not exceeded.

3. Reactor trip is actuated if any two out of three ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with axial power distribution to ensure that the allowable fuel power rating is not exceeded.

4. A high pressure reactor trip, actuated from any two out of three pressure channels, is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

5. A high pressurizer water level reactor trip, actuated from any two out of three level channels, is actuated at a setpoint. This affords additional protection for control rod assembly withdrawal incidents.

6. In addition to the above listed reactor trips, there are the following control rod assembly withdrawal blocks.

   (a) High nuclear power (one out of four)
   (b) High overpower ΔT (two out of three)
   (c) High overtemperature ΔT (two out of three)

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of Reactor Coolant System conditions is illustrated in Section 7. Figure 7.2-8 represents the allowable conditions of reactor vessel average temperature and ΔT with the design power distribution in a two-dimensional plot. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors, so that under nominal conditions trip would occur well within the area bounded by these lines.

The utility of the diagram just described is in the fact that the operating limit imposed by any given DNBR can be represented as a line on this coordinate system. The DNB lines represent the locus of conditions for which the DNBR equals 1.3. All points below and to the left of the line for a given pressure have a DNBR greater than 1.3. The diagram shows that DNB is prevented for all cases if the area enclosed within the maximum protection lines is not traversed by the applicable DNBR line at any point.

The region of permissible operation (power, pressure and temperature) is completely bounded by the combination of reactor trips: nuclear overpower (fixed setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints). These trips are designed to prevent a DNBR of less than 1.30.

Method of Analysis

The purpose of this analysis is to demonstrate the manner in which the above protective systems function for various reactivity insertion rates from different initial conditions. Reactivity insertion rates and initial conditions govern which protective function occurs first.

Analysis is performed using several digital computer codes. First, the actual core limits are determined employing the W-3 DNB correlation described in Section 3. Protection lines, illustrated in Figure 7.2-1 are then selected and incorporated into a transient analysis by a detailed digital simulation of the unit.

In order to obtain conservatively low DNBR's, the following assumptions are made:

1. Initial conditions assume maximum power and reactor coolant temperatures and minimum pressure; i.e. the power is assumed 2% high, the average temperature is assumed 4°F high, and the pressure is assumed 30 psi low. This gives the minimum initial margin to DNB.

2. A zero moderator coefficient of reactivity was assumed corresponding to the beginning of core life. A conservatively small (in absolute magnitude) Doppler power reactivity coefficient was used (see Figure 14.2.2-8). The assumed reactivity coefficients result in a minimum of negative feedback reactivity and, therefore, higher peak powers and temperatures.

3. The reactor trip on high nuclear power is assumed to be actuated at a conservative value of 118 percent of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors. The delays for trip signal actuation are assumed at their maximum values, i.e.0.5 seconds for the high nuclear power trip and 3.5 seconds for the ΔT trip.

4. The rate of negative reactivity insertion corresponding to the trip of the control rod assemblies is based on the assumption that the highest worth control rod assembly is stuck in its full withdrawn position.

The effect of the control rod assembly movement on axial core power distribution is accounted for by its effect of causing a decrease in overtemperature ΔT and overpower ΔT trip setpoints proportionate to the decrease in margin to DNB.

Results

Figures 14.2.2-1 and 14.2.2-2 show the response of nuclear power, average coolant temperature, pressure, and DNBR to a rapid control rod assembly withdrawal (6.0×10^{-4} Δk/sec) incident starting from full power. The reactivity insertion rate is greater than that for the two highest worth banks both assumed in their highest incremental worth region. It is also greater (by more than a factor of 10) than the maximum insertion rate of the part length control rod assemblies which is 5.25×10^{-5} Δk/sec. Reactor trip on high nuclear power occurs approximately 2.4 seconds after start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result. A larger margin to DNB is maintained, the minimum DNBR being 1.53.

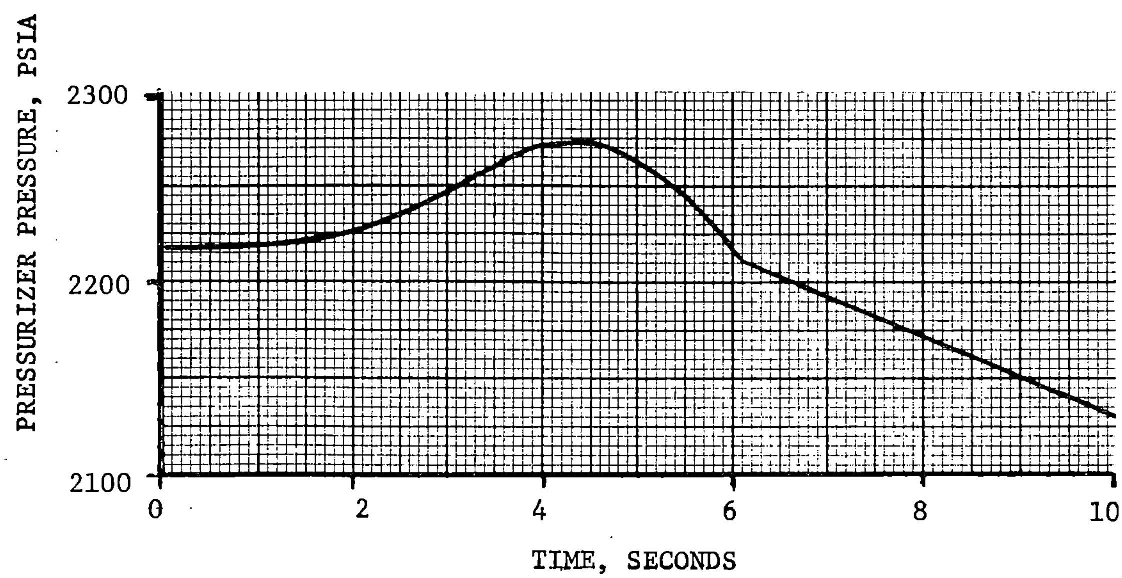
The response of nuclear power, average coolant temperature, pressure, and DNBR for a slow control rod assembly withdrawal (2.0×10^{-5} Δk/sec) from full power is shown in Figures 14.2.2-3 and 14.2.2-4. Reactor trip on overtemperature ΔT occurs after approximately 48 seconds. The rise in temperature and pressure is larger than for the rapid control rod assembly withdrawal. The minimum DNBR reached during the transient is 1.36.

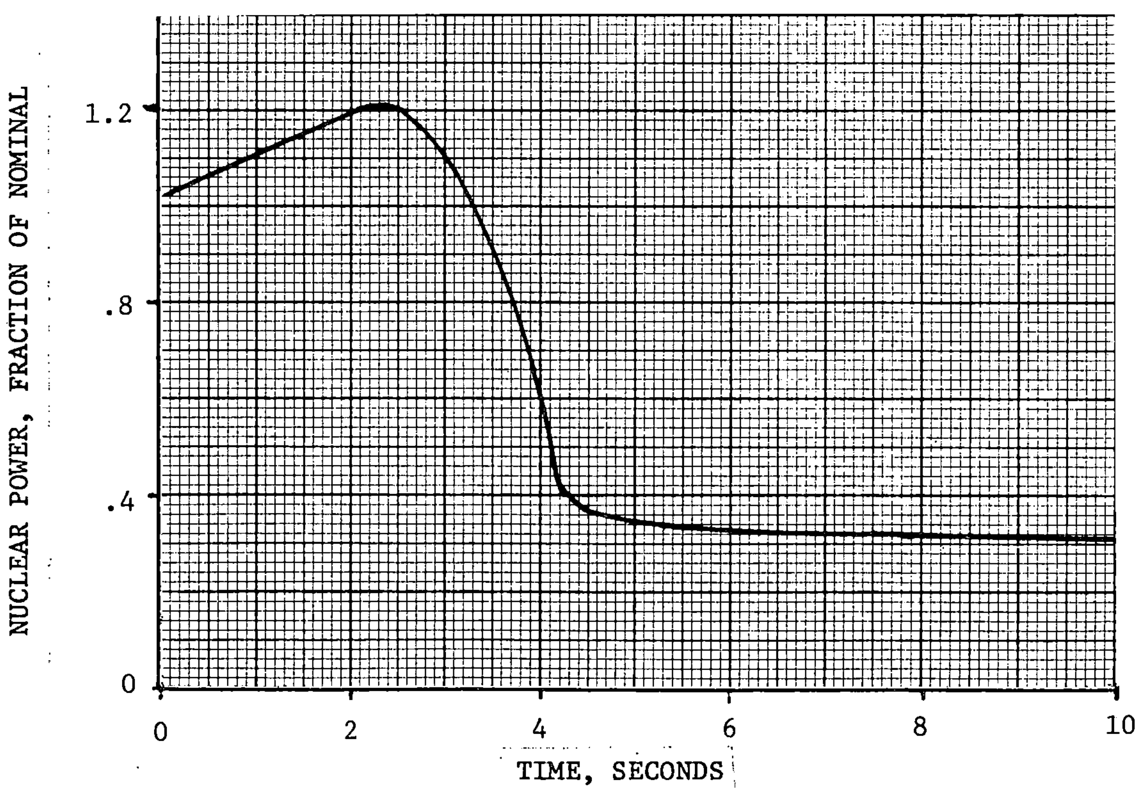
Figure 14.2.2-5 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation. It can be seen that two reactor trip channels provide protection over the whole range of reactivity rates. These are the high nuclear flux and overtemperature ΔT trip channels. The cross over point between the two zones of effectiveness occurs when the reactivity rate is 3.4×10^{-5} Δk/sec. The minimum DNBR is never less than 1.34.

Figures 14.2.2-6 and 14.2.2-7 show the minimum DNBR as a function of reactivity insertion rate for control rod assembly withdrawal incidents starting at 60 and 10 percent power respectively. The results are very similar to the 100 percent power case, except as the initial power is decreased, the range over which the overtemperature ΔT trip operates is increased. In neither case does the DNBR fall below 1.30.

Conclusions

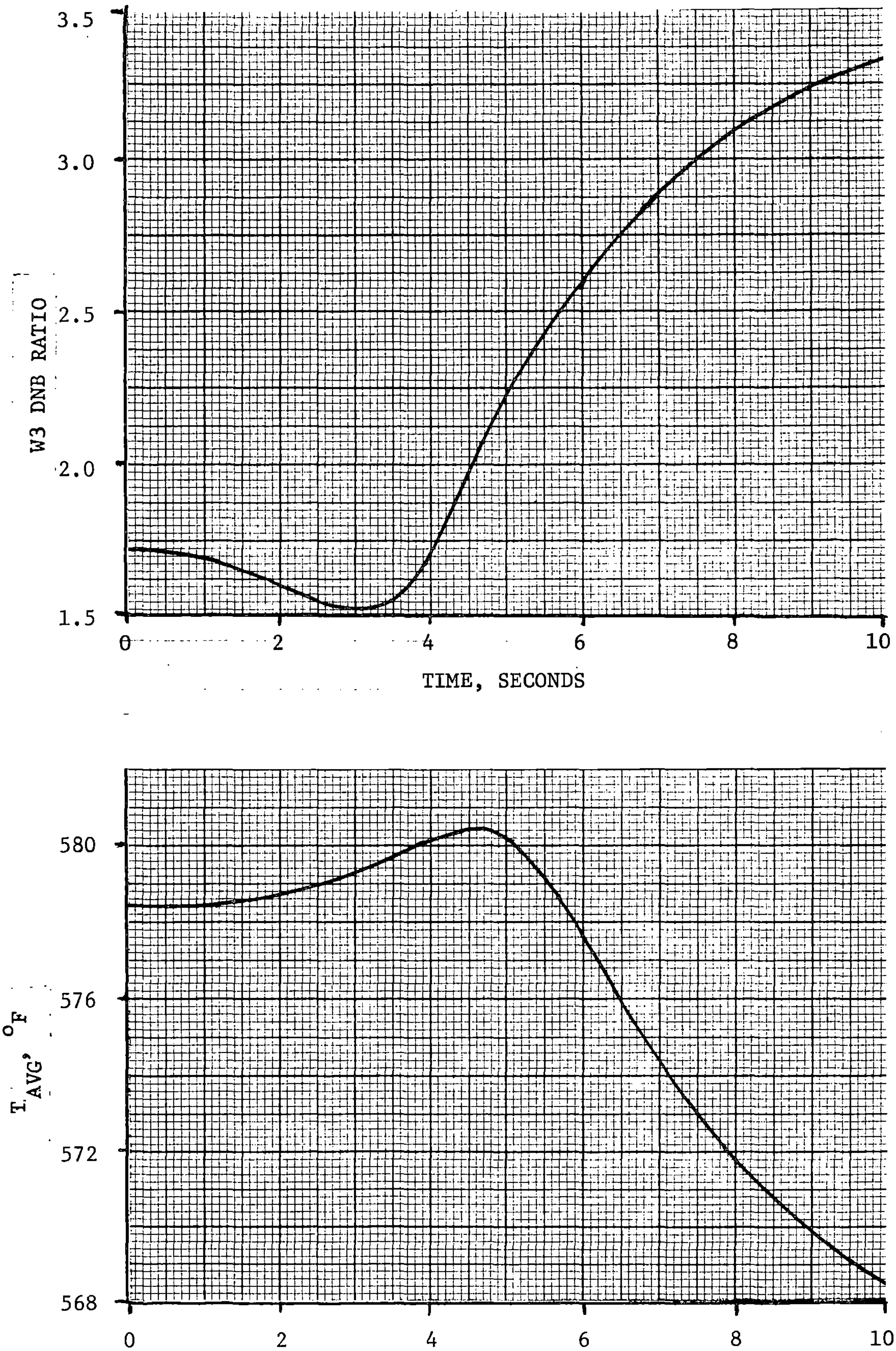
In the unlikely event of a control rod assembly withdrawal incident during power operation, the core and Reactor Coolant System are not adversely affected since the minimum value of the DNBR reached is in excess of 1.3 for all control rod assembly reactivity rates. Protection is provided by the nuclear flux overpower, and the overtemperature ΔT trips. The preceding sections have described the effectiveness of these protection channels.



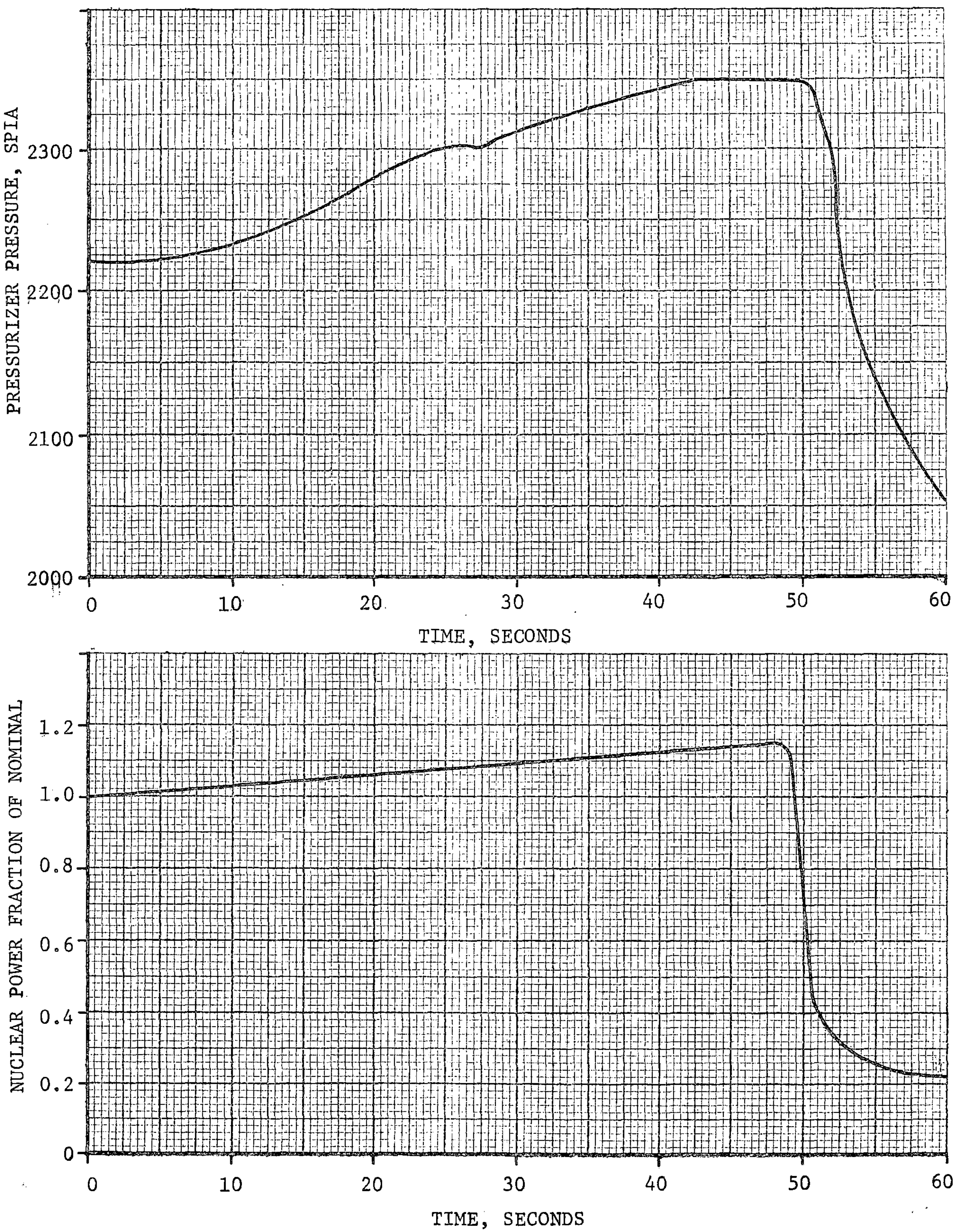


TRANSIENT RESPONSE FOR UNCONTROLLED
ROD WITHDRAWAL FROM FULL POWER
TERMINATED BY NUCLEAR OVERPOWER TRIP
(6×10^{-4} Δk/sec)

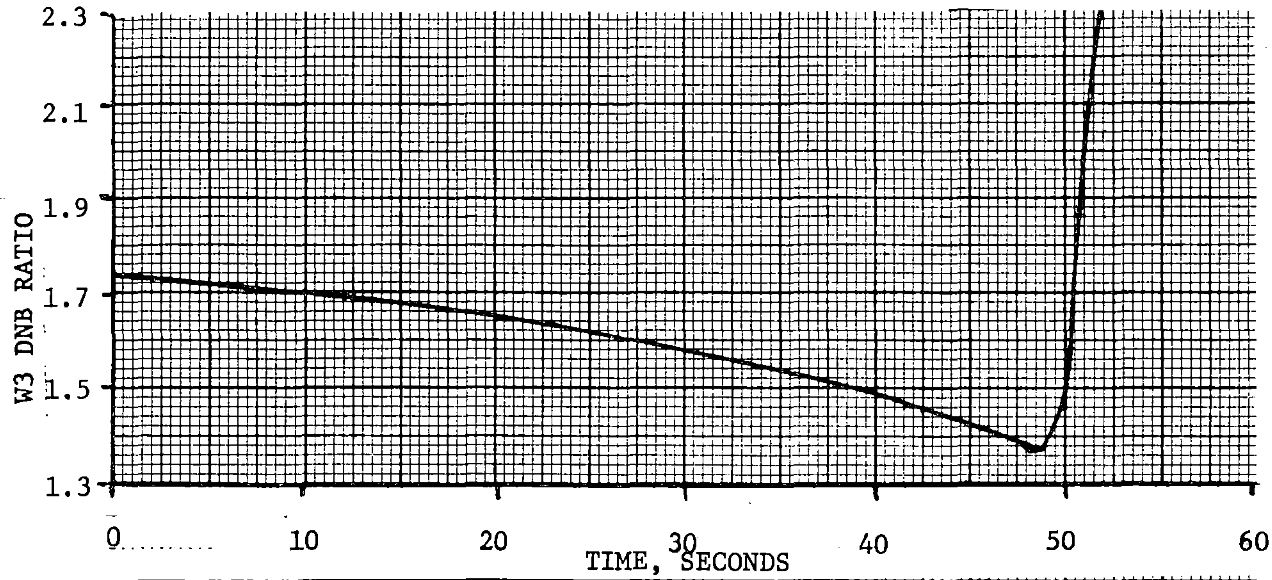
FIGURE 14.2.2-2
12-1-69

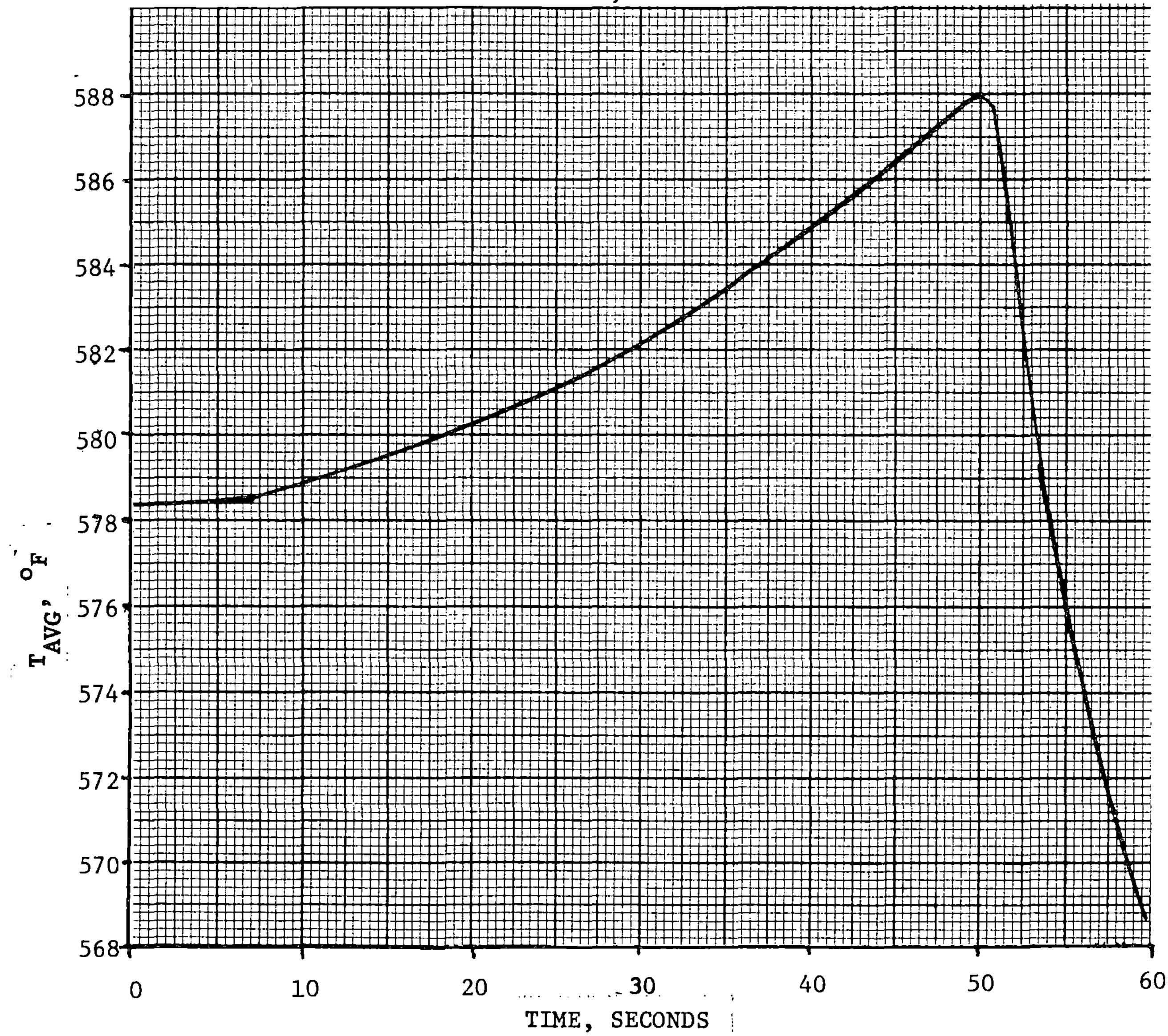


TRANSIENT RESPONSE FOR UNCONTROLLED ROD WITHDRAWAL FROM FULL POWER TERMINATED BY NUCLEAR OVERPOWER TRIP ($6 \times 10^{-4} \Delta K/SEC$)

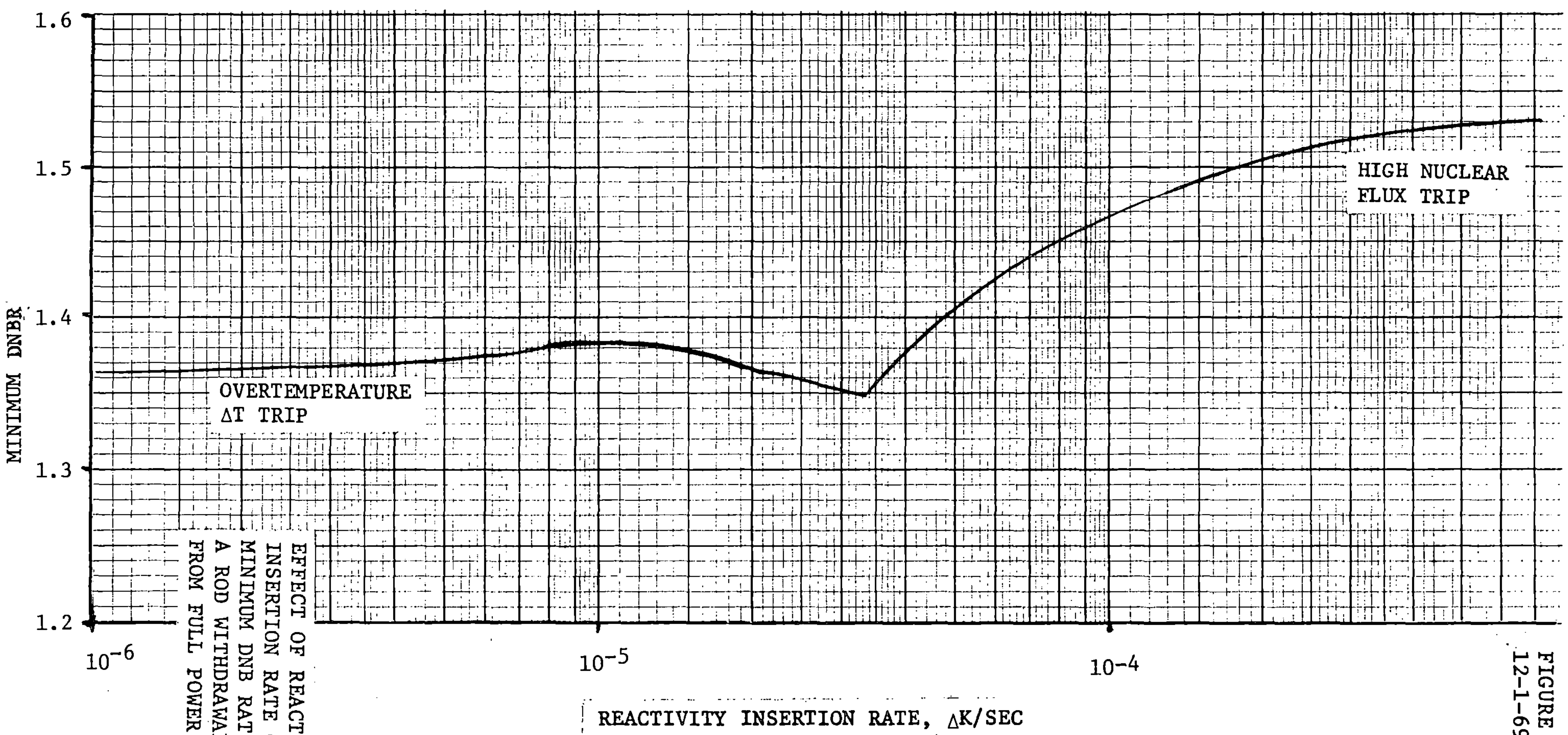


TRANSIENTS RESPONSE FOR UNCONTROLLED ROD WITHDRAWAL FROM FULL POWER TERMINATED BY OVERTEMPERATURE ΔT TRIP (2×10^{-5} Δk/sec)



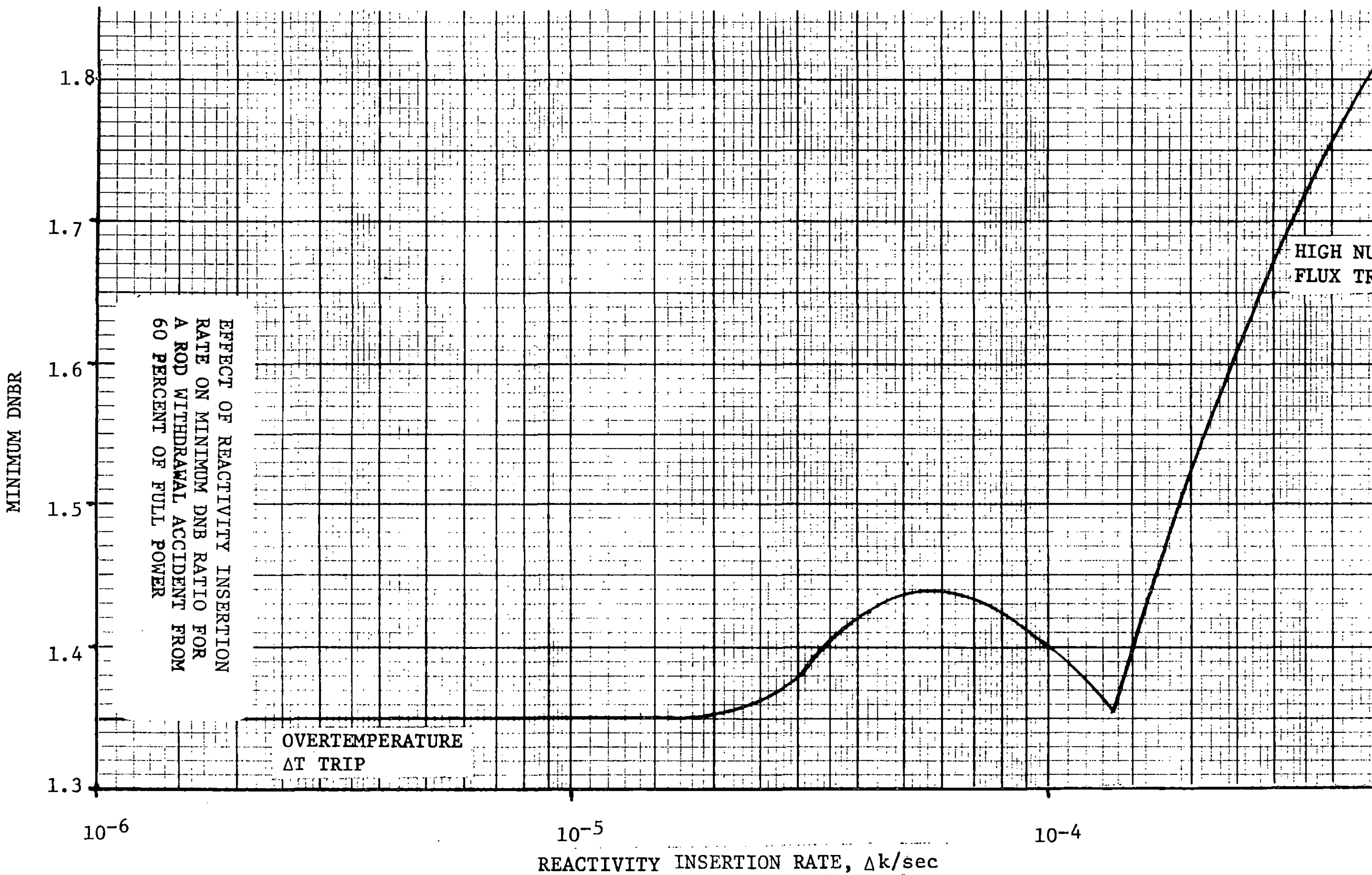


TRANSIENT RESPONSE FOR UNCONTROLLED ROD WITHDRAWAL FROM FULL POWER TERMINATED BY OVERTEMPERATURE ΔT TRIP (2×10^{-5} Δk/sec)



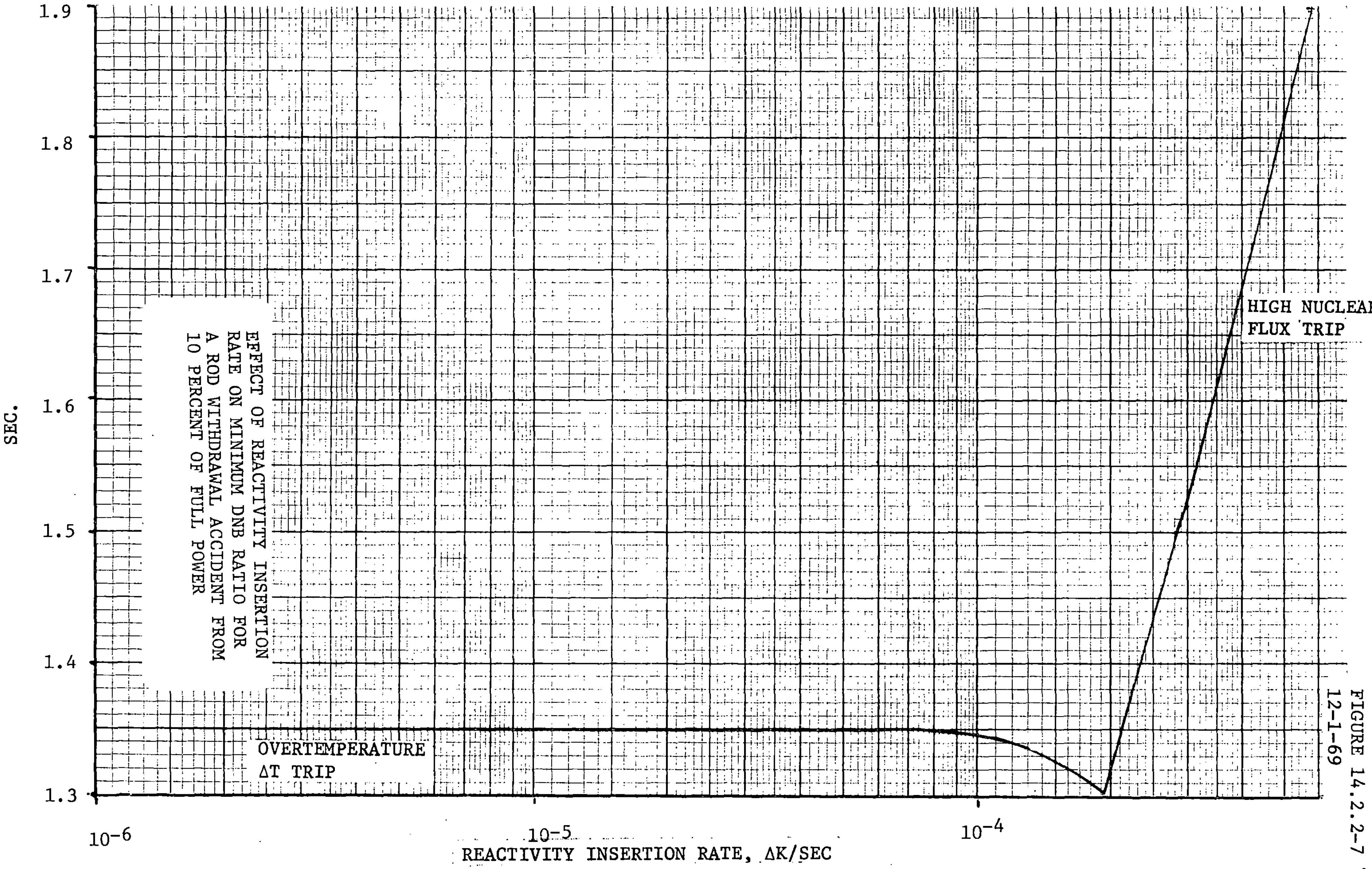
EFFECT OF REACTIVITY INSERTION RATE ON MINIMUM DNB RATIO FOR A ROD WITHDRAWAL ACCIDENT FROM FULL POWER CONDITIONS

FIGURE 14.2.2-5
12-1-69



EFFECT OF REACTIVITY INSERTION RATE ON MINIMUM DNB RATIO FOR A ROD WITHDRAWAL ACCIDENT FROM 60 PERCENT OF FULL POWER

FIGURE 14.2.2-6
12-1-69

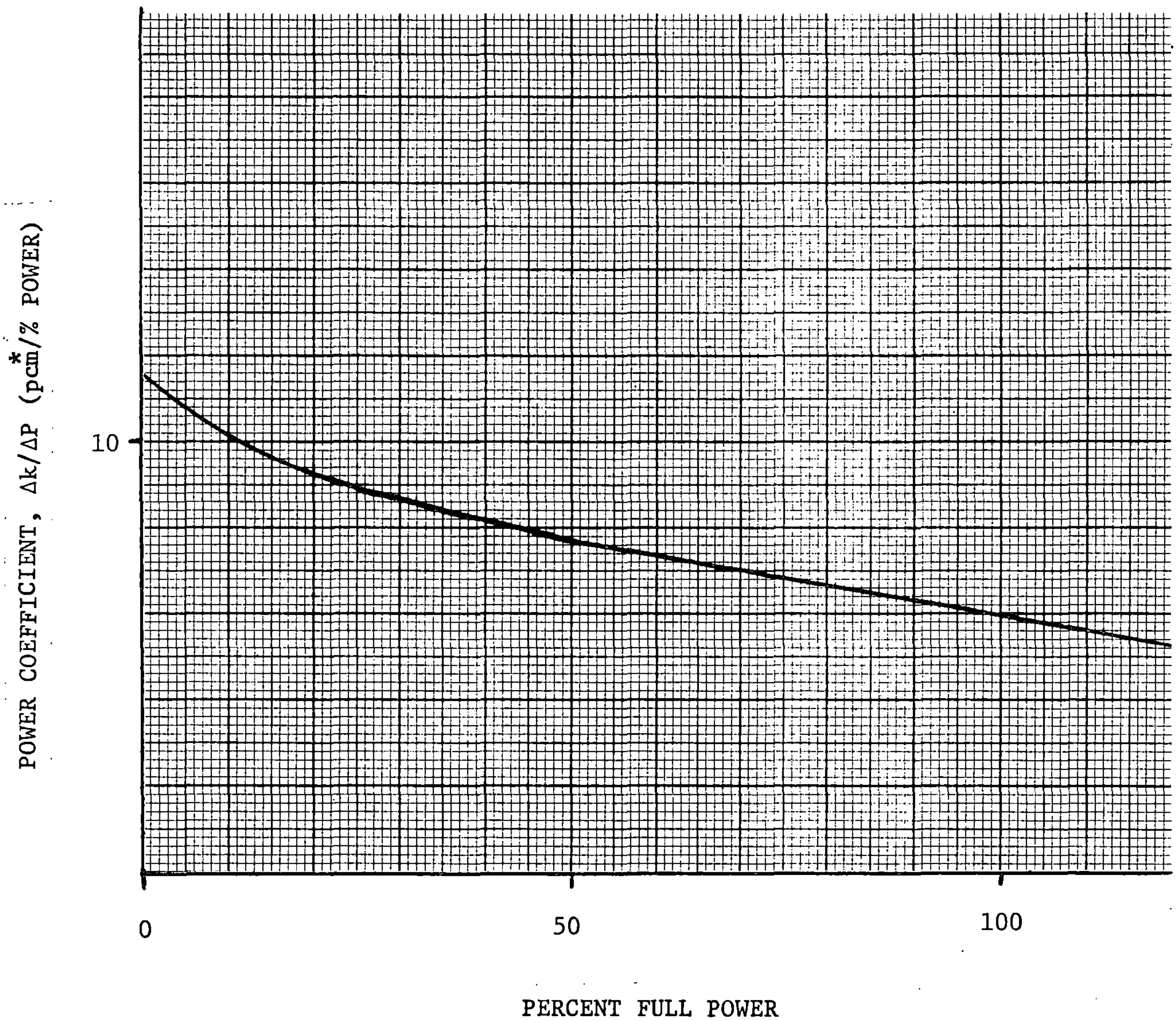


EFFECT OF REACTIVITY INSERTION RATE ON MINIMUM DNB RATIO FOR A ROD WITHDRAWAL ACCIDENT FROM 10 PERCENT OF FULL POWER

FIGURE 14.2.2-7

12-1-69

FIGURE 14.2.2-8
12-1-69



POWER COEFFICIENT (DOPPLER CONTRIBUTION ONLY)
VS. POWER LEVEL FOR AN UNCONTROLLED
CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER

* 1 pcm = $10^{-5}\Delta k$

14.2.3 MALPOSITIONING OF THE PART LENGTH CONTROL ROD ASSEMBLIES

Part Length Control Rod Assemblies

The part length control rod assemblies used to optimize the axial power distribution and to control potential axial xenon oscillations.[1] A failure of the power supply will not cause these assemblies to drop into the core. Since the assemblies are controlled manually, the malfunctions that can occur are due to operator action or inaction.

The maximum assembly speed and incremental rod worth (see Sections 14.2.1 and 14.2.2) of the part length control rod assemblies are such that if the operator continuously moves the assemblies erroneously the consequences are less severe than for the full length control rod assembly withdrawal cases considered in Sections 14.2.1 and 14.2.2.

Since the part length control rod assemblies are used to counter xenon oscillations having a period of about 30 hours, the part length assemblies need to be moved only at several hour intervals. It may be necessary to move the part length control rod assemblies during load changes, but that is not expected to be the normal case.

The instrumentation system provides sufficient information to the operator for positioning the part length control rod assemblies by displaying the top and bottom core power imbalance (out-of-core ion chamber current difference) in each quadrant of the core on the control board. Out of limit signals are generated should the relative readings differ by a preset amount. The Reactor Protection System automatically resets the overpower and overtemperature trip settings to a level consistent with the power distribution. A reduction in the power capability might be necessary until the normal power distribution is restored. The Control and Protection System automatically initiates a turbine runback and full length control rod assembly withdrawal stop to prevent an unnecessary trip caused by the trip setpoint reductions. The Control and Protection System thus ensures that the core limits are not reached as a result of operator inaction. Once the operator takes the necessary steps to rectify the maldistribution in axial flux, the turbine load can be increased to the previous level.

The operator is provided with actual assembly position indication for each part length control rod assembly and the demanded position of the part length control assembly rod bank on the control board.

REFERENCE: Section 14.2.3

1. Westinghouse Proprietary, "Power Distribution Control in Westinghouse Pressurized Water Reactors," WCAP-7208 (1968).

14.2.4 CONTROL ROD ASSEMBLY DROP

Dropping of a full length control rod assembly could occur only when the drive mechanism is de-energized. This would result in a power reduction and an increase in the hot channel factor. If no protective action occurred, the Reactor Control System would restore the power to the level which existed before the incident. This would lead to a reduced safety margin or possibly DNB, depending upon the magnitude of the hot channel factor.

If a control rod assembly drops into the core during power operation, it would be detected by either a rod bottom signal device or by the use of the out of core ion chambers.[1] The rod bottom signal device provides an individual position indication signal for each control rod assembly. Initiation of this signal is independent of lattice location, reactivity worth, or power distribution changes inherent with the dropped control rod assembly. The other independent indication of a control rod assembly drop is obtained by using the out of core power range channel signals. This rod drop detection circuit is actuated upon sensing a rapid decrease in local flux such as could occur from depression of flux in one region by a dropped control rod assembly. This detection circuit is designed such that normal load variations do not cause it to be actuated.

A rod drop signal from any control rod assembly position indication channel, or from one or more of the four power range channels, initiates protective action by reducing turbine load by a preset adjustable amount and blocking of further automatic control rod assembly withdrawal. The turbine runback is redundantly obtained by acting upon the turbine load limit and on the turbine load reference. The control rod assembly stop is also redundantly actuated.

Method of Analysis

The transient following a dropped control rod assembly incident is determined by a detailed digital simulation of the unit. The dropped assembly is assumed to cause a step decrease in reactivity and the core power generation is determined using a point neutron kinetics model. The overall unit response is calculated by simulating the turbine load runback and blocking of automatic control rod assembly withdrawal. The analysis is performed for the case in which the load cutback nearly matches the power decrease from the negative reactivity for a dropped assembly (-2.6×10^{-3} Δk), and also for the case in which the load cutback is greater than that required to match the worth of the dropped assembly (-1.0×10^{-3} Δk). In both cases the load is assumed to be cut back from 102 to 76 percent of full load at a conservatively slow rate of one percent per second. The actual amount of load cutback to be used is determined during initial startup experiments and is set to match the power reduction caused by the highest worth dropped control rod assembly.

The most negative values of moderator and Doppler temperature coefficients of reactivity are used in this analysis resulting in the highest heat flux during the transient. These are a moderator temperature coefficient of -3.5×10^{-4} $\Delta k/°F$ and a Doppler coefficient of -1.6×10^{-5} $\Delta k/°F$. A control bank worth of 4×10^{-5} $\Delta k/in$ is assumed as equilibrium conditions are restored. The incident is analyzed at the following initial conditions:

Power:	102% nominal	(2490 MWt)
T_{avg}:	Nominal +4°F	(578.4°F)
Pressure:	Nominal -30 psi	(2220 psia)

Results

Figures 14.2.4-1 and 14.2.4-2 illustrate the transient response following a dropped control rod assembly of -2.6×10^{-3} Δk. The coolant average temperature decreases initially due to the fact that more energy is taken out of the primary system than is produced in the core, then increases as the load is reduced and as the nuclear power increases because of the moderator negative reactivity feedback. The temperature then decreases slowly to a new steady-state value. The peak heat flux following the initial response to the dropped control rod assembly is 93 percent of nominal. At the time of the peak heat flux, the core average temperature has dropped by 3.8°F and the pressure by 40 psi.

Figures 14.2.4-3 and 14.2.4-4 illustrate the transient response following a dropped control rod assembly of $-1. \times 10^{-3}$ Δk. Again the coolant average temperature decreases initially, and then increases because of the negative feedback and the load cutback. The equilibrium temperature is achieved in about four minutes. For this case the peak heat flux following the initial response to the dropped assembly is 98.0 per cent of nominal. At the same time the core average temperature drops by 1.0°F and the pressure by 12 psi.

An analysis has been made of the amount of flux tilt that can be tolerated without core damage for the maximum full power operating conditions (2490 MWt power, core water inlet temperature of 547°F, primary pressure of 2220 psia). The effect of the flux tilt was represented by an increase in the radial heat flux hot channel factor. It was found that this factor could be increased by 13 percent before reaching a DNBR of 1.30. During initial startup experiments, it is verified that the flux tilt caused by the worst dropped rod, coupled with the thermal flux, coolant temperature, and primary system pressure responses, will not result in a condition of DNB.

Conclusions

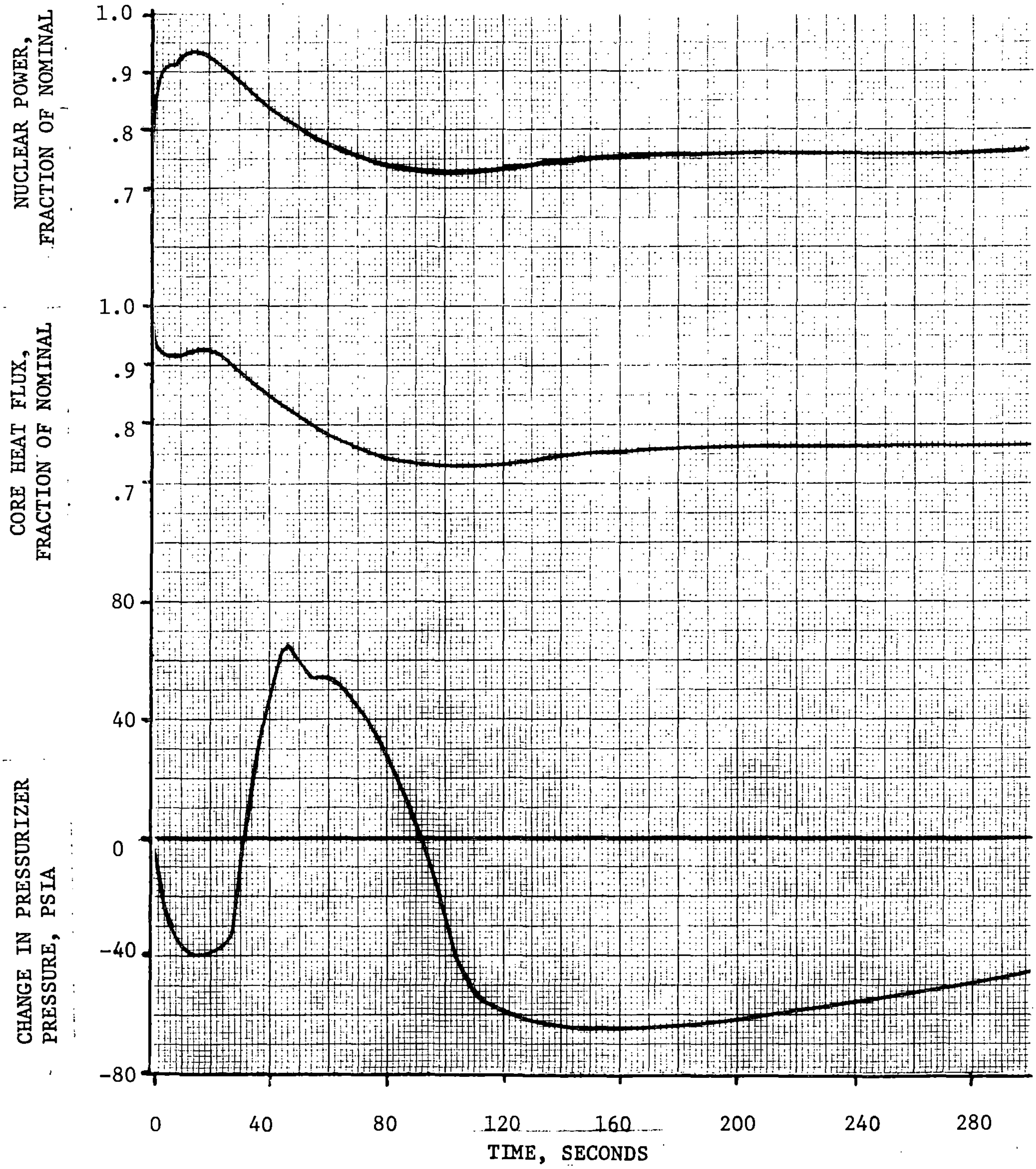
Protection for a dropped control rod assembly is provided by automatic turbine power runback and blocking of automatic control rod assembly withdrawal. The magnitude of the power cutback is determined during the initial startup tests. As the analyses presented show, the protection system, in conjunction with the load cutback, protects the core from DNB for a power tilt of 13 percent at maximum full power conditions. At the reduced power condition following the control rod assembly drop, this allowable tilt is even greater.

The power tilt is experimentally determined and the protection system set to maintain a DNBR greater than 1.30.

REFERENCES: Section 14.2.4

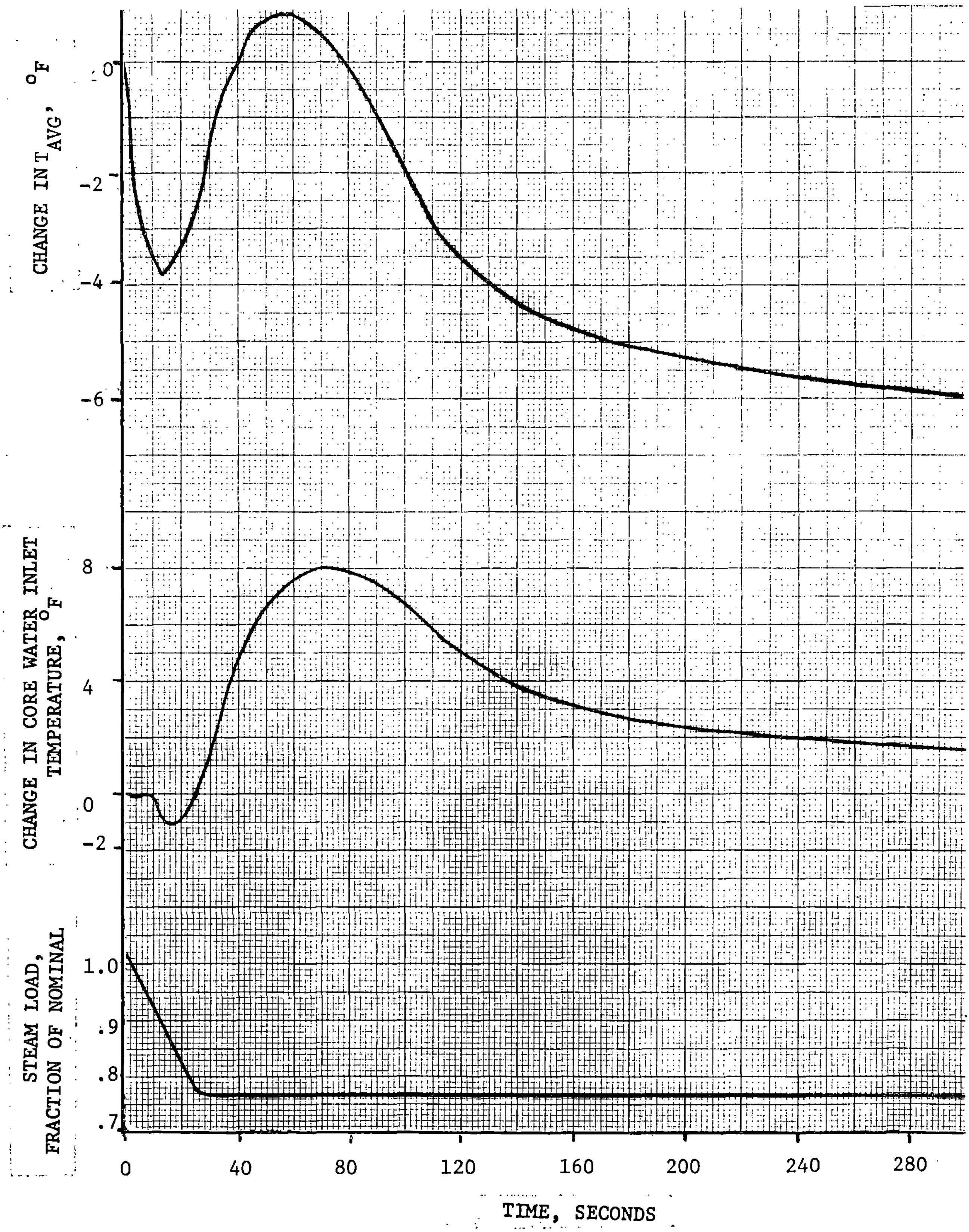
1. Westinghouse Proprietary, "Power Distribution Control in Westinghouse Pressurized Water Reactors," WCAP-7208 (1968).

FIGURE 14.2.4-1
12-1-69



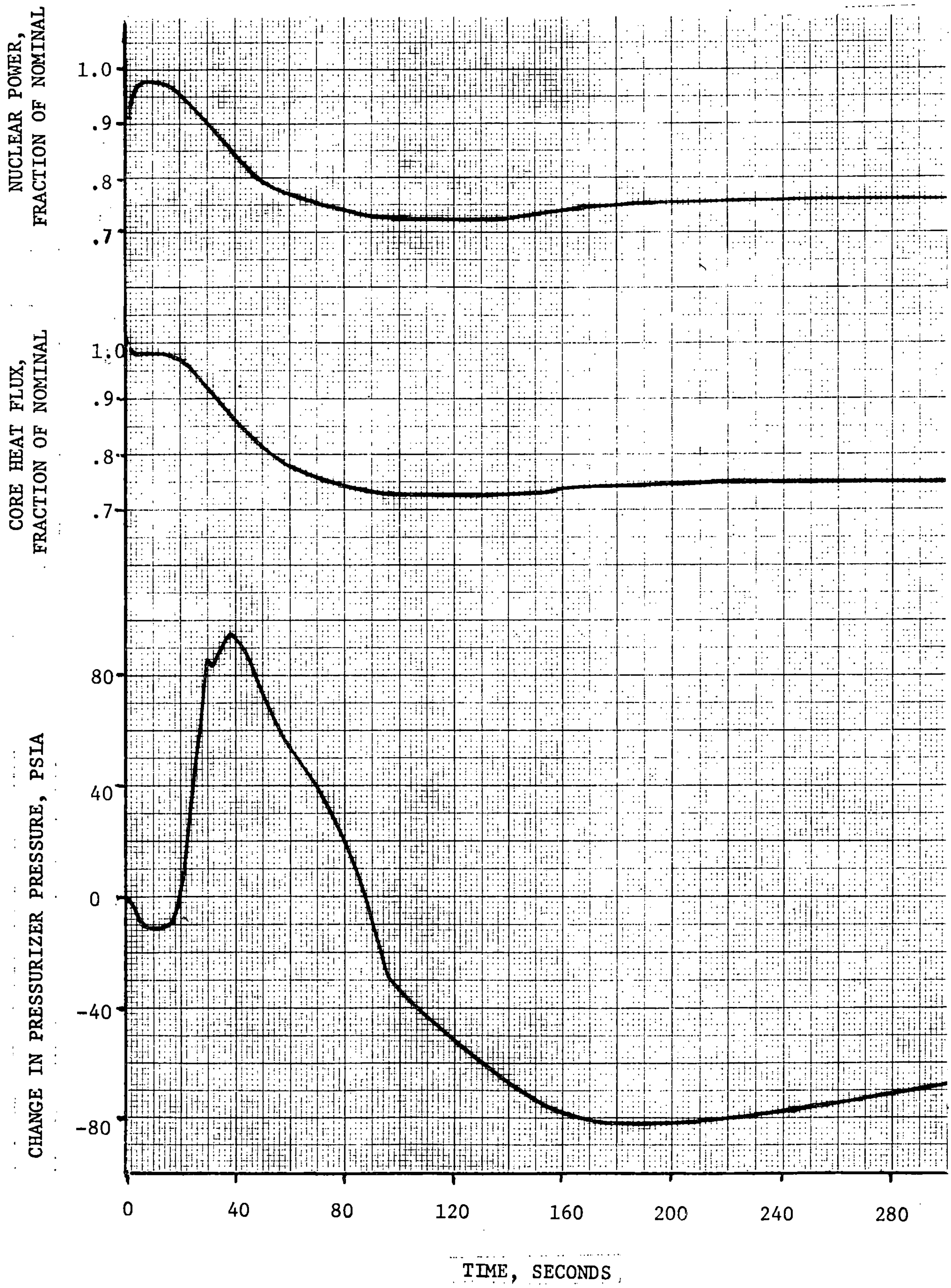
RESPONSE TO A DROPPED CONTROL ROD ASSEMBLY OF WORTH $-2.6 \times 10^{-3}\ \Delta k$ WITH LOAD CUTBACK

FIGURE 14.2.4-2
12-1-69



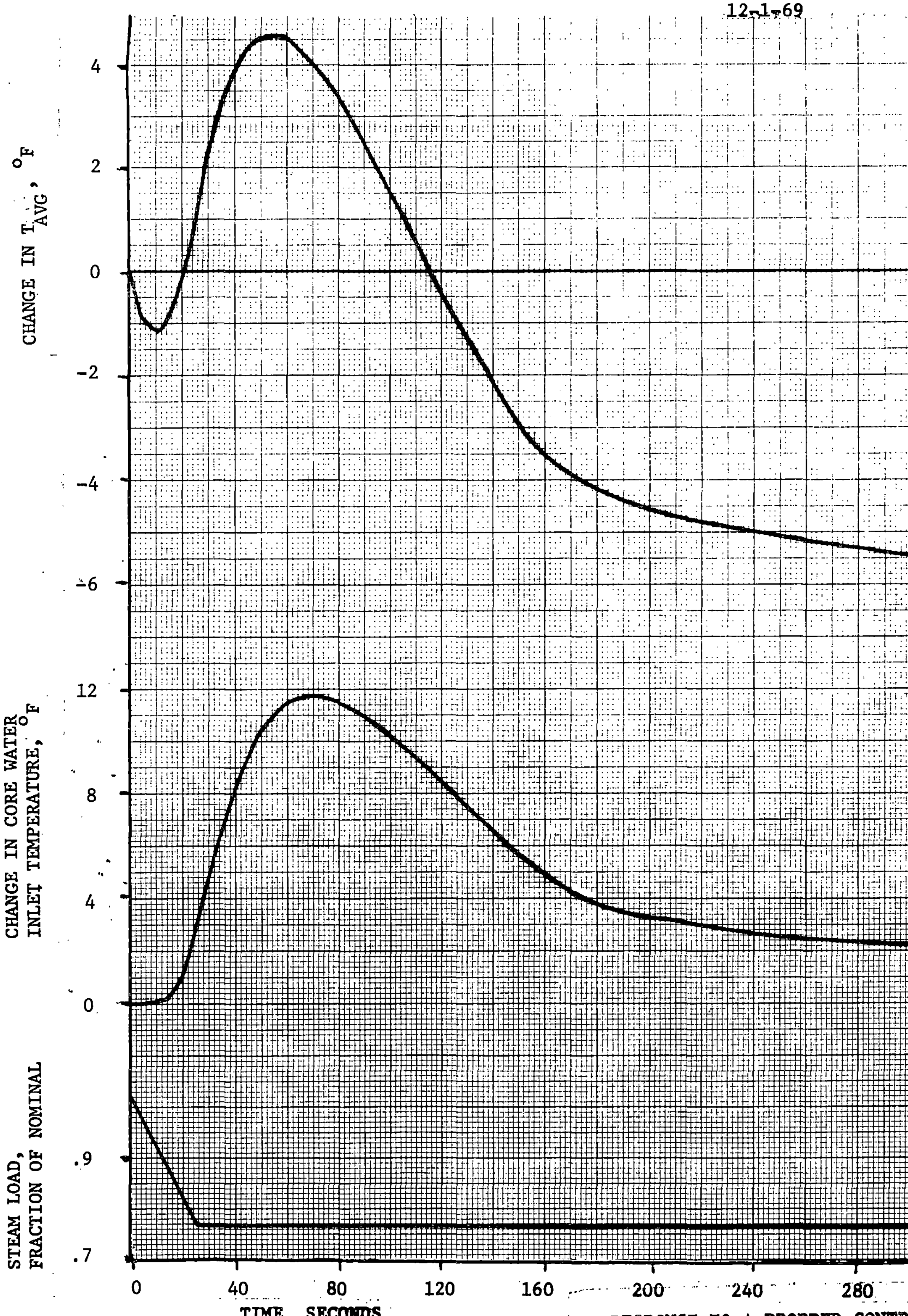
RESPONSE TO A DROPPED CONTROL ROD ASSEMBLY OF WORTH - 2.6×10^{-3} Δk WITH LOAD CUTBACK

FIGURE 14.2.4-3
12-1-69



RESPONSE TO A DROPPED ROD OF WORTH
$-1.0 \times 10^{-3} \Delta k$ WITH LOAD CUTBACK

FIGURE 14.2.4-4
12-1-69



RESPONSE TO A DROPPED CONTROL ROD ASSEMBLY OF WORTH -1.0 x 10^{-3} Δk WITH LOAD CUTBACK

14.2.5 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION

Reactivity can be added to the core by feeding primary grade water into the Reactor Coolant System via the reactor makeup portion of the Chemical and Volume Control System. The normal dilution procedures call for a limit on the rate and magnitude for any individual dilution, under strict administrative controls. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the Reactor Coolant System. The Chemical and Volume Control System is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the primary water makeup control valve provides the only supply of makeup water to the Reactor Coolant System which can dilute the reactor coolant. Inadvertent dilution can be readily terminated by closing this value. In order for makeup water to be added to the Reactor Coolant System, at least one charging pump must also be running in addition to the reactor makeup water pumps.

The rate of addition of unborated water makeup to the Reactor Coolant System when it is not at pressure is limited by the capacity of the primary water supply pumps. The maximum addition rate in this case is 300 gpm with both primary water supply pumps running. The 300 gpm reactor makeup water delivery rate is based on a pressure drop calculation comparing the pump curves with the system resistance curve. This is the maximum delivery based on the unit piping layout. Normally, only one primary water supply pump is operating while the other is on standby. With the Reactor Coolant System at pressure, the maximum delivery rate is limited by the charging pumps. Assuming all three charging pumps are operating, the maximum deliver rate is 165 gpm. Normally only one charging pump is in operation.

The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flow rates of boric acid and primary grade water on the control board. In order to dilute, two separate operations are required. First, the operator must switch from the automatic makeup mode to the dilute mode. Second, the start button must be depressed. Omitting either step would prevent dilution. This makes the possibility of inadvertent dilution very remote.

Information on the status of reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of pumps in the Chemical and Volume Control System. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

To cover all phases of unit operaiton, boron dilution during refueling, startup, and power operation are considered in this analysis.

14.2.5.1 Method of Analysis and Results

Dilution During Refueling

During refueling the following conditions exist:

1. One residual heat removal pump is operating to ensure continuous mixing in the reactor vessel.

2. The valve in the seal injection water header to the reactor coolant pumps is closed.

3. The valves on the suction side of the charging pumps are adjusted for addition of concentrated boric acid solution.

4. The boron concentration of the refueling water is 2000 ppm, corresponding to a shutdown of at least 10 per cent Δk with all control rods in; periodic sampling ensures that this concentration is maintained.

5. The source range detectors outside the reactor vessel are active and provide an audible count rate. During initial core loading, sources are installed in the core and B^{10} detectors connected to instrumentation giving audible count rates are installed within the reactor vessel to provide direct monitoring of the core.

A minimum water volume in the Reactor Coolant System of 3345 ft^3 is considered. This corresponds to the volume necessary to fill the reactor vessel above the nozzles to ensure mixing via the residual heat removal loop. A maximum dilution flow of 300 gpm, limited by the capacity of the two primary water supply pumps, and uniform mixing are also considered.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High count rate is alarmed in the reactor containment and the Main Control Room. The count rate increase is proportional to the subcritical multiplication factor. At 1500 ppm, for example, the core is approximately 5 percent shutdown and the count rate is increased by a factor of 2 over the count rate at 2000 ppm.

The boron concentration must be reduced from 2000 ppm to approximately 1000 ppm before the reactor will go critical. This would take 57 minutes or about 1 hour. This is ample time for the operator to recognize a high count rate signal and isolate the reactor makeup water source by closing valves and stopping the primary water supply pumps.

Dilution During Startup

Prior to startup, the Reactor Coolant System is filled with borated (2000 ppm) water from the refueling water storage tank. Core monitoring is by external BF_3 detectors. Mixing of the reactor coolant is accomplished by operation of the reactor coolant pumps. High source level and all reactor trip alarms are effective.

In the analysis, a maximum dilution flow of 300 gpm is again considered. The volume of reactor coolant is approximately 7600 ft^3 which is the active volume of the Reactor Coolant System excluding the pressurizer.

The minimum time required to reduce the reactor coolant boron concentration to 1000 ppm, where the reactor would go critical with all control rod assemblies in, is 98 minutes or about 1.5 hours. Once again this should be more than adequate time for the operator to recognize the high count rate signal and terminate the dilution flow.

Dilution at Power

With the unit at power and the Reactor Coolant System at pressure, the dilution rate is limited by the capacity of the charging pumps. The effective reactivity addition rate for a boron dilution flow of 165 gpm (3 charging pumps in operation) at 574.4°F is shown as a function of Reactor Coolant System boron concentration in Figure 14.2.5-1. This figure includes the effect of increasing boron worth with dilution. The reactivity rate used in the following evaluation is 1.02×10^{-5} Δk/sec based on a conservatively high value for the expected boron concentration (1500 ppm) at power.

With the reactor in automatic control at full power, the power and temperature increase from boron dilution results in the insertion of the control rod assemblies and a decrease in shutdown margin. Continuation of dilution and control rod assembly insertion would cause the assemblies to reach the minimum limit of the rod insertion monitor in approximately 6.5 minutes assuming the control rod assemblies to be initially at a position providing the maximum operational maneuvering band consistent with maintaining a minimum control bank incremental rod worth. Before reaching this point, however, two alarms would be actuated to warn the operator of the accident condition. The first of these, the low insertion limit alarm, alerts the operator to initiate normal boration. The other, the extra low insertion limit alarm, alerts the operator to follow emergency boration procedures. The low alarm is set sufficiently above the low-low alarm to allow normal boration without the need for ememgency procedures.

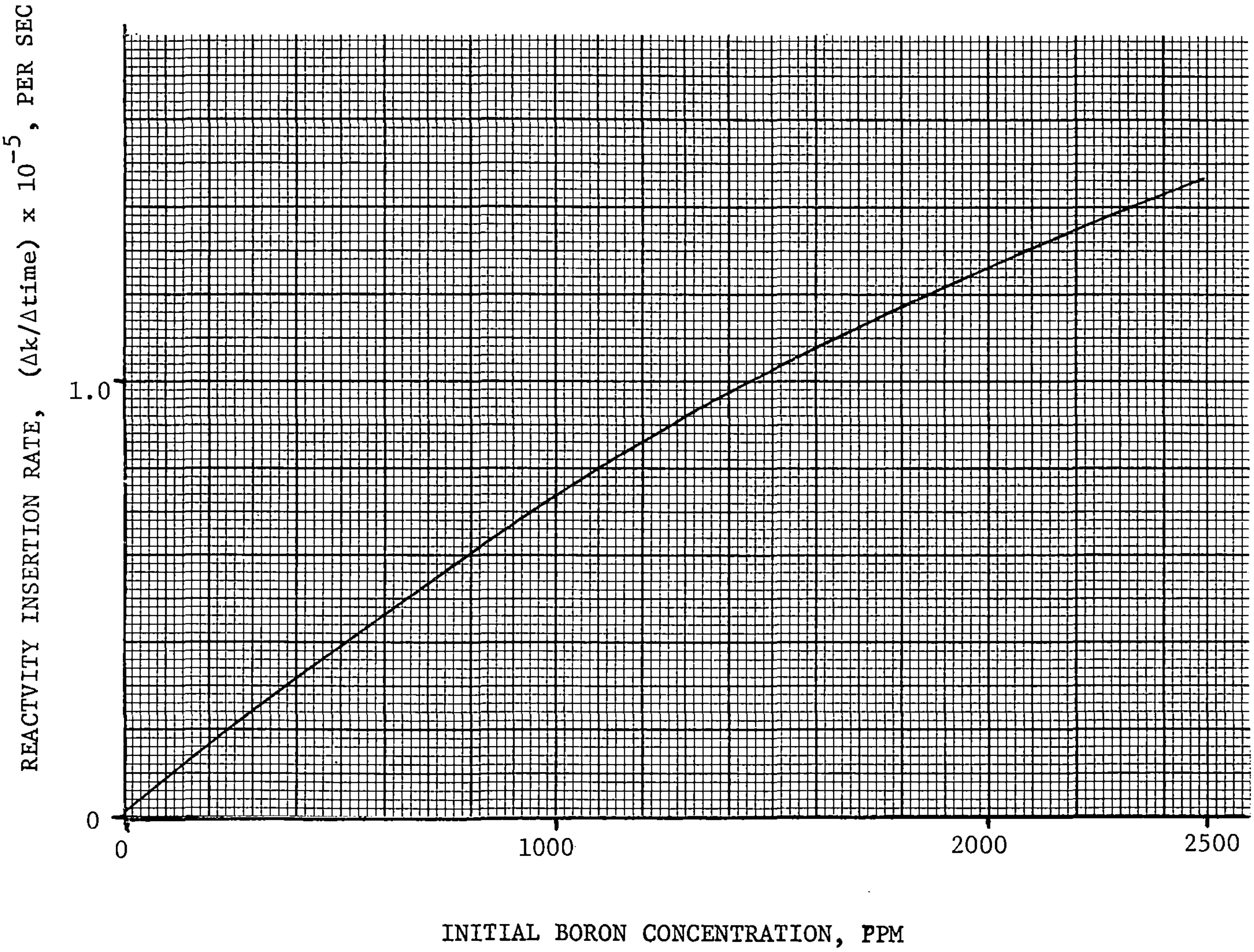
If dilution continues after reaching the low-low alarm, it takes approximately 16.3 minutes after the low-low alarm before the total shutdown margin (assuming one percent) is lost due to dilution. Therefore, adequate time is available following the alarms for the operator to determine the cause, isolate the primary grade water source, and initiate reboration.

If the reactor is in manual control and the operator takes no action, the power and temperature will rise to the overtemperature ΔT trip setpoint in 1.3 minutes as determined by a detailed digital simulation of the unit. The boron dilution accident in this case is essentially identical to a control rod assembly withdrawal accident. The reactivity insertion rate of 1.02×10^{-5} ΔK/sec from the boron dilution is well within the range of reactivity insertion rates considered in section 14.2.2, "Uncontrolled Control Rod Withdrawal at Power". Prior to the overtemperature ΔT trip, an overtemperature ΔT alarm and turbine runback would be actuated. There are approximately 15 minutes available after a reactor trip before the reactor can return to critical assuming a 1% shutdown margin at the beginning of the dilution.

Conclusions

Because of the procedures involved in the dilution process, an erroneous dilution is considered incredible. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and take corrective action before excessive shutdown margin is lost.

FIGURE 14.2.5-1
2-13-70



VARIATION IN REACTIVITY INSERTION RATE
WITH INITIAL BORON CONCENTRATION FOR
A BORON DILUTION RATE OF 165 GPM

14.2.6 STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

The unit can be operated with an inactive reactor coolant loop in either of two ways. The pump in the inactive loop can be shutdown with the loop stop valves in the normal fully open position. In this case, because of normal flow developed differential pressure across the reactor vessel, a small reverse flow is developed in the inactive loop. The unit can also be operated with one or both loop stop valves in the inactive loop closed. In this case there is no reverse flow in the inactive loop and the unit operates the same as a unit with only two loops. In the following sections the startup of an inactive reactor coolant loop is considered both for the case of the loop stop valves initially open and for the case of the loop stop valves initially closed.

14.2.6.1 Startup of an Inactive Reactor Coolant Loop-Loop Stop Valves Initially Open

If the unit is operated with the pump in one loop shut down and with the stop valves open there is a reverse flow through the loop. The cold leg temperature in the inactive loop is identical to the cold leg temperatures of the active loops and to the reactor core inlet temperature. If the reactor is operated at power with all main steam trip valves and nonreturn valves open, there is a temperature drop across the steam generator in the inactive loop, and the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Administrative procedures require that the unit be brought to a load of less than 25% of full power prior to starting a pump in an inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature. The starting of the idle reactor coolant pump without bringing the inactive loop hot leg temperature closer to the core inlet temperature would result in the injection of cold water into the core. The cold water causes a rapid reactivity and power increase.

Assumptions and Method of Analysis

The following assumptions are made:

1. Following the start of the idle pump, the inactive loop flow reverses and accelerates to its nominal full flow value instantaneously, i.e., there is no slip and the time to accelerate the pump and coolant is assumed to be zero.

2. A conservative negative moderator coefficient of -3.5×10^{-4} $\Delta k/°F$ is assumed.

3. A conservative low Doppler power coefficient of -0.7×10^{-4} $\Delta k/\%$ is assumed.

4. The reactor is assumed to be initially at 62% of 2441 MWt with the secondary side of all three steam generators at the same pressure and reverse reactor coolant flow through the idle loop steam generator. Sixty two percent power is the maximum steady-state power level allowed with two loops in operation and the loop stop valves in the inactive

loop open. The 62% includes 2% allowance for calibration and instrument errors. The high initial power assumed is conservative since it gives the greatest temperature difference between the core inlet temperature and the inactive loop hot leg temperature.

5. The initial Reactor Coolant System average temperature in the active loops is at the nominal full power value plus 4°F. This is a conservatively high value for the initial average temperature including instrument errors and results in the minimum margin to core DNB limits.

6. The initial Reactor Coolant System pressure is at the nominal value minus 30 psi. This is conservatively low value for the initial pressure including instrument errors and results in the minimum margin to core DNB limits.

7. The initial reactor coolant loop flows are at the values with one pump shut down and two pumps running. The core flow is 65% of the value at nominal full flow.

A detailed digital simulation of the unit including heat transfer to the steam generators of the active and inactive loops and Reactor Coolant System loop flow transit times was used to study the transient following the startup of an idle pump.

Results

The results following the startup of the idle pump with the above listed assumptions are shown in Figures 14.2.6-1 and 14.2.6-2. The nuclear power increases to the trip point (assumed to be 118%) in 0.5 second. The nuclear power increase causing the trip occurs as a result of the increase in core flow which causes a decrease in core average temperature before the core inlet temperature begins to decrease. The thermal flux response, of interest for DNB considerations, indicates that the peak average thermal heat flux reaches only 80 percent of the nominal value. The core average temperature, core inlet temperature, and Reactor Coolant System pressure during the transient are also shown. Since the peak heat flux remains below the nominal value, the core is not adversely affected by the transient. The minimum DNBR during the transient has been calculated to be 2.35 and occurs at 1.3 seconds.

The actual transient effects will be less severe than shown in Figures 14.2.6-1 and 14.2.6-2 because of alleviating factors which have not been taken into account. For example, the time constant of the pump is likely to be about 10 seconds which means that the change in core temperature occurs more gradually than shown in the figures. Furthermore, until the idle loop flow reaches at least 87% of its nominal value, the reactor is tripped at a power level considerably lower than the 118% used in the above analysis.

Conclusion

The transient results for the startup of an inactive reactor coolant loop with the loop stop valves initially open show that there is a considerable margin to a limiting DNBR of 1.3.

14.2.6.2 Startup of an Inactive Reactor Coolant Loop - Loop Stop Valves Initially Closed

If the stop valves in one loop are closed, the isolated section of the loop could cool down below the temperature of the active loops. Administrative procedures require that the unit be brought to zero load, the temperature of the isolated loop brought to within 10°F of the active loops, and the boron concentration of the isolated loop verified prior to opening the loop valves and returning the loop to service.

Interlocks are provided to ensure that an inadvertant startup of an isolated loop which has a lower temperature or lower boron concentration than the core and active loops is relatively slow. The interlocks ensure that flow from the isolated loop to the remainder of the Reactor Coolant System takes place through the relief line bypassing the cold leg stop valve for a period of one hour before the cold leg stop valve can be opened. The flow through the relief line is made low (no more than 400 gpm) so that the temperature and boron concentration in the isolated loop are brought to equilibrium with the remainder of the system at a relatively slow rate should the administrative procedures be violated and an attempt made to open stop valves when the isolated loop temperature or boron concentration is lower than in the core and active loops.

Interlocks are provided to:

1. Prevent opening of a hot leg loop stop valve unless the cold leg loop stop valve in the same loop is fully closed.

2. Prevent starting a reactor coolant pump unless:

   (a) The cold leg loop stop valve in the same loop is fully closed and the loop bypass valve is open, or

   (b) Both the hot leg loop stop valve and cold leg loop stop valve are fully open.

3. Prevent opening of a cold leg loop stop valve unless:

   (a) The hot leg loop stop valve in the same loop has been fully opened for 1 hour.

   (b) The bypass valve in the loop has been opened for 1 hour.

   (c) Flow has existed through the relief line for 1 hour.

   (d) The cold leg temperature is within 20°F of the highest cold leg temperature in other loops and the hot leg temperature is within 20°F of the highest hot leg temperature in the other loops.

The interlocks are a part of the Reactor Protection System and include the following redundancy:

1. Two independent limit switches to indicate that a valve is fully open.

2. Two independent limit switches to indicate that a valve is fully closed.

3. Two differential pressure switches in each line which bypasses a cold leg loop stop valve to determine that flow exists in the line. Flow through the line indicates:

   (a) The valves in the line are open.

   (b) The pump in the isolated loop is running.

The interlocks meet the IEEE 279 criteria and, therefore, cannot be negated by a single failure. The interlock on hot leg temperatures is a backup for the interlock on cold leg temperatures. Thus, the single failure criterion applies to the combination and not to each separately.

With the above protection system interlocks, the following procedure is necessary in order to re-open loop stop valves once either stop valve in a loop has left the fully open position:

a. The cold leg loop stop valve must be fully closed before the hot leg stop valve can be returned to its fully open position.

b. Flow must have existed from the isolated portion of the system to the remainder of the system (maximum rate is 400 gpm) for at least 1 hour through the line bypassing the cold leg stop valve and the isolated loop and active loop temperatures must agree to within 20°F before the cold leg loop stop valve can be opened.

Assumptions and Method of Analysis

The startup of an inactive reactor coolant loop with the loop stop valves initially closed has beed analyzed assuming the inactive loop to be at a boron concentration of 0 ppm while the active portion of the system is at 1500 ppm, a conservatively high value for beginning of life. The flow through the relief line is assumed at its maximum value of 400 gpm.

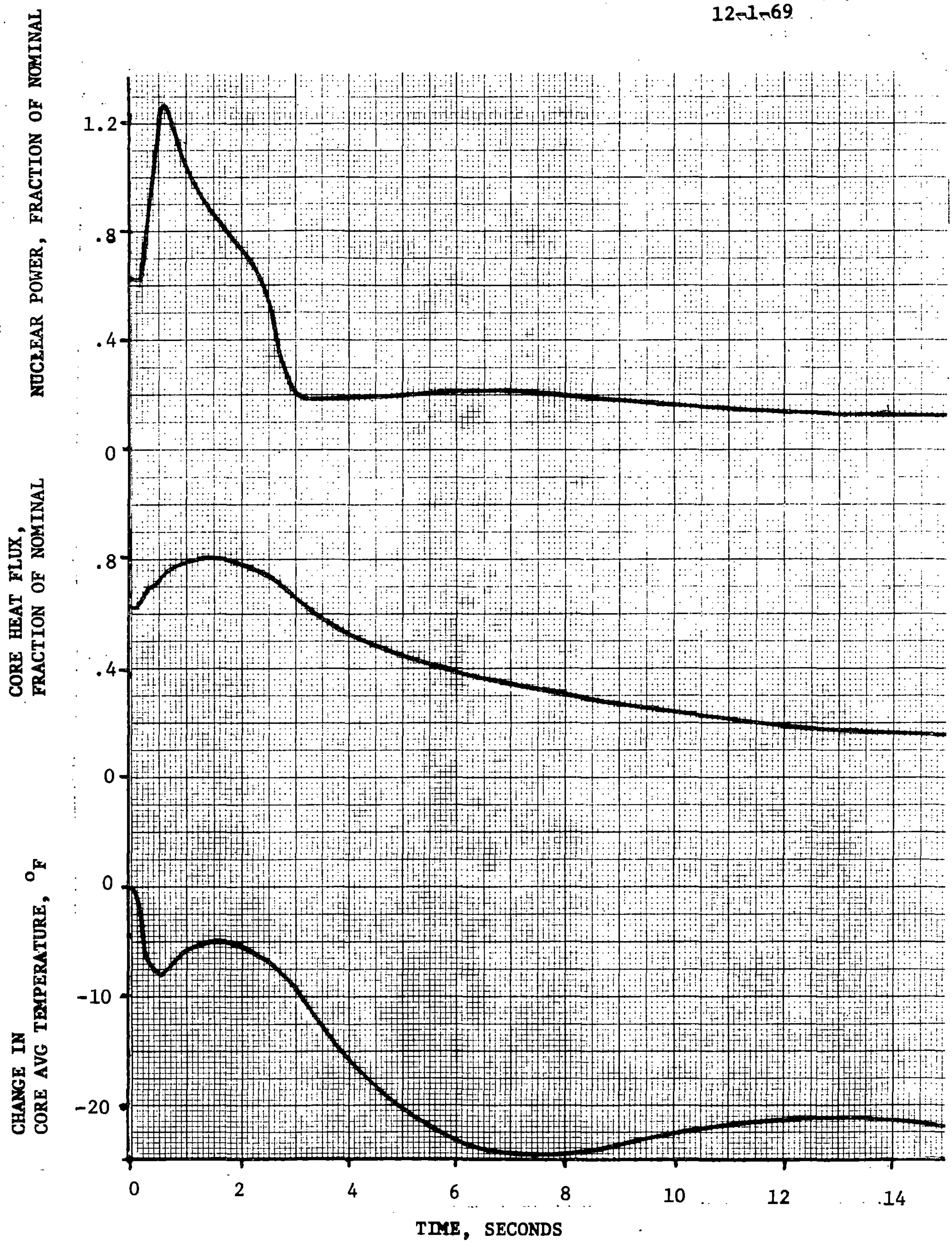
Results

Even with the assumption that administrative procedures are violated to the extent that an attempt is made to open the loop stop valves with 0 ppm in the inactive loop while the remaining portion of the system is at 1500 ppm, the dilution of the boron in the core is slow. The initial reactivity insertion rate is calculated to be 3.2×10^{-5} Δk/sec, considerably less than the reactivity insertion rates considered in sections 14.2.1 and 14.2.2. It takes 6 minutes after the beginning of dilution before the total shutdown margin is lost assuming one percent shutdown margin at the beginning of the accident. This is ample time for the operator to recognize a high count rate signal and terminate the dilution by turning off the pump in the inactive loop or by borating to counteract the dilution.

The reactivity addition at end of life due to an attempt to open stop valves when the inactive loop temperature is less than the core temperature is smaller than the reactivity addition considered in the above beginning of life case.

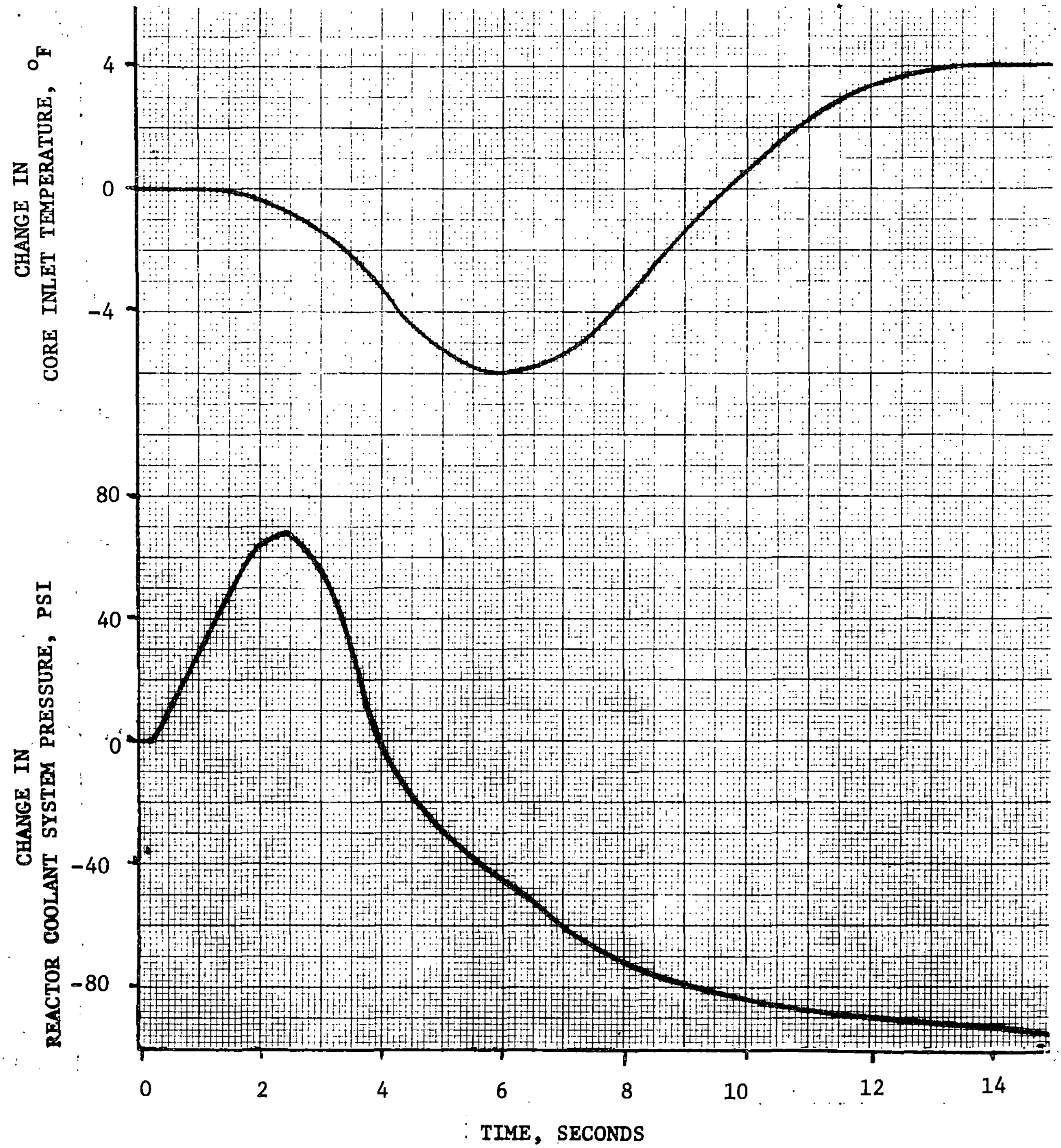
Conclusions

The transient results for the startup of an inactive reactor coolant loop with the loop stop valves initially open show that there is considerable margin to a limiting DNBR of 1.3. With the loop stop valves initially closed, the redundant interlocks provided in the Reactor Protection System ensure that the temperature and boron concentration in an isolated loop are brought to equilibrium with the remainder of the system at a slow rate. Should administrative procedures be violated and an attempt made to open stop valves when the isolated loop temperature or boron concentration is lower than that in the core, the reactivity addition rate is slow enough to allow the operator to take corrective action before excessive shutdown margin is lost.



STARTUP OF AN INACTIVE REACTOR COOLANT LOOP, LOOP STOP VALVES INITIALLY OPEN

FIGURE 14.2.6-2
12-1-69



STARTUP OF AN INACTIVE REACTOR COOLANT LOOP, STOP VALVES INITIALLY OPEN

14.2.7 EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS

Reductions in feedwater temperature or additions of excessive feedwater are means of increasing core power above full power. Such transients are attenuated by the thermal capacity in the secondary plant and in the Reactor Coolant System. The overpower - overtemperature protection (nuclear over-power and ΔT trips) prevents any power increase which could lead to a DNBR less than 1.30.

An extreme example of excess heat removal from the primary system is the transient associated with the accidental opening of the feedwater bypass valve which diverts flow around the low pressure feedwater heaters. The function of this valve is to maintain net positive suction head on the main feedwater pump in the event that the heater drain pump flow is lost, e.g., during a large load decrease.

In the event of an accidental opening, there is a sudden reduction in inlet feedwater temperature to the steam generators. This increased sub-cooling will create a greater load demand on the primary system.

Method of Analysis

Two cases have been analyzed to demonstrate the unit behavior in the event of a sudden feedwater temperature reduction resulting from accidental opening of the feedwater bypass valve. The first case was for an uncontrolled reactor with a zero moderator coefficient, since this represents a condition where the unit has the least inherent transient capability. The second case was for a controlled reactor with a large negative moderator coefficient ($3.5 \times 10^{-4}\ \Delta k/°F$).

Initial pressurizer pressure (2220 psia), reactor coolant average temperature (578.4°F), and power (102% of full power = 2490 MWt are assumed at extreme values consistent with steady-state, full power operation to allow for calibration and instrument errors. This results in the minimum margin to core DNB at the start of the transient. The analyses were performed using a detailed digital simulation of the unit including core kinetics, Reactor Coolant System, and the Steam and Feedwater System behavior.

Results

Figures 14.2.7-1 and 14.2.7.-2 show the transient without automatic reactor control and with a zero moderator coefficient of reactivity. As expected, the average reactor coolant temperature and pressurizer pressure show a fairly rapid decreases as the secondary heat extraction remains greater than the core power generation. The core power level increases slowly and eventually comes to equilibrium at a value of 103.9% nominal. There is an increased margin to DNB because of the accompanying reduction in average temperature. The reactor would not trip. There is a small increase in ΔT as the heat transfer increases through the steam generator.

Figures 14.2.7-3 and 14.2.7-4 illustrate the transient with automatic reactor control. A large negative moderator coefficient ($-3.5 \times 10^{-4}\ \Delta k/°F$) is assumed, which acts to increase power. The core power increase from the negative moderator coefficient retards the decrease in coolant average temperature and pressurizer pressure. Steady-state conditions are reached with a minimum DNBR of 1.68.

An evaluation of the accidental full opening of a feedwater control valve at full power has shown that the consequences of this incident are no more severe than those resulting from the opening of the feedwater bypass valve.

The reactivity insertion rate at no load following an excessive feedwater accident has also been calculated with the following assumptions:

1. A step increase in feedwater flow to one steam generator from 0 to the nominal full load flow.

2. The most negative reactivity moderator coefficient at end of life. The value used in the calculation was that corresponding to Figure 14.3.5-1 which is for a rodded core. The value when just critical at no load will be less negative.

3. A constant feedwater temperature of 70°F.

4. Neglect of the heat capacity of the Reactor Coolant System and steam generator shell thick metal.

5. Neglect of the energy stored in the fluid of the unaffected steam generators.

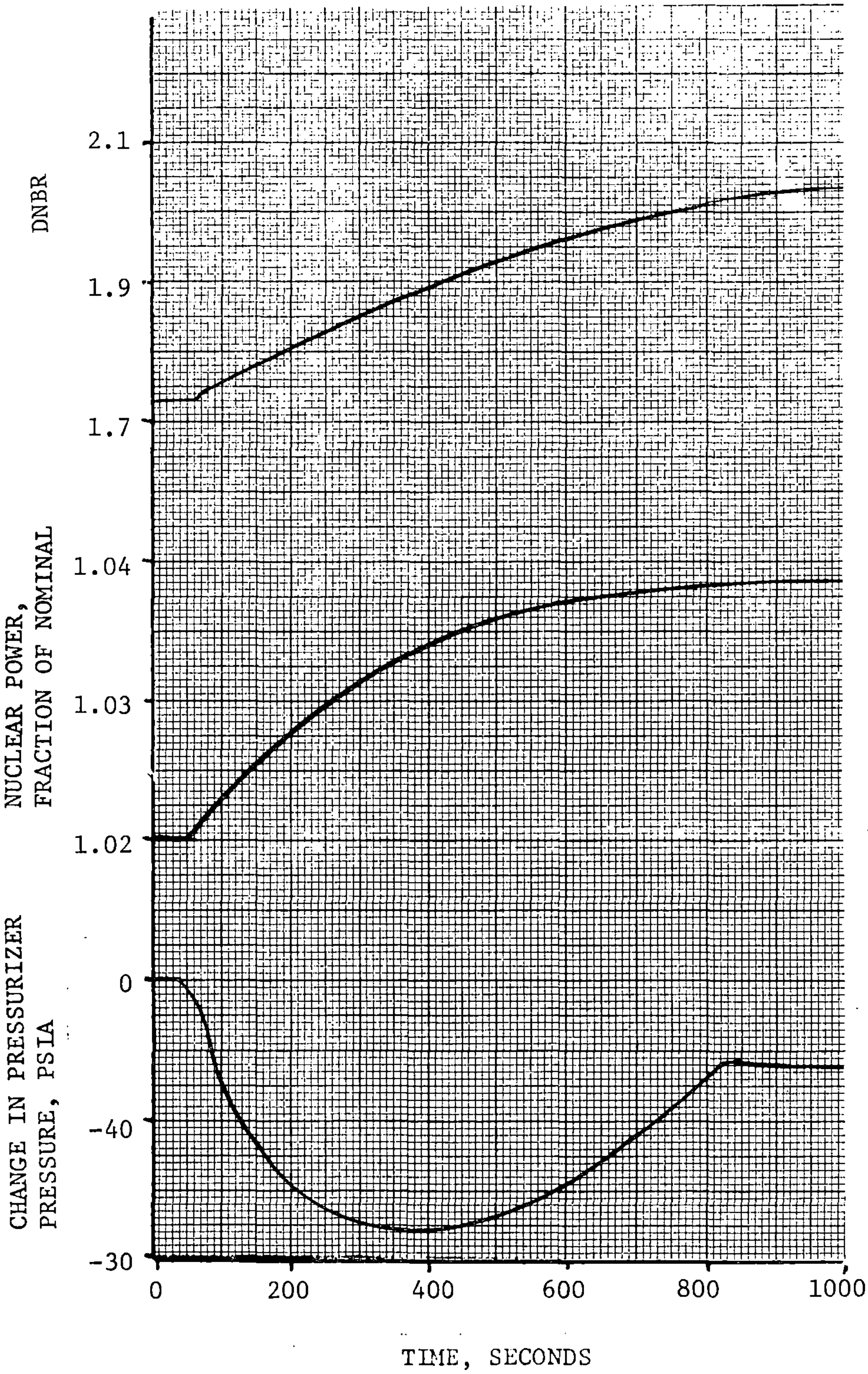
The maximum reactivity insertion rate was calculated to be 3.9×10^{-4} Δk/sec which is less than the maximum reactivity insertion rate analyzed in Section 14.2.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical Condition. It should be noted that if the incident occurs with the unit just critical at no load, the reactor may be tripped by the power range flux level trip (low setting) set at approximately 25%. As shown in Section 14.2.1 there is a large margin to DNB with the above calculated reactivity insertion rate.

Continuuous addition of cold feedwater after a reactor trip is prevented since the reduction of Reactor Coolant System temperature, pressure, and pressurizer level leads to the actuation of safety injection on low pressurizer pressure and level. The safety injection signal trips the main feedwater pumps and closes the feedwater pump discharge valves as well as closing the main feedwater control valves.

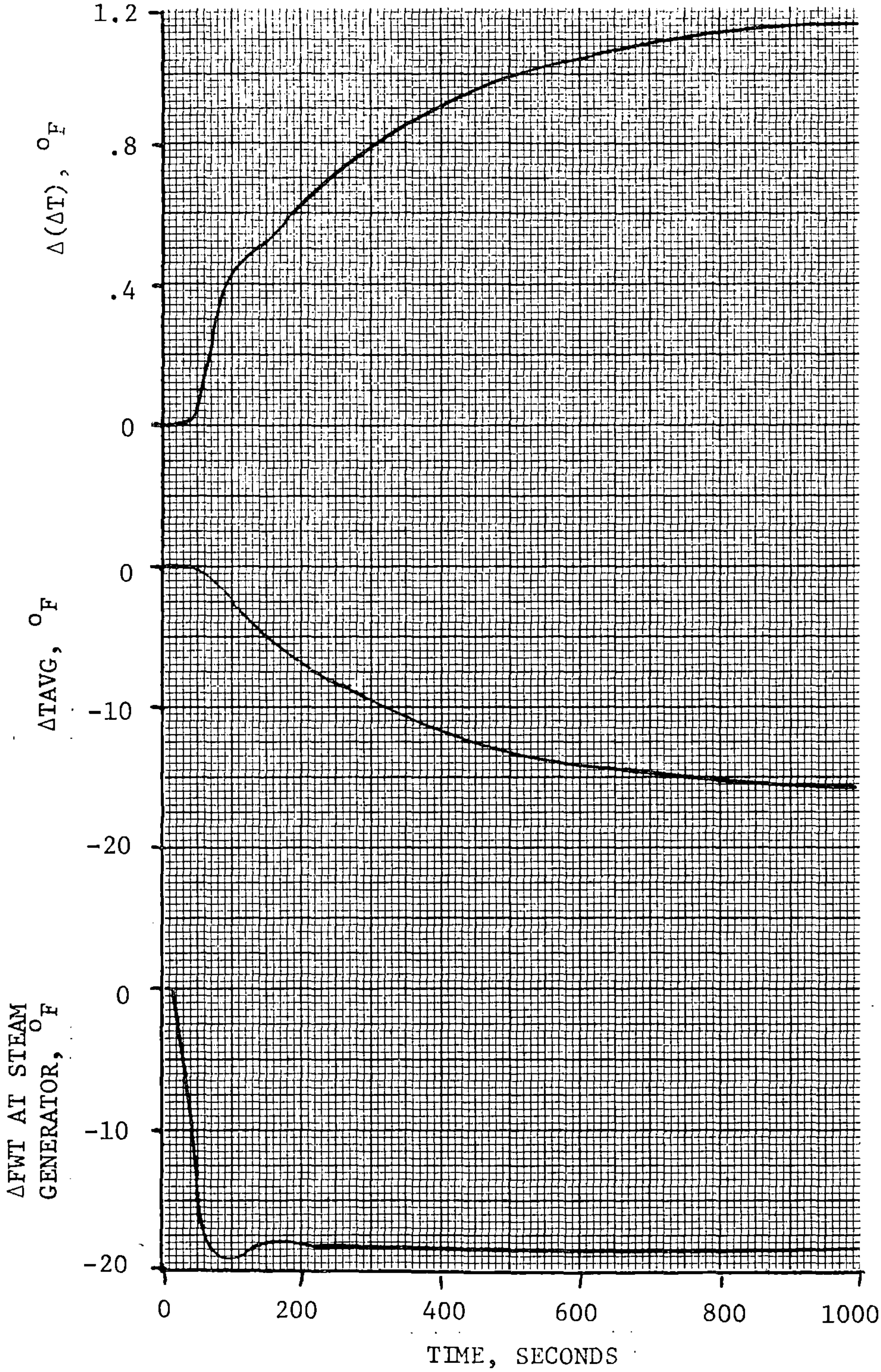
Conclusions

Representative transient results for excessive load increases due to cold feedwater addition have been shown which indicate that a core power increase is accompanied by an average temperature decrease. This has the effect of maintaining considerable margin to a limiting DNBR of 1.3. It has been shown that the maximum reactivity insertion rate which occurs at no load following excessive feedwater addition is less than the maximum value considered in the analysis of a control rod assembly withdrawal incident from a subcritical condition. It has further been shown that automatic action occurs to prevent continuous cold feedwater addition after a unit trip.

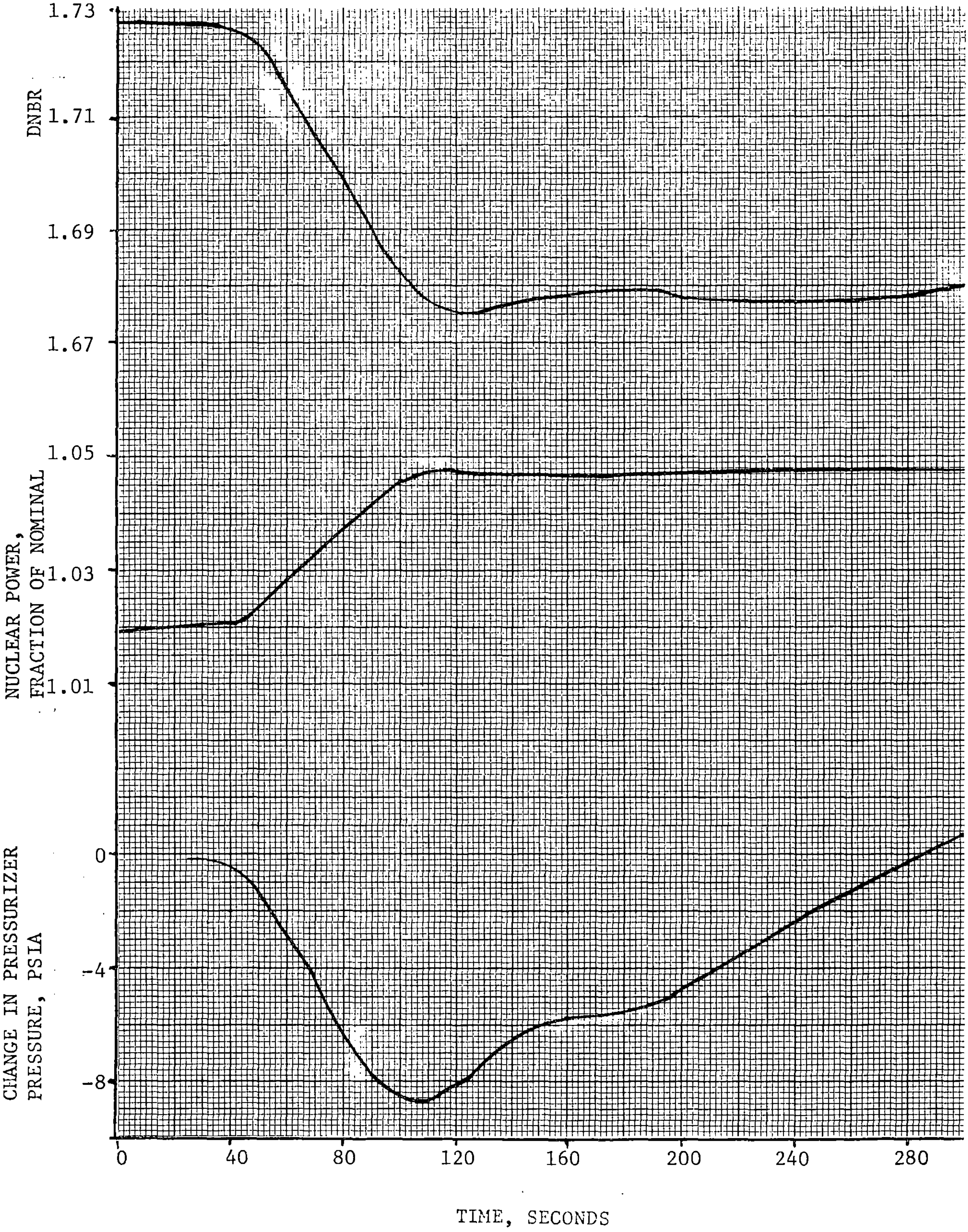
There is, thus, no radioactive release and no public hazard in the event of a reduction in feedwater temperature or an excessive feedwater incident.



FEEDWATER BYPASS OPENING BOL,
NO REACTOR CONTROL

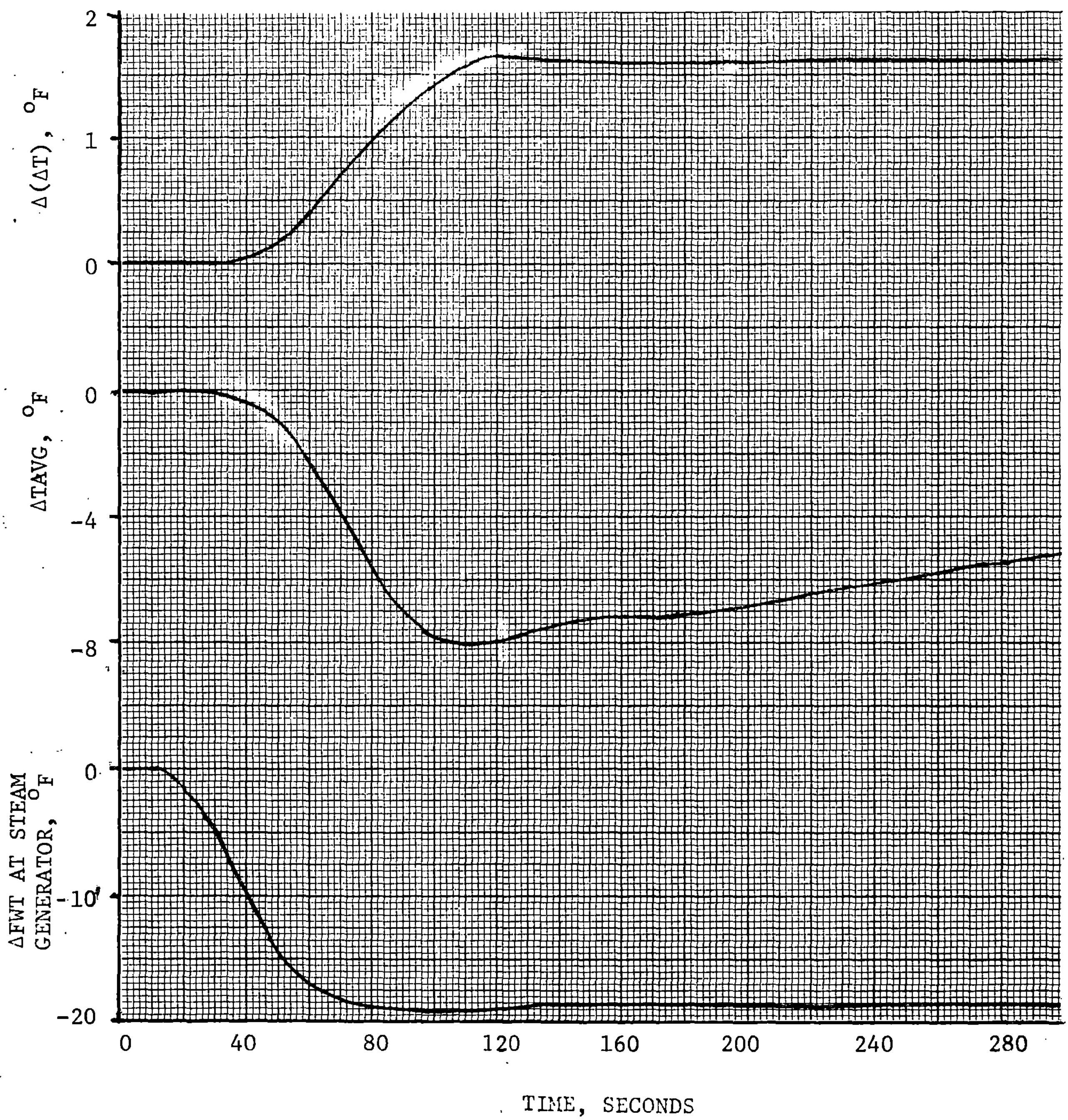


FEEDWATER BYPASS OPENING BOL,
NO REACTOR CONTROL



FEEDWATER BYPASS OPENING BOL,
WITH REACTOR CONTROL

FIGURE 14.2.7-4
12-1-69



FEEDWATER BYPASS OPENING EOL,
WITH REACTOR CONTROL

14.2.8 EXCESSIVE LOAD INCREASE INCIDENT

An excessive load increase incident is defined as a rapid increase in steam generator steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The Reactor Control System is designed to accommodate a 10 percent step load increase or a 5 percent per minute ramp load increase (without a reactor trip) in the range of 15 to 100 percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the Reactor Protection System. If the load increases exceeds the capability of the Reactor Control System, the transient is terminated in sufficient time to prevent the DNBR from being reduced below 1.30, since the core is protected by the combination of the nuclear overpower trip and the overpower - overtemperature trips as discussed in Section 7. An excessive load increase incident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam bypass control or turbine speed control.

For excessive loading by the operator or by system demand, the turbine load limiter keeps maximum turbine load from exceeding 100% rated load.

During power operation, steam bypass to the condenser is controlled by reactor coolant condition signals; i.e., high reactor coolant temperature indicates a need for steam bypass. A single controller malfunction does not cause steam bypass; an interlock is provided which blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

Method of Analysis

Four cases have been analyzed to demonstrate the unit behavior for a 10% step increase from rate load. The first two cases were for a manual controlled reactor at beginning of life (B.O.L., α_m = zero $\Delta k/°F$ and end of life (E.O.L., $\alpha_m = -3.5 \times 10^{-4}$ $\Delta k/°F$) conditions, α_m is the moderator reactivity coefficient. Beginning of life represents a condition where the unit has the smallest moderator temperature coefficient of reactivity and therefore the least inherent transient capability. The last two cases were analyzed for an automatic-control situation at B.O.L. and E.O.L. conditions. A conservative limit on the turbine valve opening was assumed corresponding to 1.2 times nominal steam flow at nominal steam pressure. Initial pressurizer pressure (2220 psia), reactor coolant average temperature (578.4°F), and power (102% of full power = 2490 MWt.) are assumed at extreme values consistent with steady-state, full power operation, to allow for calibration and instrument errors. This results in the minimum margin to core DNB at the start of the transient. The analyses were performed using a detailed digital simulation of the unit including core kinetics, Reactor Coolant System, and the Steam and Feedwater System behavior.

Results

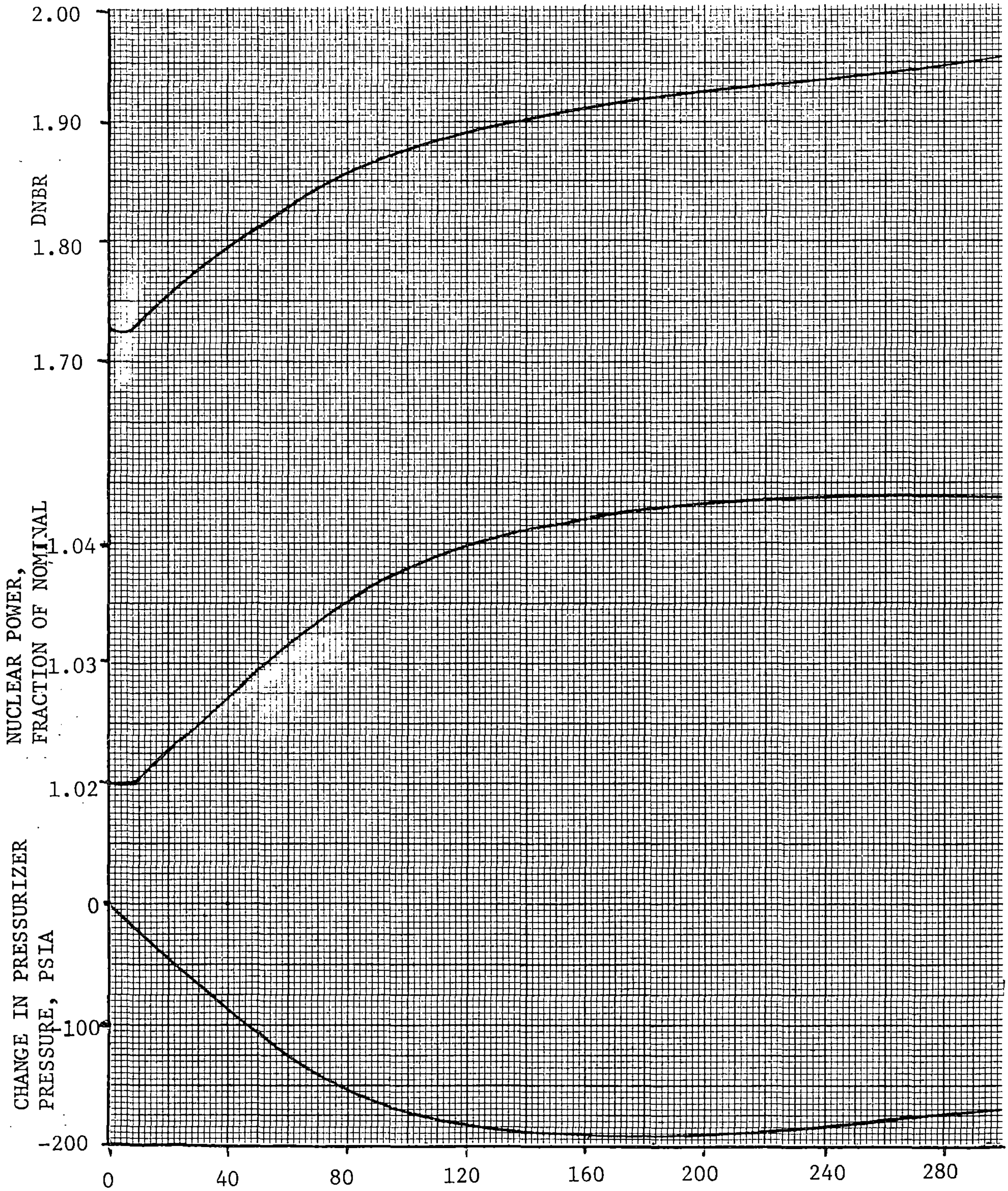
Figures 14.2.8-1 through 14.2.8-4 illustrate the transient with the reactor in the manual control mode. As expected, for the BOL case with a very slight power increase, the average temperature shows a large decrease. This results in a DNBR which increases above its initial value. For the EOL case, there is a much larger increases in reactor power due to the moderator feedback. A minimum DNBR of 1.55 is reached for the EOL, manually controlled case.

Figures 14.2.8-5 through 14.2.8-8 illustrate the transient assuming the reactor is in automatic control. Both the BOL and EOL cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For the BOL case, steady-state conditions are reached with a minimum DNBR of 1.53. The EOL transient shows a greater increase in power than the BOL transient due to the additional moderator feedback. A minimum DNBR of 1.53 is reached for the EOL case.

Conclusions

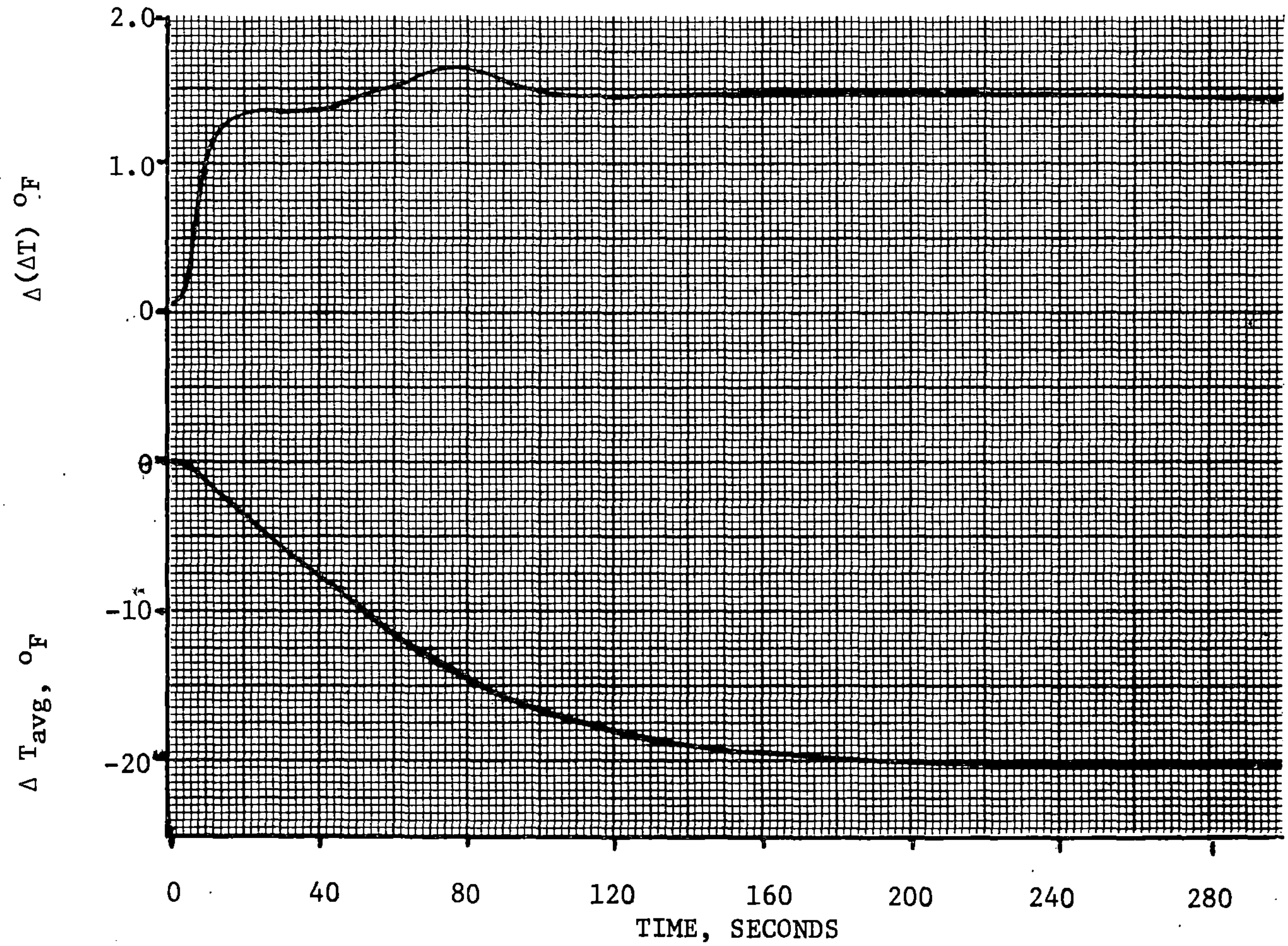
The four cases analyzed show a considerable margin to the limiting DNBR of 1.3. It is concluded that unit integrity is maintained throughout lifetime for the excessive load increase incident.

FIGURE 14.2.8-1
12-1-69



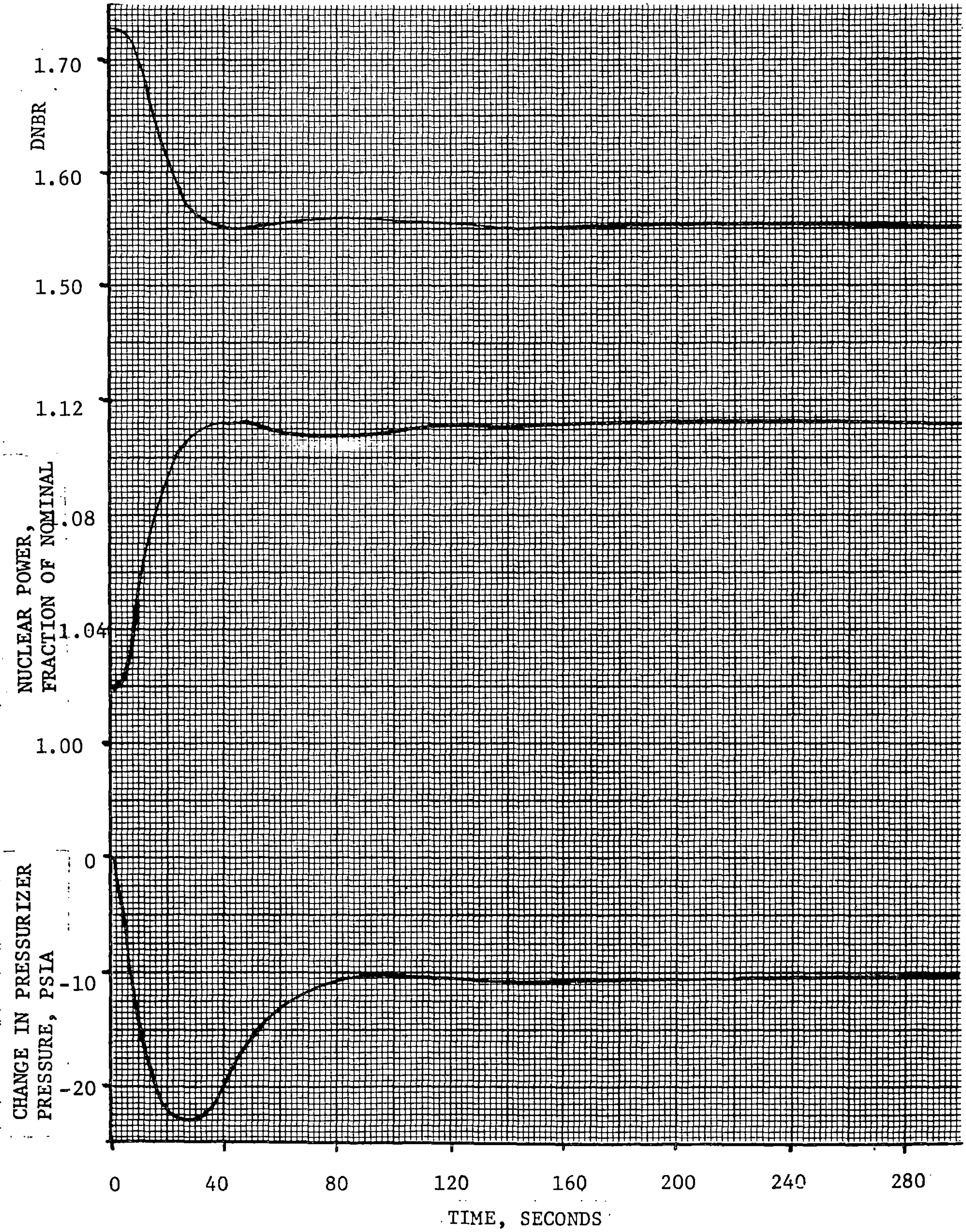
10% STEP LOAD INCREASE
BOL, NO CONTROL

FIGURE 14.2.8-2
12-1-69



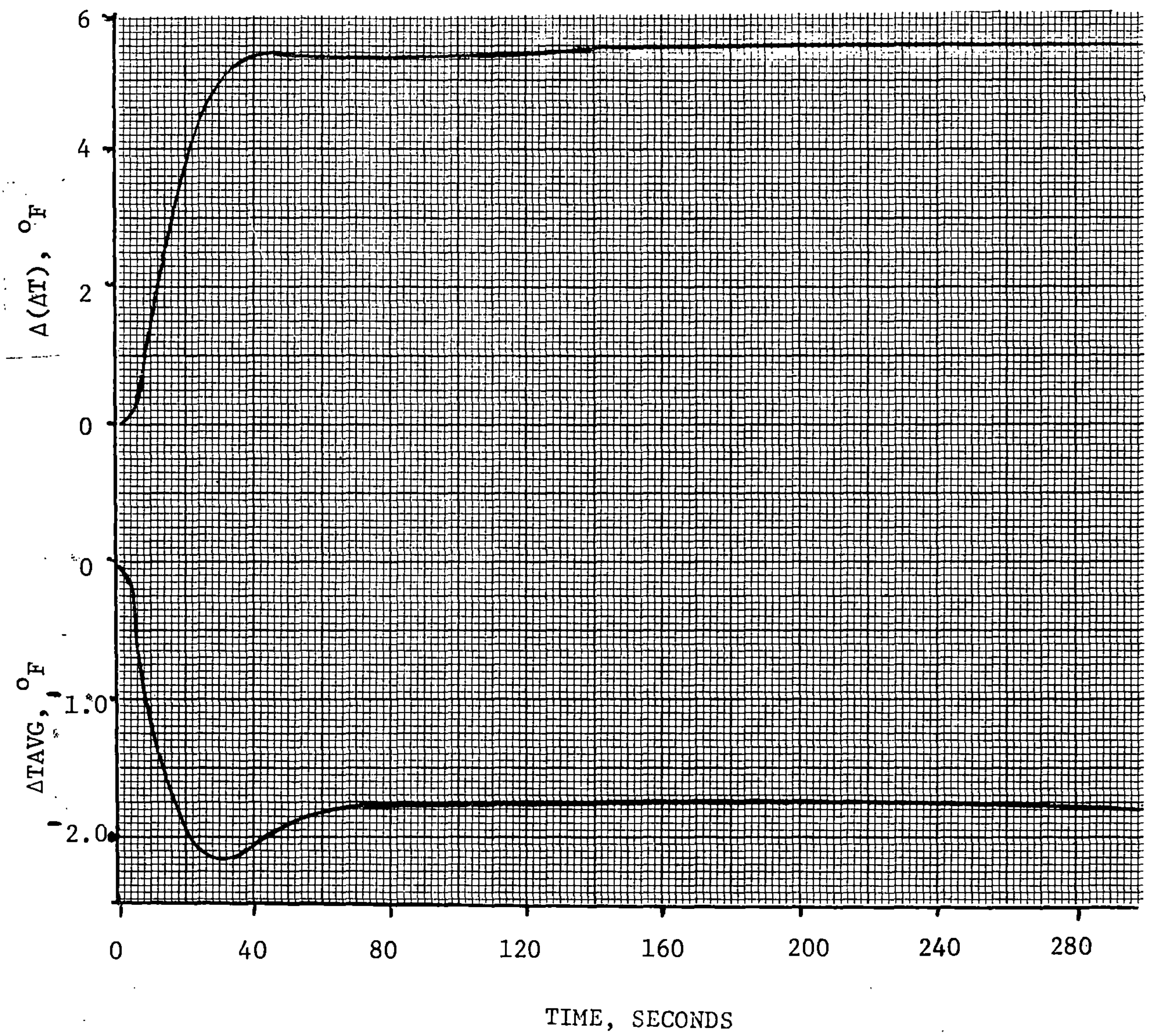
10% STEP LOAD
INCREASE - BOL, NO CONTROL

FIGURE 14.2.8-3
12-1-69



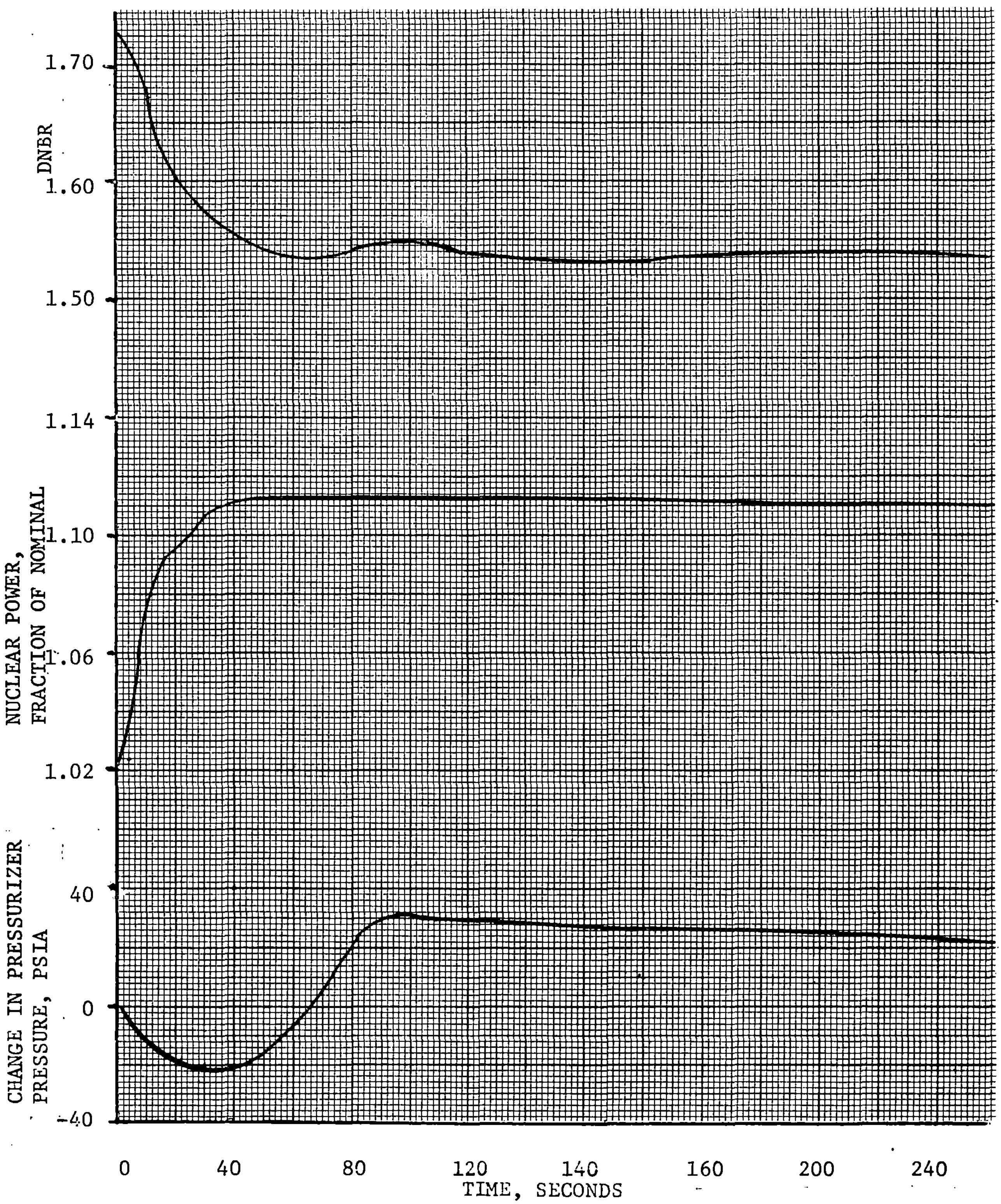
10% STEP LOAD INCREASE
EOL, NO CONTROL

FIGURE 14.2.8-4
12-1-69



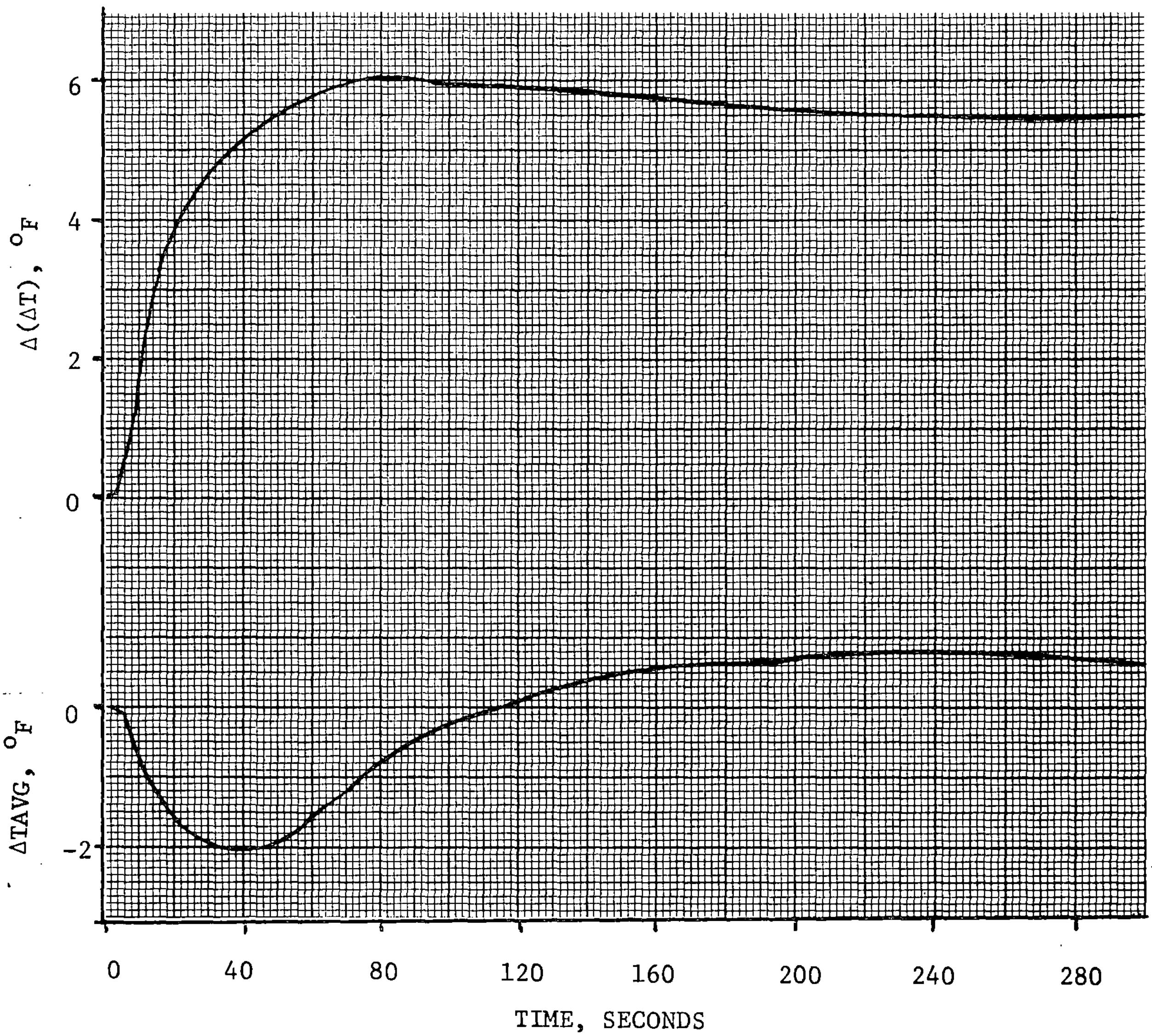
10% STEP LOAD INCREASE
EOL, NO CONTROL

FIGURE 14.2.8-5
12-1-69



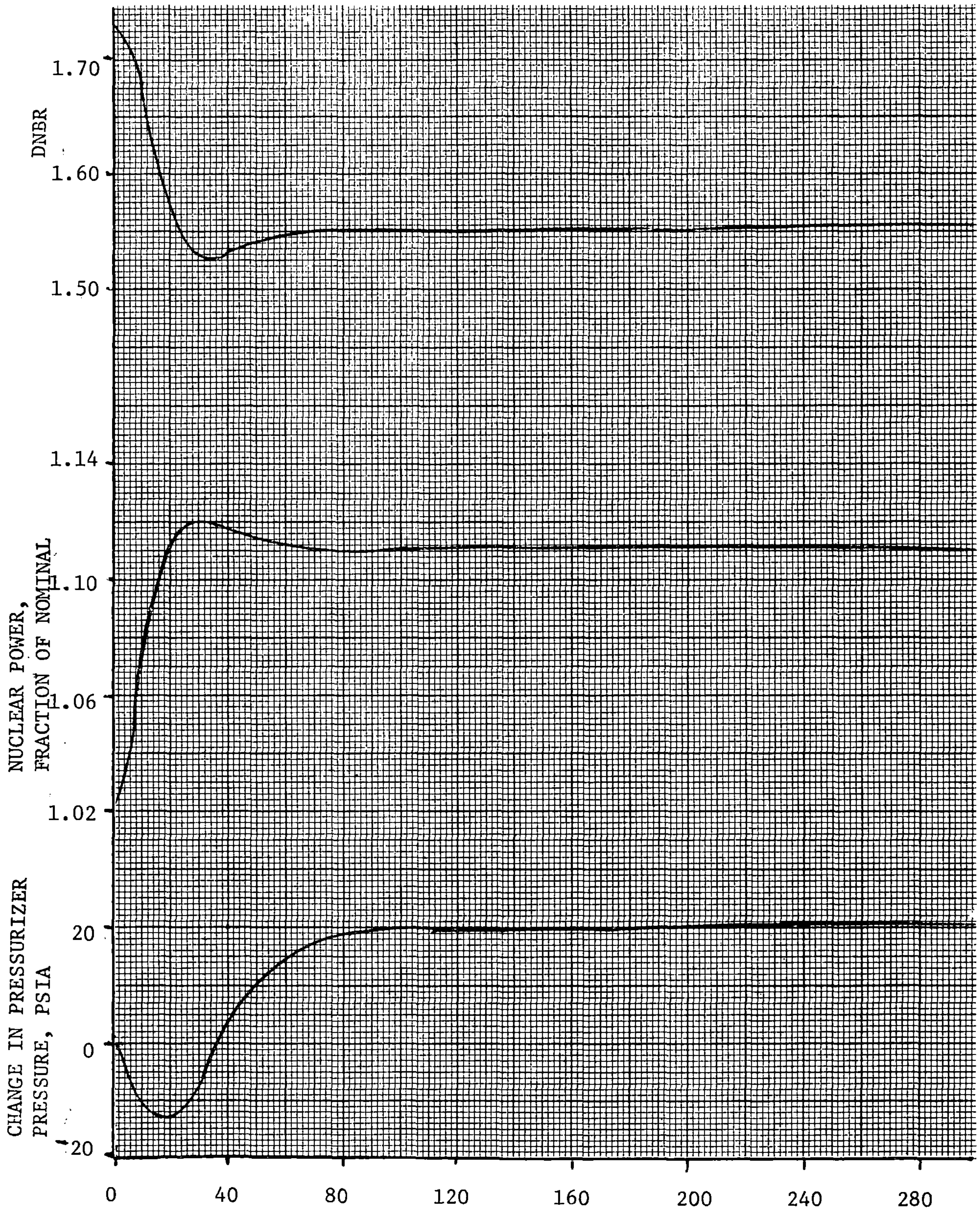
10% STEP LOAD INCREASE -
BOL, WITH CONTROL

FIGURE 14.2.8-6
12-1-69



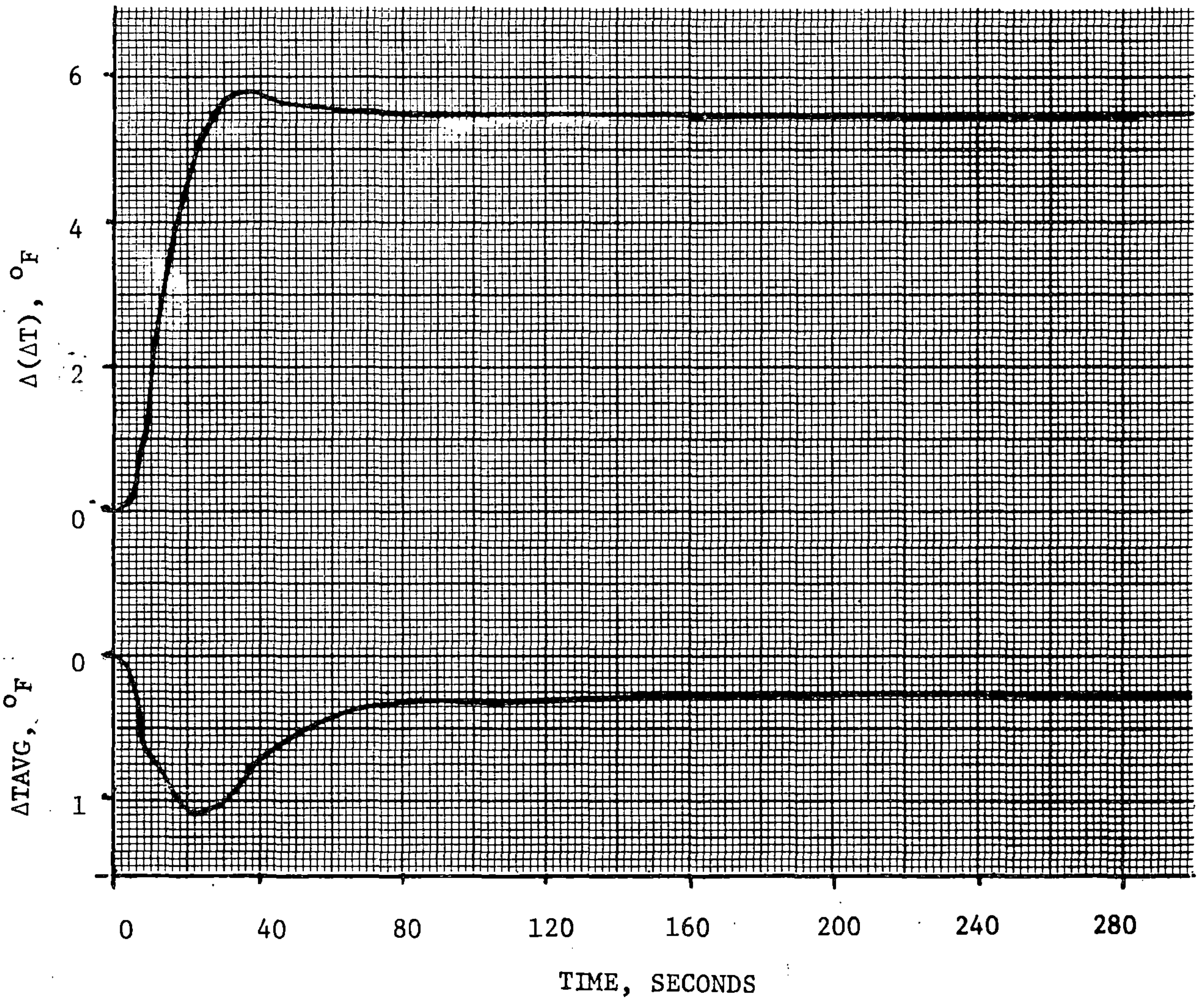
10% STEP LOAD,
INCREASE - BOL, WITH CONTROL

FIGURE 14.2.8-7
12-1-69



10% STEP LOAD INCREASE - EOL, WITH CONTROL

FIGURE 14.2.8-8
12-1-69



10% STEP LOAD,
INCREASE - EOL, WITH CONTROL

14.2.9 LOSS OF REACTOR COOLANT FLOW

Flow Coast-Down Incidents

A loss of coolant flow incident can result from a mechanical or electrical failure in a reactor coolant pump, or from an interruption in the power supply to these pumps. If the reactor is at power at the time of the incident, the immediate effect of loss of coolant flow is a rapid increase in coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly. The following trip circuits provide the necessary protection against a loss of coolant flow incident:

1) Low voltage or low frequency on pump power supply buses
2) Pump circuit breaker opening
3) Low reactor coolant flow

These trip circuits and their redundancy are further described in Section 7.2, Reactor Control and Protection System.

Simultaneous loss of electrical power to all reactor coolant pumps at full power is 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 Reactor Coolant System overpressure and the DNBR from being reduced below 1.30.

Method of Analysis

The following loss of flow cases are analyzed:

1) Loss of three pumps from a nominal Reactor Coolant System heat output of 100% (2441 MWt) with three loops operating.

2) Loss of one pump from a nominal Reactor Coolant System heat output of 100% (2441 MWt) with three loops operating.

3) Loss of two pumps from a nominal Reactor Coolant System heat output of 60% (1464.6 MWt) with two loops operating; no loop stop valves closed.

4) Loss of one pump from a nominal Reactor Coolant System heat output of 60% (1464.6 MWt) with two loops operating; no loop stop valves closed.

5) Loss of two pumps from a nominal Reactor Coolant System heat output of 67% (1635.5 MWt) with two loops operating; loop stop valves closed in one loop.

6) Loss of one pump from a nominal Reactor Coolant System heat output of 67% (1635.5 MWt) with two loops operating; loop stop valves closed in one loop.

The normal power supplies for the pumps are three buses supplied by the generator. Each bus supplies power to one pump. When a generator trip occurs, the pumps are automatically transferred to a bus supplied from external power lines, and the pumps continue to supply coolant flow to the core. The simultaneous loss of power to all reactor coolant pumps is a highly unlikely event. Following any turbine trip, where there are no electrical faults which require tripping the generator from the pump supply network, the generator remains connected to the network for at least one minute. The reactor coolant pumps remain connected to the generator thus ensuring full flow for one minute after the reactor trip before any transfer is made. Since each pump is on a separate bus, a single bus fault would not result in the loss of more than one pump.

A full unit simulation is used in the analysis to compute the core average and hot spot heat flux transient responses, including flow coastdown, temperature, reactivity, and control rod assembly insertion effects.

These data are then used in a detailed thermal-hydraulic computation to compute the margin to DNB. This computation solves the continuity, momentum, and energy equations of fluid flow together with the W-3 DNB correlation discussed in Section 3.2.2. The following assumptions are made in the calculations:

Initial Operating Conditions

The initial operating conditions which are assumed are those most adverse with respect to the margin to DNB, i.e., maximum steady state power level, minimum steady state pressure, and maximum steady state inlet temperature:

Nominal 100% Power (2441 MWt) - 3 loops operating:

Power	(1.00 + .02) (2441 MWt) = 2490 MWt
Pressure	2250 - 30 = 2220 psia
Inlet Temperature	543 + 4 = 547°F

Nominal 60% Power (1454.6 MWt) - 2 loops operating; no loop stop valves closed:

Power	(0.60 + 0.02) (2441 MWt) = 1513.4 MWt
Pressure	2250 - 30 = 2220 psia
Inlet Temperature	553.4 + 4 = 557.4°F

Nominal 67% Power (1635.4 MWt) - 2 loops operating; loop stop valves closed in one loop:

Power	(0.67 + 0.02) (2441 MWt) = 1684.3 MWt
Pressure	2250 - 30 = 2220 psia
Inlet Temperature	549.3 + 4 = 553.3°F

The above assumed initial operating conditions include allowance for calibration and instrument errors. A maximum power level of 62% (including errors) is assumed for two loop operation with the loop stop valves open and a maximum power level of 69% (including errors) is assumed for two loop operation with the stop valves in one loop closed. For two loop operation, the corresponding steady state core flows are 65% (of the value with all loops in service) with the loop stop valves open and 74% (of the value with all loops in service) with the loop stop valves closed. The maximum power levels assumed for two loop operation, therefore, result in power to flow ratios which are less than the power to flow ratio at nominal full power conditions.

Reactivity Coefficients

The highest absolute magnitude of the Doppler (-1.6×10^{-5} Δk F^{-1}) and the lowest absolute magnitude of the moderator zero Δk F^{-1}) temperature coefficients are assumed since these result in the maximum hot spot heat flux during the initial part of the transient, when the minimum DNB ratio is reached.

Reactor Trip

Following the loss of all pumps at power a reactor trip is actuated by either low voltage or low frequency since the incident is due to the simultaneous loss of power to all pump busses. Both the low voltage and low frequency trip circuit meet the IEEE 279 criterion and therefore cannot be negated by a single failure. The time from the loss of power to all pumps to the initiation of control rod assembly motion to shut down the reactor is taken as 1.2 seconds. This is a conservative assessment of the delay.

A low flow trip is actuated following loss of one pump. The low flow trip setting is 90 percent of full loop flow; the trip signal is assumed to be initiated at 87 percent of full loop flow allowing 3 percent for flow instrumentation errors. The time from the initiation of the low flow signal to initiation of control rod assembly motion is 0.6 seconds. Upon reactor trip it is also assumed that the most reactive control rod assembly is stuck in its fully withdrawn position, hence resulting in a minimum insertion of negative reactivity. The negative reactivity insertion upon trip is conservatively based on a 1 percent shutdown margin at no load conditions.

Flow Coastdown

Reactor coolant flow coastdown curves are shown on Figures 14.2.9-1, 14.2.9-4, 14.2.9-7, 14.2.9-10, 14.2.9-13 and 14.2.9-16. These curves are based on high estimates of loop pressure losses.

Results

Loss from 100% power (nominal) with three loops operating (Figure 14.2.9-2) shows the neutron flux, the average heat flux, and the hot channel heat flux response for the loss of three pumps.

Figure 14.2.9-3 shows the DNB ratio as a function of time for this case. A minimum W-3 DNB ratio value of 1.46 is reached 2.35 seconds after initiation of the incident.

Figure 14.2.9-5 shows the transient for loss of one pump and Figure 14.2.9-6 shows the DNBR as a function of time for this case. The minimum value of the DNBR is 1.62 and occurs 3 seconds after initiation of the transient.

Loss from 60% power (nominal) with two loops operating, no loop stop valves closed:

Figure 14.2.9-8 shows the transient for loss of two pumps. Figure 14.2.9-9 shows the DNBR as a function of time for this case. The minimum value of the DNBR is 2.73 and occurs 2.4 seconds after initiation of the transient.

Figure 14.2.9-11 shows the transient for loss of one pump. Figure 14.2.9-12 shows the DNBR as a function of time for this case. The minimum value of the DNBR is 2.78 and occurs 3.4 seconds after initiation of the transient.

Loss from 67% (nominal) with two loops operating, loop stop valves closed in one loop:

Figure 14.2.9-14 shows the transient for loss of two pumps. Figure 14.2.9-15 shows the DNBR as a function of time for this case. The minimum value of the DNBR is 2.33 and occurs 2.1 seconds after initiation of the transient.

Figure 14.2.9-17 shows the transient for loss of one pump. Figure 14.2.9-18 shows the DNBR as a function of time for this case. The minimum value of the DNBR is 2.32 and occurs 3.6 seconds after initiation of the transient.

Conclusions

Since DNB does not occur in any loss of coolant flow incident, there is no cladding damage and no release of fission products into the reactor coolant. Therefore, once the fault is corrected, the unit can be returned to service in the normal manner. The absence of fuel failures would, of course, be verified by analysis of reactor coolant samples.

Locked Rotor Incident

A transient analysis is performed for the hypothetical instantaneous seizure of a reactor coolant pump rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon unit trip.) The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the steam generator causes an insurge into the pressurizer and a pressure increase throughout the Reactor Coolant System. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect as well as the pressure-reducing effect of the spray is not included in this analysis.

Method of Analysis

The following cases are analyzed:

1 Locked Rotor Incident from a nominal Reactor Coolant System heat output of 100% (2441 MWt) with three loops operating.

2 Locked Rotor Incident from a nominal Reactor Coolant System heat output of 60% (1464.6 MWt) with two loops operating, no loop stop valves closed.

3 Locked Rotor Incident from a nominal Reactor Coolant System heat output of 67% (1635.5 MWt), loop stop valves in one loop closed.

Initial Conditions

At the beginning of the postulated locked rotor incident, i.e., at the time the shaft in one of the reactor coolant pumps is assumed to seize, the unit is assumed to be in operation under the most severe steady state operating conditions. The unit is assumed to be operating at 102% of nominal full power (2441 MWt) with three pumps operating. With 2 pumps operating, the unit is assumed to be operating at 62% of nominal full power with no loop stop valves closed and at 69% of nominal full power with the loop stop valves in one loop closed. These operating conditions are based on the maximum expected calorimetric error of 2%.

Inlet temperature is assumed to be 4°F above its programmed value to allow for 2°F deadband on control rod assembly motion and a maximum temperature error of 2°F. Nominal inlet temperature with three pumps operating is 543°F. With two loops operating and no loop stop valves closed this temperature is assumed to be 553.4°F and to be 549.3°F with stop valves in one loop closed. When the peak pressure is evaluated the initial reactor coolant pressure is conservatively estimated as 30 psi above nominal pressure (2250 psia) to allow for errors in the pressurizer measurement and control channels in order to get the highest possible rise in coolant pressure during the transient. When evaluating the DNBR, the 30 psi permissible error was conservatively considered to be below nominal pressure in order to give low initial DNBR.

Evaluation of the Pressure Transient

A digitial code was used to determine the peak pressure in the Reactor Coolant System under the postulated incident conditions and to obtain the nuclear power as a function of time which is used later on in the analysis.

After pump seizure, nuclear power is rapidly reduced because of the control rod assembly insertion upon unit trip. In this analysis, the time from pump seizure to initiation of control rod assembly motion was taken as 0.9 seconds. Shutdown reactivity is conservatively based on a 1 percent shutdown margin at no load conditions.

No credit was taken for the pressure-reducing effect of the pressurizer relief valves, pressurizer spray, steam dump, or controlled feedwater flow after unit trip. Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves start operating at 2500 psia and their capacity for steam relief is 30.41 ft^3/sec.

Evaluation of DNB in the Core During the Incident

Heat flux transients following the pump seizure were evaluated by a detailed digital model with the input of the nuclear power, the pressure and the coolant conditions previously calculated as functions of time. The model is similar to the model incorporated in the LOCTA code but features a larger number of lumps in the fuel. This study used 6 lumps for the fuel and one for the clad.

Calculations of the extent of DNB in the core during the incidents were performed using a multichannel THINC-III model with the heat flux, the coolant flow decay and the coolant conditions input as a function of time. Six concentric channels were used for this study.

In order to estimate the severity of the incident in the core as far as the integrity of the fuel rods is concerned, the thermal behavior of the fuel located at the hot spot after DNB was investigated using the detailed digital model mentioned above with a film boiling heat transfer calculation. Results obtained from an analysis of this "hot spot" condition represent the upper limit with respect to clad temperature, approach to clad melting and zirconium-steam reaction. The steady state conditions at the hot spot in the core just prior to the incident are shown below:

	3 pumps operating	2 pumps operating No loop stop valves closed	2 pumps operating Loop stop valves in one loop closed
Heat flux, Btu/hr-ft^2	544,780	331,140	368,530
Avg. pellet temp, °F	2261	1776	1880
Avg. clad temp., °F	712	690	699
System pressure, psia	2220	2220	2220
Coolant mass flow rate, lbs/hr-ft^2	2.29×10^6	1.49×10^6	1.69×10^6

Film Boiling Coefficient

The film boiling coefficient is calculated in the digital program by an empirical equation. This equation is described in the WCAP-7247 Report (Post DNB Heat Transfer During Blow-down).

The steam properties are evaluated at film temperature (avg. between wall and bulk temperatures). The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at this time.

The system pressure, bulk density and mass flow rate are an input to the program as a function of time.

For this analysis, the initial values of the pressure and the bulk density were used throughout the transient, since they were the most conservative.

For conservatism, DNB was assumed to start at the beginning of the incident and the heat transfer coefficient between clad and water was reduced suddenly from its steady-state value to 0.7 times the film boiling value at time = 0, without any period of transition boiling. The safety factor of 0.7 was assumed for conservatism in evaluating the film boiling coefficient.

Gap Coefficient

The magnitude and time dependence of the gap heat transfer coefficient between fuel and cladding has a pronounced influence on the thermal results. The larger the value of this coefficient, the more heat is transferred between pellet and clad. For the first part of the transient, a high gap coefficient produces higher clad temperatures since the heat stored and generated in the fuel pellet tries to redistribute itself in the cooler clad. This effect of the gap coefficient, however, is reversed when the clad temperature exceeds the pellet temperature in cases where the zirconium-steam reaction is present.

The effect of the gap coefficient upon the maximum clad temperature during the transient was investigated. Several cases with different gap coefficients were considered. The results are depicted in Figures 14.2.9-21, 14.2.9-27 and 14.2.9-33, and they show that the highest gap coefficient during the transient gives the highest clad temperature. Therefore, the final gap coefficient was taken to be 10600 Btu/hr-ft^2-°F.

Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). In order to take this phenomenon into account, the following correlation, which defines the rate of the zirconium-steam reaction, was introduced into the model:

$$\frac{dw^2}{dt} = 33.3 \times 10^6 \exp\left[-\frac{45,500}{1.986T}\right]$$

where w = amount reacted, mg/cm^2

t = time, sec

T = temperature, °K

The heat of reaction is 1510 cal/gm

Results

The primary coolant pressure vs. time for a locked reactor coolant pump rotor incident from a nominal 100% power with three loops operating is shown in Figure 14.2.9-20. The peak pressure reached after 2.8 sec, is 2517 psia. The minimum DNBR for this case from the W-3 correlation is shown in Figure 14.2.9-22 as a function of time; the worst DNB condition occurs about 2 seconds after the start of the incident.

Figure 14.2.9-23 shows the minimum DNBR reached during the incident as a function of number of fuel rods. It can be seen from this figure that less than 5% of the fuel rods reach a DNBR lower than 1.3.

Figure 14.2.9-24 shows the clad temperature transient with the zirconium-steam reaction at the hot spot during the incident. The maximum clad temperature is 1640°F. Since the temperature is less than 1800°F, the zirconium-steam reaction is negligible.

Figure 14.2.9-25 through Figure 14.2.9-30 show the transient response for a locked rotor incident from a nominal 60% power with two loops operating, no loop stop valves closed. The peak pressure reaches 2515 psia after 3.2 seconds (see Figure 14.2.9-26). The minimum DNBR during the incident as a function of time is depicted in Figure 14.2.9-28. At bulk qualities above 20%, the W-3 DNBR correlation is no longer valid. The DNBR at qualities above 20% has, therefore, not been plotted. In determining the number of fuel rods having a DNBR less than 1.3, it has been assumed that all fuel rods in the core regions at qualities above 20% have a DNBR less than 1.3. Figure 14.2.9-29 shows the minimum DNBR reached during the incident as a function of the number of fuel rods. It can be seen that less than 1% of the fuel rods reach a DNBR lower than 1.30. Figure 14.2.9-30 shows the clad temperature transient at the hot spot. The maximum clad temperature is 1120°F; since the temperature is less than 1800°F, the zirconium-steam reaction is negligible.

Figures 14.2.9-31 through 14.2.9-36 show the transient response for a locked rotor incident from a nominal 67% power with two loops operating and the loop stop valves in one loop closed. The peak pressure reaches 2548 psia after 2.8 sec (see Figure 14.2.9-32). The minimum DNBR during the incident as a function of time is depicted in Figure 14.2.9-34. As previously indicated, the DNBR at qualities above 20% has not been plotted. Again, it has been assumed that all fuel rods in the core regions having qualities above 20% have a DNBR less than 1.3. Figure 14.2.9-35 shows the minimum DNBR reached during the accident as a function of the number of fuel rods. It can be seen that less than 1% of the fuel rods reach a DNBR lower than 1.30. Figure 14.2.9-36 shows the clad temperature transient at the hot spot. The maximum clad temperature is 1135°F; since the temperature is less than 1800°F, the Zirconium-steam reaction is negligible.

Conclusions

1. Since the peak pressure reached during any of the transients is 2548 psia, the integrity of the Reactor Coolant System is not endangered. A pressure of 2548 psia can be considered as an upper limit because of the following conservative assumptions used in the study:

(a) Credit was not taken for any negative moderator coefficient

(b) It was assumed that the pressurizer relief valves were inoperative

(c) The steam dump was assumed to be inoperative

(d) It was assumed that the pressurizer spray was inoperative

(e) Credit was not taken for the controlled feedwater flow after trip.

2. Less than 5% of the fuel rods would exhibit a DNBR of less than 1.3 in the worst case.

3. The peak clad surface temperature of 1640°F, calculated for the hot spot during the worst transient, can also be considered an upper limit since:

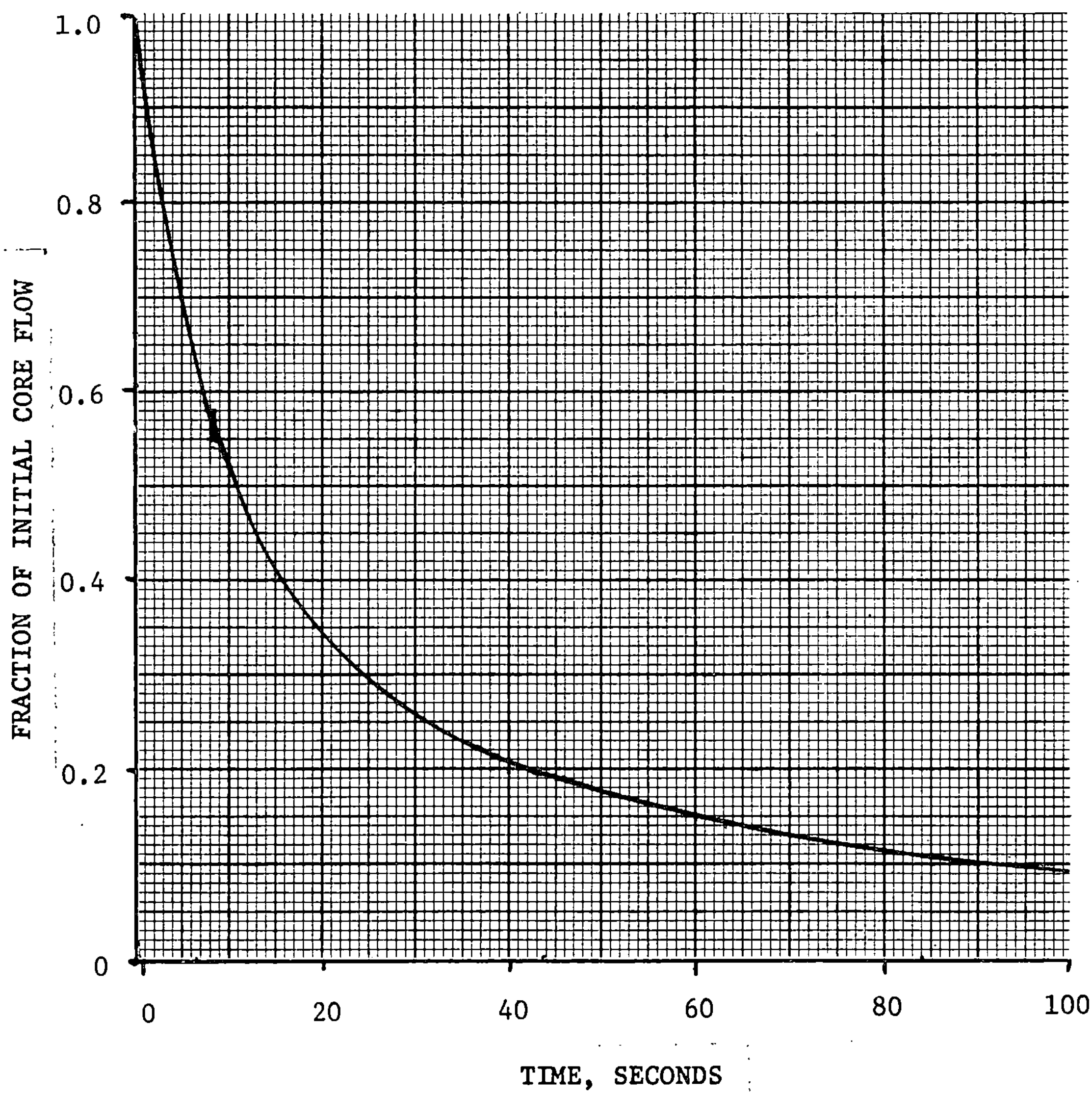
(a) The hot spot was assumed to be in DNB at the start of the incident

(b) A high gap coefficient was used

(c) A value of 0.7 times the heat transfer coefficient for film boiling was used in the study and film boiling was assumed to be fully developed from the start of the transient, i.e., no credit was taken for transition boiling.

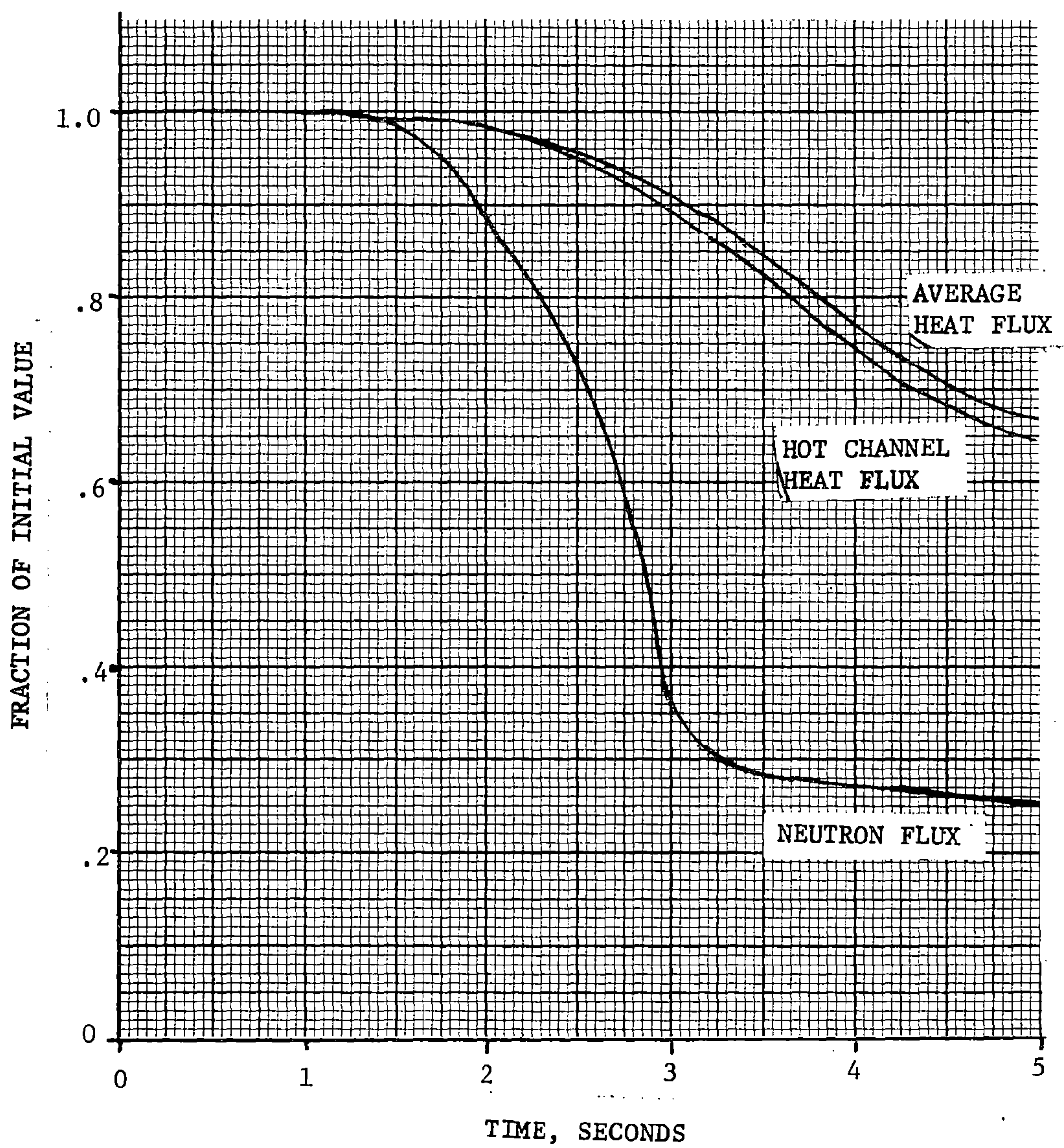
(d) The nuclear heat released in the fuel at the hot spot was based on a zero moderator coefficient.

FIGURE 14.2.9-1
12-1-69

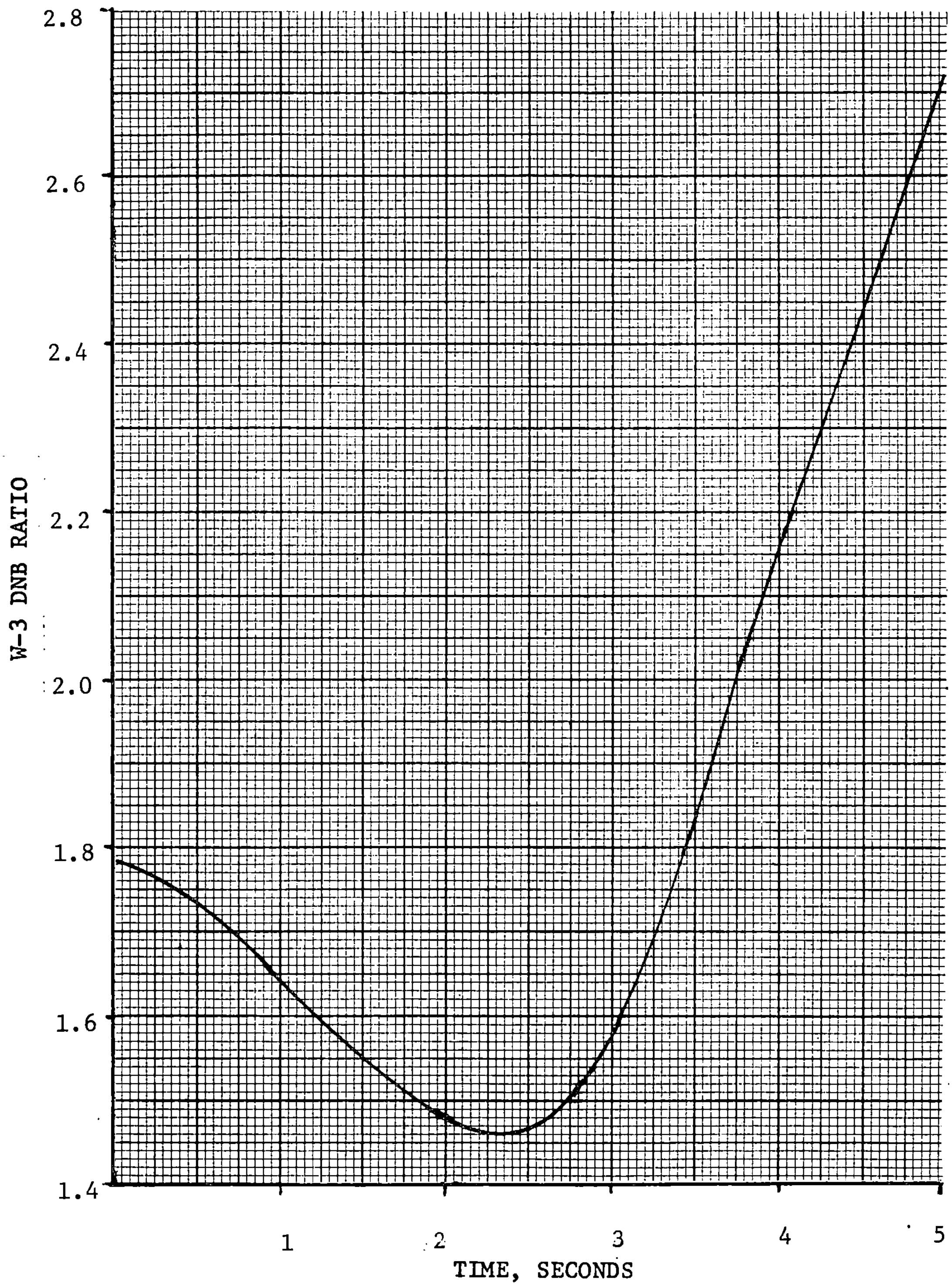


THREE LOOPS OPERATING,
THREE PUMPS LOSS OF FLOW INCIDENT

FIGURE 14.2.9-2
12-1-69

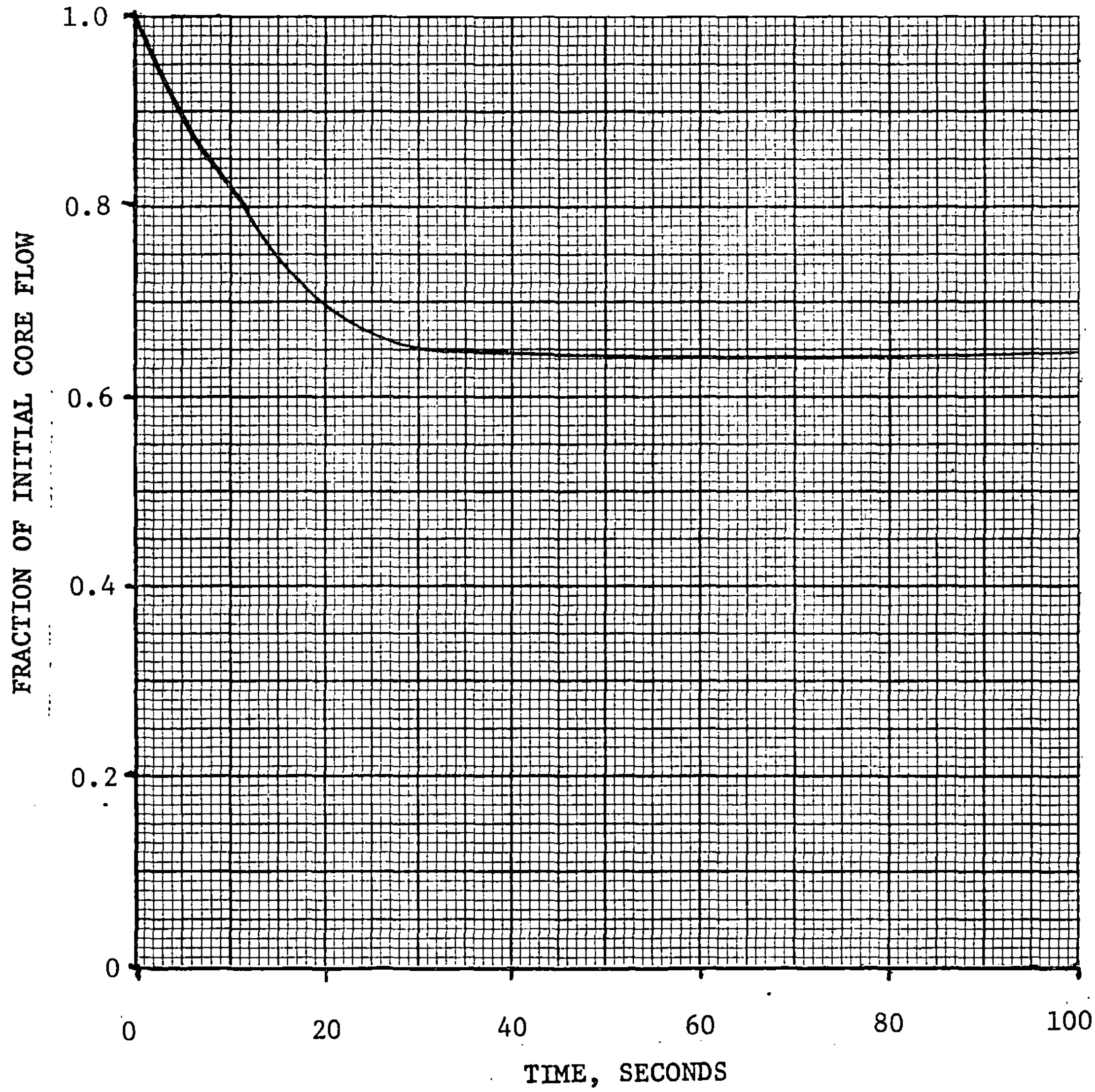


THREE LOOPS OPERATING
THREE PUMPS LOSS OF FLOW INCIDENT



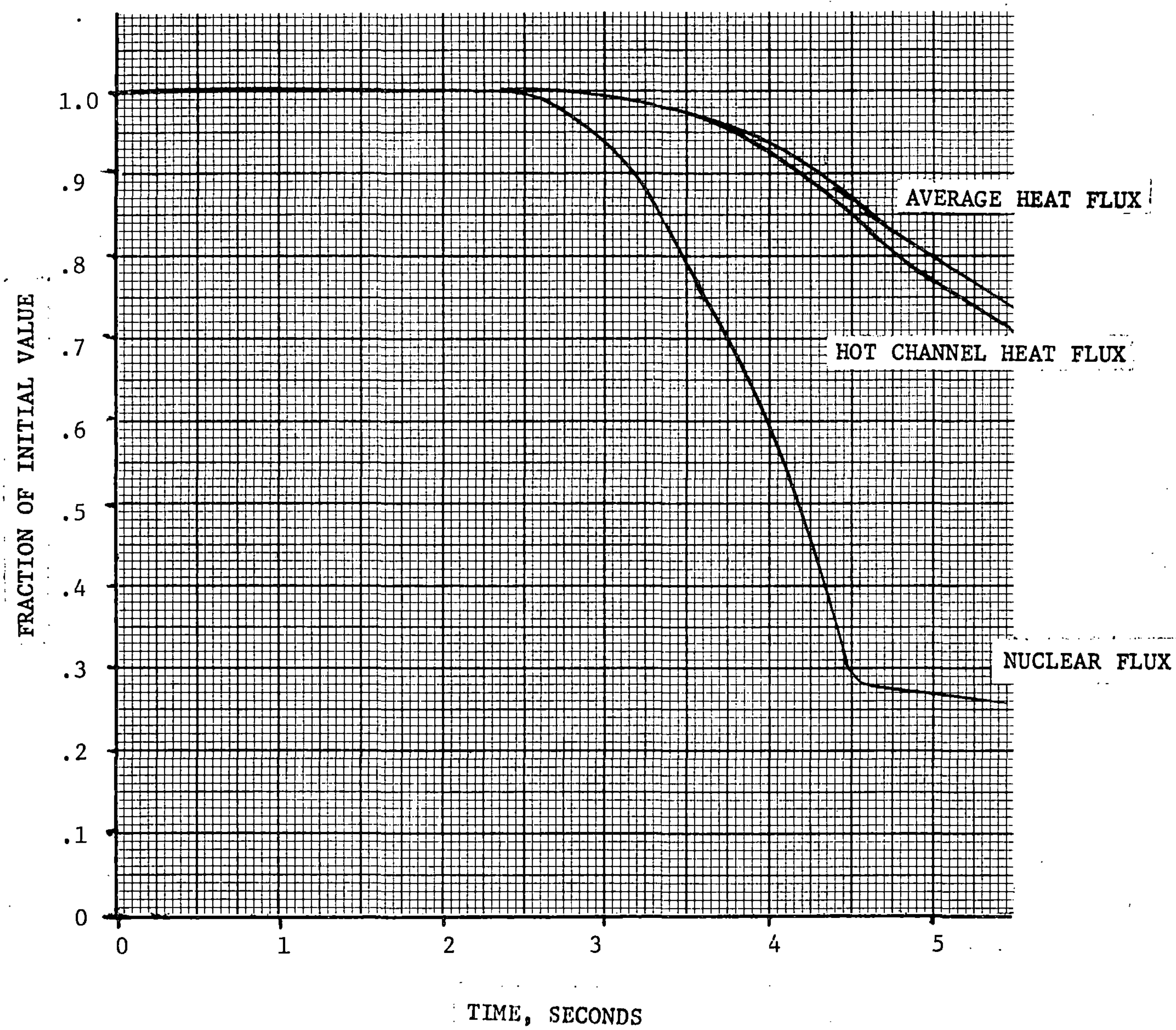
THREE LOOPS OPERATING
THREE PUMPS LOSS OF FLOW INCIDENT

FIGURE 14.2.9-4
12-1-69



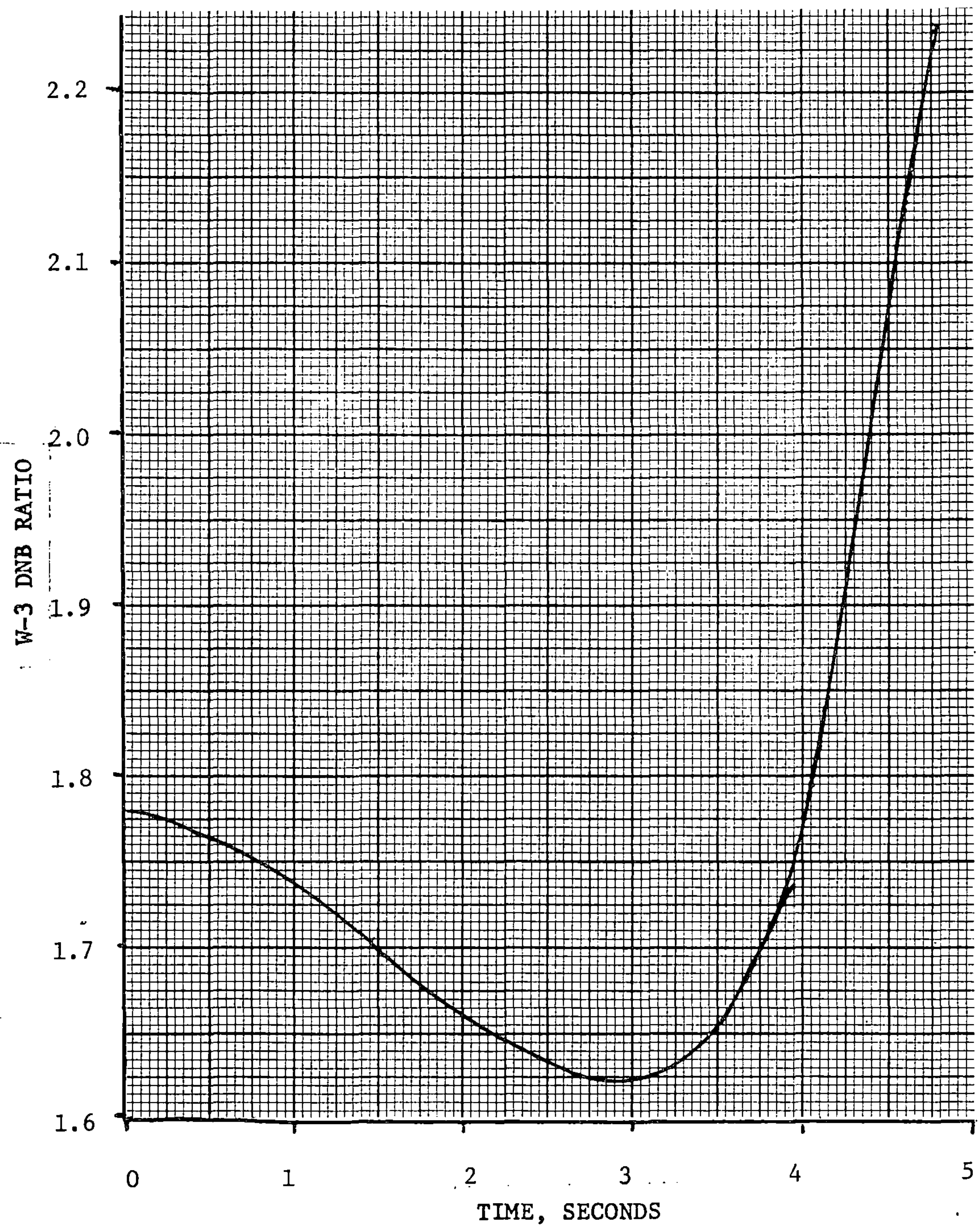
THREE LOOPS OPERATING,
ONE PUMP LOSS OF FLOW INCIDENT

FIGURE 14.2.9-5
12-1-69



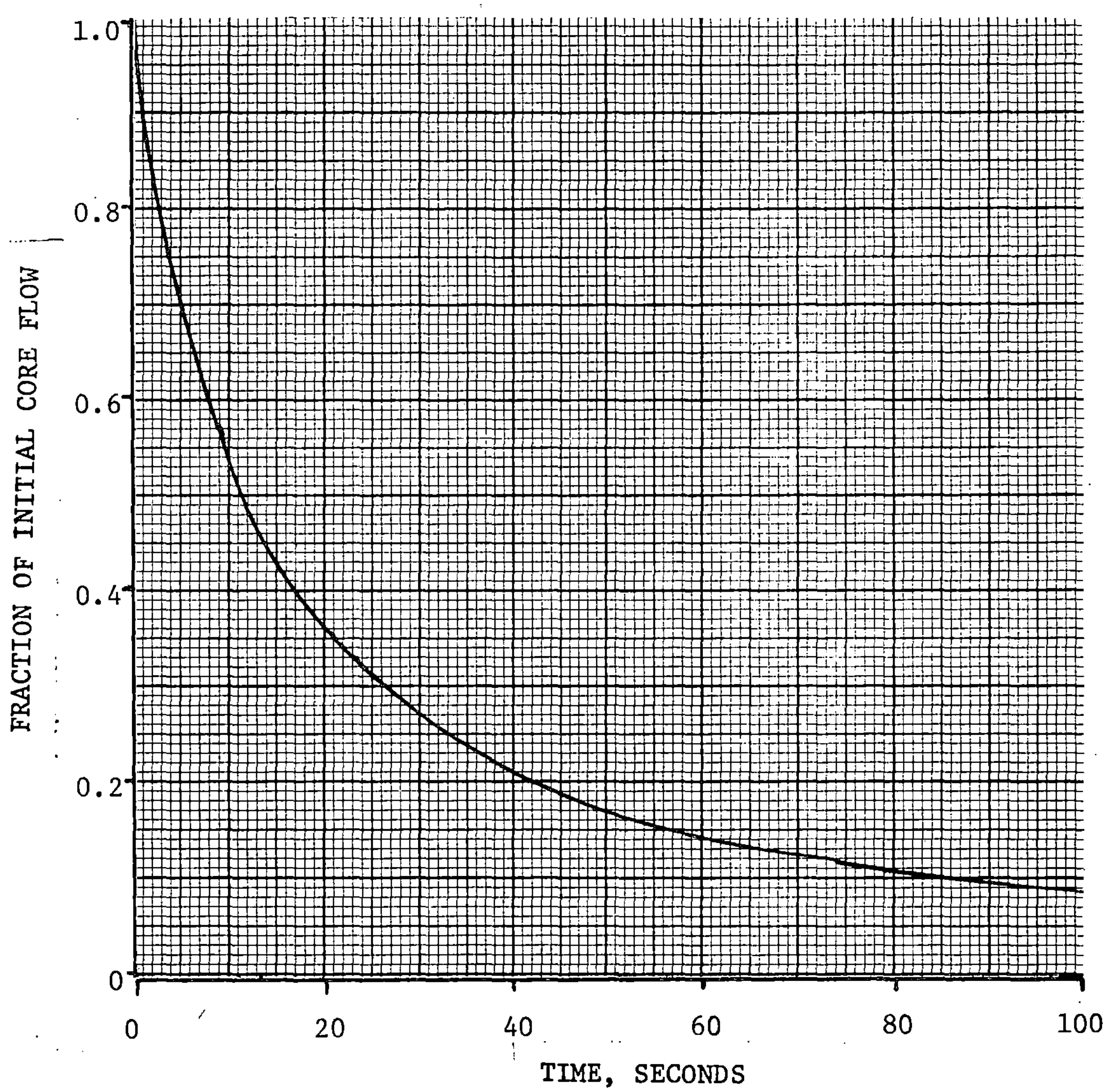
THREE LOOPS OPERATING, ONE
PUMP LOSS OF FLOW INCIDENT

FIGURE 14.2.9-6
12-1-69



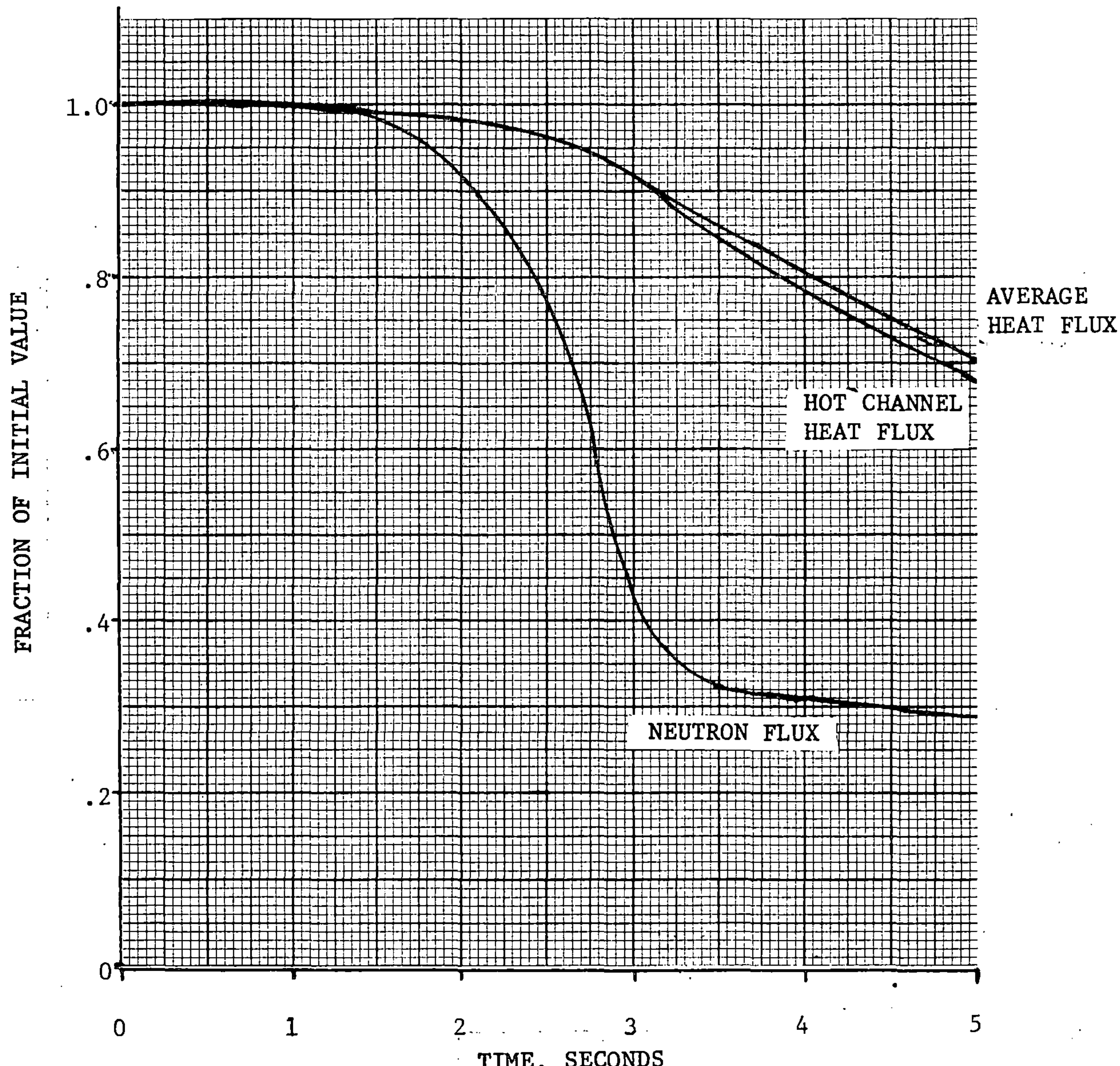
THREE LOOPS OPERATING,
ONE PUMP LOSS OF FLOW INCIDENT

FIGURE 14.2.9-7
12-1-69



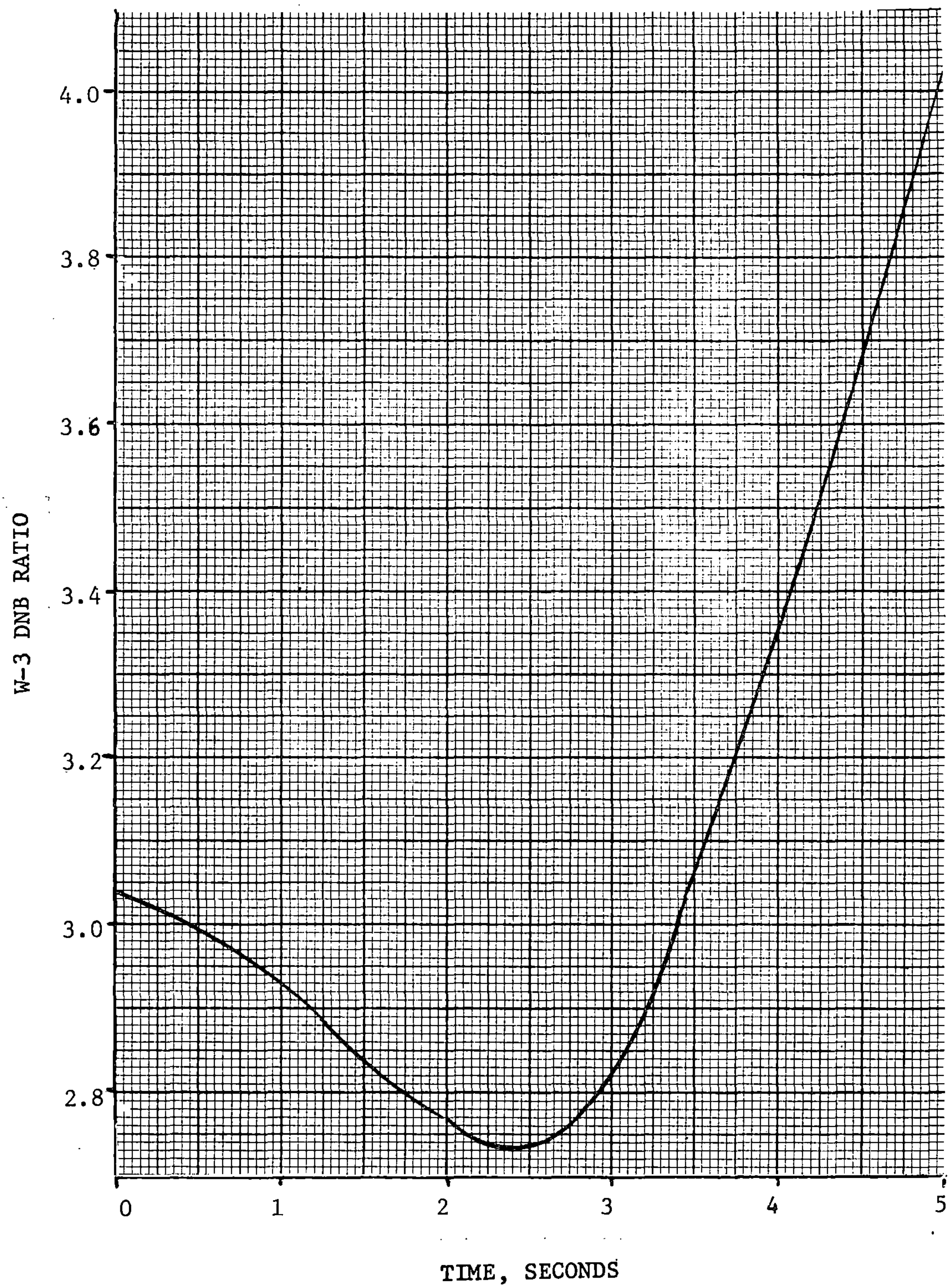
TWO LOOPS OPERATING,
NO LOOP STOP VALVES CLOSED,
TWO PUMPS LOSS OF FLOW INCIDENT

FIGURE 14.2.9-8
12-1-69



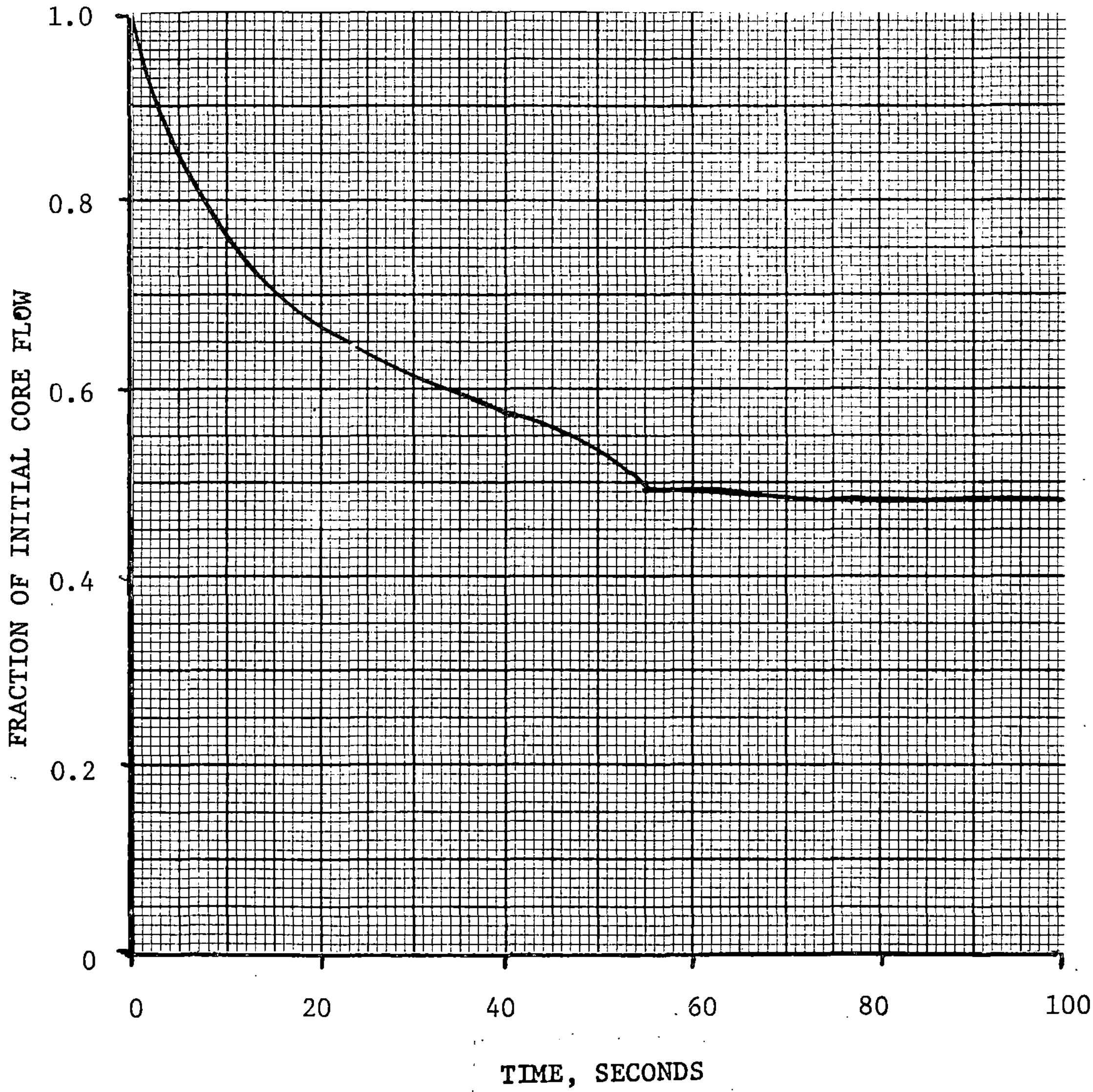
TWO LOOPS OPERATING,
NO LOOP STOP VALVES CLOSED,
TWO PUMPS LOSS OF FLOW INCIDENT

FIGURE 14.2.9-9
12-1-69



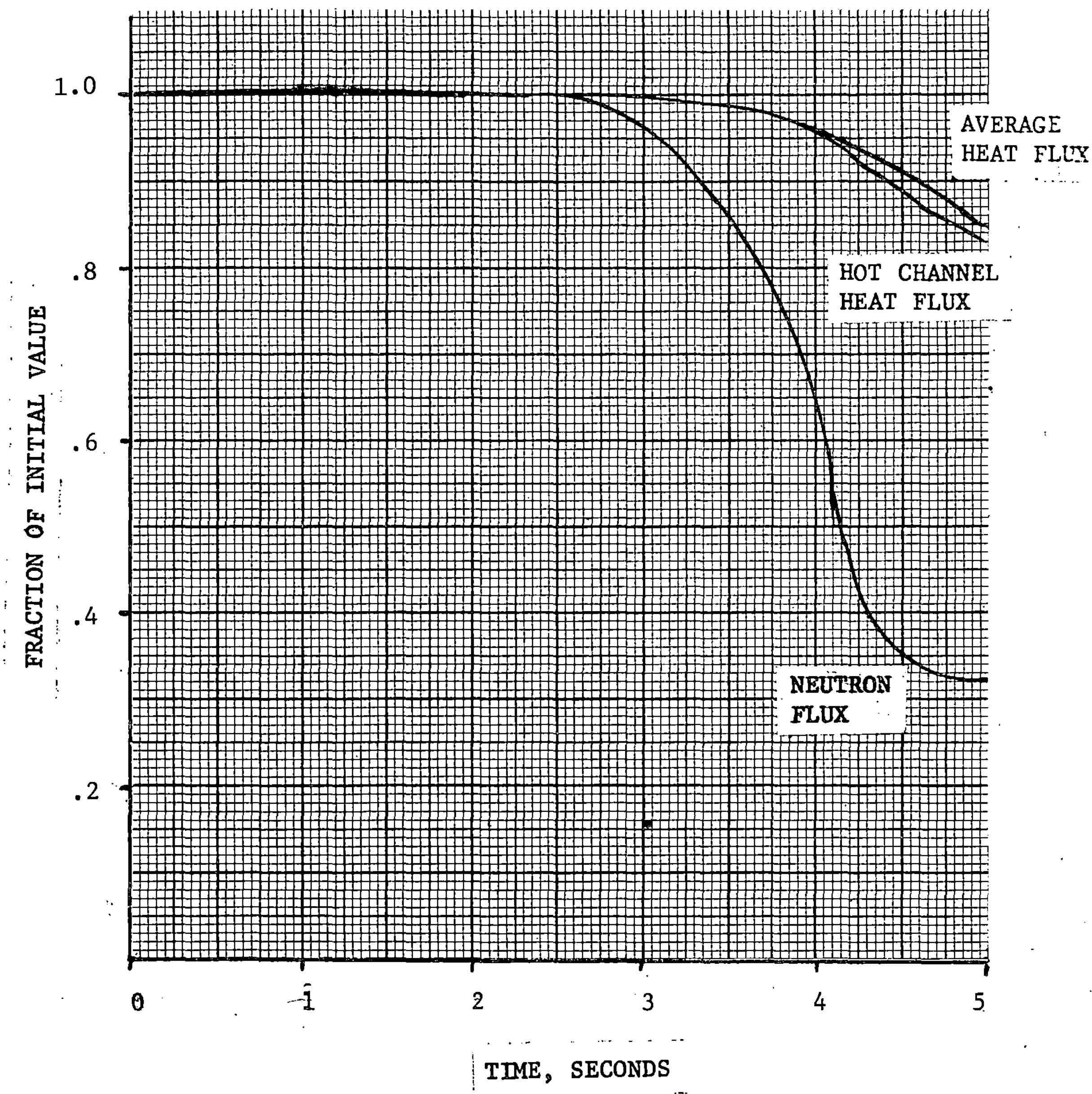
TWO LOOPS OPERATING, NO LOOP
STOP VALVES CLOSED, TWO PUMPS
LOSS OF FLOW INCIDENT

FIGURE 14.2.9-10
12-1-69



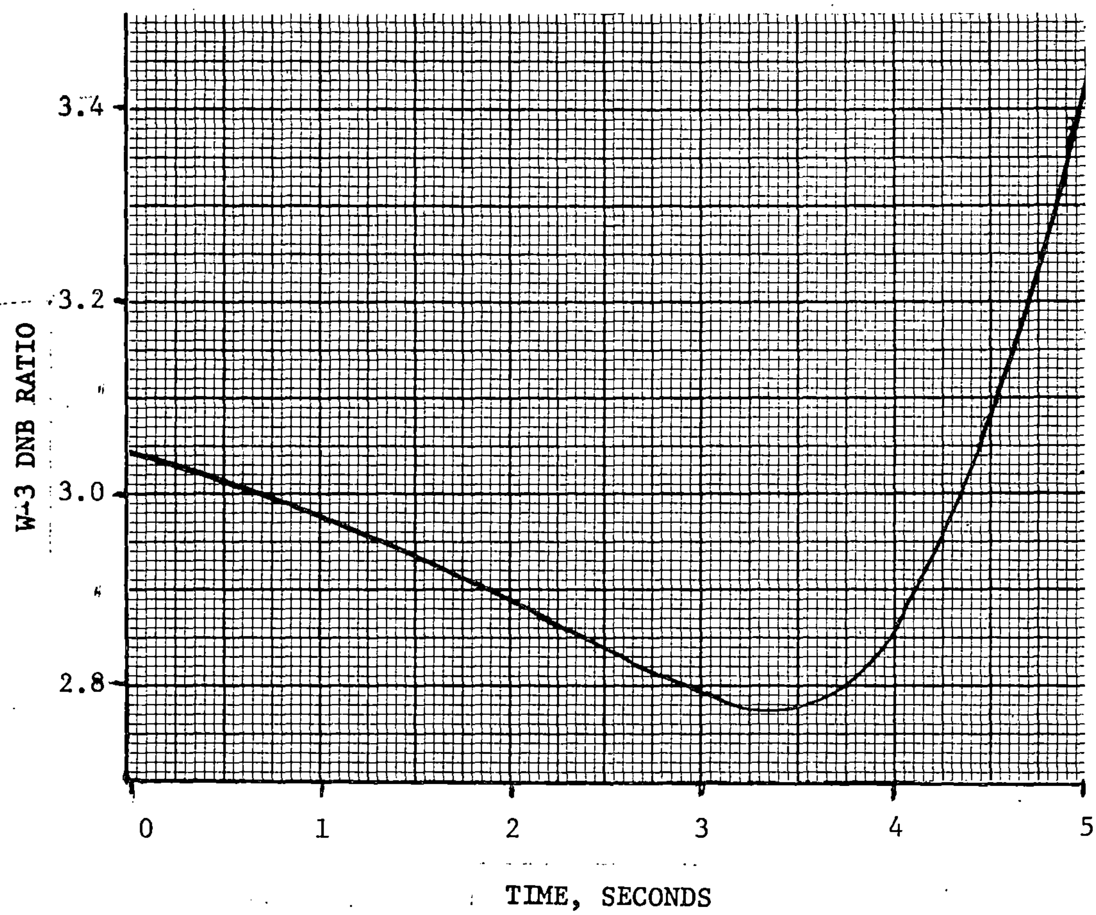
TWO LOOPS OPERATING,
NO LOOP STOP VALVES CLOSED,
ONE PUMP LOSS OF FLOW INCIDENT

FIGURE 14.2.9-11
12-1-69



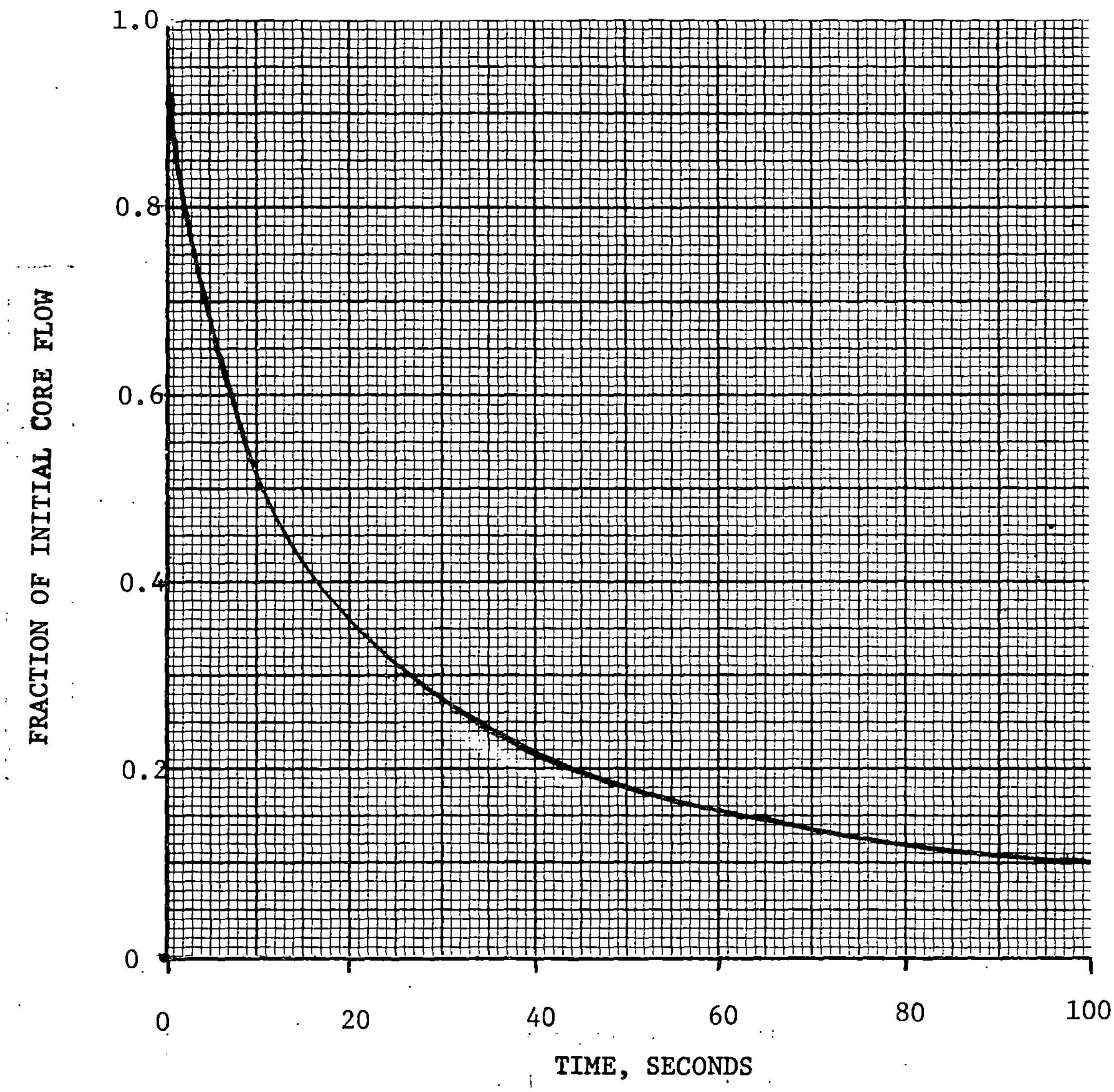
TWO LOOPS OPERATING,
NO LOOP STOP VALVES CLOSED,
ONE PUMP LOSS OF FLOW INCIDENT

FIGURE 14.2.9-12
12-1-69

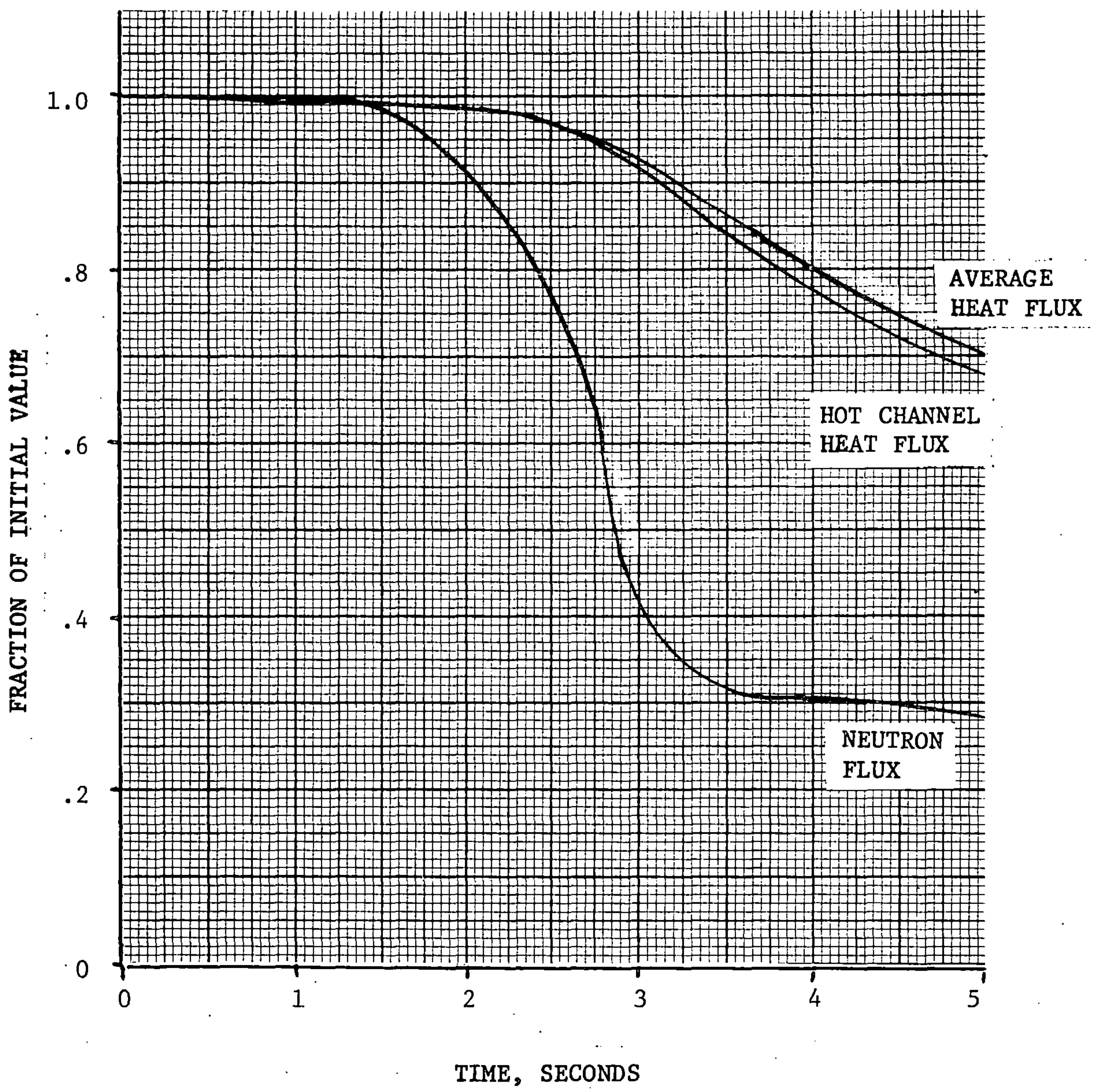


TWO LOOPS OPERATING,
NO LOOP STOP VALVES CLOSED,
ONE PUMP LOSS OF FLOW INCIDENT

FIGURE 14.2.9-13
12-1-69

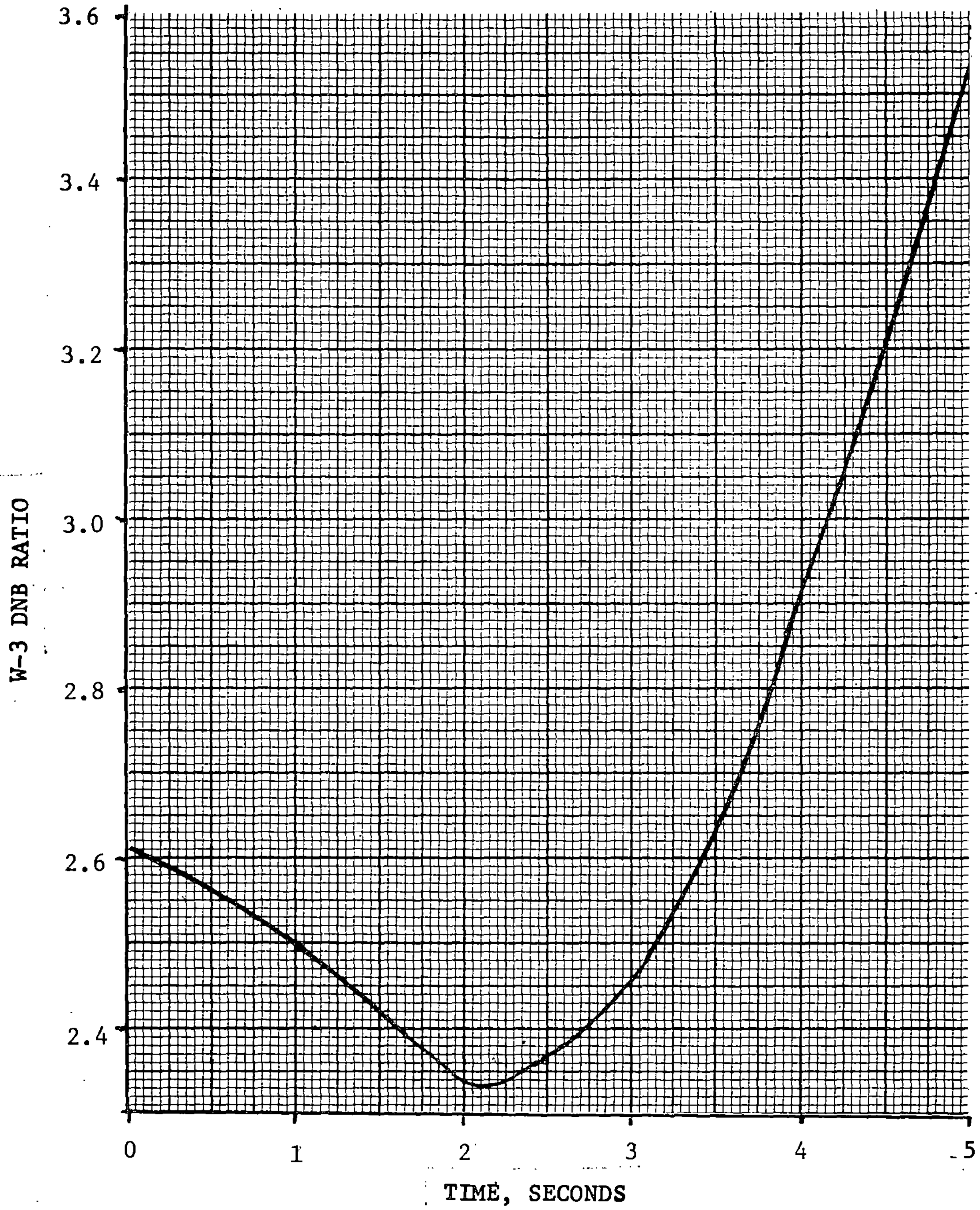


TWO LOOPS OPERATING,
LOOP STOP VALVES CLOSED IN ONE LOOP,
TWO PUMPS LOSS OF FLOW INCIDENT



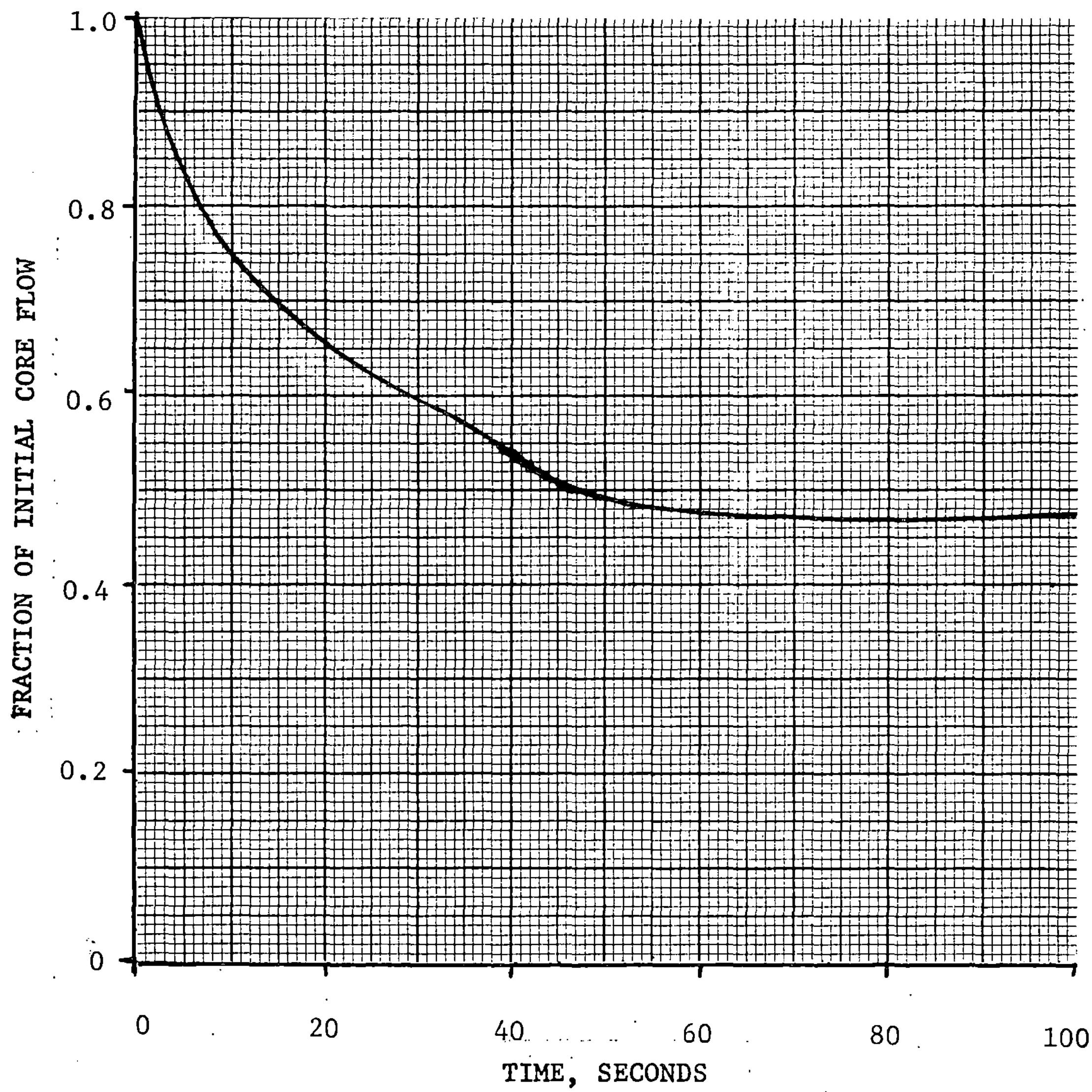
TWO LOOPS OPERATING,
LOOP STOP VALVES CLOSED IN ONE LOOP
TWO PUMPS LOSS OF FLOW INCIDENT

FIGURE 14.2.9-15
12-1-69



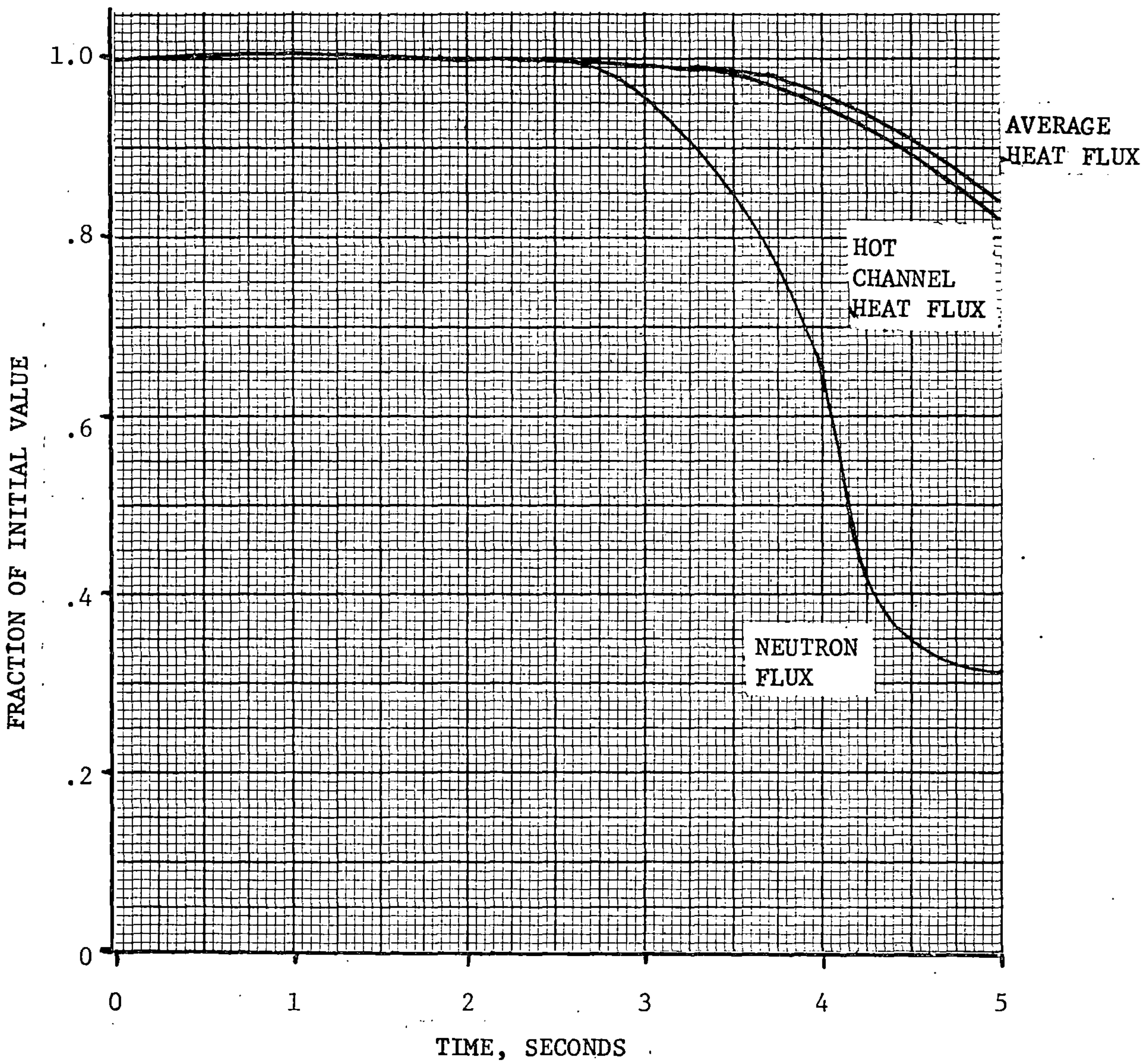
TWO LOOPS OPERATING,
LOOP STOP VALVES CLOSED IN ONE LOOP,
TWO PUMPS LOSS OF FLOW INCIDENT

FIGURE 14.2.9-16
12-1-69



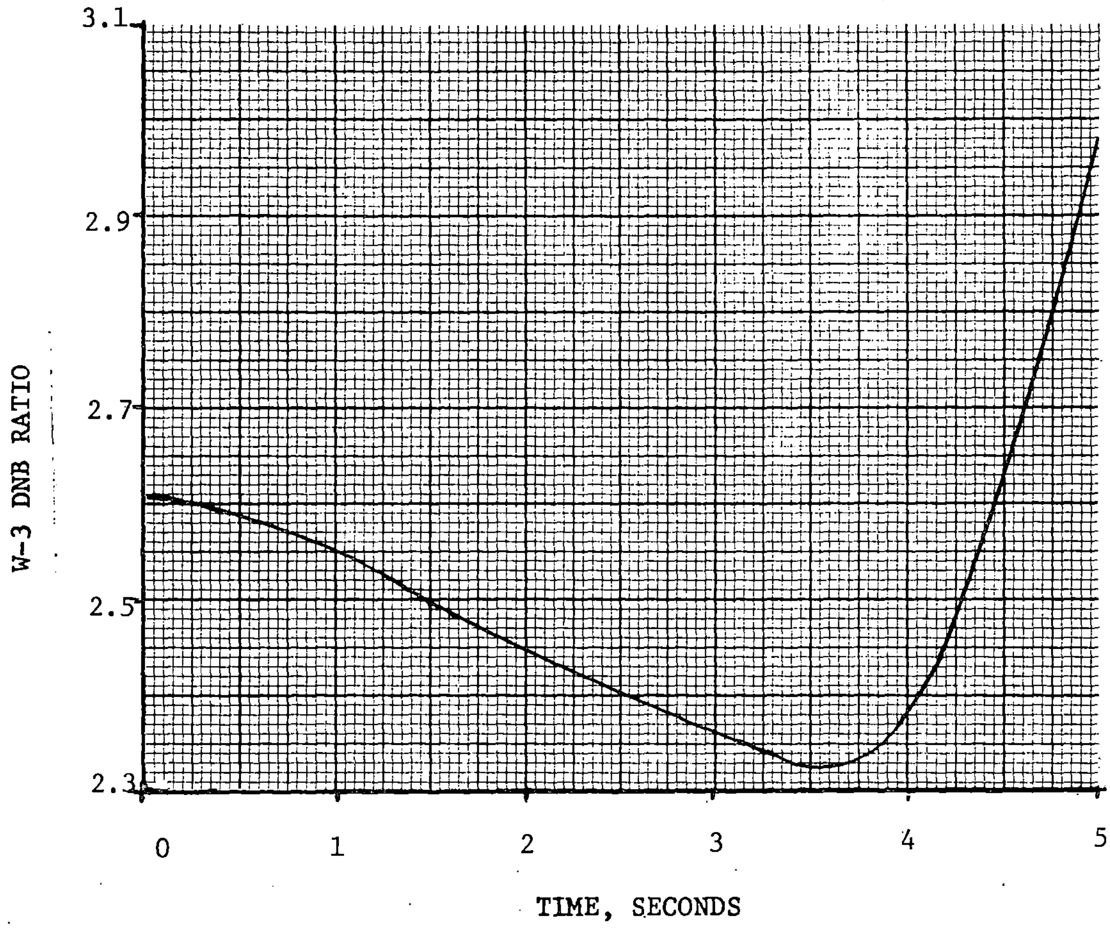
TWO LOOPS OPERATING,
LOOP STOP VALVES CLOSED IN ONE LOOP,
ONE PUMP LOSS OF FLOW INCIDENT

FIGURE 14.2.9-17
12-1-69

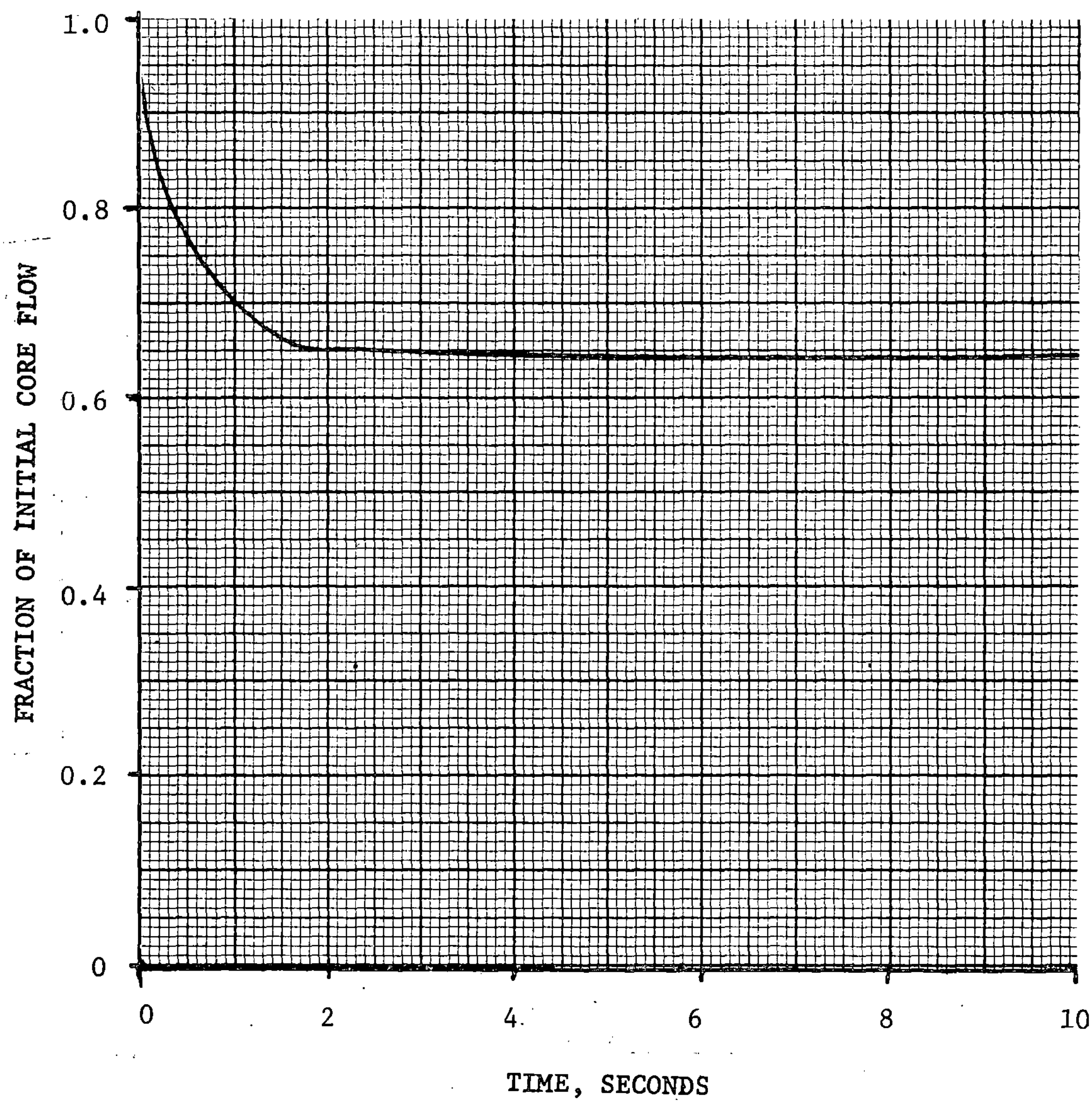


TWO LOOPS OPERATING,
LOOP STOP VALVES CLOSED IN ONE LOOP,
ONE PUMP LOSS OF FLOW INCIDENT

FIGURE 14.2.9-18
12-1-69

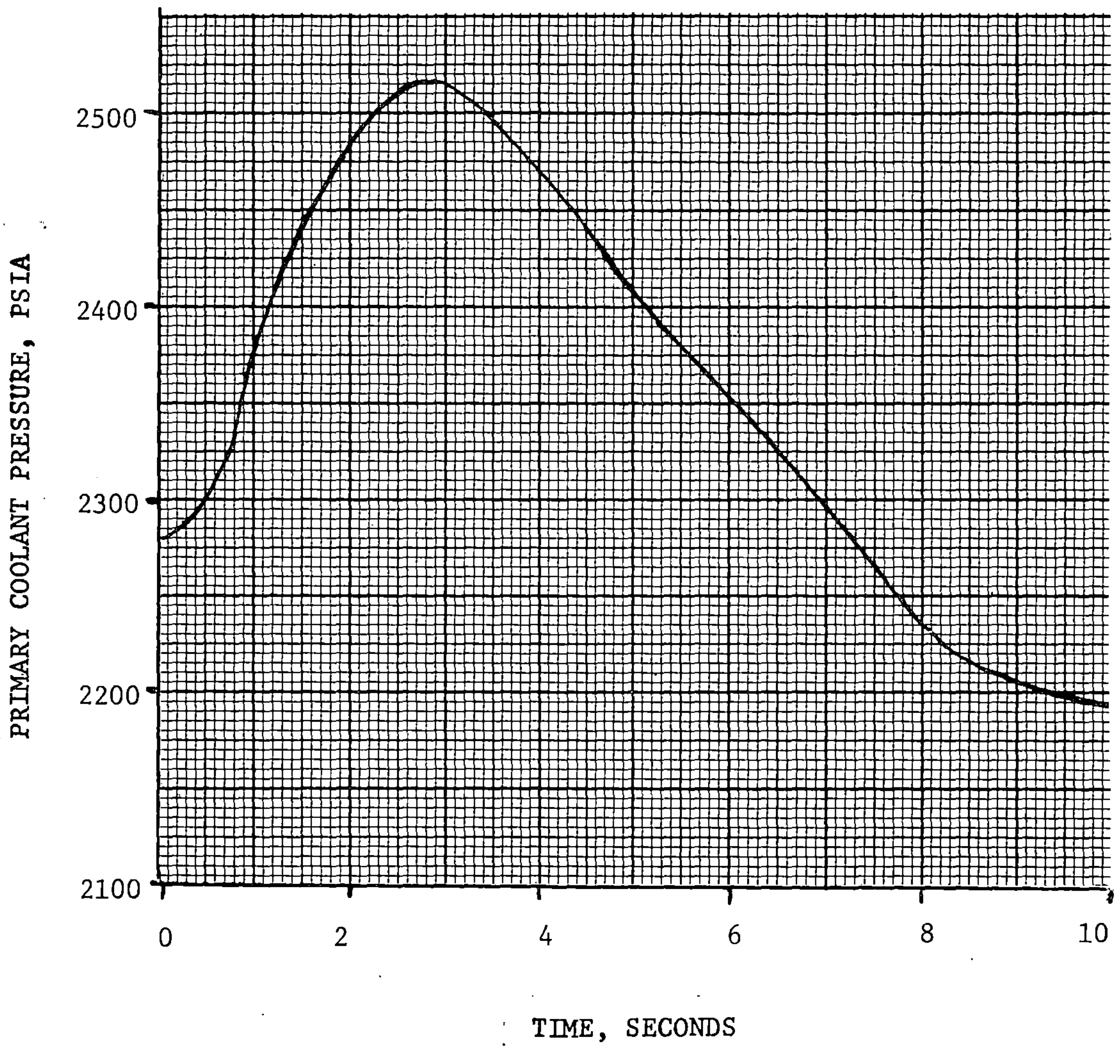


TWO LOOPS OPERATING,
LOOP STOP VALVES CLOSED IN ONE LOOP,
ONE PUMP LOSS OF FLOW INCIDENT



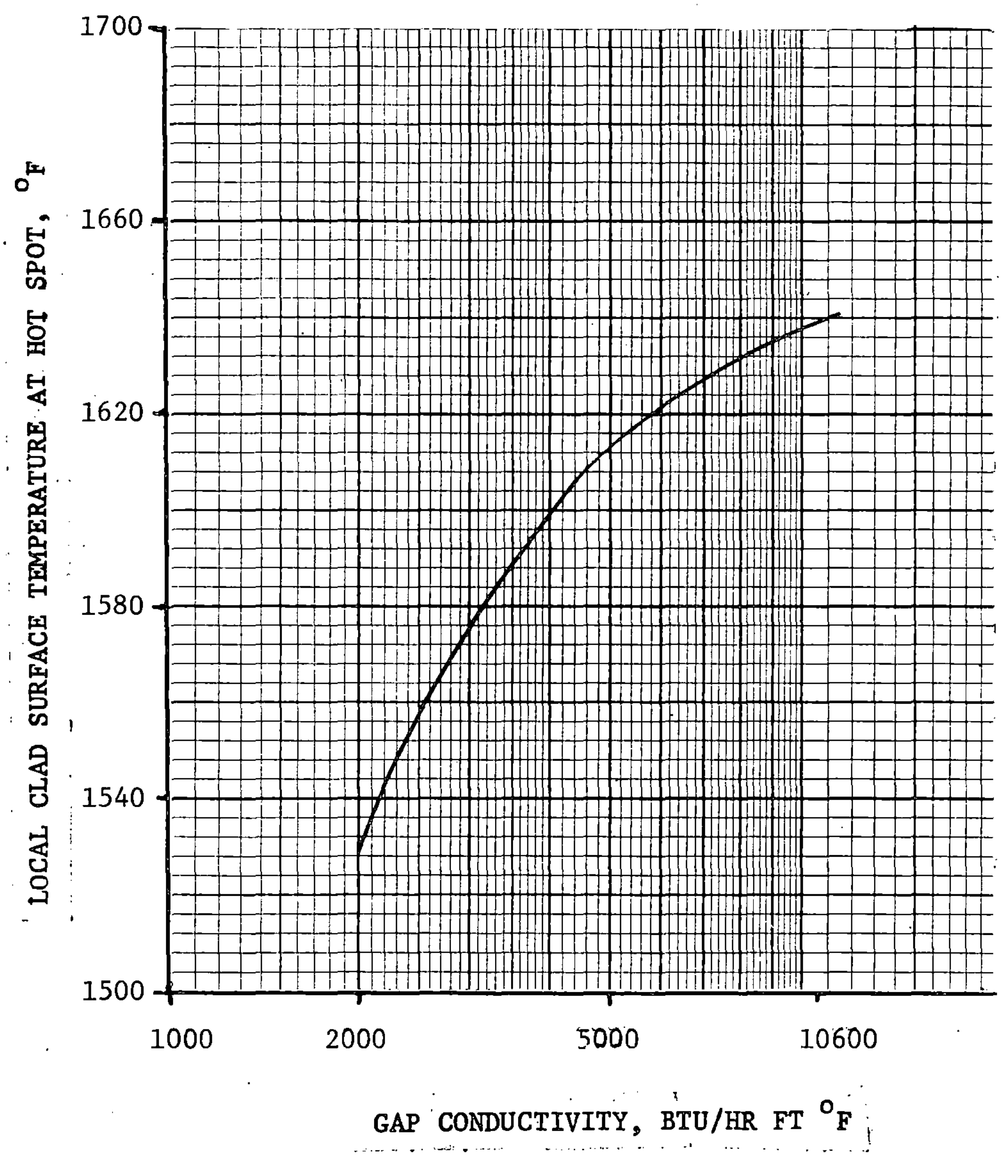
THREE LOOPS OPERATING,
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-20
12-1-69

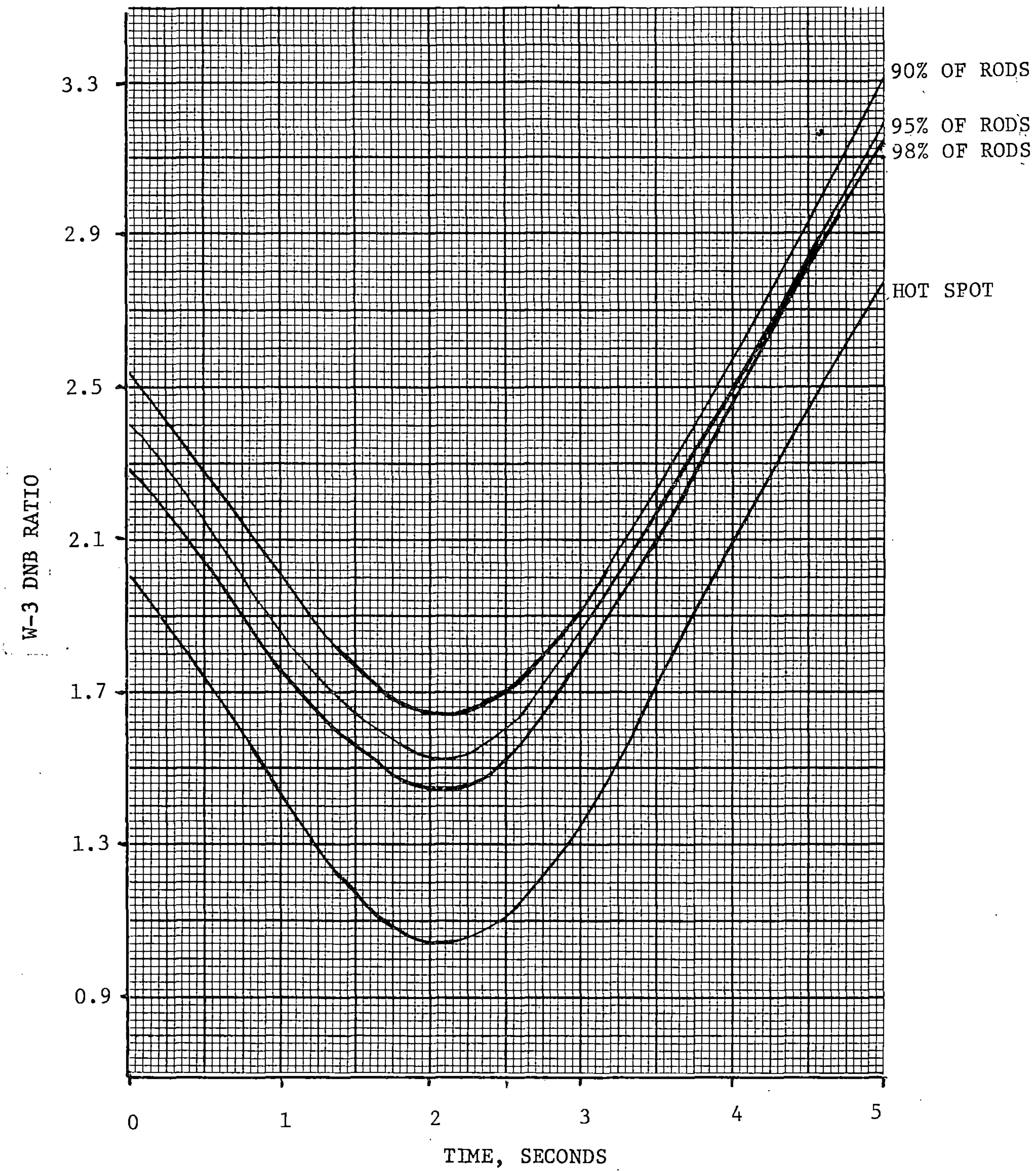


THREE LOOPS OPERATING,
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-21
12-1-69

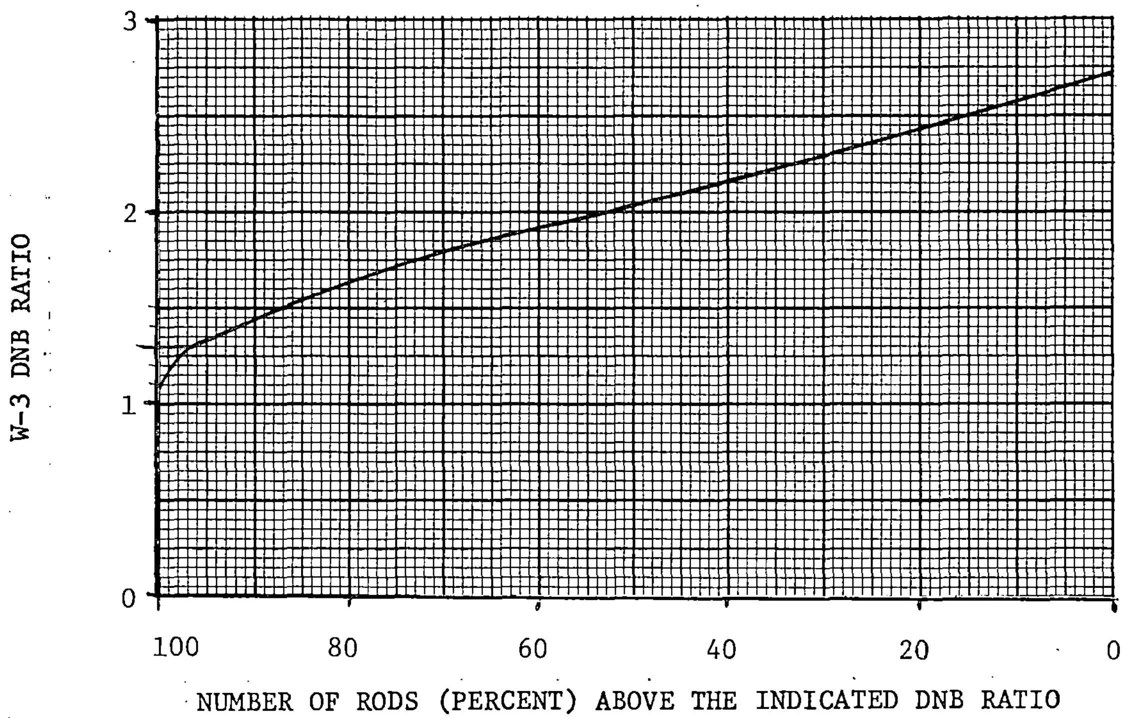


THREE LOOPS OPERATING,
LOCKED ROTOR INCIDENT



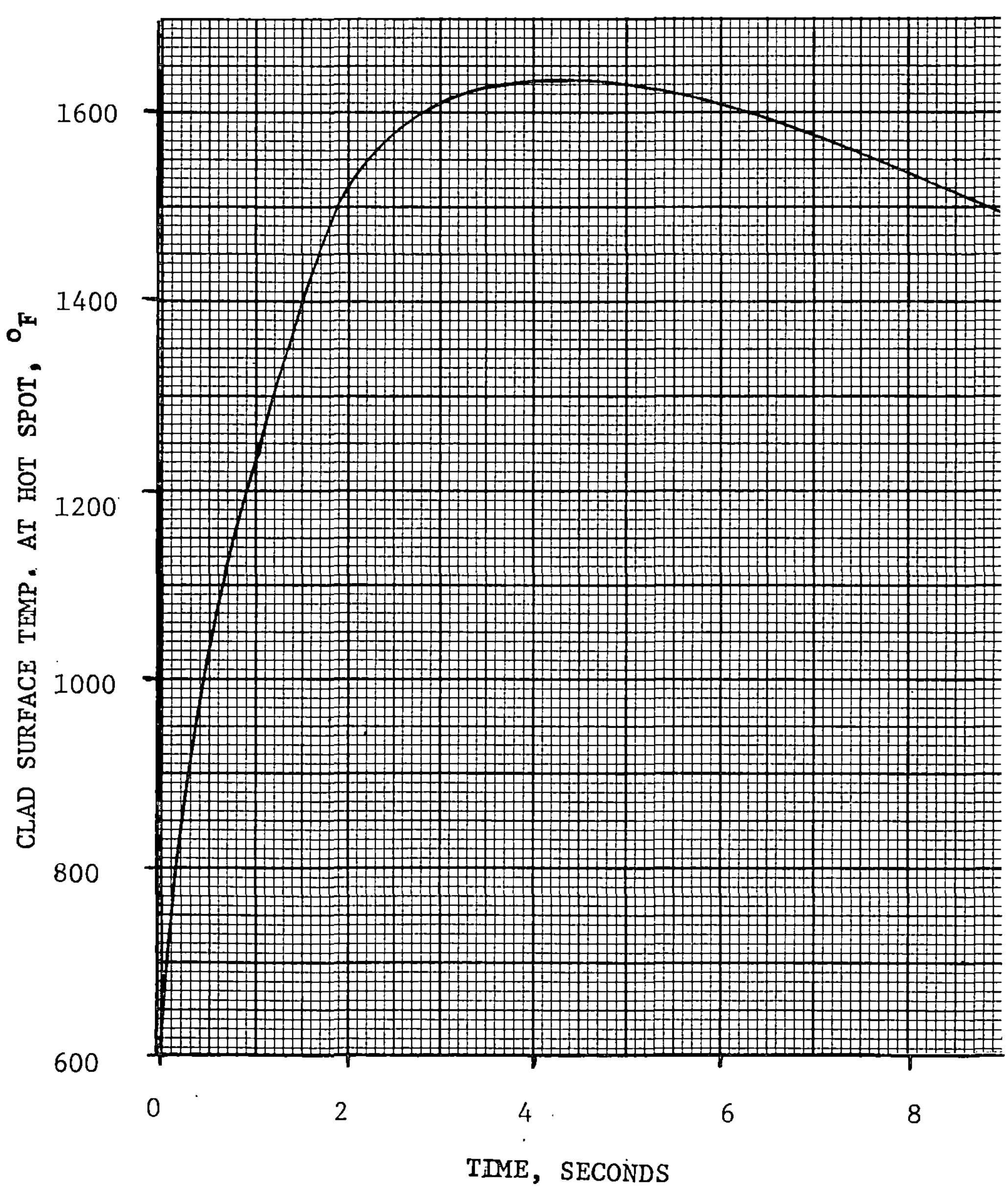
THREE LOOPS OPERATING,
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-23
12-1-69



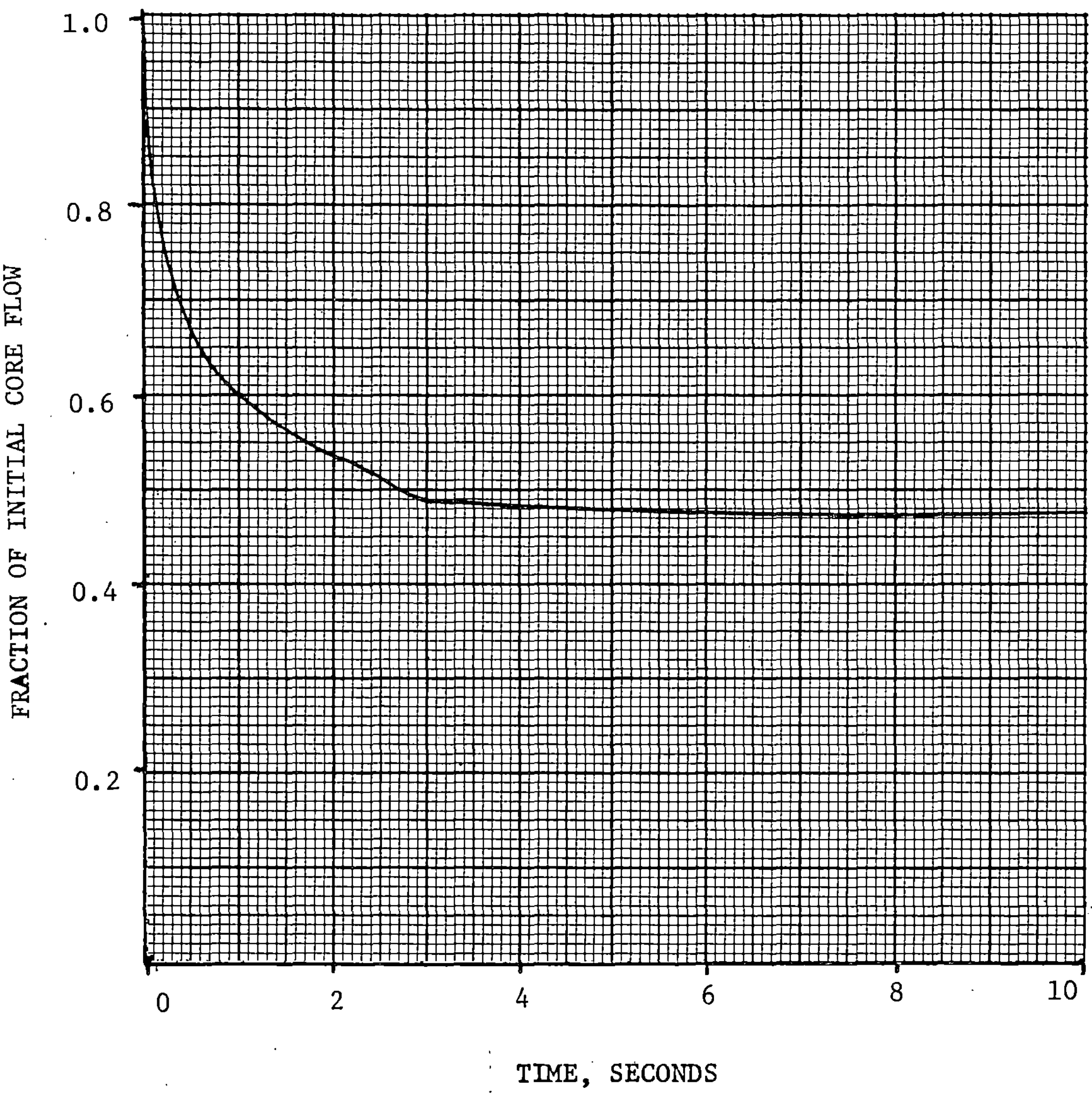
THREE LOOPS OPERATING,
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-24
12-1-69



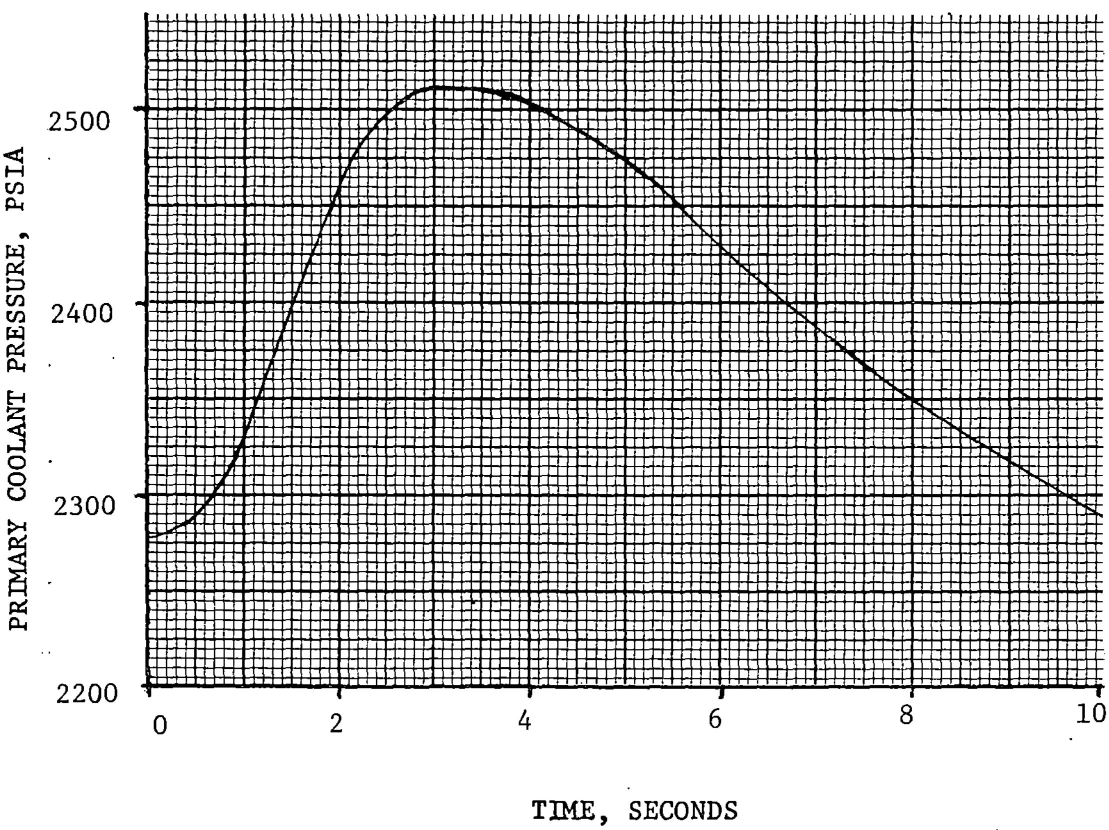
THREE LOOPS OPERATING,
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-25
12-1-69



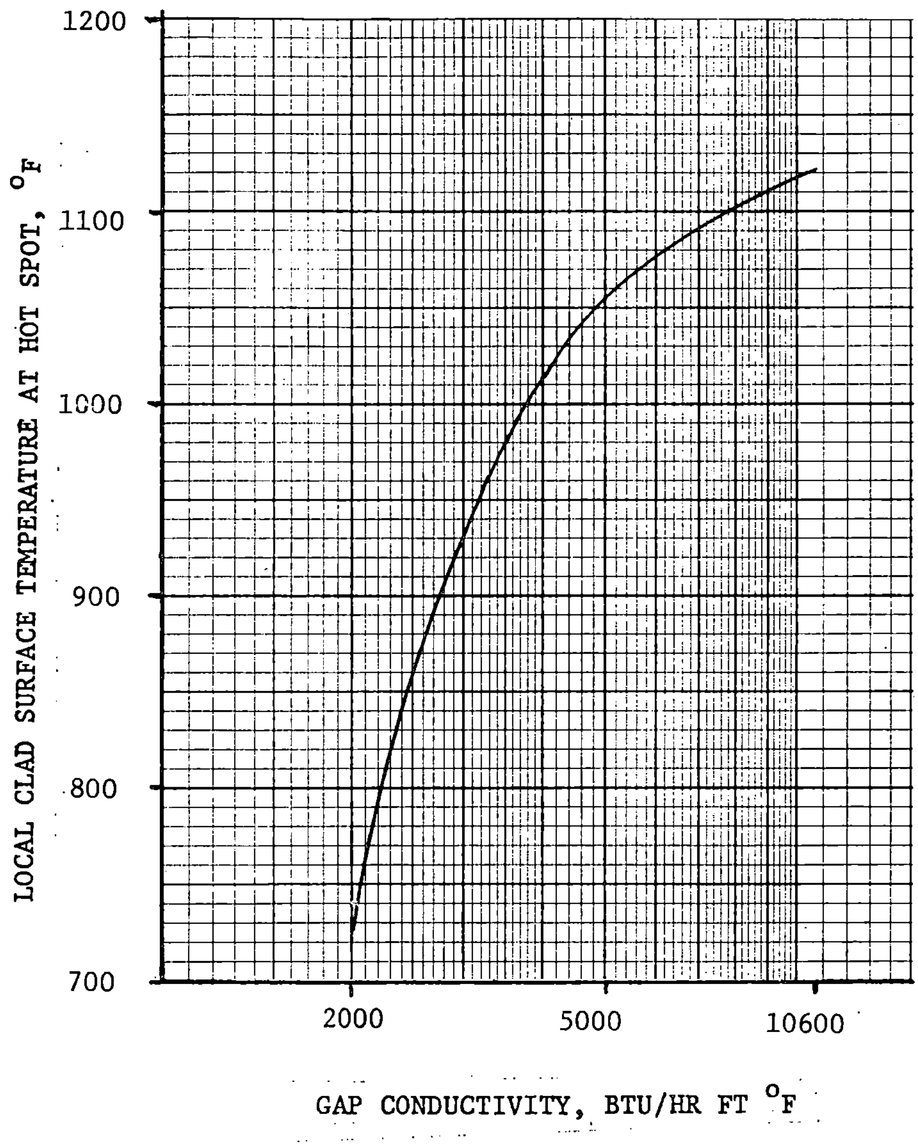
TWO LOOPS OPERATING
NO LOOP STOP VALVES CLOSED
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-26
12-1-69



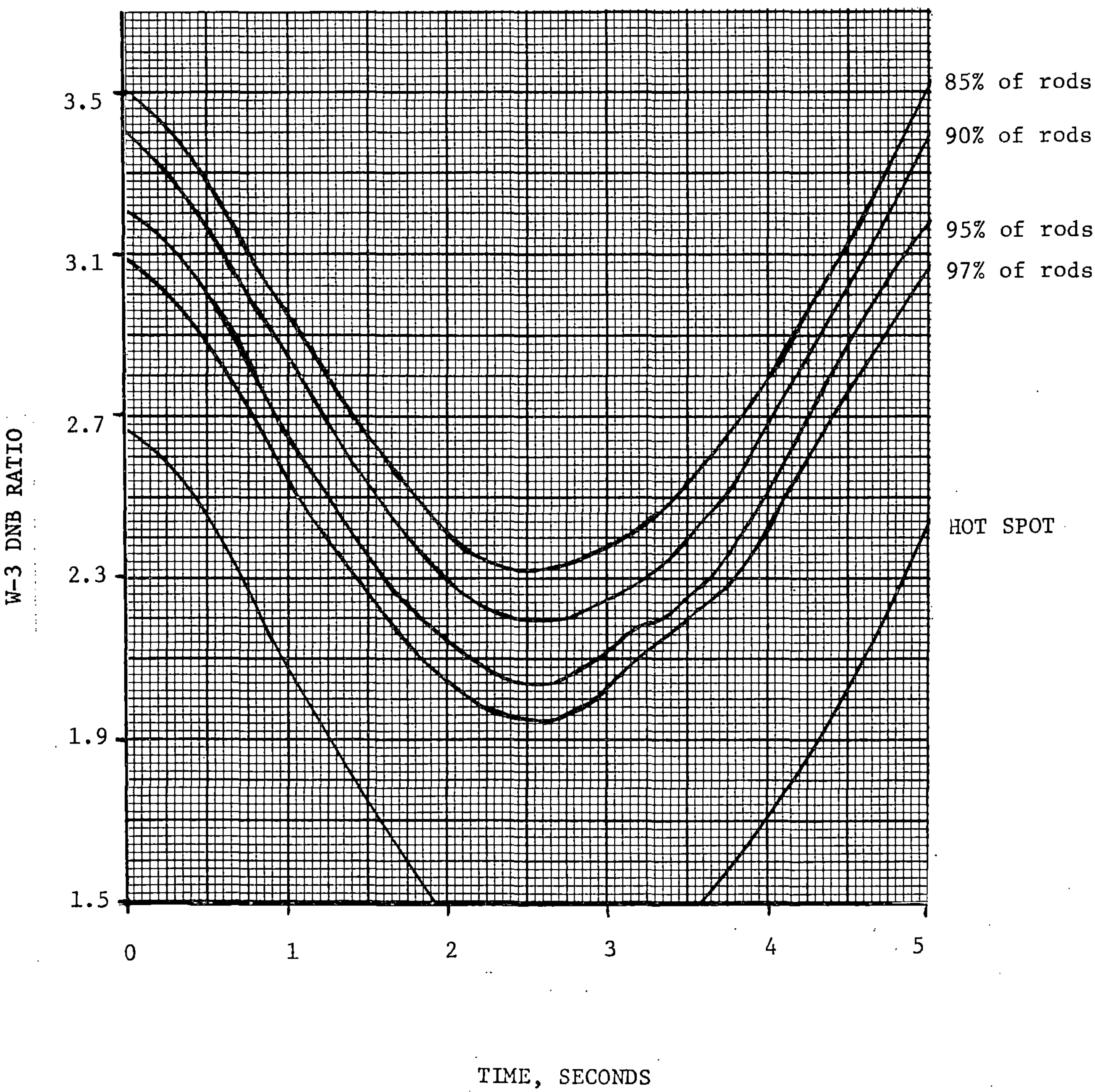
TWO LOOPS OPERATING,
NO LOOP STOP VALVES CLOSED,
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-27
12-1-69



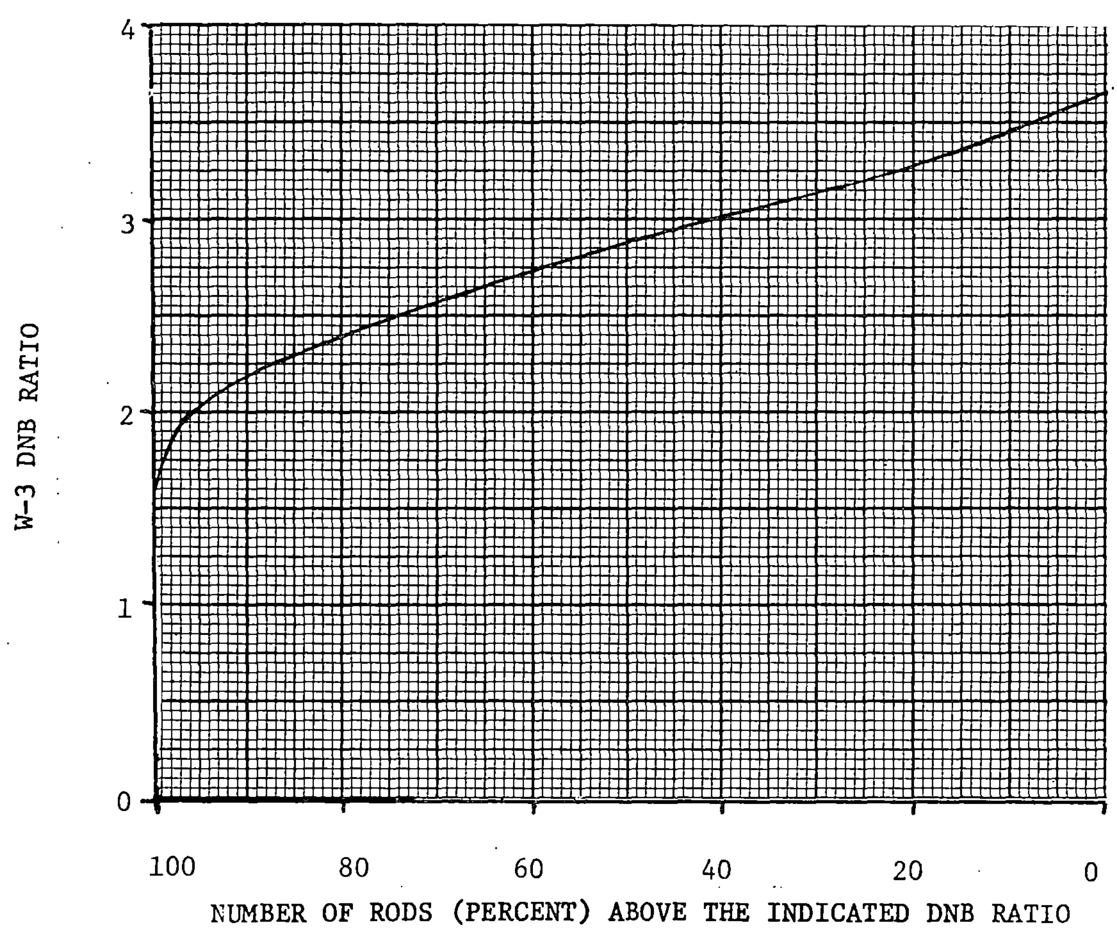
TWO LOOPS OPERATING,
NO LOOP STOP VALVES CLOSED
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-28
12-1-69



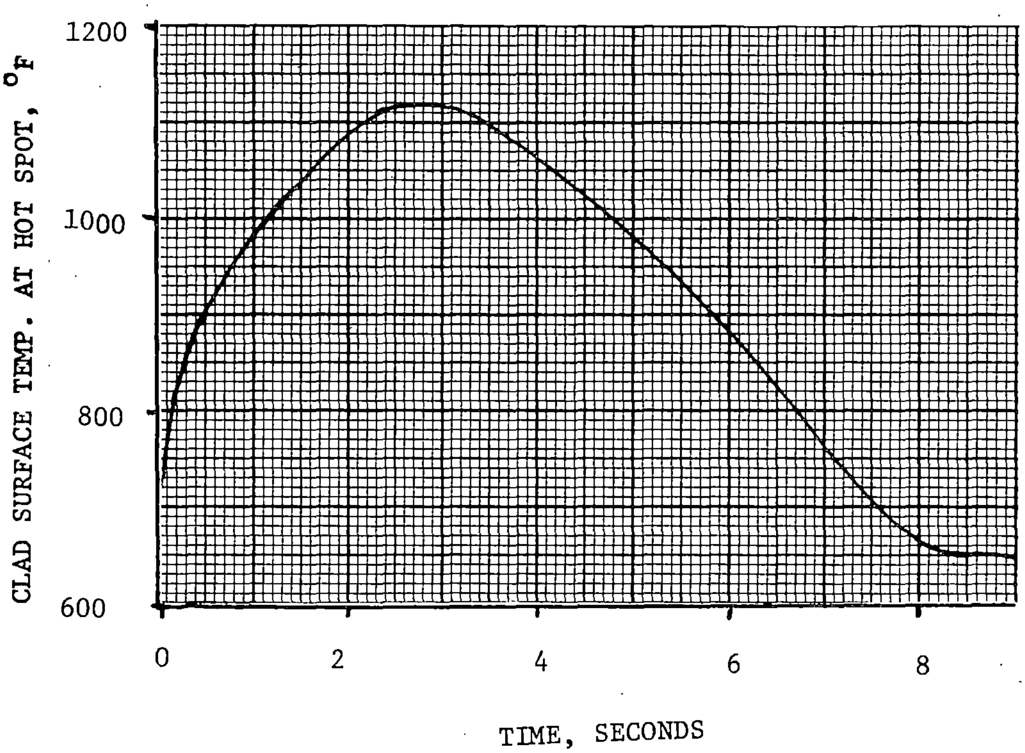
TWO LOOPS OPERATING
NO LOOP STOP VALVES CLOSED
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-29
12-1-69



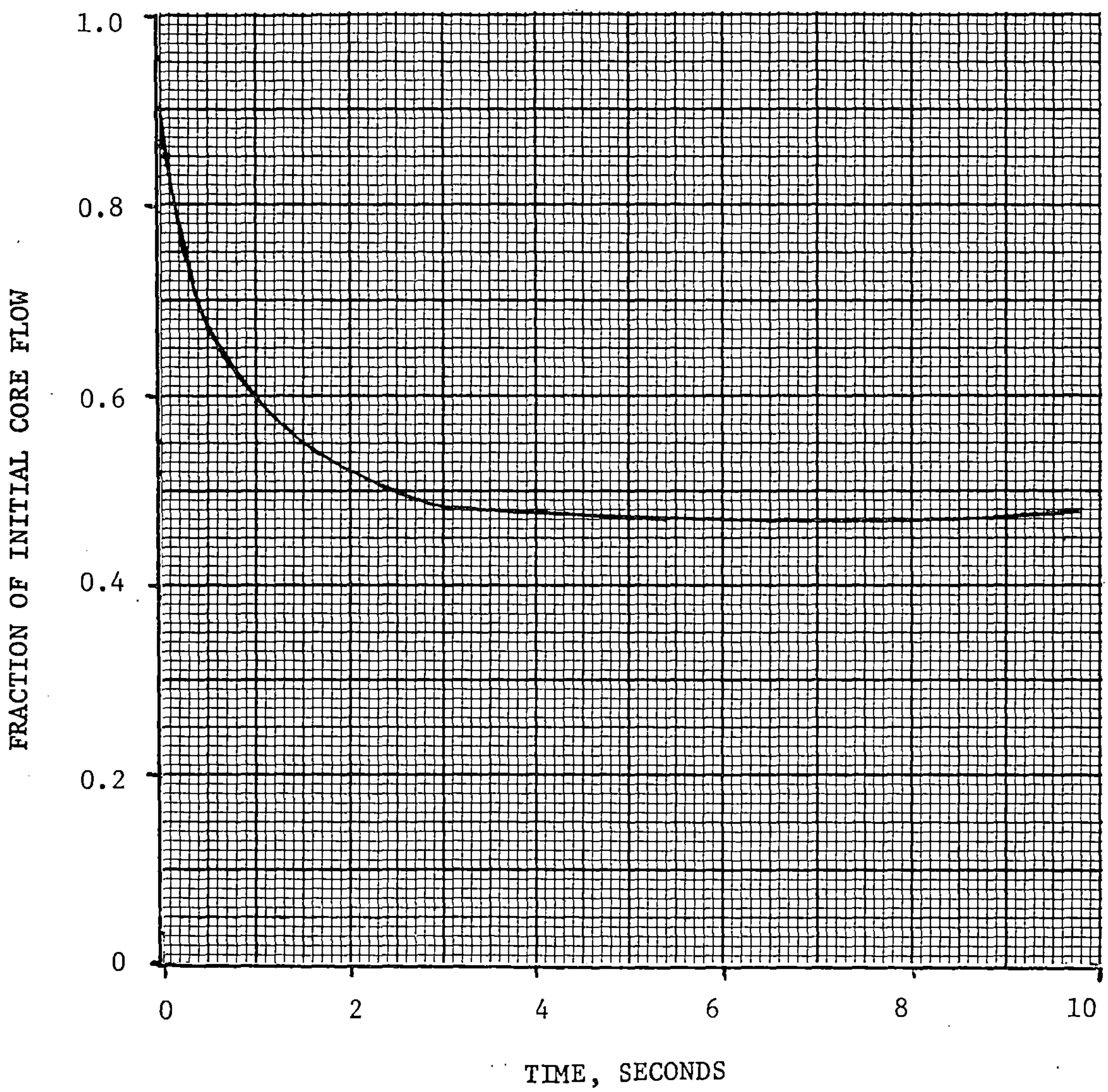
TWO LOOPS OPERATING,
NO LOOP STOP VALVES CLOSED,
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-30
12-1-69



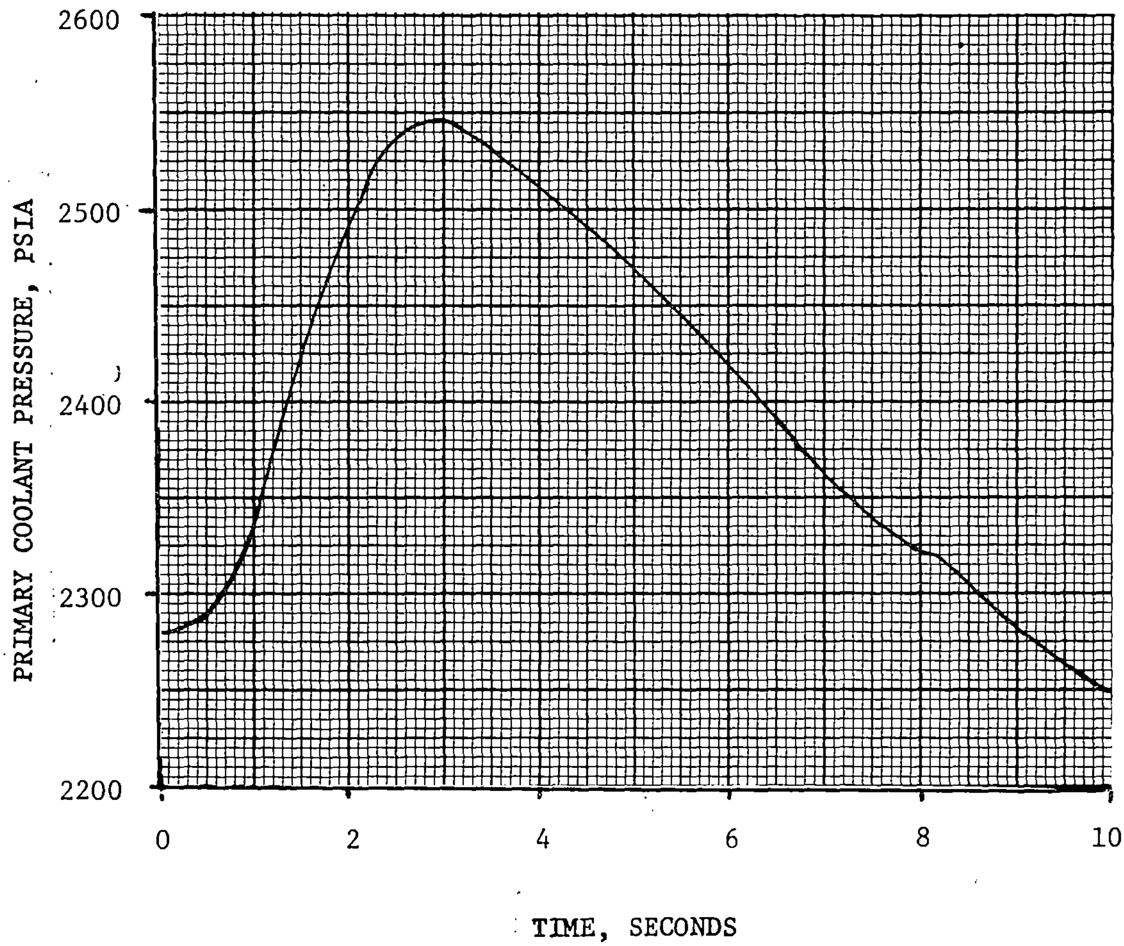
TWO LOOPS OPERATING,
NO LOOP STOP VALVES CLOSED,
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-31
12-1-69



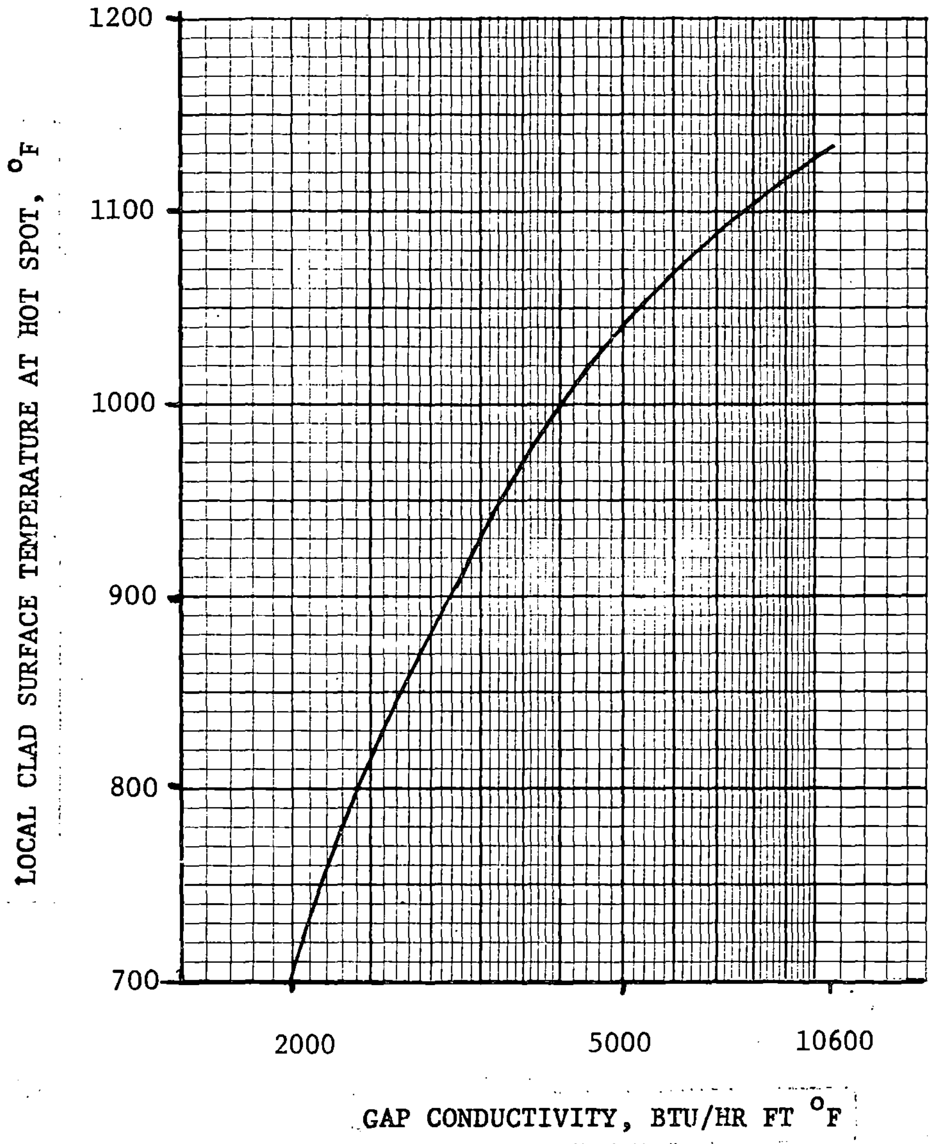
TWO LOOPS OPERATING
LOOP STOP VALVES CLOSED IN ONE LOOP
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-32
12-1-69



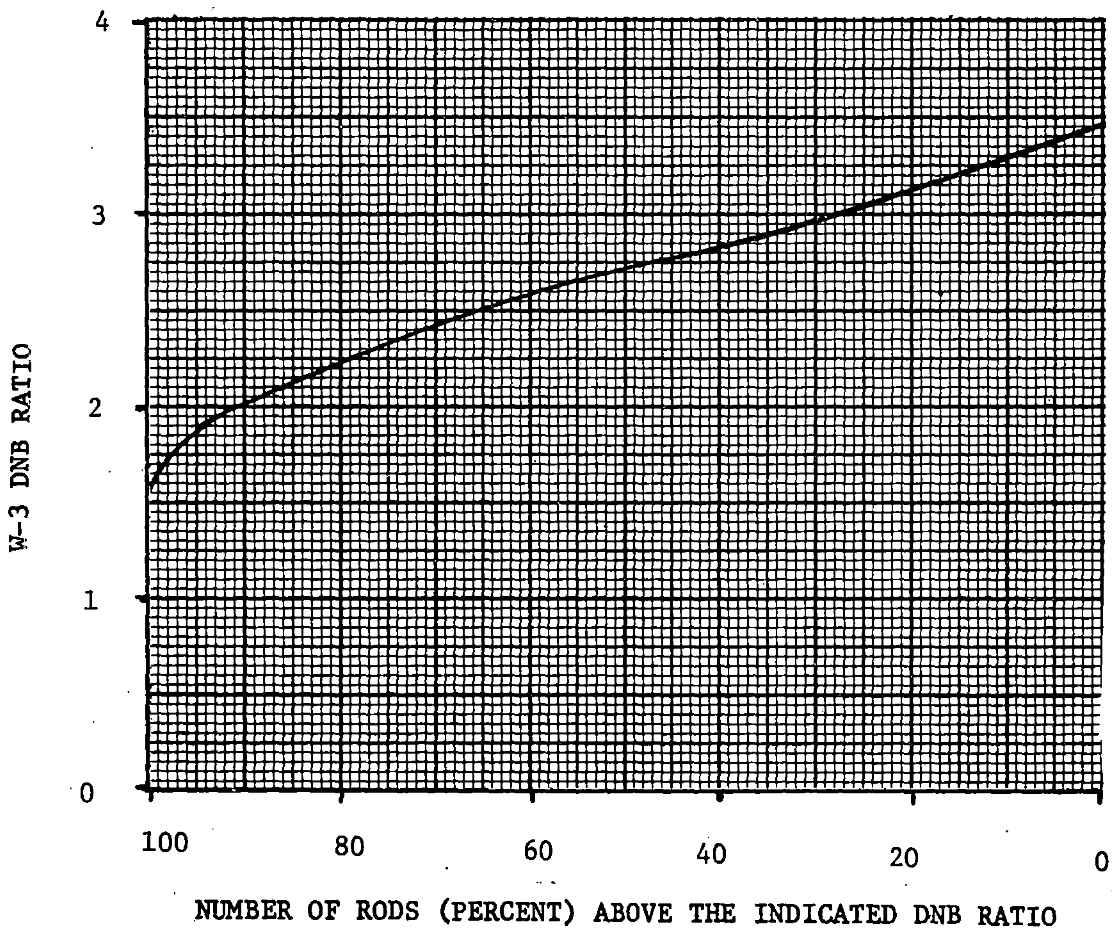
TWO LOOPS OPERATING,
LOOP STOP VALVES CLOSED IN ONE LOOP,
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-33
12-1-69



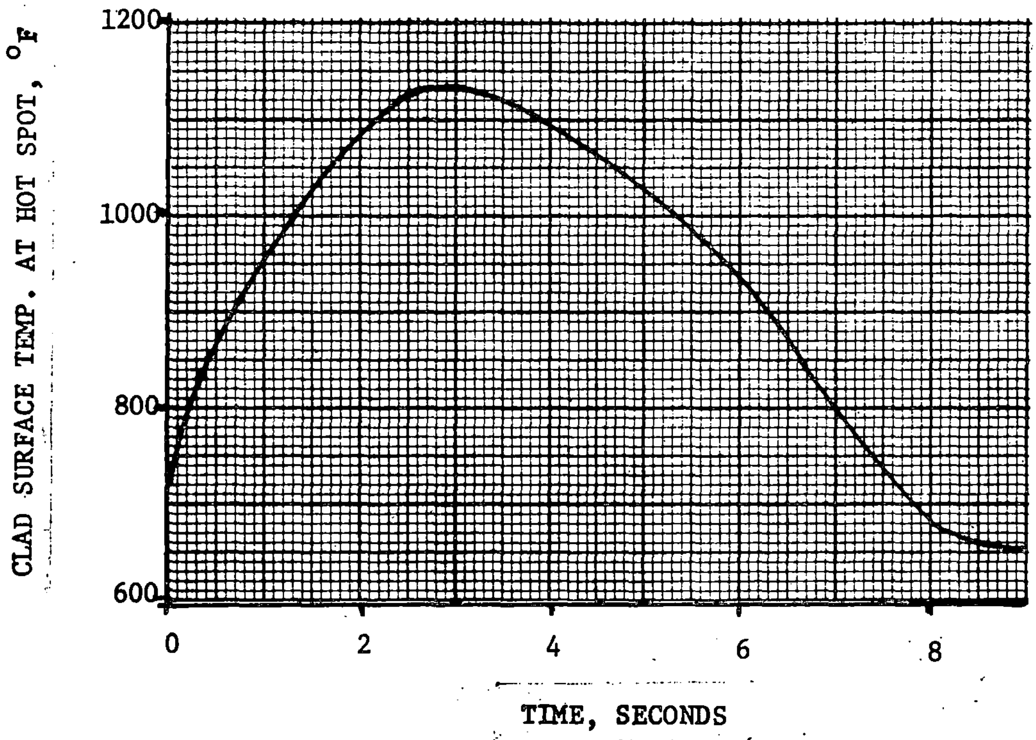
TWO LOOPS OPERATING,
LOOP STOP VALVES CLOSED IN ONE LOOP
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-35
12-1-69



TWO LOOPS OPERATING,
LOOP STOP VALVES CLOSED IN ONE LOOP,
LOCKED ROTOR INCIDENT

FIGURE 14.2.9-36
12-1-69



TWO LOOPS OPERATING,
LOOP STOP VALVES CLOSED IN ONE LOOP,
LOCKED ROTOR INCIDENT

14.2.10 Loss of External Electrical Load

The loss of external electrical load may result from an abnormal variation in network frequency, or other adverse network operating conditions. It may also result from a trip of the turbine generator or opening of the main breaker from the generator which fails to cause a turbine trip but causes a large,rapid Nuclear Steam Supply System load reduction by the action of the turbine control.

The unit is designed to accept a step loss of load from 100% to 50% load without actuating a reactor trip. The automatic steam bypass system, with 40% steam dump capacity to the condenser, is able to accommodate this load rejection by reducing the severity of the transient imposed upon the Reactor Coolant System. The reactor power is reduced to the new equilibrium power level at a rate consistent with the capability of the Rod Control System. The pressurizer relief valves may be actuated, but the pressurizer safety valves and the steam generator safety valves do not lift for the 50% step loss of load with steam dump.

In the event the steam bypass valves fail to open following a large load loss or in the event of a complete loss of load with steam dump operating, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal or the high pressurizer level signal. The steam generator shell side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the Reactor Coolant System and steam generator against overpressure for all load losses without assuming the availability of the steam bypass system. The steam dump valves will not be opened for load reductions of 10% or less. For larger load reductions they may open.

The most likely source of a complete loss of load on the Nuclear Steam Supply System is a trip of the turbine-generator. In this case, there is a direct reactor trip signal (unless below approximately 10% power) derived from either the turbine autostop oil pressure or a closure of the turbine stop valves. Reactor coolant temperatures and pressure do not significantly increase if the steam bypass system and pressurizer pressure control system are functioning properly. However, in this analysis, the behavior of the unit is evaluated for a complete loss of load from 102% of full power without a direct reactor trip primarily to show the adequacy of the pressure relieving devices and also to show that no core damage occurs. The Reactor Coolant System and Main Steam System pressure relieving capacities are designed to ensure safety of the unit without requiring the automatic rod control, pressurizer pressure control and/or steam bypass control systems.

Method of Analysis

The total loss of load transients are analyzed by employing a detailed digital computer program. The program describes the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level. The objectives of this analysis are to demonstrate margins of core protection and to demonstrate the adequacy of the unit pressure relieving devices. The variables shown in the figures, for example, were calculated by the program.

Initial Operating Conditions

The initial reactor power and Reactor Coolant System temperatures are assumed at their maximum values consistent with steady state, full power operation, including allowances for calibration and instrument errors. The initial Reactor Coolant System pressure is assumed at the minimum value consistent with steady state, full power operation, including allowances for calibration and instrument errors. This results in the maximum power difference for the load loss, and the minimum margin to core protection limits at the initiation of the total loss of load accident.

Moderator and Doppler Coefficients of Reactivity

The total loss of load is analyzed for both beginning-of-life and end-of-life conditions. At beginning-of-life a zero value of moderator coefficient is used and at end-of-life a -3.5×10^{-4} $\Delta k/°F$ value of moderator coefficient is used. A conservatively negative value of Doppler power coefficient is used for all cases.

Reactor Control

Two cases for both the beginning and end of life are analyzed as follows:

1. The reactor is assumed to be in normal automatic control with the control rod assemblies in the minimum incremental worth region.

2. The reactor is assumed to be in manual control with no control rod assembly insertion until a reactor trip occurs.

Steam Release

No credit is taken for the operation of any of the steam bypass (dump) valves or power operated steam generator relief valves. The steam generator pressure rises to the safety valve set point where steam release through safety valves limits secondary steam pressure at the set point.

Pressurizer Spray and Power Operated Relief Valves

Two cases for both the beginning and end of life are analyzed as follows:

1. Full credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting coolant pressure.

2. No credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting coolant pressure.

Results

The transient responses for a total loss of load from 102% full power operation are shown for four cases, two cases for the beginning of core life and two cases for the end of core life, in Figures 14.2.10-1 through 14.2.10-8.

Figures 14.2.10-1 and 14.2.10-2 show the transient responses for the total loss of load at beginning-of-life with zero moderator temperature coefficient assuming full credit for the pressurizer spray, pressurizer power operated relief valves, and automatic control rod assembly insertion. No credit is taken for steam dump. It can be seen from the transients that the power operated relief valve capacities are not large enough to prevent a high pressure trip for this case. The high pressure trip (assumed at a pressurizer pressure of 2425 psia) occurs 11.9 seconds after the loss of load, and the pressurizer pressure rises to a maximum of 2460 psia before decreasing, following the reactor trip. The minimum DNBR occurs at the beginning of the transient and is 1.73, which is well above the 1.3 design value. The steam generator safety valve setpoint is reached at 10 seconds after the start of the transient and the valves are required to relieve a peak flow of 80% of the flow at rated full power at 17 seconds.

Figures 14.2.10-3 and 14.2.10-4 show the responses for the total loss of load at end of life with the most negative moderator temperature coefficient (-3.5×10^{-4} $\Delta k/°F$). The rest of the unit operating conditions are the same as the case above. The pressurizer pressure increases to a peak of 2357 psia. The combination of pressurizer spray, pressurizer power operated relief valves, and control rod assembly insertion were sufficient to prevent the pressurizer pressure from reaching the high pressure trip point. The pressure decreases rapidly after about 20 seconds as a result of the large reduction in nuclear power. The increase in coolant average temperature is about 18°F. The DNBR increases throughout the transient and never drops below the initial value of 1.73. The steam generator safety valve setpoint is reached at 10 seconds after the start of the transient and the valves are required to relieve a peak flow of 72% of the flow at rated full power at about 20 seconds.

The safety valves were not actuated in the transients shown in Figures 14.2.10-1 through 14.2.10-4 which include the effects of pressurizer spray, pressurizer power operated relief valves, and automatic control rod assembly insertion.

The total loss of load accident was also studied assuming the unit to be initially operating at 102% of full power with manual rod control. In this case there is no control rod assembly insertion until a trip occurs. In addition, no credit is taken for the pressurizer spray, pressurizer power operated relief valves, or steam dump. The reactor is tripped on high pressure (assumed at 2425 psia) Figures 14.2.10-5 and 14.2.10-6 show the beginning of life transient with zero moderator coefficient. The nuclear power remains essentially constant at 102% full power until the reactor is tripped at 6.2 seconds after loss of load. The peak pressurizer pressure is 2540 psia and the maximum surge rate is 17.3 ft^3/sec. This is compared to a pressurizer safety valve capacity of approximately 30.5 ft^3/sec. The steam generator safety valve setpoint is reached at 10 seconds after the start of the transient and the valves are required to relieve a peak flow of 67% of the flow at rated full power at 12 seconds. Figures 14.2.10-7 and 14.2.10-8 are the transients at end of life with the other assumptions being the same as in Figures 14.2.10-5 and 14.2.10-6. The peak pressurizer pressure is 2534 psia and the maximum surge rate is 15.1 ft^3/sec. In this case, the steam generator safety valve setpoint is reached at 10 seconds and the valves must relieve a peak flow of 65% at 12 seconds.

Conclusions

The analysis indicates that a total loss of load without a direct or immediate reactor trip presents no hazard to the integrity of the Reactor Coolant System or the Main Steam System. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within design limits. The integrity of the core is maintained by the high pressurizer pressure reactor trip. The minimum DNBR never fell below 1.73 which is well above the 1.3 design value.

FIGURE 14.2.10-1
12-1-69

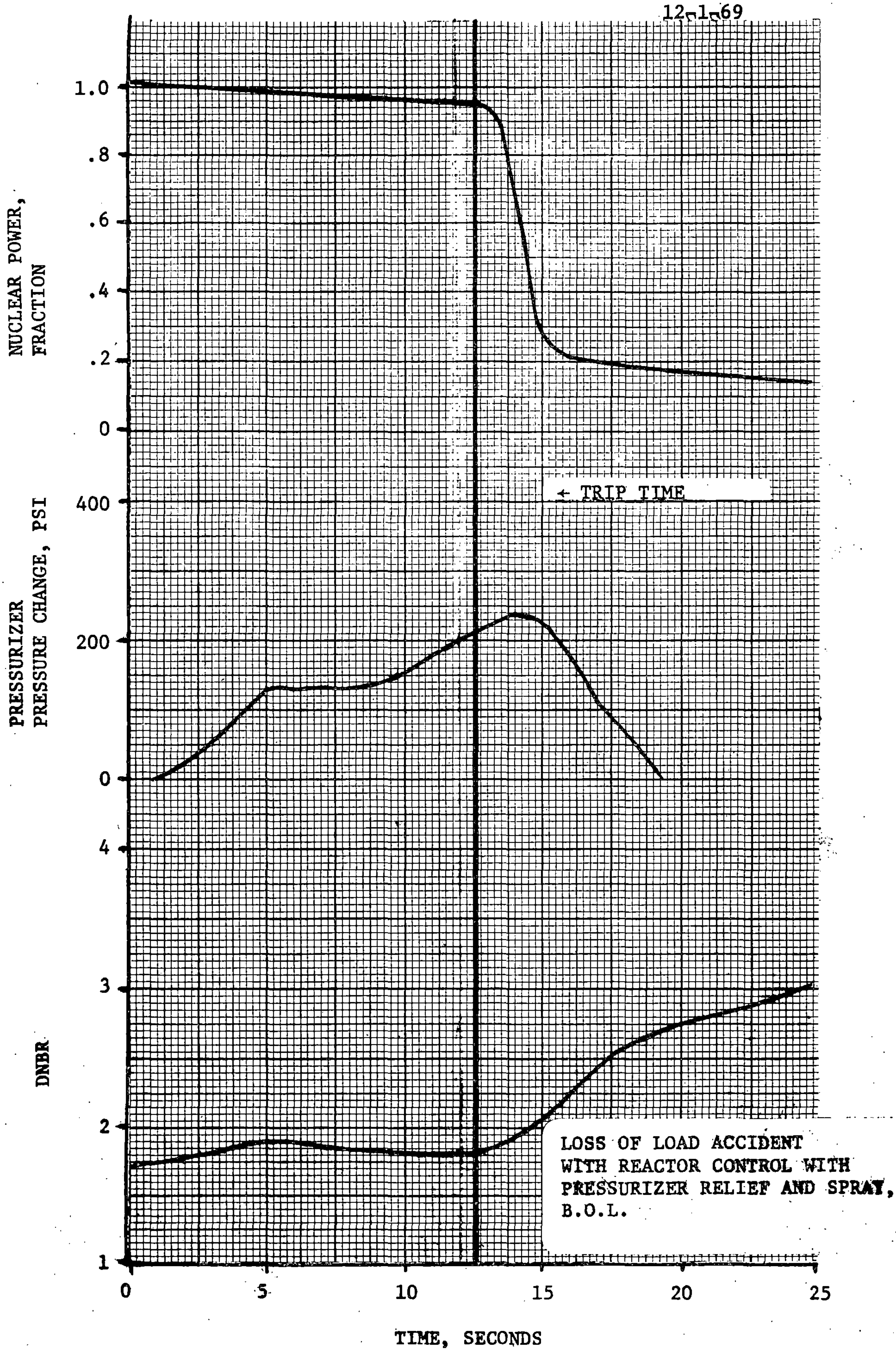
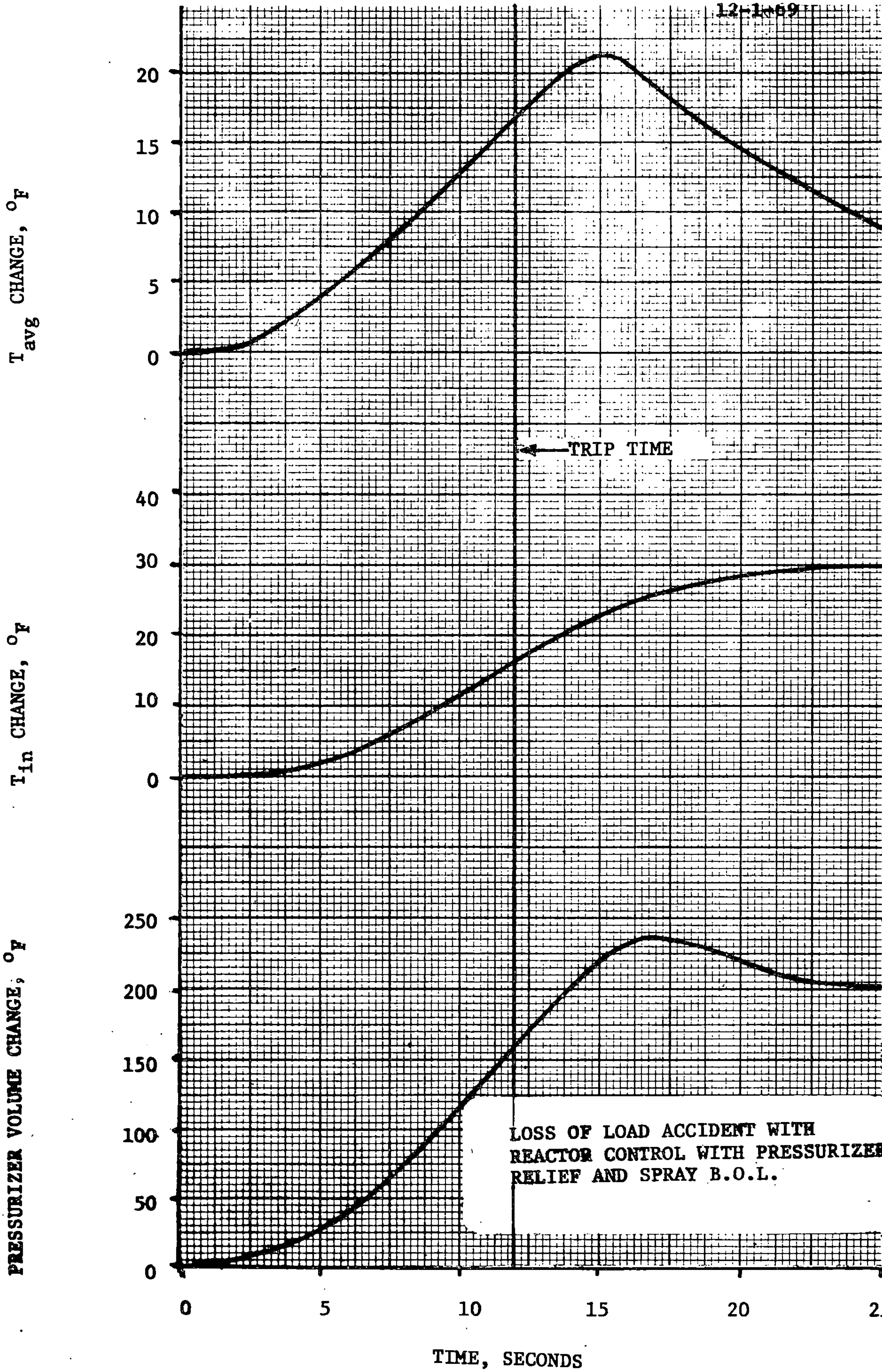


FIGURE 14.2.10-2
12-1-69



LOSS OF LOAD ACCIDENT WITH REACTOR CONTROL WITH PRESSURIZER RELIEF AND SPRAY B.O.L.

FIGURE 14.2.10-3
12-1-69

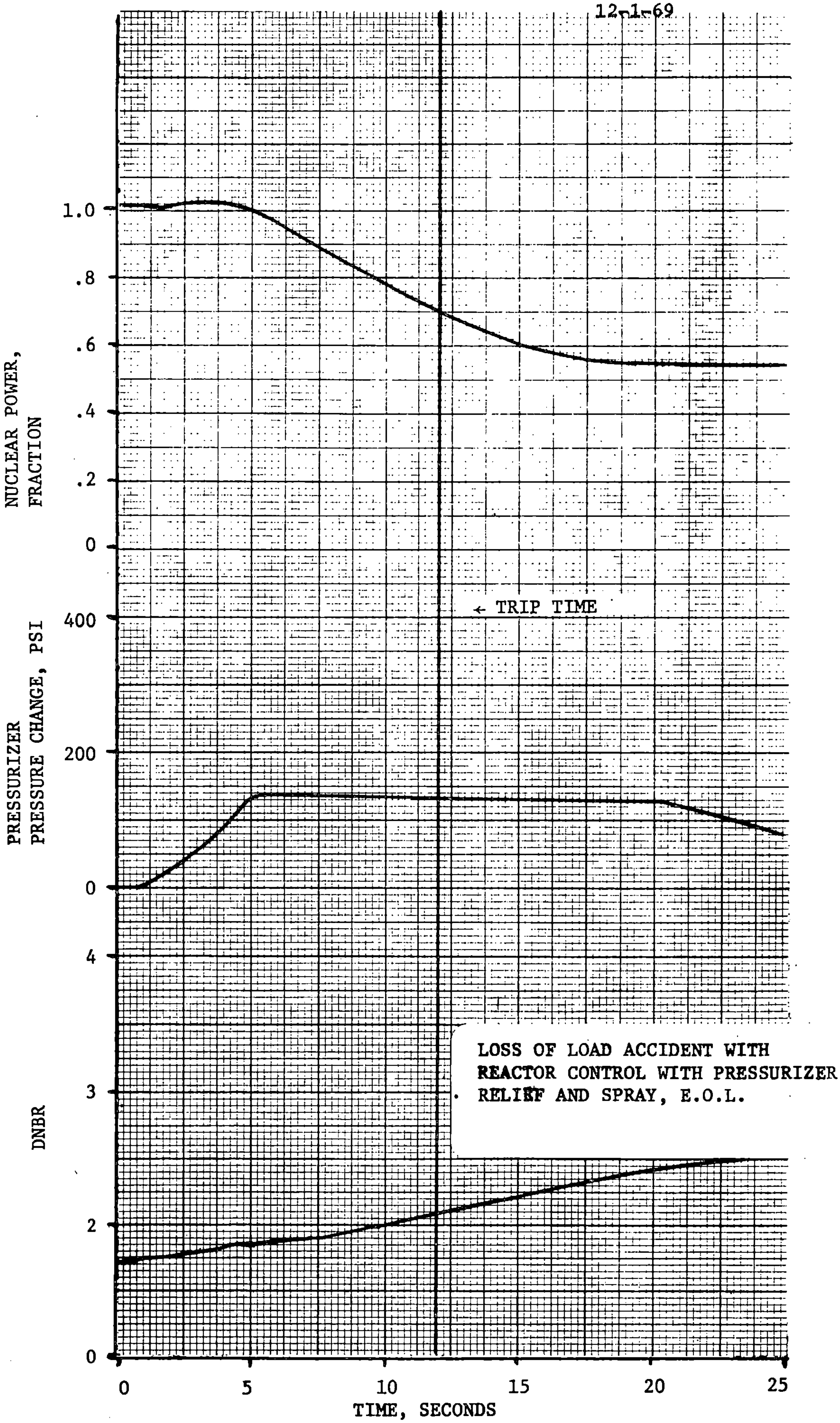
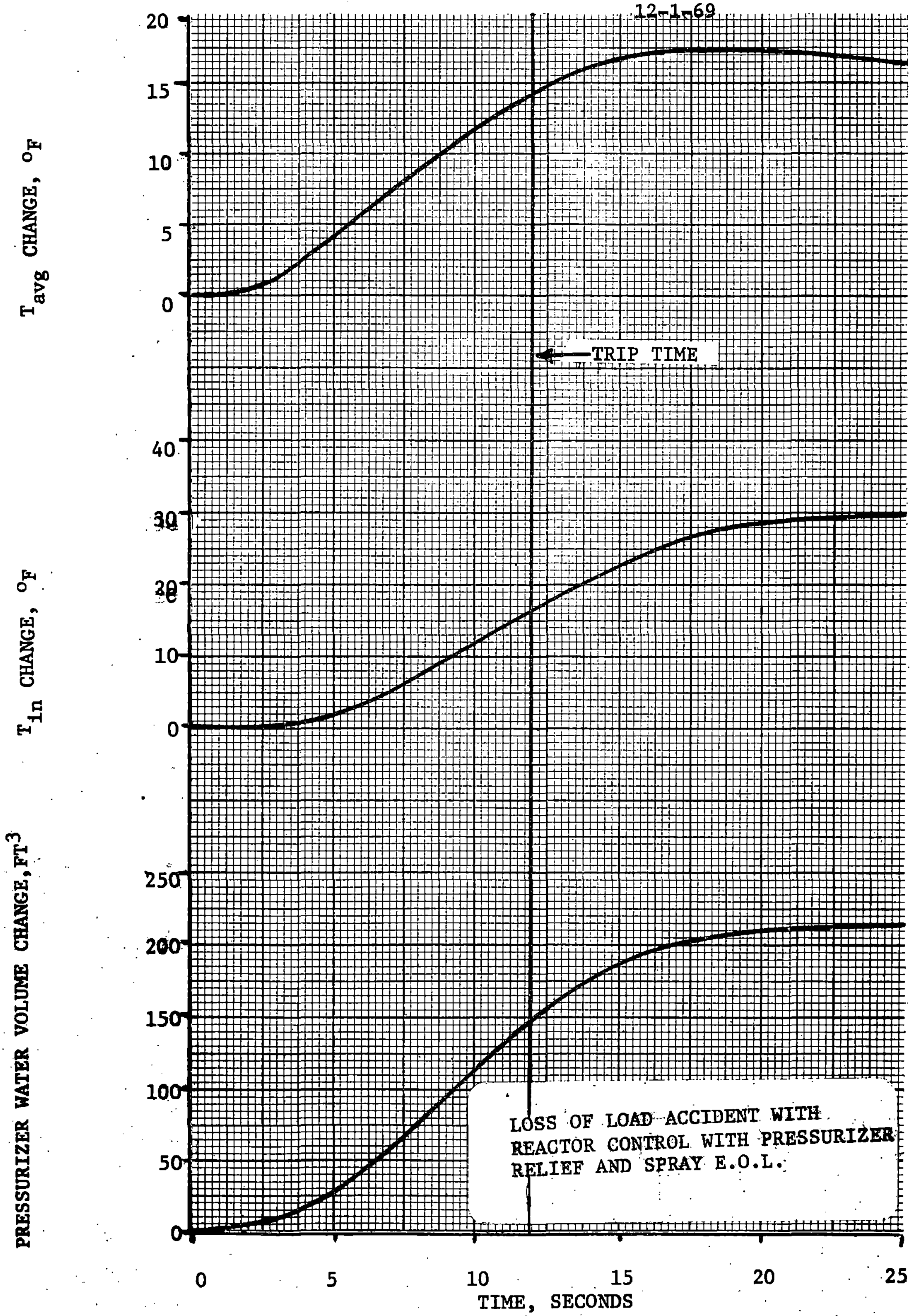


FIGURE 14.2.10-4
12-1-69



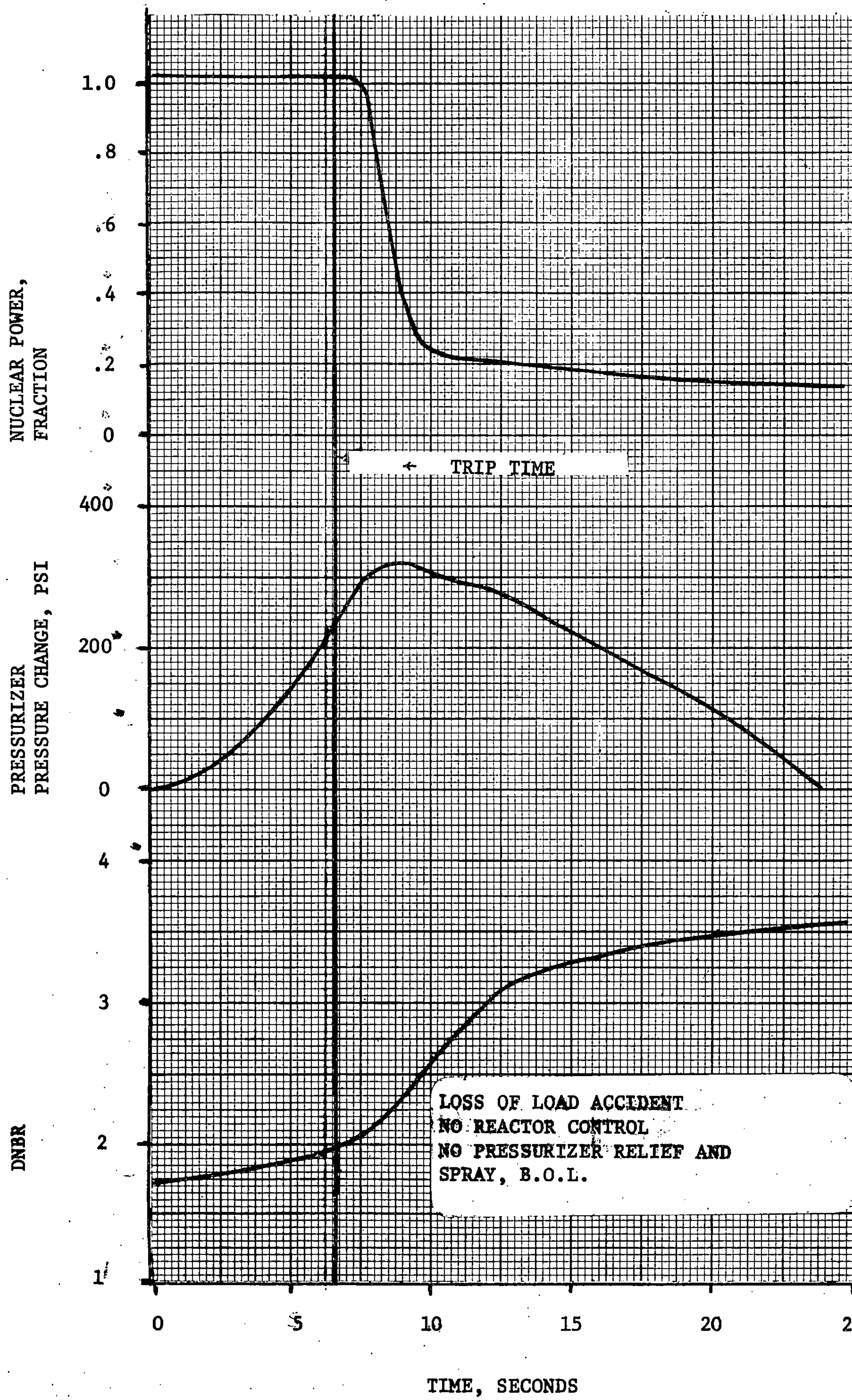

NUCLEAR POWER,
FRACTION
1.0
.8
.6
.4
.2
0
← TRIP TIME
PRESSURIZER
PRESSURE CHANGE, PSI
400
200
0
DNBR
4
3
2
1
LOSS OF LOAD ACCIDENT
NO REACTOR CONTROL
NO PRESSURIZER RELIEF AND
SPRAY, B.O.L.
0
5
10
15
20
25
TIME, SECONDS

FIGURE 14.2.10-6
12-1-69

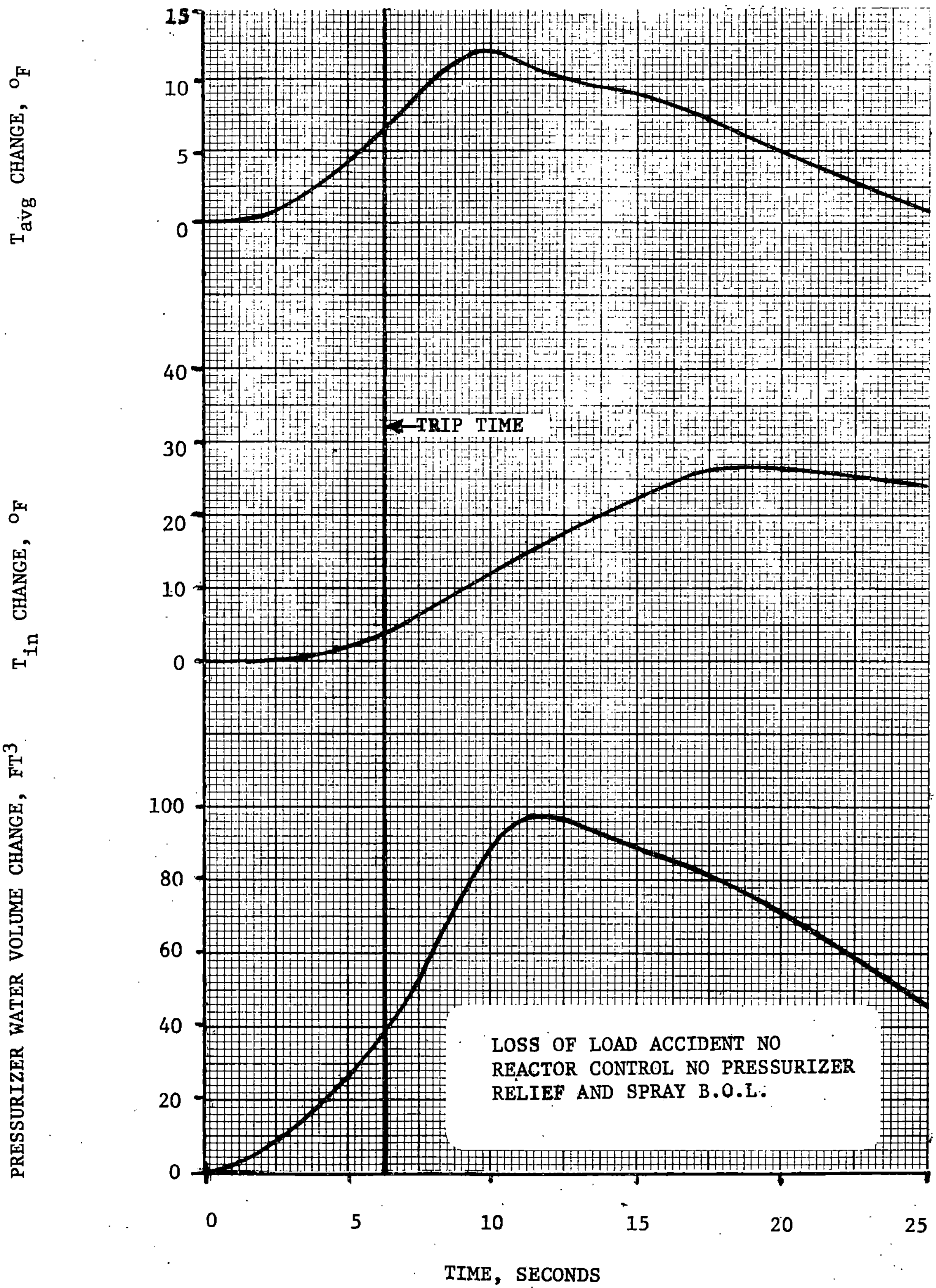


FIGURE 14.2.10-7
12-1-69

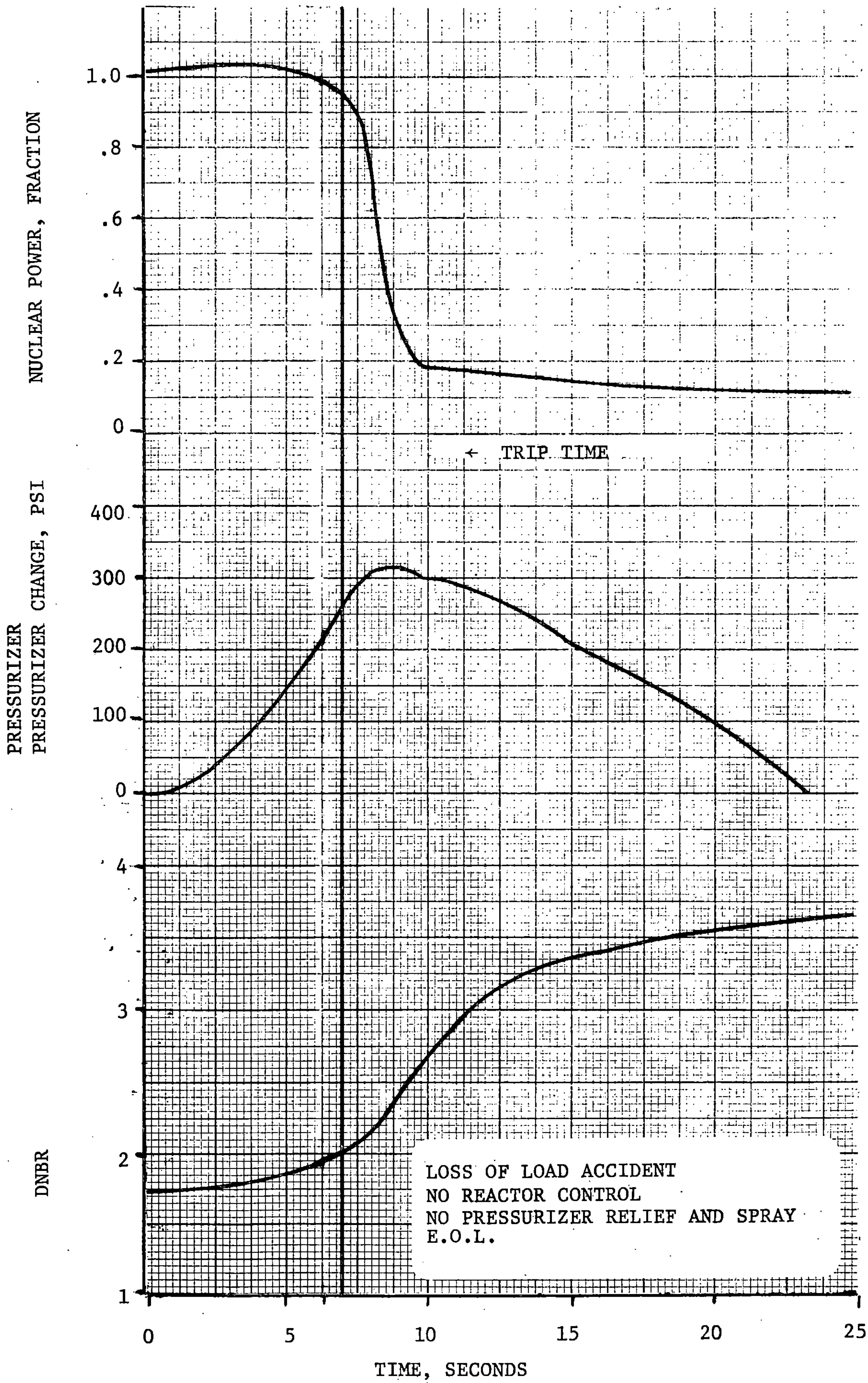
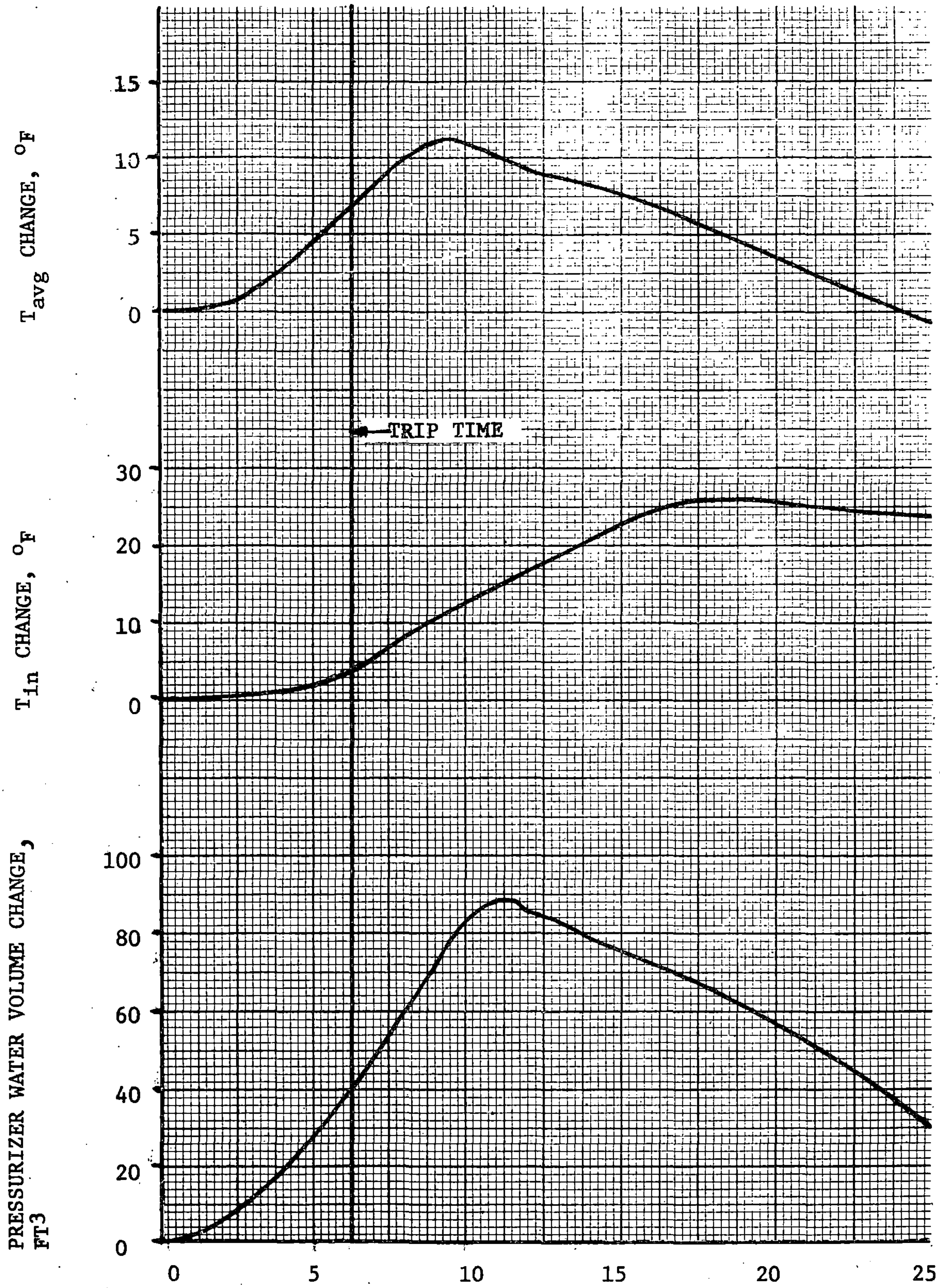


FIGURE 14.2.10-8
12-1-69



LOSS OF LOAD ACCIDENT
NO REACTOR CONTROL
NO PRESSURIZRR RELIEF AND SPRAY
E.O.L.

14.2.11 LOSS OF NORMAL FEEDWATER

A loss of normal feedwater (from a pipe break, pump failures, valve malfunctions, or loss of outside a-c power) results in a loss in capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this incident, reactor core damage could possibly occur from a sudden loss of heat sink. If an alternate supply of feedwater were not supplied to the unit, residual and sensible heat following reactor trip would heat the Reactor Coolant System water to the point where water relief from the pressurizer relief valves occurs. Loss of significant water from the Reactor Coolant System could conceivably lead to core damage.

The following provides the necessary protection against a loss of normal feedwater.

1. Reactor trip on low-low water level in any steam generator.

2. Reactor trip on main steam flow-feedwater flow mismatch in coincidence with low water level in any steam generator.

3. Two motor driven auxiliary feedwater pumps (350 gpm each) which are started automatically on:

(a) Low-low level in any steam generator, or

(b) Opening of both feedwater pump circuit breakers, or

(c) Any safety injection signal, or

(d) Loss of all a-c power, or

(e) Manually

4. One turbine driven pump (700 gpm) which is started automatically on

(a) Low-low level in 2/3 steam generators, or

(b) Loss of voltage on all 4160 V. busses, or

(c) Manually

The motor driven auxiliary feedwater pumps are supplied by the diesel generators if a loss of outside power occurs, and the turbine-driven pump uses steam from the steam generators. The turbine exhausts the steam to the atmosphere. The auxiliary feedwater pumps take suction directly from the buried 100,000 gal. condensate storage tank for delivery to the steam generators.

The above provides functional diversity in equipment and control logic to ensure that reactor trip and automatic auxiliary feedwater flow will occur following any loss of normal feedwater including that caused by a loss of a-c power.

Method of Analysis

The analysis was performed using a digital simulation of the unit to show that following a loss of normal feedwater, the auxiliary feedwater system is adequate to remove stored and residual heat.

The following assumptions were made:

1. The steam generator water level (in all steam generators) at the time reactor trip occurs is at the lowest level which will result in reactor trip and automatic initiation of the auxiliary feedwater flow. The initial water level in the analysis is assumed to be at the lower narrow range level tap.

2. The unit is initially operating at 102% of 2546 MWt (the maximum calculated turbine rating).

3. A heat transfer coefficient in the steam generators assuming Reactor Coolant System natural circulation.

4. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip.

5. Only one motor driven auxiliary feedwater pump is available at one minute after the incident.

6. Auxiliary feedwater is delivered to only two steam generators.

7. Secondary system steam relief through the self actuated safety valves (steam relief will, in fact, be through the power operated relief valves or condenser dump valves for most cases of loss of normal feedwater. However, these were assumed to be unavailable in the analysis).

Results

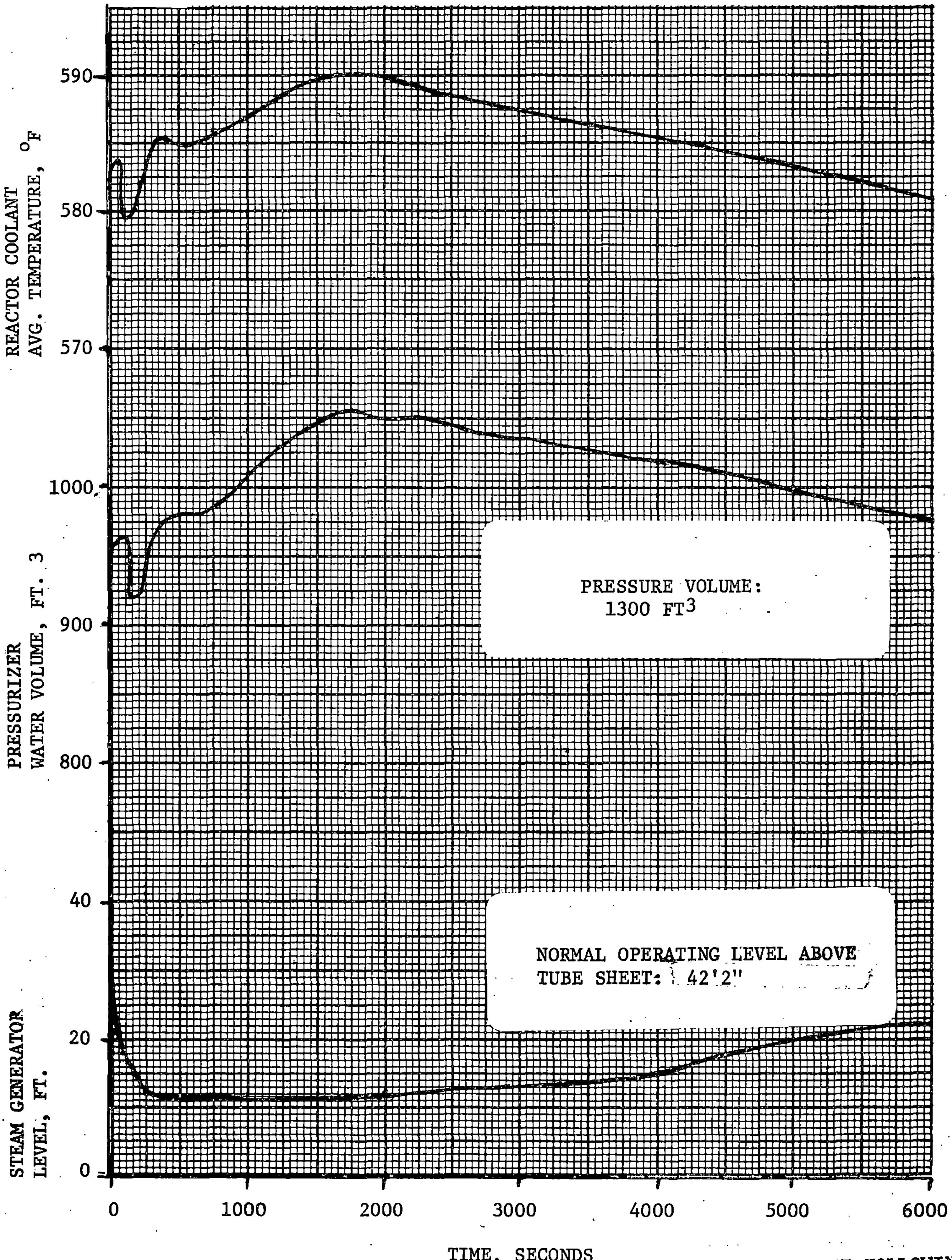
Figure 14.2.11-1 shows the unit parameters following a loss of normal feedwater incident with the assumptions listed above. Following the reactor and turbine trip from 1.02 x 2546 MWt, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. One minute following the initiation of the low-low level trip the auxiliary feedwater pump is automatically started reducing the rate of water level decrease. The capacity of the auxiliary feedwater pump is such that the water level in the steam generators being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the primary system relief or safety valves.

From Figure 14.2.11-1 it can be seen that at no time is the tube sheet uncovered in the steam generators receiving auxiliary feedwater flow and at no time is there water relief from the pressurizer. If the auxiliary feed delivered is greater than that of one motor driven pump, the initial reactor power is less than 102% of 2546 MWt, or the steam generator water level in one or more steam generators is above the low-low level trip point at the time of trip then the result will be a steam generator minimum water level higher than shown and an increased margin to the point at which reactor coolant water relief from the pressurizer occurs.

Conclusions

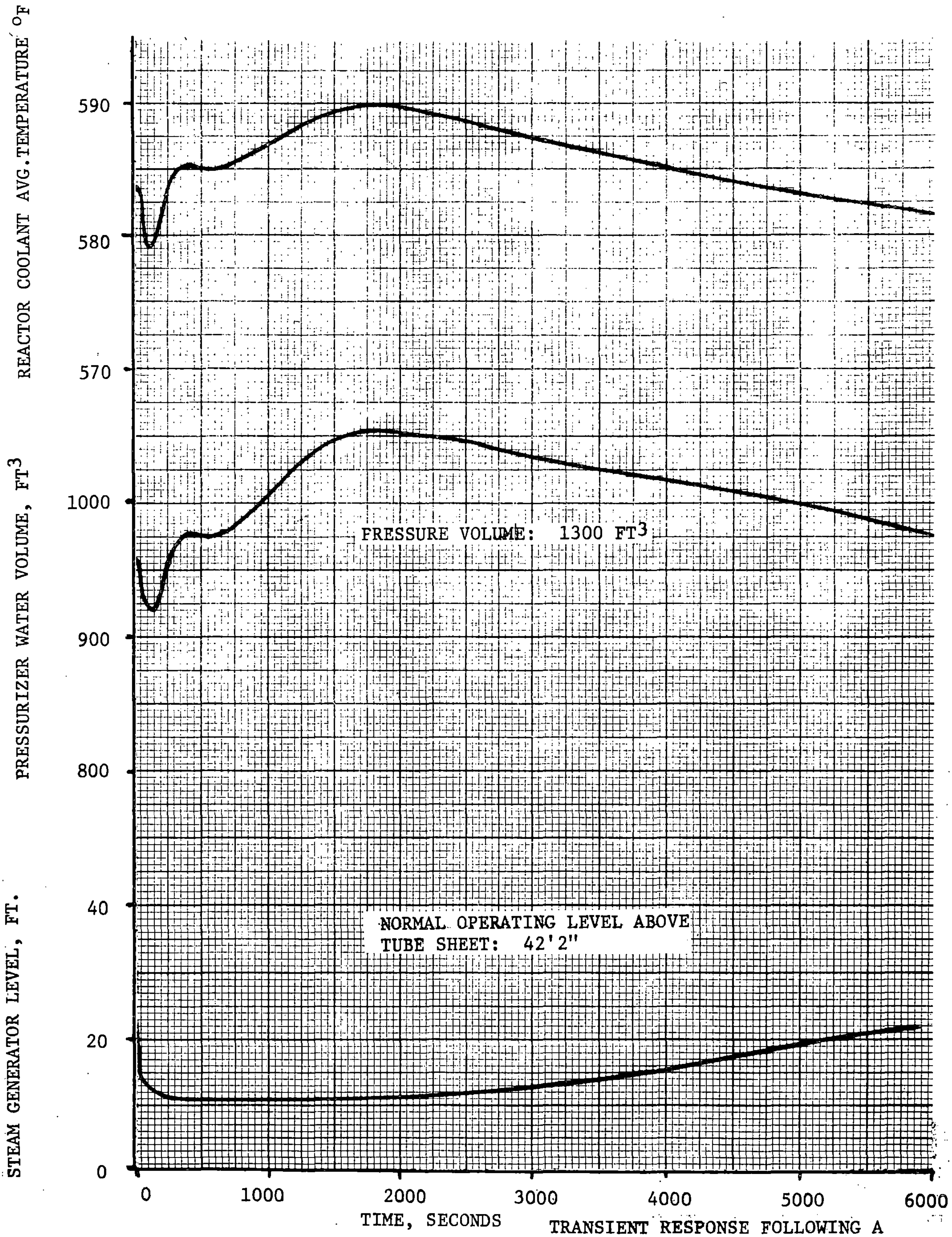
The loss of normal feedwater does not result in any adverse condition in the core, because it does not result in water relief from the pressurizer relief or safety valves, nor does it result in uncovering the tube sheets of the steam generators being supplied with water.

FIGURE 14.2.11-1
12-1-69



TRANSIENT RESPONSE FOLLOWING A LOSS OF NORMAL FEEDWATER WITH ONE 350 GPM AUXILIARY FEED PUMP DELIVERING TO TWO STEAM GENERATORS AT ONE MINUTE

FIGURE 14.2.11-2
12-1-69



TRANSIENT RESPONSE FOLLOWING A LOSS OF NORMAL FEEDWATER WITH ONE 350 GPM AUXILIARY FEED PUMP DELIVERING TO TWO STEAM GENERATORS AT ONE MINUTE

14.2.12 LOSS OF ALL AC POWER TO THE STATION AUXILIARIES

In the event of a complete loss of off-site power and turbine trip, there would be a loss of power to the unit auxiliaries, i.e. the reactor coolant pumps, main feedwater pumps, etc. The events following a loss of a-c power with turbine trip are described in the sequence below.

1. Unit vital instruments are supplied by the emergency power sources.

2. As the steam system pressure subsequently increases, the steam system power relief valves are automatically opened to the atmosphere. (Steam bypass to the condenser is assumed not available since the steam bypass is not required for reactor protection.)

3. If the steam flow rate through the power relief valves is not sufficient (or if the power relief valves are not available), the steam generator self-actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual heat produced in the reactor.

4. As the no load temperature is approached, the steam power relief valves (or self actuated safety valves if the power relief valves are not available for any reason) are used to dissipate the residual heat and to maintain the unit at the hot shutdown condition.

5. The emergency diesel generators will start on loss of voltage on the emergency 4160 volt busses to supply unit vital loads.

The auxiliary feedwater system is started automatically as discussed in Section 14.2.11. The steam driven auxiliary feedwater pump uses main steam and exhausts to the atmosphere. The motor driven auxiliary feedwater pumps are supplied by power from the diesel generators. The pumps take suction directly from the buried, 100,000 gal. condensate storage tank for delivery to the steam generators. The auxiliary feedwater system ensures feedwater supply of at least 350 gpm upon loss of power to the station auxiliaries, since the auxiliary steam turbine driven feedwater pump has a capacity of 700 gpm and the motor driven auxiliary feedwater pumps have a capacity of 350 gpm each.

The steam driven pump can be tested at any time by admitting steam to the turbine driver. The motor driven pumps also can be tested at any time. The valves in the system can be operationally tested at any time.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. The natural circulation flow was calculated for the conditions of equilibrium flow and maximum loop flow impedance. The model used has given results within 15% of the measured flow values obtained during natural circulation tests conducted at the Yankee-Rowe plant and has also been confirmed at San Onofre and Connecticut Yankee. The natural circulation flow ratio as a function of reactor power is given in Table 14.2.12-1.

The average temperature, pressurizer water volume, and steam generator level assuming the most conservative initial unit conditions and equipment availability are shown in Figure 14.2.11-1 for a loss of normal feedwater including a loss of outside a-c power, and Reactor Coolant System natural circulation. It is shown in Section 14.2.11 that a loss of normal feedwater from any cause including a loss of outside a-c power does not result in water relief from the pressurizer relief or safety valves.

Conclusion

The loss of a-c power to the station auxiliaries does not cause any adverse condition in the core since it does not result in water relief from the pressurizer relief or safety valves.

TABLE 14.2.12-1

NATURAL CIRCULATION REACTOR COOLANT FLOW VS REACTOR POWER

Reactor Power % Full Power	Reactor Coolant Flow % Nominal Flow
3.5	5.0
3.0	4.7
2.5	4.4
2.0	4.1
1.5	3.8
1.0	3.3

14.2.13 LIKELIHOOD OF TURBINE-GENERATOR UNIT OVERSPEED

The present technology of rotor forging machinery and inspection techniques guarantees practically defect-free turbine rotors. Further, Westinghouse conservative design eliminates any harmful stress-concentration point, as the no-failure record of Westinghouse turbine-generator confirms.

Due to the redundancy and reliability of the turbine control protection system and of the Main Steam System, the probability of a unit overspeeding above the design value, i.e., 130%, is very remote.

A description and operation of the electro-hydraulic governing system is given in Section 7.3.

Due to conservative design, very careful rotor forgings procurement, precision machining rigid inspection and reliable turbine control, Westinghouse turbine-generator units have never experienced a massive failure.

A survey of the available literature on turbine-generator unit failure shows that the last massive failure of a turbine generator unit occurred more than eleven years ago. The causes of failure were identified at that time, and provisions were adopted to prevent the recurrence of massive failures. The record since that time demonstrates the soundness of these provisions and correct design.

The no-failure record of Westinghouse turbine generator units, plus the experience gained from the reference incidents, together with the improvement in the design and inspection techniques in the past eleven years indicates that the likelihood of massive turbine-generator failure is extremely remote. With regard to design and inspection techniques, it is worthwhile to mention that a technical committee of forging suppliers and equipment manufacturers was formed about 15 years ago under ASTM to study turbine and generator rotor failures. This group developed the high-toughness NiCrMoV material, now used in all turbine rotors and disks. This Task Force[1] has been very active in making additional improvements in quality and soundness of large forgings and is still in being.

The survey of the literature on massive turbine failures in the last 25 years indicates that all of them occurred between 1953 and 1958.

This survey has pointed out that the rare events of a catastrophic failure of turbines fell into one of two categories:

1. Failure by overstressing arising from accidental and excessive overspeed; and

2. Failure, due to defects in the material, occurring at about normal speed.

No failure falling in the first category occurred in the USA. The only two documented examples occurred in the United Kingdom. Both incidents were caused by the main steam admission valves sticking in the open position after full load rejection, because of impurities in the turbine control and lubrication oil. The probability of this occurrence in this unit is very remote as previously pointed out.

In addition to the provisions in the design of the turbine control and protection system during unit operation, valves are exercised on a periodic basis, to further preclude the possibility of a valve stem sticking. Analysis of oil samples are performed regularly.

The turbine is periodically overspeeded to check the tripping speed. The remaining tripping devices are routinely checked.

The causes of the failures that fall in the second category, i.e., failures due to defects in the material occuring about normal speed, are summarized in Appendix 14 A. The causes of failure were completely identified and, if the ultrasonic test were used as one of the bases for rejection or acceptance of forging, many of them would not have occurred. Further, the stress concentration points that initiated failure in some units are strictly correlated to the peculiar design characteristics of those units. As pointed out in Appendix 14 A, these discontinuities are not present in Westinghouse units.

Westinghouse specifies the quality and method of manufacturing of the purchased forgings. Written specifications cover the manufacturing process, the chemical and mechanical properties, the tests to be performed, etc. Specifically, the tests performed are both destructive and non-destructive in nature. The destructive tests include tension tests, impact tests, and transition temperature measurement tests. The tension specimens are taken in a radial and/or longitudinal direction. The tensile properties are determined in accordance with ASTM A-370 on a Standard Round 1/2 inch Diamter 2 Inch Gage Length Test specimen. The yield strength is taken as the load per unit of original cross section at which the material exhibits an offset of 0.2 percent of the original length. The Charpy impact specimens are taken in a radial direction, and the minimum impact strength at room temperature measured. The transition temperature is determined from 6 specimens tested at different temperatures in accordance with ASTM A-443. The specimens are taken in a radial direction and machined in such a manner that the V-notch is parallel to the forging axis. Two specimens are machined from each test bar. All specimens are taken following heat treatment. Curves of impact strength and percent brittle failure versus test temperature are drawn. The non-destructive tests include bore inspection, sulfur printing, magnetic particle test, thermal stability test, and ultrasonic tests.

The bores are visually inspected and the walls of the finished bores shall be free from cracks, pipe shrinkage, gas cavities, non-metallic inclusions, injurious scratches, tool marks and similiar defects.

A magnetic particle test is made on each forging to demonstrate the freedom from surface discontinuities. The end faces of the main body and down over and beyond the fillets joining the main body to the shaft portions are magnetic particle tested. The bore is also magnetic particle tested at a high sensitivity level in accordance with ASTM A-275. These inspections are done by Westinghouse inspectors prior to Westinghouse accepting these forgings. After final machining by Westinghouse, rotors are again magnetic particle inspected on the external surfaces by Westinghouse.

The face of the test prolongations at each end of the rotor body or an area on the end faces of the rotor body equivalent to the test prolongations is sulfur printed to determine the freedom from undue ingot corner segregation and excessive sulfide inclusions.

A thermal stability test is performed on the forging at the place of manufacture after all heat treatment has been completed.

The forgings are ultrasonically inspected at the place of manufacture by Westinghouse inspectors.

Based on conservative design, precision machining, reliable turbine control system, careful rotor forging procurement and rigid inspection, the probability of a combination of excessive overspeed, new-born large forging defects, and operating temperature below the transition temperature is considered practically zero.

REFERENCES

1. Curran, R. M., "History of the Special ASTM Task Force on Large Turbine and Generator Rotors," ASTM Meeting, Purdue University, 1965.

14.3.1 STEAM GENERATOR TUBE RUPTURE

14.3.1.1 General

The accident examined is the complete severance of a single steam generator tube adjacent to the tube sheet. It is assumed that the accident takes place at power and while the reactor coolant is contaminated with fission products corresponding to continuous operation with one percent of fuel rods defective. The accident leads to contamination of the secondary system due to leakage of radioactive coolant from the Reactor Coolant System and in the event of a co-incident loss of outside power there will be, in addition, a discharge of activity to the atmosphere through the steam generator safety and/or power operated relief valves.

The steam generator tube material is Inconel 600 and as the material is highly ductile it is considered that the assumption of a complete severance is extremely conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the Steam and Power Conversion System is subject to continual surveillance and an accumulation of minor leaks which approach the equivalent of one complete tube rupture is not permitted during unit operation.

The main objective of the Operator should be to determine that a steam generator tube rupture has occurred and to identify and isolate the faulty steam generator on as short a time scale as possible in order to minimize contamination of the secondary system and ensure termination of activity discharge to the atmosphere. The recovery procedure can be carried out in such a time scale as to ensure that break flow to the secondary system is terminated before water level in the affected steam generator can rise into the main steam pipe and it can be seen that sufficient indications and controls are provided to enable the Operator to carry out these functions satisfactorily.

14.3.1.2 Method of Analysis and Description of Accident

The following sequence of events is initiated by a tube rupture:

1. Pressurizer low pressure and low level alarms are actuated, and prior to unit trip, charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow mismatch before trip as feedwater flow to the affected steam generator is reduced due to the additional break flow which is now being supplied to that steam generator.

2. Loss of reactor coolant inventory leads to falling pressure and level in the pressurizer and eventually a reactor trip signal is generated by low pressurizer pressure. Automatic unit cooldown following reactor trip leads to a rapid change of pressurizer level, and the safety injection signal, initiated by co-incident low pressurizer pressure and level, follows soon after the reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates auxiliary feedwater addition.

3. The steam generator blowdown liquid monitor and the air ejector radiation monitor alarms, indicating the passage of reactor coolant into the secondary system, and a transfer signal is initiated which causes the air ejector exhaust from the condenser to be discharged to the containment, thereby terminating any direct atmospheric release.

4. The unit trip automatically shuts off steam supply to the turbine and if outside power is available the condenser bypass valves open permitting steam dump to the condenser. In the event of a coincident station blackout, the condenser bypass valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase and discharge steam to the atmosphere through the steam generator safety and/or power operated relief valves.

5. Following a unit trip, the continued action of auxiliary feedwater supply and borated safety injection flow (supplied from the Refueling Water Storage Tank) provides a heat sink which eventually absorbs decay heat. Thus, steam bypass to the condenser, or in the case of loss of outside power steam relief to atmosphere, is discontinued on a time scale which is dependent on the exact amount of emergency equipment (safety injection pumps and auxiliary feedwater pumps) operating.

6. Safety injection flow results in increasing pressurizer water level. The time after trip at which the operator can clearly see returning level in the pressurizer is also dependent upon the amount of operating auxiliary equipment.

14.3.1.3 Results

It is expected that in the event of a steam generator tube rupture accidentthe the operator is able to determine rapidly the accident type and prepare to isolate the affected steam generator when it has been identified by stopping the reactor coolant pump and closing the reactor coolant isolation valves in the proper loop. In this way further discharge of activity to the secondary system is terminated. However, for the purpose of this analysis and in order to determine the maximum potential release of active steam to the atmosphere, it was assumed that a steam generator tube rupture also occurred with a loss of outside power co-incident with initiation of the safety injection signal. It was further assumed that no effort was made to close the loop isolation valves or the valves failed to close. The amount of steam discharged through the secondary safety valves is therefore governed by the time required for auxiliary feedwater flow and safety injection flow to absorb the core decay heat and core stored heat.

Four transient studies were performed to determine the effect of operation of one and two safety injection pumps, respectively, together with the addition of 50% and 100% of auxiliary feedwater flow. In the event of loss of outside power the emergency equipment is operated by diesel generator power and suitable delays were incorporated in the transient to allow for the delays in starting diesel generators, connecting the emergency equipment to the 480 V buses and opening valves in the safety injection lines.

The results are shown in Figures 14.3.1-1 to 14.3.1-4. Following the reactor trip the steam generator shell side temperature rapidly increases to the saturation temperature corresponding to the safety valve pressure setting and steam discharge continues until decay heat is quenched by safety injection flow and auxiliary feedwater flow. The amount of steam discharged and the time at which discharge is terminated is shown in Table 14.3.1-1. The percentage of initial reactor coolant inventory which has been discharged to the affected steam generator shell side during active steam discharge can be seen in the figures. In the later part of the transient the reactor coolant is repressurized by safety injection action to an extent which depends on the various combination of emergency equipment. This in turn affects the long term integrated break flow.

The environmental consequences of a tube rupture accident corresponding to cases 1 through 4 are discussed in Sections 14.3.1.4 below and the results shown in Table 14.3.1-1 and Figures 14.3.1-1 to 14.3.1-4 are incorporated in the analysis.

TABLE 14.3.1-1

STEAM RELEASE TO ATMOSPHERE FROM SAFETY VALVES

Case No.	1	2	3	4
Number of Safety Injection Pumps	2	2	1	1
Auxiliary Steam Generator Feed Flow, %	100	50	100	50
End of Steam Discharge sec. after trip	104	396	100	744
Mass of Steam Discharged, lbs.	13,800	19,800	14,340	20,300

14.3.1.4 Environmental Consequences of Tube Rupture

If offsite power is available, all volatile activity is released into the containment and there is no offsite dose. With offsite power the activity release is only through the faulted steam generator and is limited by the concentration of activity in the reactor coolant that is blown down to the secondary side of the steam generator prior to the termination of atmospheric relief. As shown in Figures 14.3.1-1 through 4, case 4 with one safety injection pump and 1/2 auxiliary flow results in the highest integrated break flow during the period of steam release shown on Table 14.3.1-1.

Activities released to the secondary side of the steam generator, on the basis of this coolant release (case 4, 10% of the primary system inventory) and operation with one percent fuel defects, is as follows:

Isotope	Activity, Curies
I-131	44
I-132	16
I-133	71
I-134	9.9
I-135	37.
Kr 85	63
Kr-85m	29.8
Kr-87	20
Kr-88	73.
Xe-133	4917.
Xe-133m	49.
Xe-135	136.

Because the iodine is soluble in water, considerable gaseous separation will occur. Assuming equilibrium was reached between the liquid and vapor iodine concentrations, the effective partition factor for the steam-to-water concentration would be 4×10^{-3} (Reference 1). Further separation would occur in the dryer and moisture separator sections of the steam generator. (A factor of 2.5×10^{-3} is used for the design moisture entrainment for full steam flow.) Realizing, however, that the actual situation would be a transient condition rather than steady-state, a conservative value of 10^{-2} has been chosen to represent the overall partition factor. Using this effective partition factor of 1.0×10^{-2}, for iodine and an X/Q of 8.14×10^{-4} sec/m^3, the activity released to the atmosphere through the safety valves results in a thyroid dose of about 0.28 rem and a whole body dose of about 0.30 rem at the site boundary.

It is concluded that a steam generator tube rupture will not result in excessive radiation exposure at the site boundary.

14.3.1.5 Recovery Prodcedure

The immediately apparent symptoms of a tube rupture accident, such as falling pressurizer pressure and level and increased charging pump flow, can also be symptoms of small steam line breaks and loss-of-coolant accidents. It is therefore important that the operator determine that the accident is the rupture of a steam generator tube in order to carry out the correct recovery procedure. This accident is uniquely identified by a condenser air ejector radiation alarm or a steam generator blowdown radiation alarm, and the operator does not proceed with the following recovery procedure unless these alarms are observed. In the event of a relatively large rupture, such as that analyzed above and shown in Figures 14.3.1-1 to 14.3.1-4, it is clear soon after trip that the level in one steam generator is rising more rapidly than in the others and this also is a unique indication of a tube rupture accident.

The operator carries out the following procedures which lead to isolation of the faulty steam generator and subsequently to unit cooldown.

1. Before the faulty steam generator is identified, auxiliary feedwater flow is regulated to all steam generators to maintain the minimum water level reached as no load temperature and pressure are established.

2. If outside power is available, the operator verifies that condenser steam dump maintains the no-load T_{avg}, and transfers steam dump to steam header pressure control.

3. As water level returns in the pressurizer, all but one safety injection pump is stopped to minimize break flow to the secondary system.

4. When the faulty steam generator is identified by rising water level, the reactor coolant pump in the associated loop is stopped and the loop isolation valves are closed. As soon as the

affected steam generator pressure is reduced below 1100 psig, the main steam stop valve is closed and this completes isolation of the faulty steam generator.

5. Auxiliary feedwater flow to the faulty steam generator is stopped.

6. If the affected steam generator has not been discovered by the time level returns in the pressurizer it is then identified by sampling the steam generator secondary side. Thereafter actions 4 and 5 above are completed.

7. If outside power is available but the loop isolation valves failed to close, the steam header pressure is reduced to 850 psig with condenser steam dump. This cools the entire system below 1100 psig at which time the main steam stop valve on the affected steam line can be closed, steam dump being continued from the other steam generators.

8. If outside power is not available, atmospheric steam dump from the unaffected steam generators is used to establish 850 psig in the unaffected steam generators with the pressure maintained by regulating the power relief valves. At the same time the reactor coolant system pressure is decreased to 1000 psig using pressurizer relief valves and spray. These actions automatically reduce the pressure in the faulty steam generator below 1100 psig and steam line isolation can then be carried out as above.

9. When the above actions have been carried out as necessary, and when outside power is (again) available, establish the reactor systems necessary for a normal shutdown and proceed with unit shutdown to the cold condition.

After the residual heat removal system is in operation, the condensate accumulated in the secondary system can be examined and processed through the Waste Disposal System.

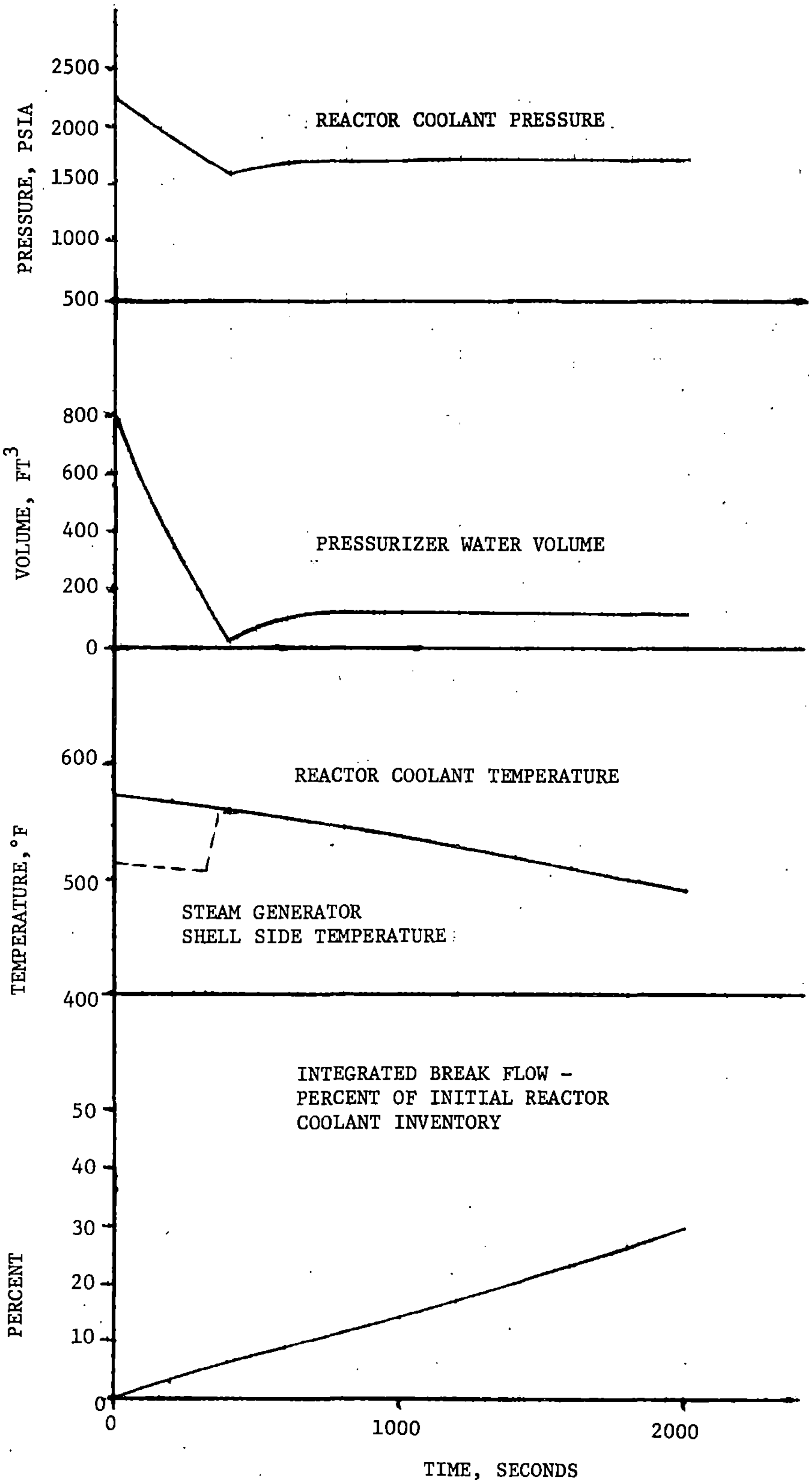
There is ample time available to carry out the above recovery procedure such that isolation of the affected steam generator is established before water level rises into the main steam pipes. A conserative estimate of water inventory in the faulty steam generator indicates that at least four minutes after reactor trip are required for the water level to reach the narrow range level tap. At this point the operator starts to regulate water level as demanded in the operating procedure. In the case of the faulty steam generator this involves terminating auxiliary feedwater flow as soon as it is apparent that water level is continuing to increase under the influence of break flow. From this point a further thirty minutes would be required for water level to rise into the main steam pipes even with three safety injection pumps operating.

Therefore, with careful operator vigilance there is no problem in recovery to the point where isolation of the faulty steam generator can be achieved prior to excessive water level being attained.

REFERENCES

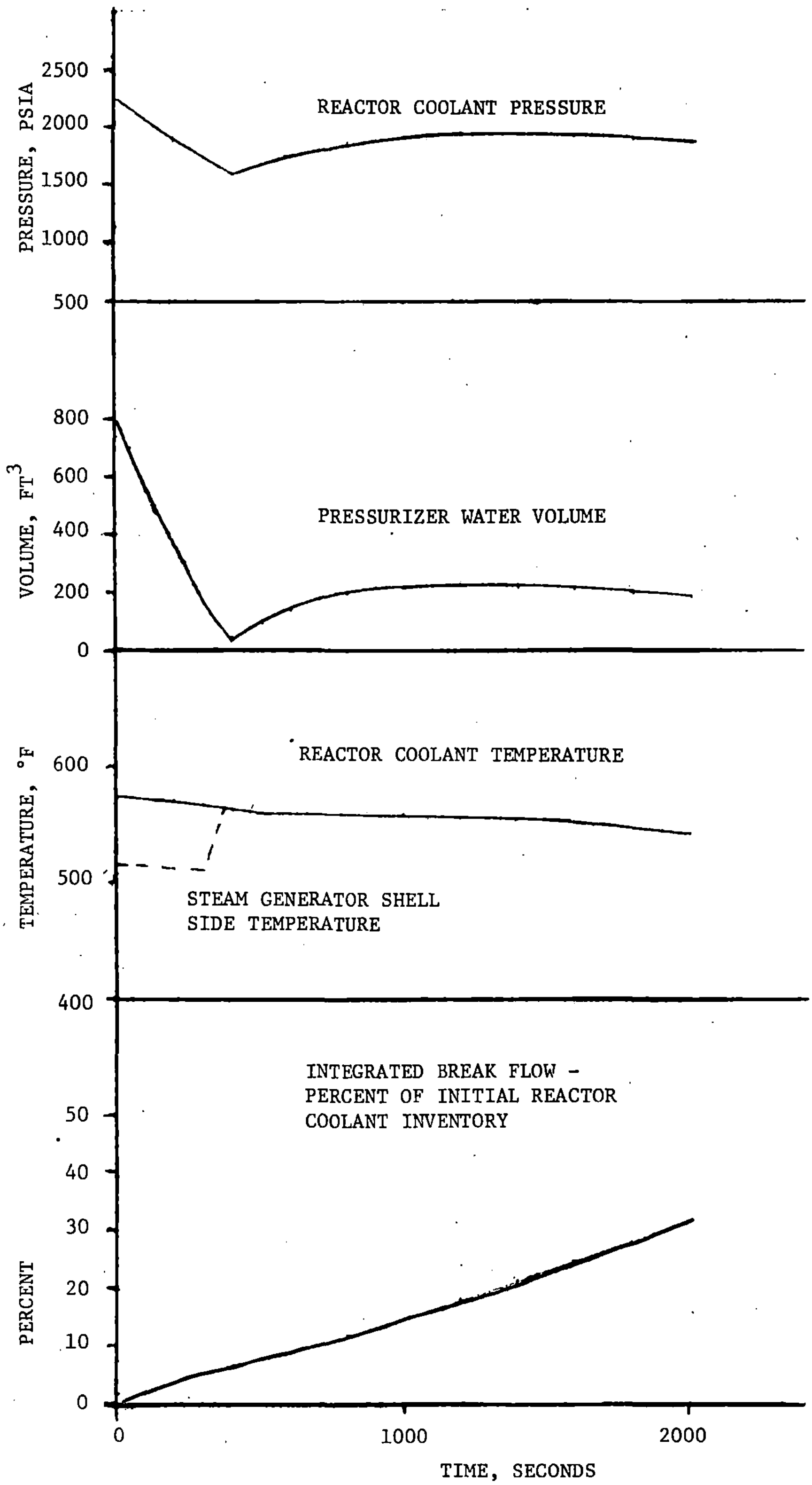
1. M. A. Styrikovich, et al., Transfer of Iodine from Aqueous Solutions to Saturated Vapor, (UDC 621.039.526.5) Translated from Atomnaya Energiya Vol. 17 No. 1, pp. 45-49 (July, 1964).

Figure 14.3.1-1
12-1-69



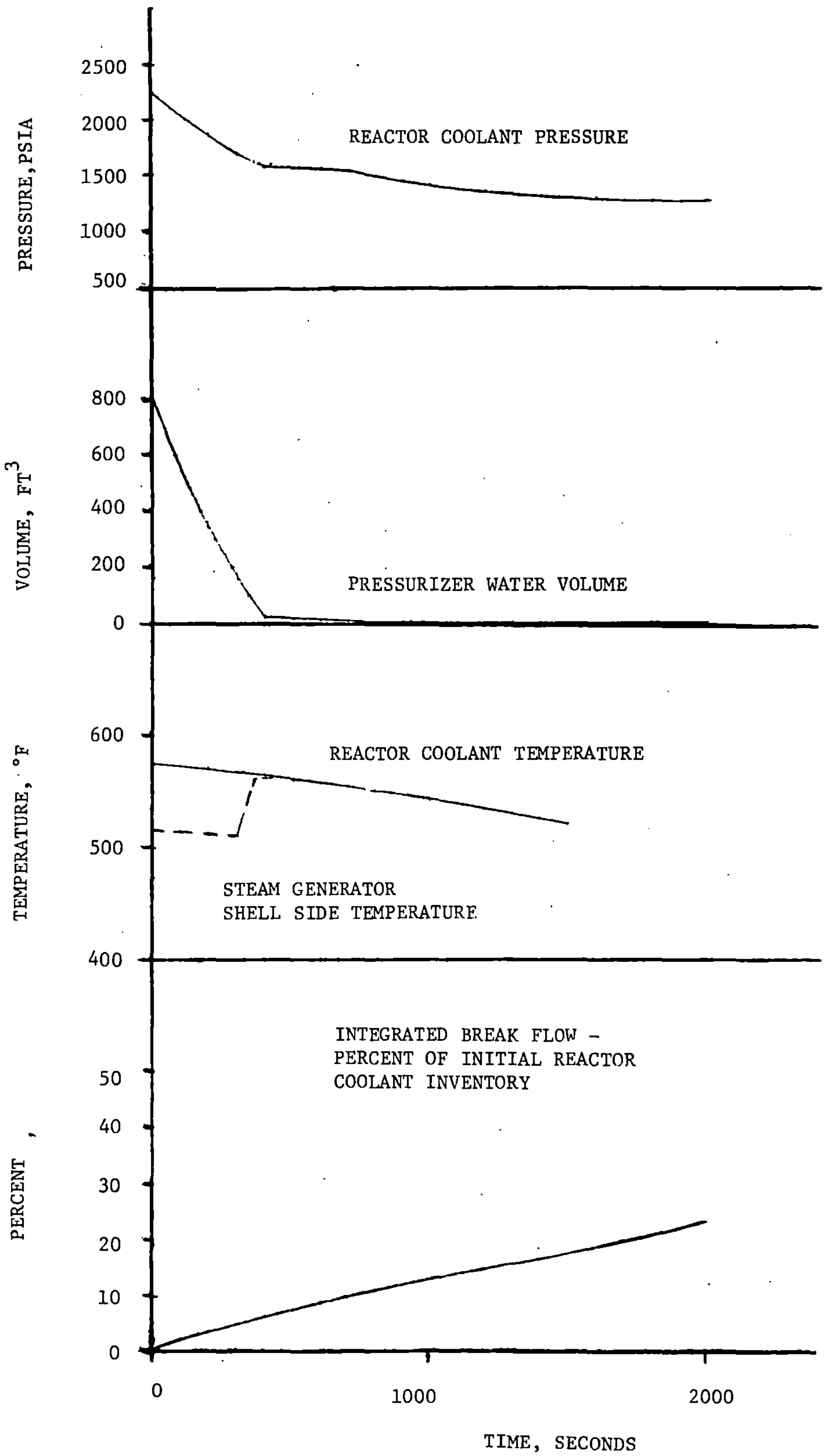
TUBE RUPTURE STUDY
CASE 1 - 2 SI PUMPS
FULL AUXILIARY FLOW

Figure 14.3.1-2
12-1-69

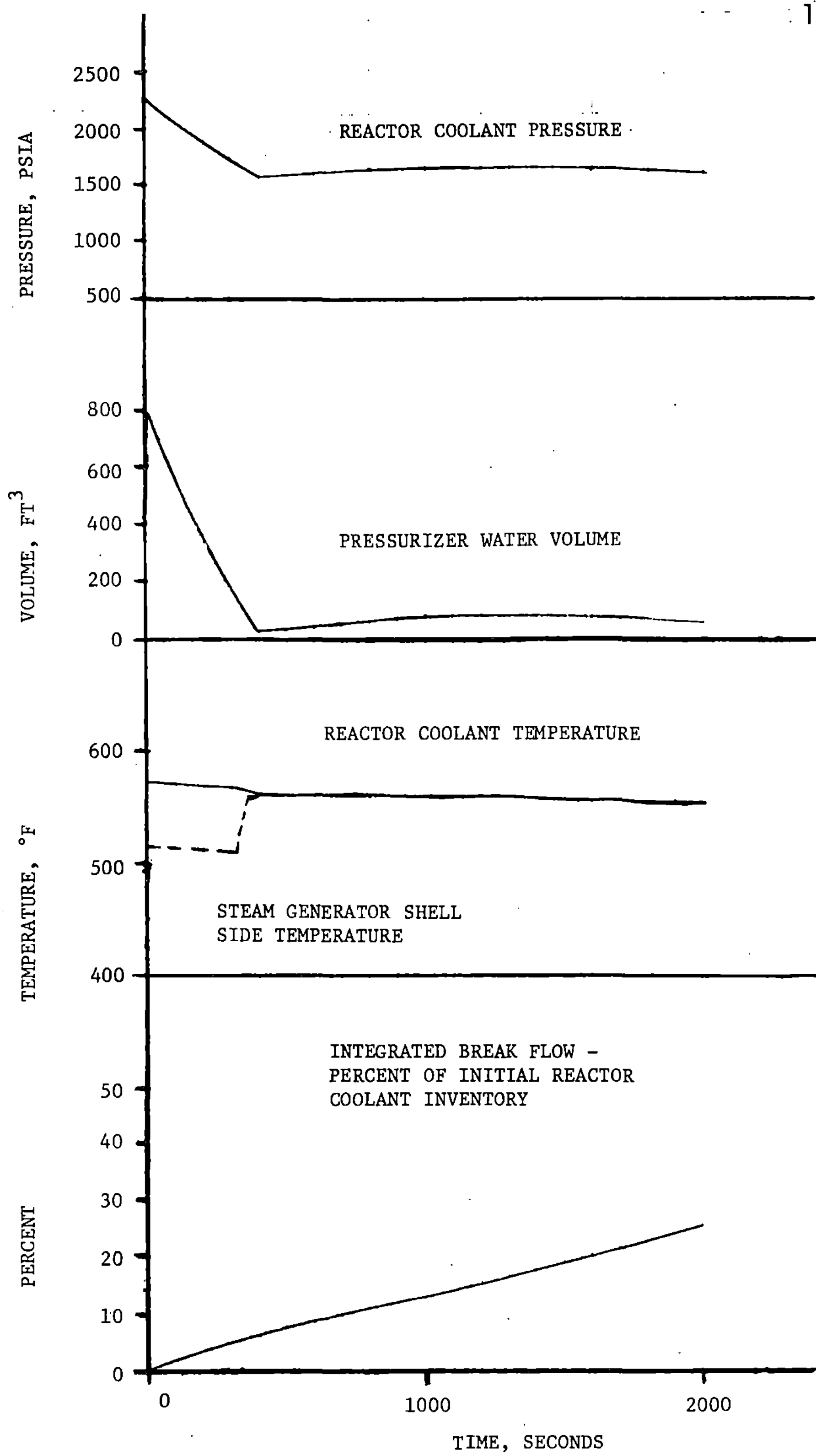


TUBE RUPTURE STUDY
CASE 2 - 2 SI PUMPS,
1/2 AUXILIARY FLOW

Figure 14.3.1-3
12-1-69



TUBE RUPTURE STUDY
CASE 3 - 1 SI PUMP,
FULL AUXILIARY FLOW



TUBE RUPTURE STUDY
CASE 4 - 1 SI PUMP,
1/2 AUXILIARY FLOW

d. Three out of four high containment pressure signals.

2) The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring upon actuation of the Safety Injection System.

3) Redundant isolation of the steam generator feedwater lines. Sustained high feedwater flow would cause additional cooldown, thus, in addition to the normal control action which closes the main feedwater valves, any safety injection signal rapidly closes all feedwater control valves, trips the steam generator feedwater pumps, and closes the feedwater pump discharge valves.

4) Trip of the fast acting main steam line trip valves (designed to close in less than 5 seconds) on:

a. High steam flow in two out of three main steam lines (one out of two per line) in coincidence with either low Reactor Coolant System average temperature (two out of three) or low steam line pressure (two out of three).

b. Three out of four high containment pressure signals.

Each main steam line has a fast closing trip valve and a non-return valve. These six valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For example, for a break upstream of the trip valve in one line, closure of either the non-return valve in that line or the trip valves in the other lines prevent blowdown of the other steam generators.

Steam flow is measured by monitoring dynamic head in nozzles inside the main steam pipes. The nozzles are of smaller diameter than the main steam pipe and are located inside the containment near the steam generators. They also serve to limit the maximum steam flow for any break further downstream. In particular, the nozzles limit the flow for all breaks outside the containment.

14.3.2.1 Method of Analysis

The analysis of the main steam pipe rupture has been performed to determine:

1) The core heat flux and Reactor Coolant System temperature and pressure resulting from the cooldown following the steam line break. A full unit digital computer simulation has been used.

2) The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital computer calculation has been used to determine if DNB occurs for the core conditions computed in (1) above. This calculation solves the continuity, momentum, and energy equations of fluid flow in the core and with the Westinghouse APD Subcooled Correlation (see reference in paragraph 7 below) determines the margin to DNB.

14.3.2 RUPTURE OF A MAIN STEAM PIPE

A rupture of a main steam pipe is assumed to include any accident which results in an uncontrolled steam release from a steam generator. The release can occur due to a break in a pipe line or due to a valve malfunction. The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of reactor coolant temperature and pressure. With a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive control rod assembly is assumed stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power even with the remaining control rod assemblies inserted. A return to power following a main steam pipe rupture is a potential problem mainly because of the high hot channel factors which exist when the most reactive rod is assumed stuck in its fully withdrawn position. Assuming the worst combination of circumstances which could lead to resumption of power generation following a main steam line break, the core is ultimately shut down by the boric acid in the Safety Injection System.

The analysis of a main steam pipe rupture is performed to demonstrate that:

1) Assuming a stuck control rod assembly with or without offsite power, and assuming a single failure in the engineered safety features there is no consequential damage to the primary system and the core remains in place and intact.

2) With no stuck control rod assembly, and all equipment operating at design capacity, insignificant (or no) cladding rupture occurs.

3) There will be no return to criticality for any single active failure in the Main Steam System. The single active failure is the opening, with failure to close, of the largest of any single steam bypass, relief or safety valve.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis shows that no DNB occurs for any rupture assuming the most reactive control rod assembly stuck in its fully withdrawn position.

The following systems provide the necessary protection against a main steam pipe rupture:

1) Safety Injection System actuation from any of the following*:

   a. One out of three pressurizer coincident low pressure and low level signals.

   b. Two out of three differential pressure signals between any main steam line and the main steam header.

   c. High steam flow in two out of three main steam lines (one out of two per line) in coincidence with either low Reactor Coolant System average temperature (two out of three) or low main steam line pressure (two out of three).

* The details of the logic used to actuate Safety Injection are discussed in Section 7.

The following assumptions were made:

1) A 1.77% $\frac{\Delta k}{k}$ shutdown reactivity from all but one control rod assembly at no load conditions. This is the end of life design value including design margins with the most reactive control rod assembly stuck in its fully withdrawn position. The actual shutdown capability is expected to be significantly greater.

2) The negative moderator coefficient corresponding to the end of life core with all but the most reactive control rod assembly inserted. The variation of the coefficient with temperature and pressure has been included. The reactivity versus temperature at 1000 psia corresponding to the negative moderator coefficient used is shown in Figure 14.3.2-1. In computing the power generation following a steam line break, the local reactivity feedback from the high neutron flux in the region of the core near the stuck control rod assembly has been included in the overall reactivity balance. The local reactivity feedback is composed of Doppler reactivity from the high fuel temperatures near the stuck control rod assembly and moderator feedback from the high water temperature near the stuck control rod assembly. For the cases analyzed where steam generation occurs in the high flux regions of the core, the effect of void formation on the reactivity has also been included. The effect of power generation in the core on overall reactivity is shown in Figure 14.3.2-2. The curve assumes end of life core conditions with all control rod assemblies in except

the most reactive control rod assembly which is assumed stuck in its fully withdrawn position (completely removed from core).

3) Minimum safety injection capability corresponding to the most restrictive single failure in the engineered safeguards equipment. In the case of continued availability of outside power the failure of a single safety injection pump is assumed. However, in the case of loss of outside power, the worst failure is the loss of one diesel generator which can result in a safety injection capability corresponding to the operation of only one high head safety injection pump. 20,000 ppm boron was assumed in the Safety Injection System. The time delays required to sweep the low concentration boric acid from the safety injection piping prior to the delivery of the 20,000 ppm boron have been included in the analysis.

4) A steam generator heat transfer coefficient (UA) of 14800 BTU/sec °F. This is considered conservative since no allowance for reduction of the heat transfer UA as the water level falls into the tube region has been made. Furthermore, higher steam generator UA values result in lower reactor coolant temperature at full power which in turn result in an increase in the available shutdown margin at zero load.

5) Hot channel factors corresponding to one stuck control rod assembly -- the control rod assembly giving the highest factor at end of life. The hot channel factors account for the void existing local to the

stuck control rod assembly at the pressure that occurs during the return to power phase following the steam break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck control rod assembly. The hot channel factors depend upon the core temperature, pressure, and flow and, thus, are different for each case studied. The values used for each case are given in Table 14.3.2-1. The calculations used to obtain the hot channel factors again assume end of life core conditions with all control rod assemblies in except the most reactive control rod assembly.

6) Five combinations of break sizes and initial unit conditions have been considered in determining the core power and Reactor Coolant System transient.

a) Complete severence of a main steam pipe outside the containment, downstream of the steam flow measuring nozzle, initially at no load conditions with outside power available.

b) Complete severence of a main steam pipe inside the containment at the outlet of the steam generator initially at no load conditions with outside power available.

c) Case (a) above with loss of outside power simultaneous with the steam break.

TABLE 14.3.2-1

CORE PARAMETERS* AND NUCLEAR HOT CHANNEL FACTORS USED IN STEAM BREAK ANALYSES

Case Number	1	2	3	4	5	6
Inlet Temp., °F	377	338	312	414	395	335
Pressure, psig	1256	961	934	1416	970	837
Core Flow, %	38	29	23	100	100	100
Heat Flux, % of 2441 MWt	7.38	11.67	7.26	7.66	11.76	6.75
$F_{\Delta H}$	8.4	6.7	8.7	8.7	8.4	9.6
F_Z	1.87	1.41	1.56	2.35	2.03	2.42

* Values assume the end of life rodded core with one stuck control rod assembly.

d) Case (b) above with the loss of outside power simultaneous with the steam break.

e) A break equivalent to a steam flow of 247 lbs per second at 1100 psia from one steam generator with outside power available. This case was examined for one and two safety injection pumps operating.

All the cases above assume initial hot shutdown conditions with the control rod assemblies inserted (except for one stuck control rod assembly) at time zero. Should the reactor be just critical or operating at power at the time of a main steam line break the reactor is tripped by the normal overpower protection system when the power level reaches a trip point. Following a trip at power the Reactor Coolant System contains more stored energy than at no load, the average coolant temperature is higher than at no load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the main steam line break before the no load conditions of Reactor Coolant System temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes a no-load condition at time zero. However, since the initial steam generator mass is greatest at no load, the magnitude and duration of the Reactor Coolant System cooldown are less for main steam line breaks occurring at power.

7) In determination of the critical flux at which burnout could occur the "Westinghouse APD Subcooled Correlation" was used. This correlation is discussed in "Boiling Heat Transfer and Two Phase Flow" by L. S. Tong. It was considered to be the correlation which most accurately represented the range of parameters produced in the transients analyzed.

8) In computing the steam flow during a steam line break, the Moody Curve (Figure 3 of the article by F. S. Moody in Transaction of the ASME Journal of Heat Transfer, February 1965, page 134) for a pipe of zero length was used.

Results

The results presented are a conservative indication of the events which would occur assuming a main steam line rupture. The worst case assumes that all of the following occur simultaneously.

1) Minimum shutdown reactivity margin of 1.77% $\frac{\Delta k}{k}$.

2) The most negative moderator temperature coefficient for the rodded core at end of life.

3) The highest worth control rod assembly stuck in its fully withdrawn position.

4) The single most restrictive failure of the engineered safety features.

14.2.2.3 Core Power and Reactor Coolant System Transient

Figure 14.3.2-3 shows the Reactor Coolant System transient and core heat flux following a main steam pipe rupture (complete severence of a pipe) outside the containment, downstream of the flow measuring nozzle at initial no load conditions (Case A). The break assumed is the largest break which can occur anywhere outside the containment either upstream or downstream of the isolation valves. Outside power is assumed available such that full reactor coolant flow exists. The transient shown assumes the control rod assemblies inserted at time 0 (with one control rod assembly stuck in its fully withdrawn position) and steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by high differential pressure between any steam generator and the main steam header or by high steam flow signals in coincidence with either low Reactor Coolant System temperature or low steam line pressure trips the reactor. Steam release from at least two steam generators is prevented by either the non-return valves or by automatic trip of the fast acting trip valves in the steam lines by the high steam flow signals in coincidence with either low Reactor Coolant System temperature or low main steam line pressure. Even with the failure of one valve, release is limited to no more than 5 seconds for two steam generators while the third generator blows down. The main steam line trip valves are designed to be fully closed in less than 5 seconds with no flow through them. With the high flow existing during a main steam line rupture, the valves will close considerably faster since closure is flow assisted.

As shown in Figure 14.3.2-3, the core barely attains criticality with the control rod assemblies inserted (with the design shutdown, assuming one stuck control rod assembly) at 27 seconds. Boron solution at 20,000 ppm enters the Reactor Coolant System from the Safety Injection System at 27 seconds reflecting a delay of 10 seconds required to clear the Safety Injection System lines of low concentration boric acid. The remainder of the delay time consists of 7 seconds to receive and actuate the safety injection signal and 10 seconds to completely open valve trains in the safety injection lines. Since the safety injection pump accelerates to full speed in less than the time required to open the valve train, and since it is expected that the safety injection signal will be generated in less than 7 seconds, the overall delay time is considered conservative.

The calculation assumes the boric acid is mixed with and diluted by the water flowing in the Reactor Coolant System prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the Reactor Coolant System and in the Safety Injection System. The variation of mass flow rate in the Reactor Coolant System due to water density changes is included in the calculation as is the variation of flow rate in the Safety Injection System due to changes in the Reactor Coolant System pressure. The Safety Injection System flow calculation includes the line losses in the system as well as the pump head curve.

No credit has been taken for the 2,000 ppm boron which enters the Reactor Coolant System prior to the 20,000 ppm boric acid. Since the core was slightly supercritical for less than 4 seconds in this case and with very low excess reactivity there was virtually no return to power.

Figure 14.3.2-4 shows Case B, a main steam line rupture at the exit of a steam generator at no load. The sequence of events is similar to that described above for the rupture outside the containment except that excess reactivity leads to power generation. The peak core average heat flux is 11% of the value at 2441 MWt.

Figures 14.3.2-5 and 14.3.2-6 show the responses for the previous breaks except with a loss of outside power. Reactor Coolant System flow coastdown is assumed to occur at the same time as the safety injection signal is received. The Safety Injection System delay time includes the time required to start safety injection pumps with emergency power from the diesel generators. Only one safety injection pump is assumed in Figures 14.3.2-5 and 14.3.2-6. In both of these figures, excess reactivity is attained and the peak heat fluxes are 3% and 11%, respectively, of the values at 2441 MWt.

Figure 14.3.2-7 shows the transients following a break equivalent to a steam flow of 247 lbs per sec. at 1100 psia with steam release from one steam generator (Case E). The assumed steam release is larger than or equal to the capacity of any single dump or safety valve. In this case, safety injection is initiated automatically by low pressurizer pressure and level at 211 seconds. Operation of both one and two safety injection pumps is considered. Boron solution at 20,000 ppm enters the Reactor Coolant System at 232 seconds. For these transients there is no return to criticality and the maximum reactivity attained is independent of the number of operating pumps. After 232 seconds the 20,000 ppm boron provides sufficient negative reactivity to maintain the reactor well below criticality in both cases while the steam generator empties and causes further cooldown. The cooldown for the case shown in Figure 14.3.2-7 is more rapid than the case of steam release from all steam generators through one relief, bypass, or safety valve. The transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements or the energy stored in the other steam generators. Since the transient occurs over a period of about five minutes, the neglected stored energy is likely to have a significant effect in slowing the cooldown.

It should be noted that following a main steam line break only one steam generator blows down completely. Thus, two steam generators are still available for dissipation of decay heat after the initial transient is over. In the case of loss of outside power this heat is removed to the atmosphere and the atmospheric safety valves have been sized to cover this condition.

14.3.2.4 Margin to Critical Heat Flux

Using the transients shown in Figures 14.3.2-4 and 14.3.2-6 (Case A and B) i.e., the inside containment breaks for the "with" and "without" outside power cases, the Westinghouse NES Subcooled Correlation was used to determine the margin to DNB. Three carefully chosen points from each transient, were examined and the results are presented in Table 14.3.2-1. The power and flow conditions are shown together with pressure and inlet core temperature and the nuclear hot channel factors corresponding to the end of life core with one stuck control rod assembly. It was found that all cases had a minimum DNBR greater than 2.0. The breaks outside the containment result in lower core heat fluxes than those inside the containment and thus do not cause DNB anywhere in the core.

14.3.2.5 Environmental Effects Of A Steam Line Break

No radioactivity is released to the environment because of a steam line break unless there is or has been primary to secondary system leakage in a steam generator. If leakage has occurred, the magnitude of the radioactive release depends on the activity which has built up in the secondary side of the leaking steam generator prior to a steam line break and from the activity which continues to leak out of the primary system from the time of the steam line break until the primary side of the steam generator is reduced to atmospheric pressure, a period of approximately 8 hours.

It has been determined from an analysis of this accident that about 75 percent of the resulting total thyroid dose is from Iodine-131. The remaining 25 percent of the dose is from other isotopes of iodine which have built up and from the iodines which have leaked from the reactor coolant system after the steam line break occurs. The concentration of I-131 in the secondary side of the steam generator can be directly related to the primary to secondary system leak rate by neglecting the radioactive decay and using equilibrium values. The environmental effects of a steam line break are analyzed for the following conditions in addition to those previously mentioned in this section:

One percent failed fuel.

Steam generator blowdown rate of 7 gpm.

Primary to secondary system leakage rate of 10 gpm.

Steam generator has maximum water inventory. (Zero power operation).

Ninety percent of the iodine available for release plates out.

Dose at the site boundary is calculated by:

$$\text{Dose} = \frac{(L.R.)(C_{RC})(V_{SS})\left(\frac{\chi}{Q}\right)\left(\frac{D_\infty}{A_\tau}\right)(B.R.)}{(Q_{BD})(.75)(P.F.)}$$

where:

$L.R.$ = Primary to secondary system leak rate, 10 gpm.

C_{RC} = Concentration of I-131 in the reactor coolant, 1.68 μCi/cc.

V_{SS} = volume of liquid in secondary side of a steam generator, 101 m^3

$\frac{\chi}{Q}$ = 8.14×10^{-4} sec/m^3

$\frac{D_\infty}{A_\tau}$ = 1.48×10^{6} rem/Ci for I-131

$B.R.$ = breathing rate, 3.47×10^{-4} m^3/sec from TID 14844

Q_{BD} = blowdown rate from a leaking steam generator, 7 gpm

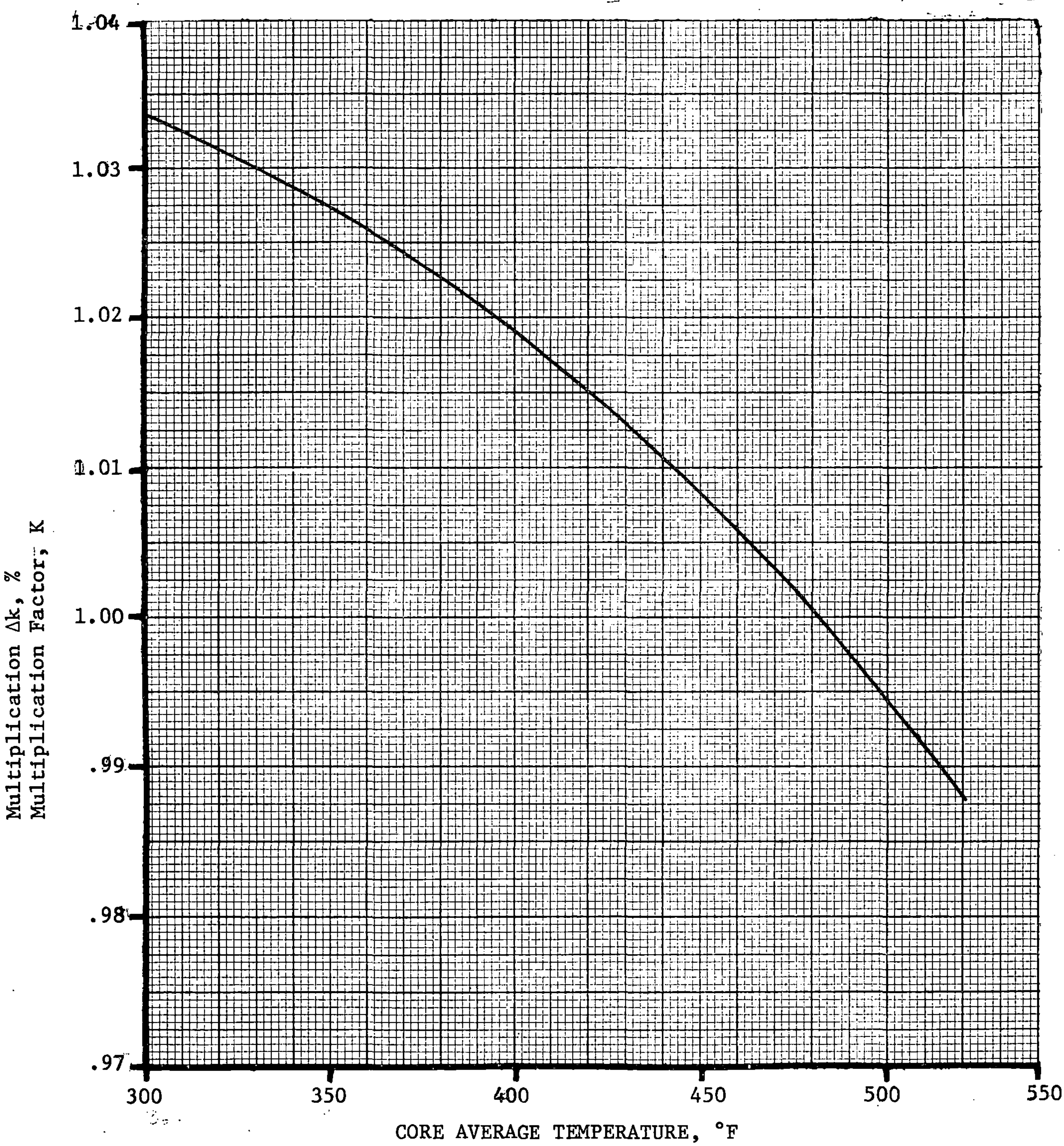
$P.F.$ = plating factor, 10.

Substituting the appropriate values into the dose equation yields 13.6 rem thyroid dose at the site boundary. This thyroid dose is considerably less than the guidelines set forth in 10CFR100.

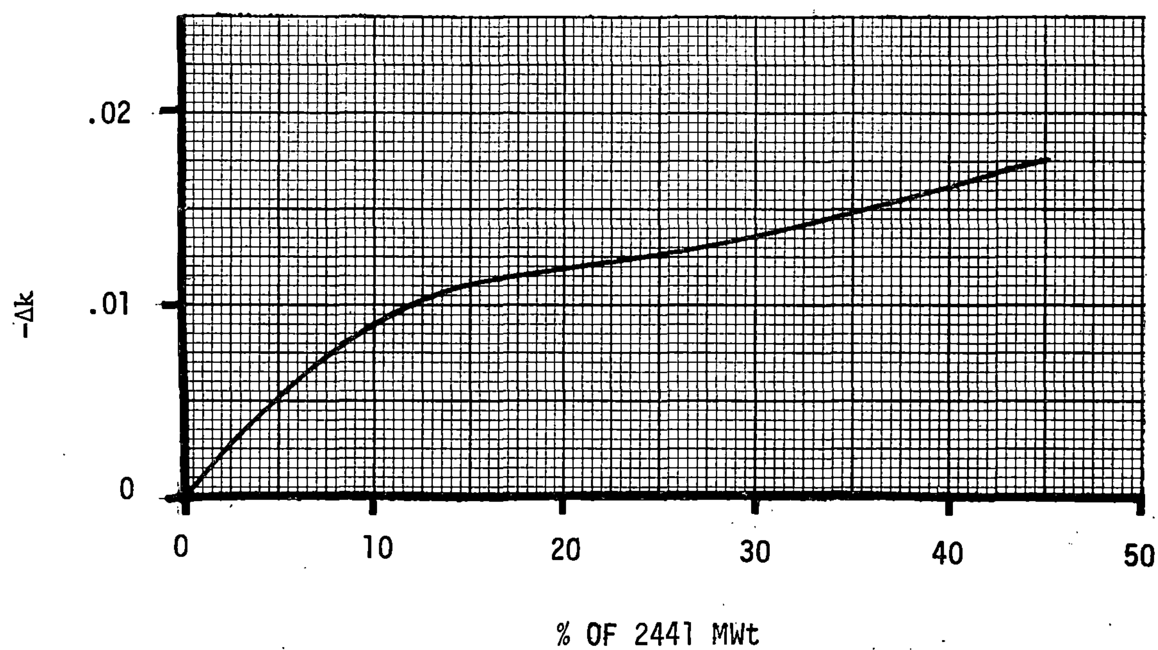
14.3.2.6 Conclusions

Although DNB and possible clad perforation (no clad melting or zirconium - water reaction) following a steam pipe rupture are not necessarily unacceptable, the above analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive rod stuck in its full withdrawn position.

FIGURE 14.3.2-1
12-1-69

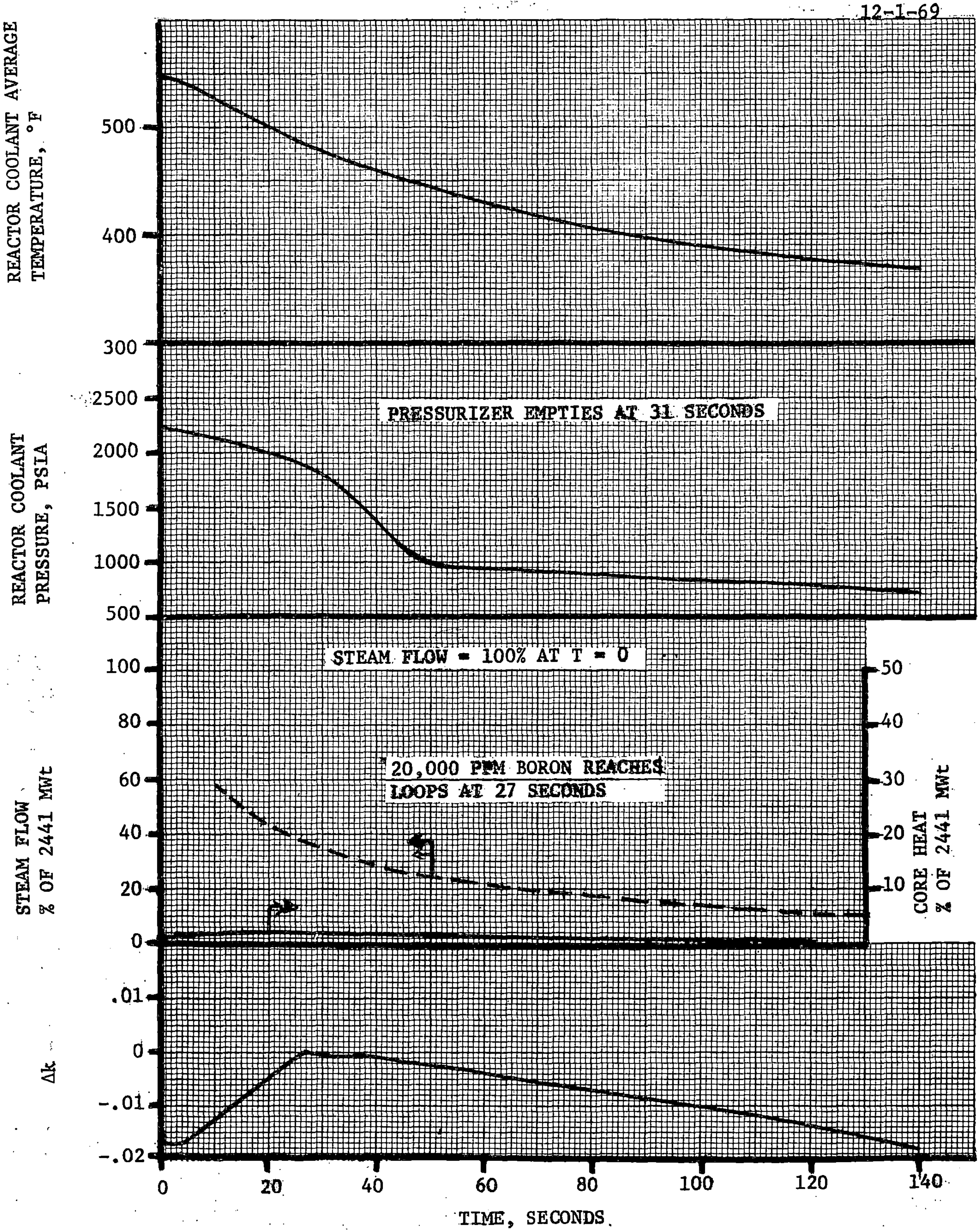


VARIATION OF REACTIVITY WITH CORE TEMPERATURE
AT 1000 PSIA FOR THE END OF LIFE RODDED CORE
WITH ONE CONTROL ROD ASSEMBLY STUCK
(ASSUMES 0 POWER)

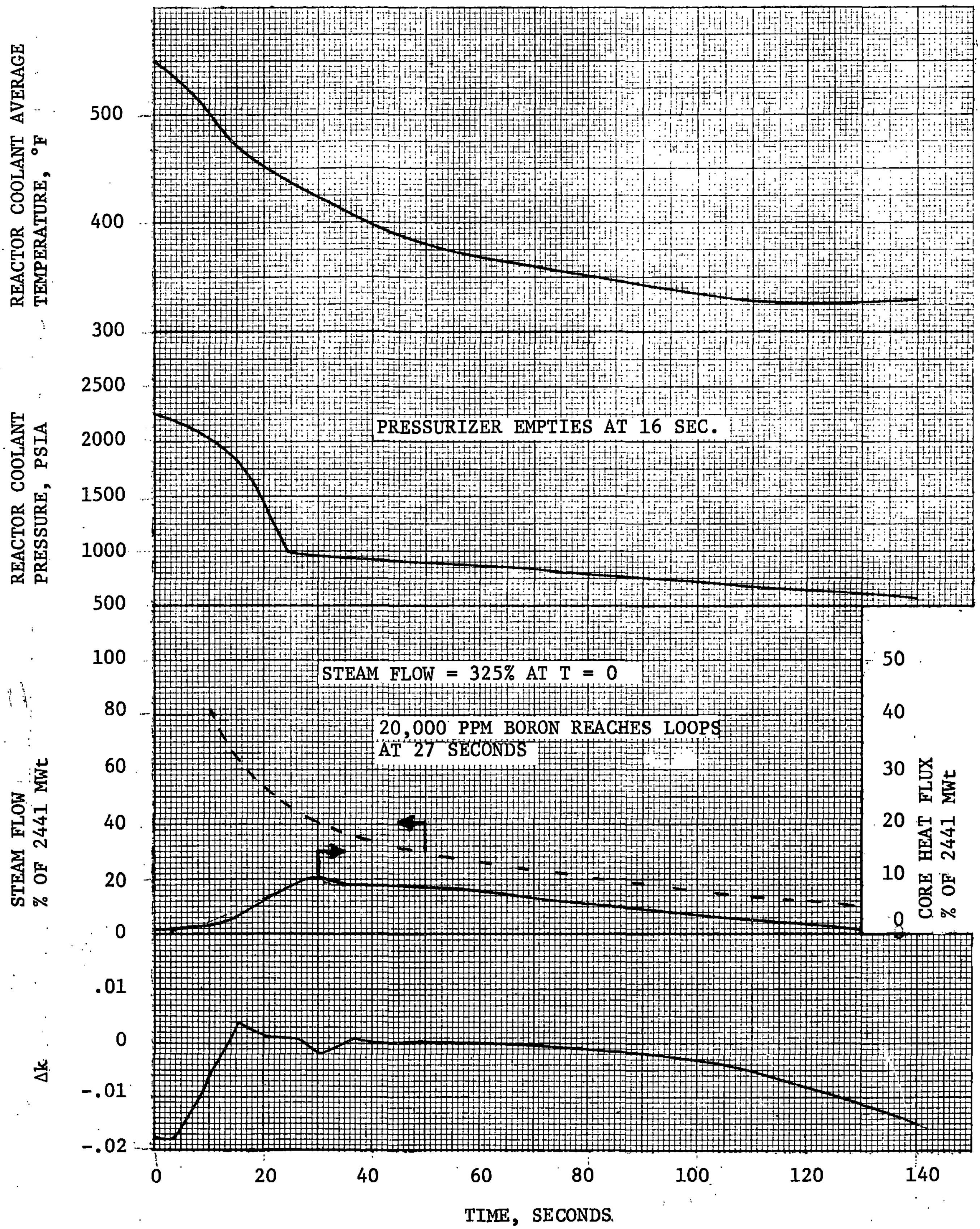


VARIATION OF REACTIVITY WITH POWER AT CONSTANT CORE AVERAGE TEMPERATURE. VALUES INDICATED WERE USED IN STEAM PIPE RUPTURE ANALYSIS FOR THE END OF LIFE RODDED CORE WITH ONE CONTROL ROD ASSEMBLY STUCK

FIGURE 14.3.2-3
12-1-69

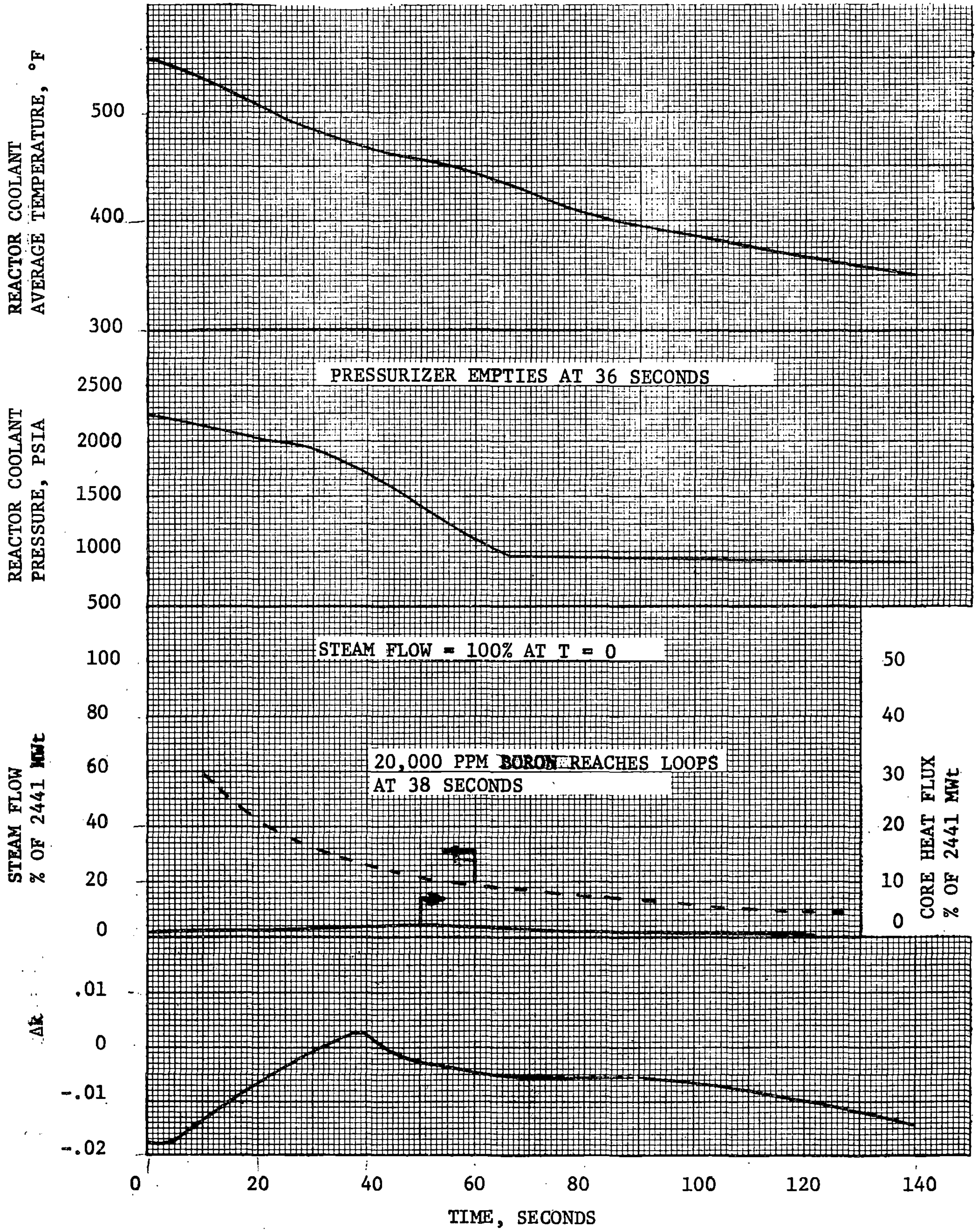


STEAM LINE BREAK DOWNSTREAM OF FLOW MEASURING NOZZLE WITH SAFETY INJECTION AND OUTSIDE POWER (CASE A)

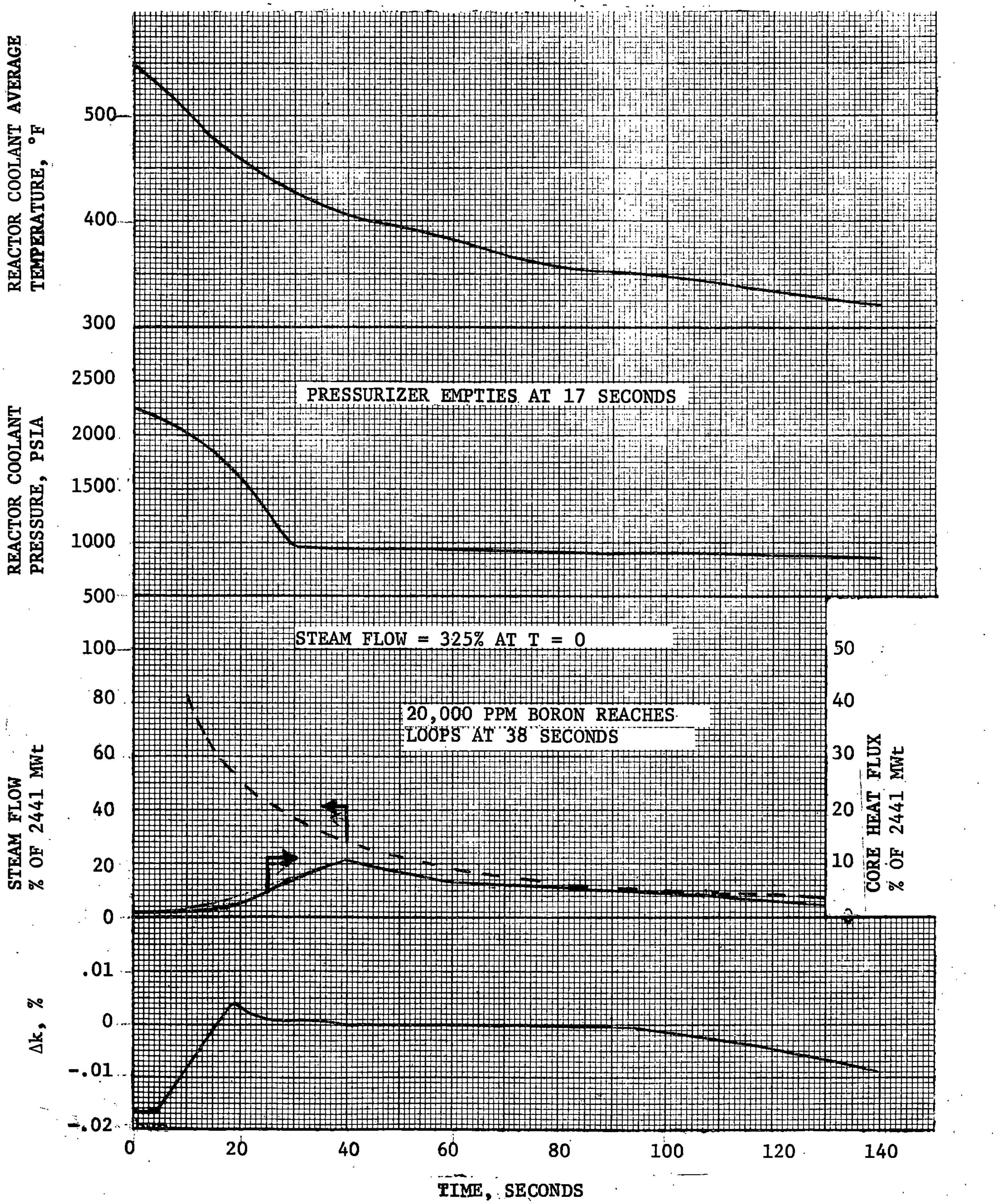


STEAM LINE BREAK AT EXIT OF STEAM GENERATOR
WITH SAFETY INJECTION AND OUTSIDE POWER
(CASE B)

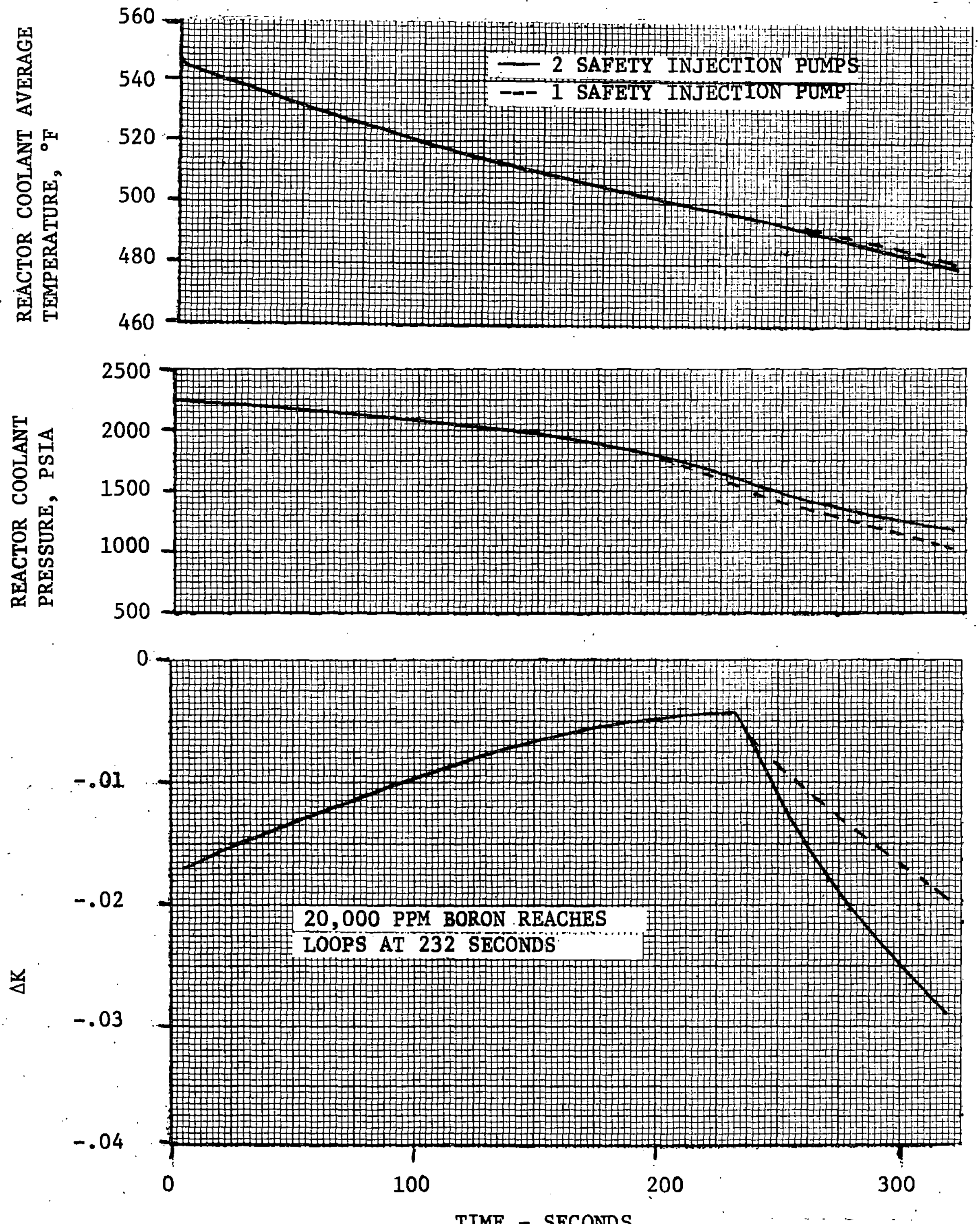
FIGURE 14.3.2-5
12-1-69



STEAM LINE BREAK DOWNSTREAM OF FLOW MEASURING
NOZZLE WITH SAFETY INJECTION, WITHOUT OUTSIDE
POWER (CASE C)



STEAM LINE BREAK AT EXIT OF STEAM GENERATOR
WITH SAFETY INJECTION WITHOUT OUTSIDE POWER
(CASE D)



STEAM LINE BREAK EQUIVALENT
TO 247 LBS/SEC AT 1100 PSIA,
WITH OUTSIDE POWER (CASE E)

14.3.3 RUPTURE OF A CONTROL ROD DRIVE MECHANISM HOUSING-CONTROL ROD ASSEMBLY EJECTION

In order for this accident to occur, a rupture of the control rod drive mechanism housing must be postulated creating a full system pressure differential acting on the drive shaft. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals.

A failure of a control rod drive mechanism housing sufficient to allow a control rod assembly to be rapidly ejected from the core is not considered credible for the following reasons:

a) Each control rod drive mechanism housing is completely assembled and shop-tested at 3450 psig.

b) The drive mechanism housings are individually hydrotested to 3107 psig as they are installed on the reactor vessel head to the head adapters, and checked during the hydrotest of the completed Reactor Coolant System.

c) Stress levels in the drive mechanism are not affected by system transients at power, or by thermal movement of the coolant loops. Moments induced by the Design Basis Earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class A components.

d) The latch mechanism housing and rod travel housing are each a single length of forged type-304 stainless steel. This material exhibits excellent notch toughness at all temperatures that are encountered.

The joints between the latch mechanism and the head adapter and between the latch mechanism and the rod travel housing are threaded joints, reinforced using canopy type seal welds.

The operation of a chemical shim unit is such that the severity of an ejection accident is inherently limited. Since control rod assemblies are used to control load variations only and core depletion is followed with boron dilution; there are only a few control rod assemblies in the core at full power. Proper positioning of these assemblies is monitored by a Main Control Room Alarm System. There are low and low-low level insertion monitors with visual and audible signals. Operating instructions require boration at the low level alarm and emergency boration at the low-low alarm. The control rod assembly position monitoring and alarm systems are described in detail in Section 7. For all cases, utilizing the flexibility in being able to select the control rod assembly groupings, radial locations and positions as a function of load, the design minimizes the peak fuel and clad temperatures.

This section describes the models used and the results obtained. Only the initial few seconds of the power transient are discussed, since the long term considerations are the same as for a loss-of-coolant accident.

Method of Analysis

The calculation of the transient is performed in two stages, first an average core calculation and then a hot region calculation. The average core is analyzed to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator density reactivity. Enthalpy and temperature transients in the hot spot are determined by adding a multiple of the average core energy generation to the hotter fuel rods and performing a transient heat-transfer calculation. The asymptotic power distribution calculated without feedback is assumed to persist throughout the transient.

Average Core

The nuclear power transients are calculated using the CHIC-KIN code developed by the Bettis Atomic Power Laboratory for similar analyses.(2) This code solves the point kinetics equations, with feedback from an axially and radially segmented fuel element. CHIC-KIN results have been compared with SPERT results for two dissimilar cores over a wide range of periods with good agreement.

The largest temperature rises, and hence the largest reactivity feedbacks occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions also have high weights. This means that the reactor feedback is larger than that indicated by a simple single channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which when applied to single channel feedbacks correct them to effective whole core feedbacks for the appropriate flux shape. The values used in the analyses are listed in Table 14.3.3-2. For this study, six delayed neutron groups were used and the fuel rod was divided into eight radial increments, with a ninth increment for the clad. Five axial segments were employed. The calculation is essentially a single channel analysis representing the core average conditions.

Prompt heat generation directly in the coolant is 2.6 percent of the nuclear power generation. This number is based on LEOPARD calculations.[3] Heat generation in the fuel pellet is assumed to occur non-uniformly radially with a slight reduction in the center due to self-shielding effects.

Hot Region

The average core energy addition calculated as described above is multiplied by the appropriate hot channel factors and the worst cases analyzed using a detailed heat transfer code. The zirconium-water reaction is explicitly represented and all material properties are represented as functions of temperature. The Tong, Sandberg and Bishop correlation (as described in Section 3) is used to determine the film boiling heat transfer coefficient after DNB. The resulting coefficient is reduced by 30% to allow for scatter in the basic data used to derive the correlation. (This corresponds to three standard deviations.)

Selection of Input Parameters

Input parameters for the analyses were conservatively selected on the basis of calculated values for this unit. The more important parameters are discussed below, and the values used given in Table 14.3.3-1.

Ejected Rod Worth

Basic values for the ejected rod worths were calculated using standard nuclear design. Since a two dimensional code was used, it was not possible to take account of the part length control rod assemblies directly. For a given ejected control rod assembly, the worth was calculated first assuming part length control rod assemblies and then with no part length control rod assemblies. The overall ejected rod worth was then assumed to be the linear average of the two values. This introduces conservatism into the analysis since the part length control rod assembly region is only one fourth of the core length and does not hold 50% of the axial reactivity weight. Also the worst ejected control rod assembly is not always fully inserted, and may not overlap the part length bank. A 20% margin is added to the ejected rod worth in order to account for possible azimuthal flux tilts.

Delayed neutron fraction β

No margin was subtracted from the calculated value of β, since margin had already been inserted into the reactivity calculation via the ejected rod worth. The values used are 0.69%ΔK at the beginning of life, and 0.52% ΔK at the end of first cycle operation.

Hot Channel Factor

In all cases the radial hot channel factor was higher in the part length region. The value used in the analyses was determined using PDQ 7, and taking no credit for the flux flattening effects of reactivity feedback.

The axial hot channel factor is highest when the part length rods are moved far away from the flux peak. However, this in general results in the minimum ejected rod worth. Also, the axial and radial peaks are not coincident under these conditions. Analyses indicate that the worst hot spot transient occurs when the part length control rod assemblies are located in the peak. Axial hot channel factors were calculated with the part length control rod assemblies in the middle of the core. This results in a conservatively high flux peaking, since to be consistent with the assumed ejected rod worth, the rods should be located in the flux peak.

The total transient hot channel factor, F_q, was obtained by multiplying the axial and radial hot channel factors. A further margin of 20% was added to account for possible azimuthal flux tilts.

Reactivity Weighting Factor

The reactivity weighting factor was reduced by 10% to allow for errors in the weighting calculation and a further 10% to allow for errors in the basic reactivity temperature coefficient calculations. This factor was applied to both the Doppler and moderator density feedbacks.

Moderator Temperature Coefficient

For conservatism, the core reactivity was assumed to be independent of moderator temperature, at the beginning of core life, even though the coefficient is always negative at operating temperature. At the end of life, the calculated reactivity versus density curves were reduced by 10% of the value at operating density. This was in addition to the two 10% margins in the weighting factor.

Heat Transfer Data

For the average core, it is conservative to assume a high gap conductance and fuel thermal conductivity at the beginning of core life, when the moderator coefficient is assumed to be zero. A value of 2.65 Btu/hr-ft-F was assumed for the fuel thermal conductivity and 10,000 Btu/hr-ft^2-F for the gap conductance. At the end of life a given quantity of heat produces more feedback in the moderator than in the fuel, and it is therefore conservative to assume a low gap conductance and fuel thermal conductivity. Values of 750 Btu/hr-ft^2-F and 2.2 Btu/hr-ft-F respectively were assumed.

The code used to determine the hot spot transient contains standard curves of thermal conductivity versus fuel temperature. This facility was used in the analyses. In order to obtain conservatively high initial fuel temperatures, a low initial gap conductance of 750 Btu/hr-ft^2-F was used in most cases. For the high power cases, the value was adjusted to yield the same center fuel temperature as that derived in the detailed thermal hydraulics calculations. During a transient the gap conductance can be expected to rise. For the possible range of gap conductances, the peak center fuel temperature is independent of the gap conductance during the transient. The cladding temperature is however strongly dependent on the gap conductance and is highest for high gap conductances. For conservatism a high value of 10,000 Btu/hr-ft^2-F has been used during transients. This value corresponds to a negligible gap resistance and a further increase would have essentially no effect on the rate of heat transfer.

Coolant Mass Flow Rates

When the core is operating at full power, all three reactor coolant pumps are always in operation. However, for zero power conditions, the system may be operating with one pump. The principal effect of operating at reduced flow is to reduce the film boiling heat transfer coefficient. This results in higher peak cladding temperatures, but does not affect the peak center fuel temperature. Reduced flow also lowers the critical heat flux. However, since DNB is always assumed at the hot spot, and since the heat flux rises very rapidly during the transient, this produces only second order changes in the cladding and center fuel temperatures. All zero power analyses for both average core and the hot spot have been conducted assuming single loop operation.

Trip Reactivity Insertion

The rods were assumed to be released 0.5 seconds after the initiation of ejection. The delay is constituted of 0.2 seconds for the instrumentation to produce a signal, 0.15 seconds for the trip breaker to open and 0.15 seconds for coil release. In calculating the shape of the insertion versus time curve all the rods are assumed to be dropping as a single bank from the fully withdrawn position. This means that the initial movement is through the low worth region at the extreme top of the core, and produces a conservatively slow reactivity insertion versus time curve.

The trip reactivity insertion is based on calculated rod worths with a 10% margin. The stuck and ejected rods together are assumed to reduce the final rod holding by twice the calculated stuck rod worth.

Lattice Deformations

Reactivity insertion as a result of lattice deformation was considered. In the region of the hot spot there will be a large power gradient. Since the fuel rods are free to move in a vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a force tending to bow the midpoint of the rods toward the hot spot. Physics calculations indicate that the net result of this would be a negative reactivity insertion. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced.

Boiling in the hot spot region will produce a net fluid flow away from that region. However, the fuel heat is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It is concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is pessimistically ignored in the following analyses.

Cases Considered

In all cases, the worst ejected rod in terms of both rod worth and hot channel factors is a control rod assembly bank, and so no separate analyses are presented for the ejection of part length control rod assemblies.

Results

Beginning of Life; Full Power

The control rod assembly program limits the control bank reactivity, holding it to 0.5% ΔK for this condition. The reactor is sub-prompt critical with the worst ejected rod worth of 0.22%ΔK. The peak power reached is 1.42 times full power and the peak hot spot clad and center fuel temperatures are 2220 °F and 4850 °F, respectively. The results are shown in Figures 14.3.3-1 through 14.3.3-3.

Beginning of Life; Zero Power

For this condition there is one control bank fully inserted and a second bank almost fully inserted. Only one pump has been assumed to be operating. The worst ejected rod worth of 0.71% ΔK results in the core becoming prompt critical. The peak hot spot cladding and center fuel temperatures are 1360 °F and 1710 °F, respectively. The results are shown in Figures 14.3.3-4 through 14.3.3-6.

End of Life; Full Power

The control rod assembly program limits the control bank reactivity, holding it to 0.3% ΔK. The worst ejected rod worth is 0.092% ΔK. The peak power reached is 1.20 times normal full power. Peak cladding and center fuel temperatures are 1930 °F and 4620 °F, respectively. The results are shown in Figures 14.3.3-3, 14.3.3-7, and 14.3.3-8.

End of Life; Zero Power

For this condition two control banks are fully inserted, and a third bank partially inserted. Assuming that only one reactor coolant pump is running, the worst ejected rod worth of 0.84% ΔK results in hot spot peak cladding and fuel temperatures of 2120 °F and 2900 °F, respectively. The results are shown in Figures 14.3.3-6 and 14.3.3-9, and 14.3.3-10.

Fission Product Release

It is assumed that fission products are released from the gaps of all fuel rods entering DNB. In all cases considered less than 15% of the rods entered DNB. (This corresponds to 2% of the core volume.) Fission product release is therefore much less than for the double ended coolant pipe break, the maximum hypothetical accident, for which over 70% of the rods are assumed to release fission products.

Pressure Surge

It is shown that there is no danger of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant. The most severe excess addition of energy to the coolant occurs for the high power end of life case. In order to estimate the magnitude of this pressure transient, average channel and hot spot heat transfer calculations were performed using a high gap conductance and without assuming DNB. The power curves used for these calculations represented a limiting case which initiated center melting at the hot spot. Using these heat flux data, a THINC 3 run was conducted to determine the volume surge without the benefit of pressure feedback. This volume surge was subsequently used as the basis for a pressure calculation. The results indicated that starting at 2250 psia, a peak pressure of about 2340 psia occurs some 1.5 seconds after a control rod assembly ejection.

Conclusions

Even on the most conservative basis, the analyses indicated no clad melting. It was concluded that there was no danger of sudden fuel dispersal into the coolant. The pressure surge was shown to be insufficient even to lift relief valves and it was concluded that there was no danger of consequential damage to the primary circuit. The amount of fission products released as a result of clad rupture during DNB is considerably less than in the case of the double ended reactor coolant pipe break.

TABLE 14.3.3-1

CONTROL ROD ASSEMBLY EJECTION DATA

Time in Life		Beginning	Beginning	End	End
Power Level		0	102% of 2546 MWt	0	102% of 2546 MWt
Ejected rod worth	%ΔK	0.71	0.22	0.84	0.092
Delayed neutron fraction	%ΔK	0.69	0.69	0.52	0.52
Feedback reactivity weighting		1.65	1.65	3.53	1.42
trip rod worth	% ΔK	2.98	4.83	0.76	4.85
Average core gap heat transfer coefficient	$Btu/hr\text{-}ft^2\text{-}°F$	10,000	10,000	750	750
Average core fuel thermal conductivity	Btu/hr-ft-°F	2.65	2.65	2.2	2.2
Initial hot spot gap heat transfer coefficient	$Btu/hr\text{-}ft^2\text{-}°F$	750	2,000	750	2,000
Transient hot spot gap heat transfer coefficient	$Btu/hr\text{-}ft^2\text{-}°F$	10,000	10,000	10,000	10,000
Initial moderator density coefficient	$\%\Delta K/gm/cm^3$	0	0	0.208	0.161
Prompt neutron lifetime	μ sec	15	15	15	15
F_q after control rod assembly ejection		5.46	5.40	16.34	4.20
F_q before control rod assembly ejection		-	2.80	-	2.80
Number of operating pumps		1	3	1	3
Max. fuel pellet average temperature	°F	1550	3629	2560	3280
Max. fuel center temperature	°F	1710	4850	2900	4620
Max. clad temperature	°F	1360	2220	2120	1930

REFERENCES

1. "Power Distribution Control of Westinghouse PWR's," WCAP-7208 (1968).

2. Redfield, J. A., "CHIC-KIN -- A Fortran Program for Intermediate and Fast Transients in a Water Moderated Reactor," WAPD-TM-479, January, 1965.

3. Barry, R. F., "The Revised LEOPARD Code - A Spectrum Dependent Non-Spatial Depletion Program," WCAP-2759 (1965).

14.3.3.1 Effects on Adjacent Housings

A control rod drive mechanism assembly is shown in Section 3. The operating coil stack assembly of this mechanism has a 10.718 in by 10.718 in cross section and 39.875 in length. The position indicator coil stack assembly is located above the operating coil stack assembly. It surrounds the rod travel housing over nearly its entire 163.25 in length. The rod travel housing outside diameter is 3.75 in and the position indicator coil stack assembly inside and outside diameters are 3.75 in and 7.0 in, respectively. This assembly consists of a Micarta tube surrounded by a continuous stack of copper wire coils. This assembly is held together by two end plates, an outer sleeve, and four axial tie rods.

Effects of Rod Travel Housing Longitudinal Failures

Should a longitudinal failure of the rod travel housing occur, the region of the Micarta tube opposite the break is stressed by the reactor coolant pressure of 2250 psia. The most probable leakage path is provided by the radial deformation of the position indicator coil assembly, resulting in the growth of axial flow passages between the rod travel housing and the Micarta tube. A radial free water jet is not expected to occur because of the small clearance between the Micarta tube and the rod travel housing, and the considerable resistance of the combination of the Micarta tube and the position indicator coils to internal pressure.

Calculations based on experimental data on the mechanical properties of Micarta and copper at reactor operating temperature show that an internal pressure of at least 2500 psia would be necessary for the combination of the Micarta tube and the coils to start leaking in a radial direction between the Micarta glass filaments.

The normal operating environment of the Micarta tube is strictly controlled during unit operation, and therefore, no deterioration of the Micarta is expected. Should for unknown reasons mechanical strength of the Micarta tube be reduced and a longitudinal crack occur in a control rod assembly housing, weepage flow between the Micarta filaments and the copper coil wires might take place, but no free jet is expected to form. The formation of a free jet implies cracking of the Micarta tube, which could occur only with internal pressures substantially in excess of reactor operating pressure. Prolonged exposure to hot water might cause deterioration of the Micarta and radial leakage might increase. Even under these conditions, however, a net radial free jet is not expected to occur.

A position indicator coil assembly has to maintain its integrity after a housing failure only until the remaining control rod assembly can be tripped into the core. Should for unknown reasons failure of the position indicator coil assembly occur after reactor trip, the resulting free radial jet from the failed housing can cause the housing to bend and contact adjacent rod travel housings. If the adjacent housings were on the periphery they might bend outward from their bases. The housing material is quite ductile and plastic hinging without cracking is expected. Rod travel housings adjacent to a failed housing in locations other than the periphery would not be bent because of the rigidity of multiple adjacent housings.

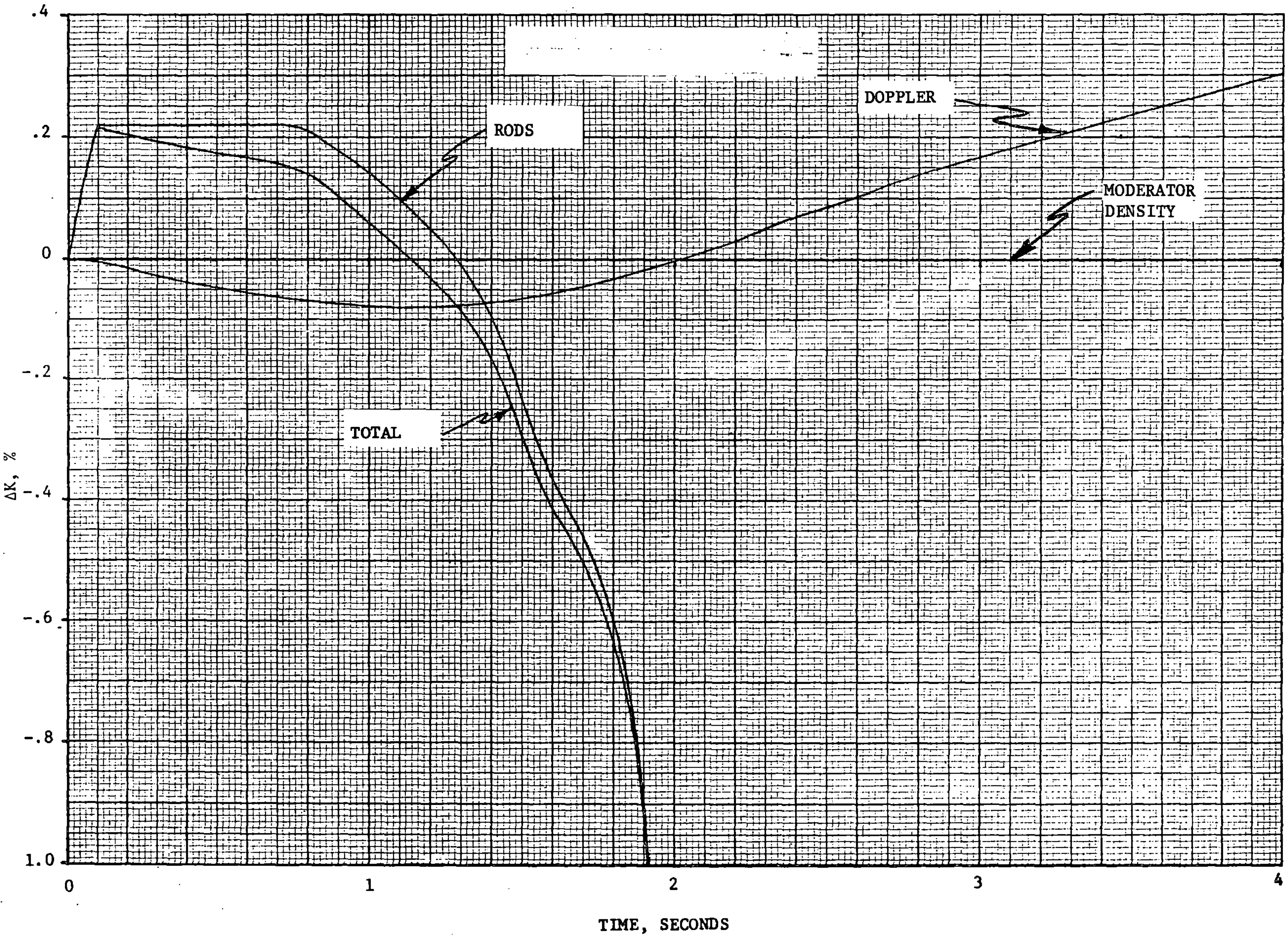
Effect of Rod Travel Housing Circumferential Failures

If circumferential failure of a rod travel housing occurs, the broken-off section of the housing is ejected vertically because the driving force is vertical. The position indicator coil stack assembly and the drive shaft tend to guide the broken-off piece upwards during its travel. Travel is limited to less than three feet by the missile shield; thereby limiting the projectile acceleration. When the projectile reaches the missile shield, it partially penetrates the shield and dissipates its kinetic energy. The water jet from the break continues to push the broken-off piece against the missile shield.

If the broken-off piece of the rod travel housing is short enough to clear the break when fully ejected, it would rebound after impact with the missile shield. The top end plates of the position indicator coil stack assemblies would prevent the broken piece from directly hitting the rod travel housing of a second drive mechanism. Even if a direct hit by the rebounding piece were to occur, the low kinetic energy of the rebounding projectile would not be expected to cause significant damage.

Summary

In summary, the considerations given above lead to the conclusion that failure of a control rod assembly housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings that would increase the severity of the initial accident.



REACTIVITY vs TIME
BEGINNING OF LIFE, FULL POWER

Figure 14.3.3-1

FUEL ROD TEMPERATURES VS. TIME

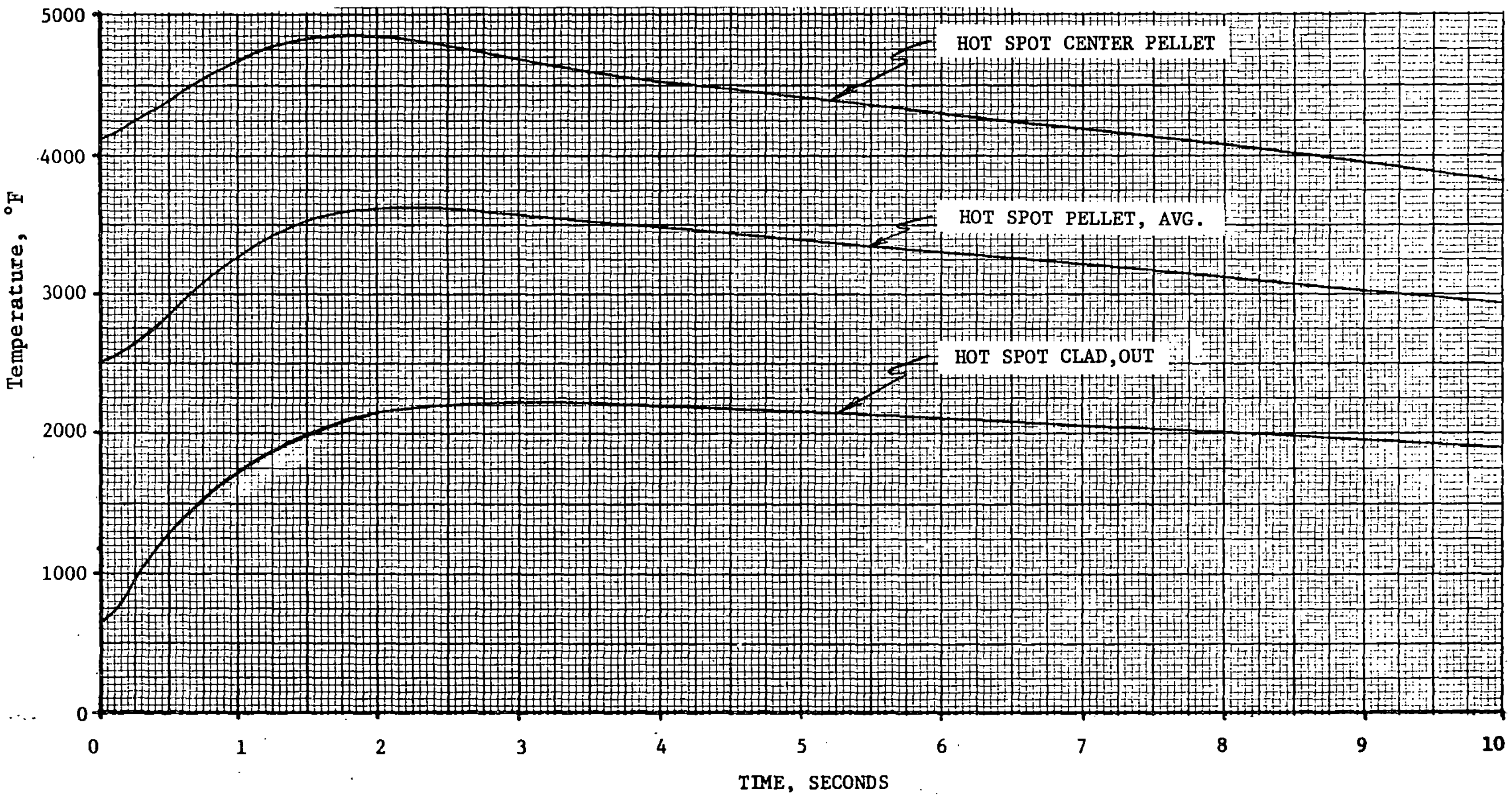


Figure 14.3.3-2
12-1-69

POWER vs TIME

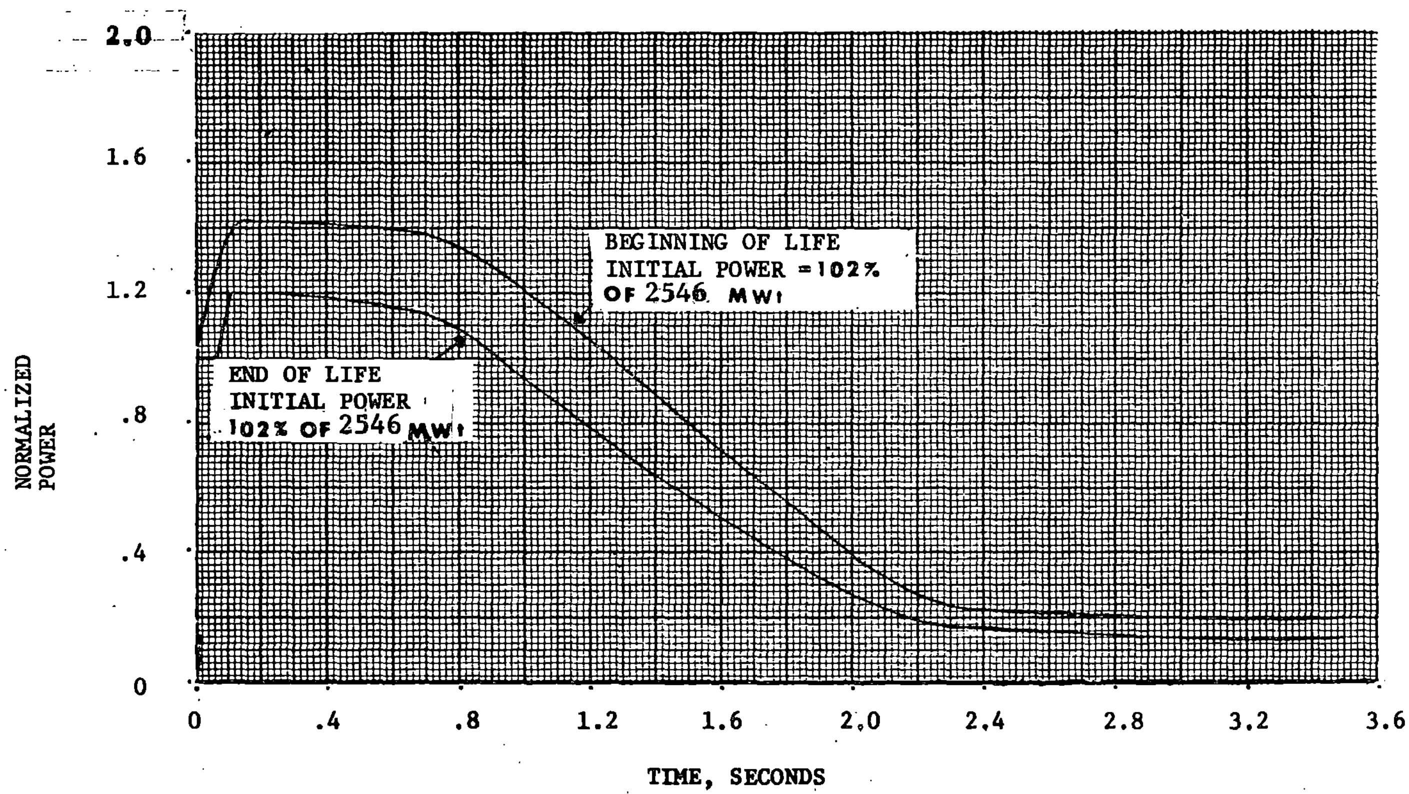
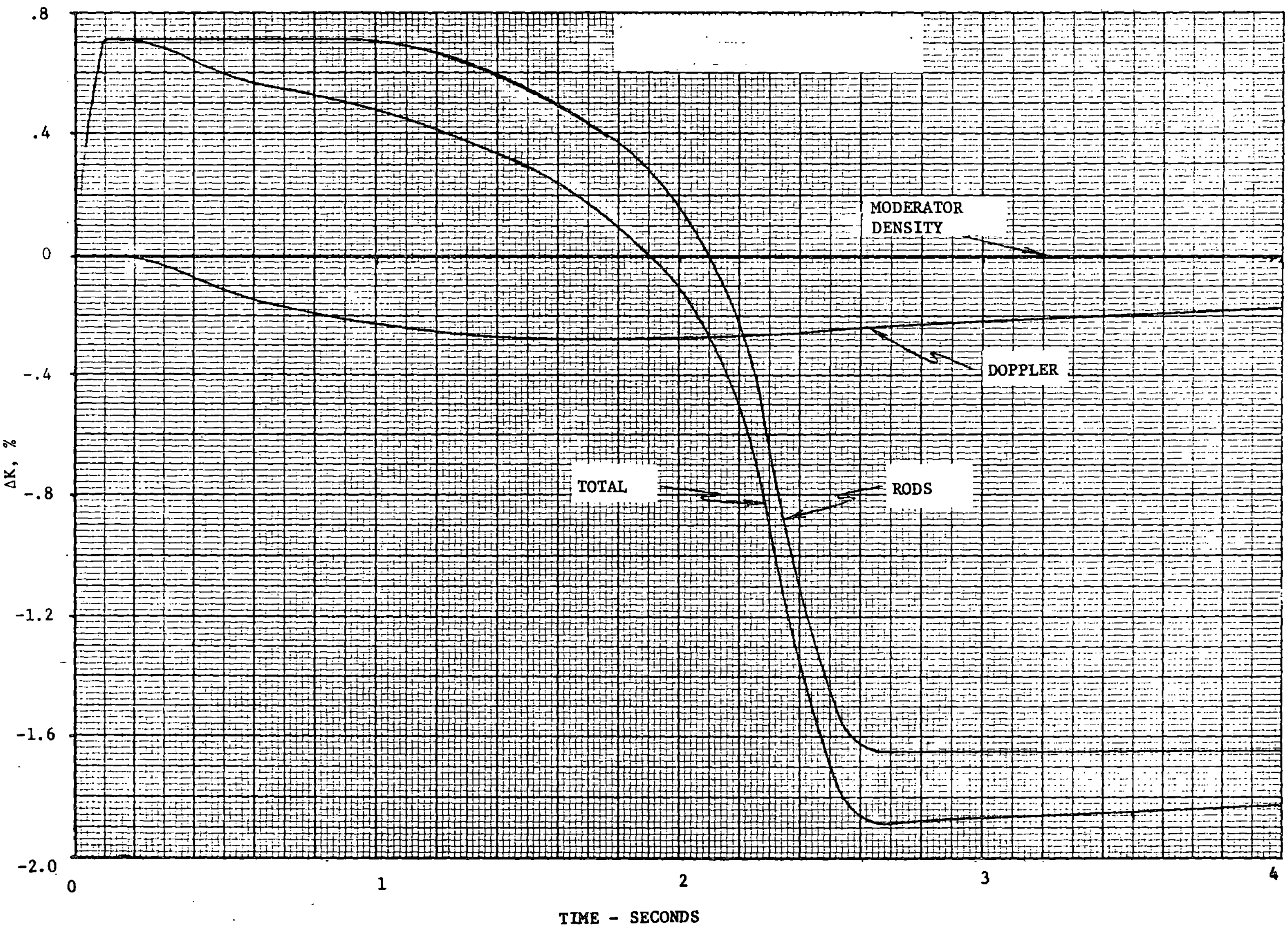


Figure 14.3.3-3
12-1-69



REACTIVITY vs TIME
BEGINNING OF LIFE, ZERO POWER

Figure 14.3.3-4
12-1-69

FUEL ROD TEMPERATURES vs TIME
BEGINNING OF LIFE, ZERO POWER
1 LOOP OPERATION

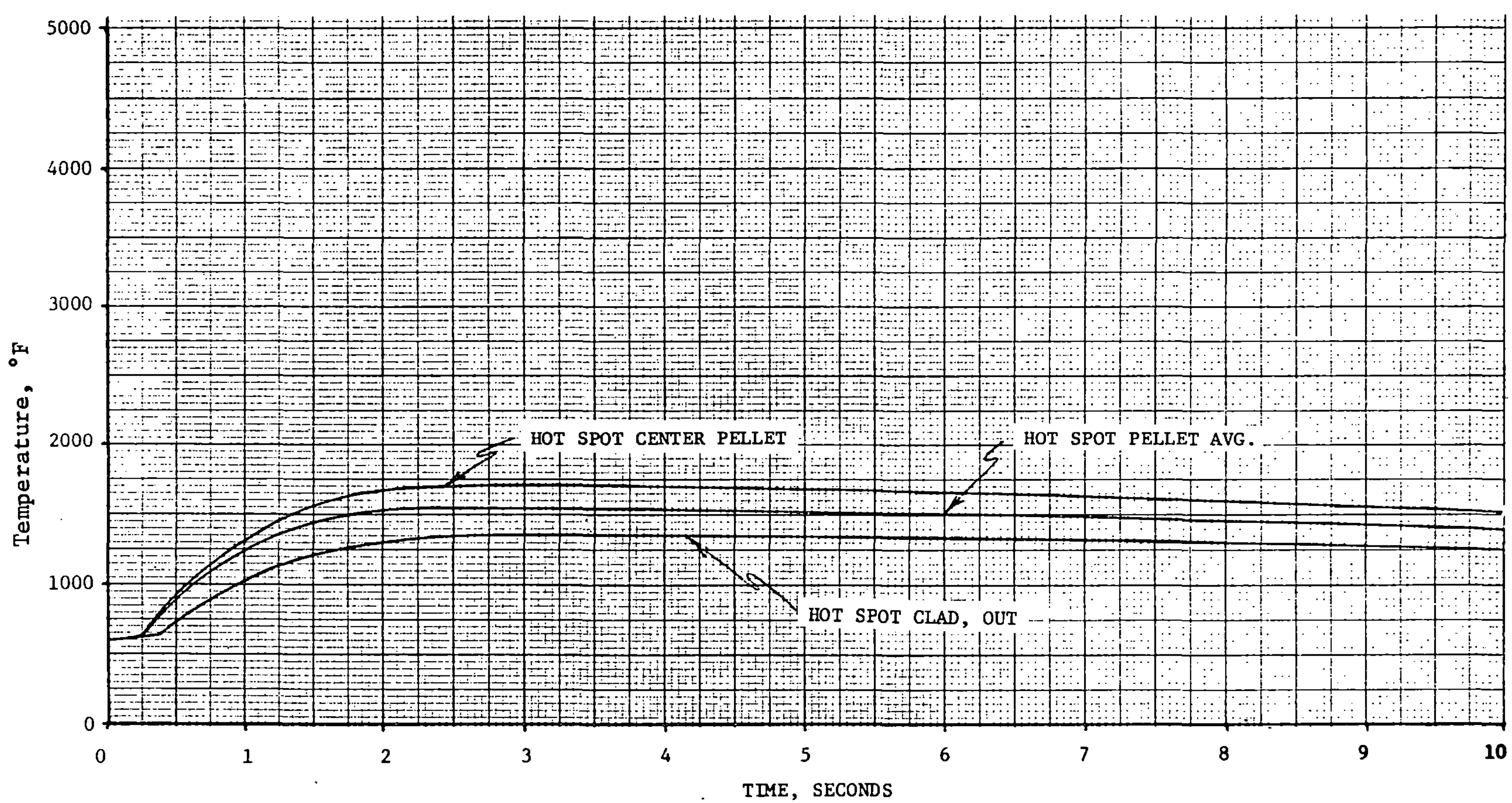
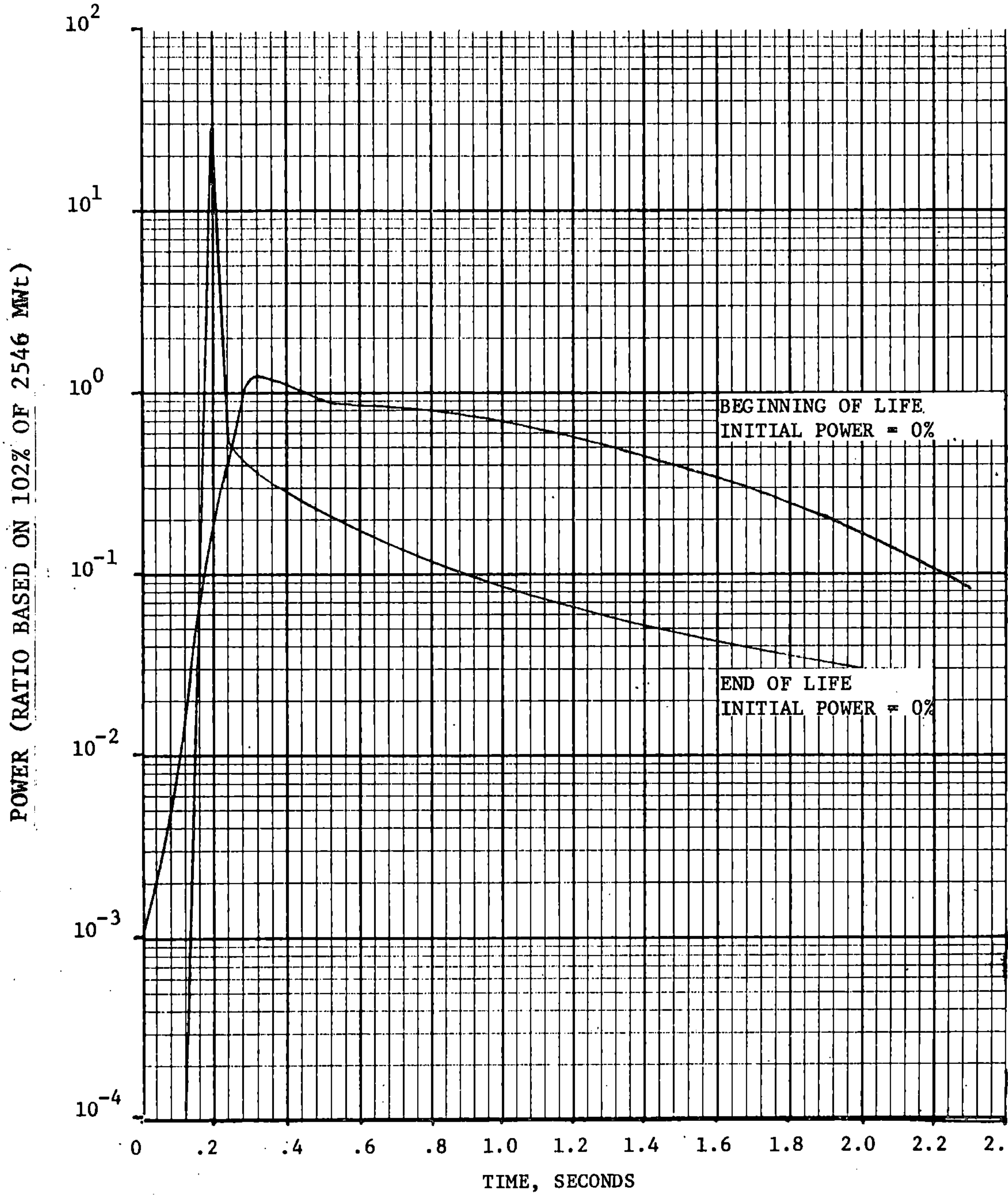


Figure 14.3.3-5
12-1-69

Figure 14.3.3-6
12-1-69



POWER vs TIME

REACTIVITY vs TIME
END OF LIFE, FULL POWER

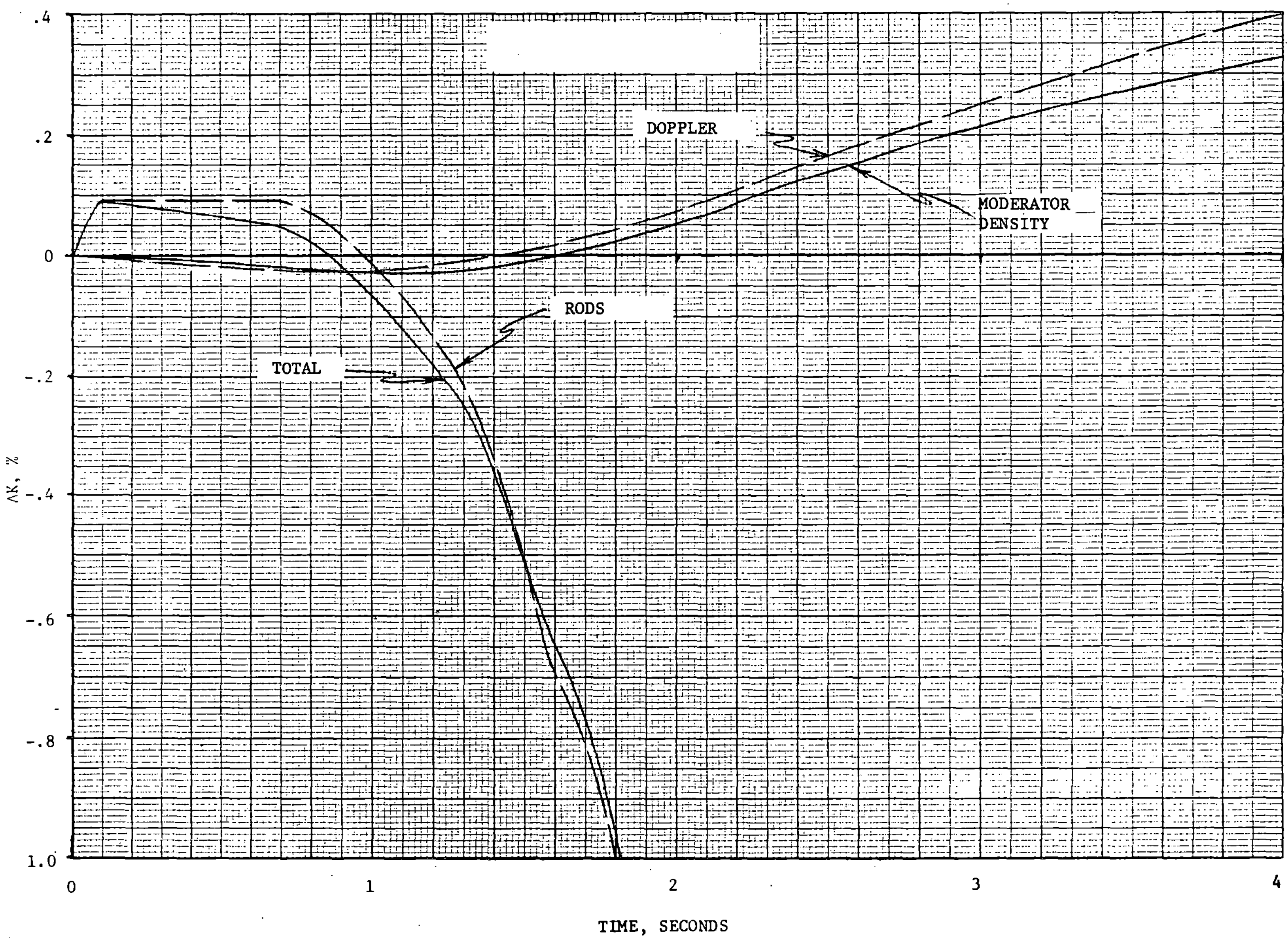


Figure 14.3.3-7
12-1-69

FUEL ROD TEMPERATURES vs TIME
END OF LIFE, FULL POWER
3 LOOP OPERATION

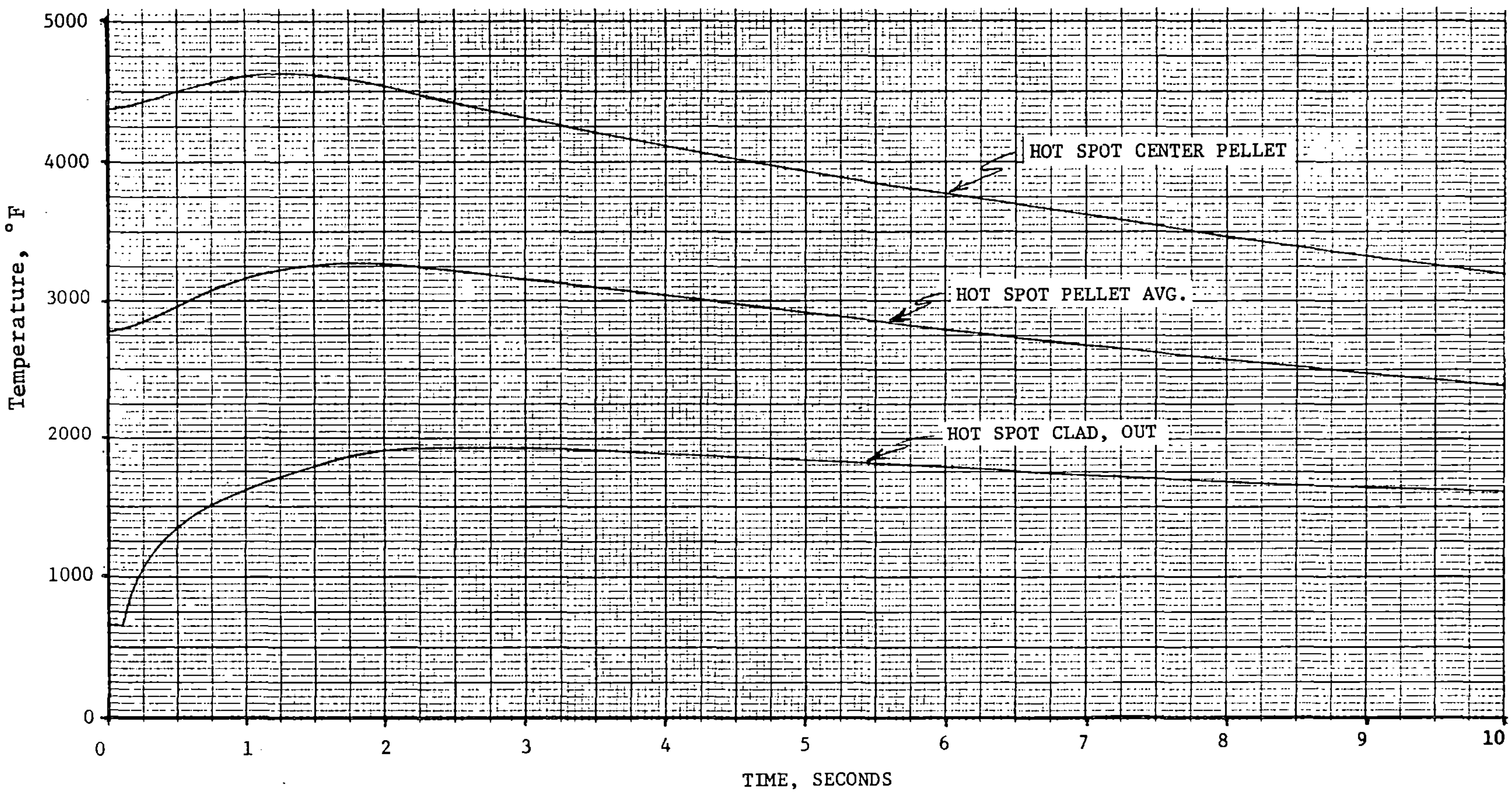


Figure 14.3.3-8
12-1-69

REACTIVITY vs TIME
END OF LIFE, ZERO POWER

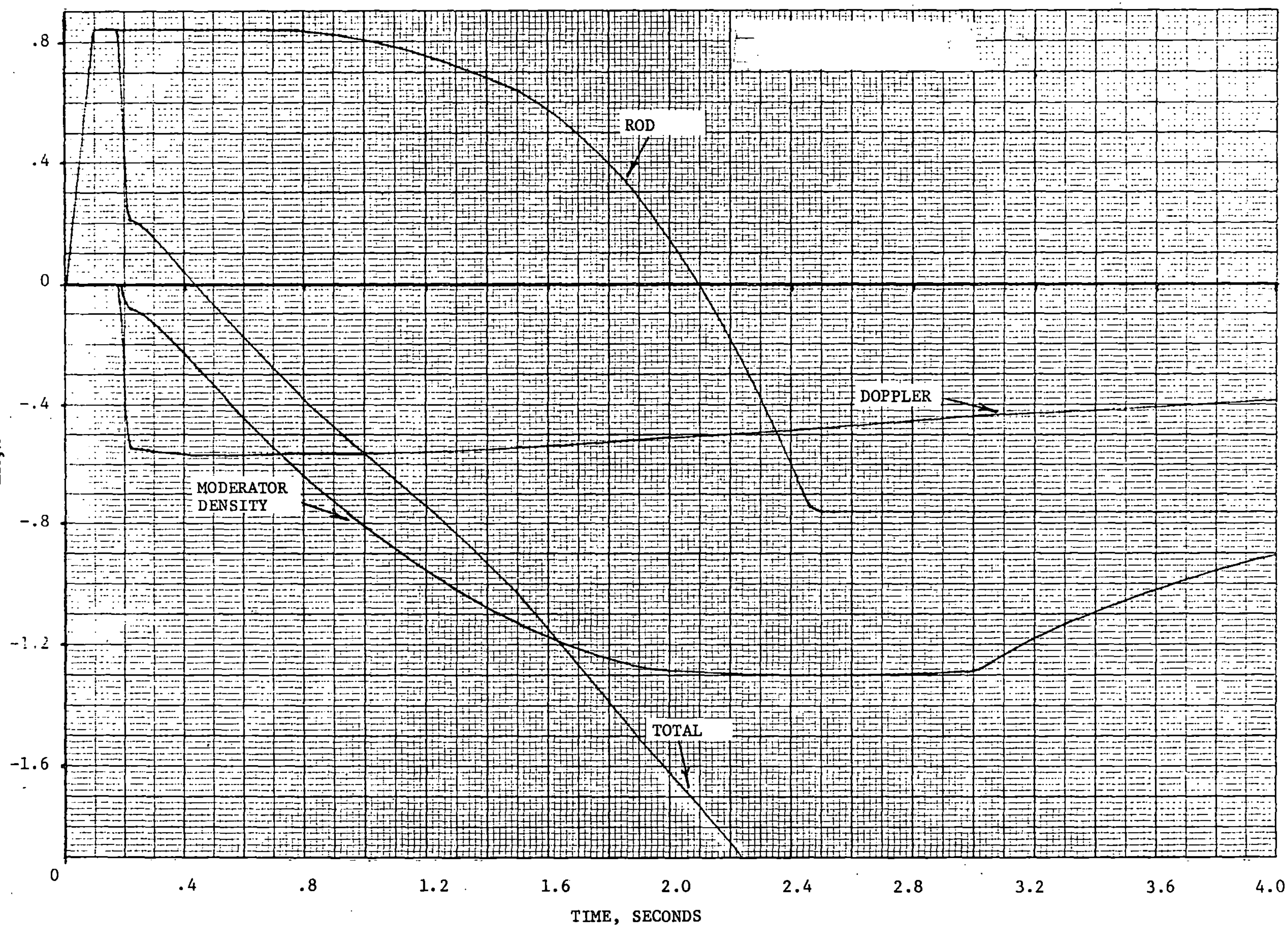
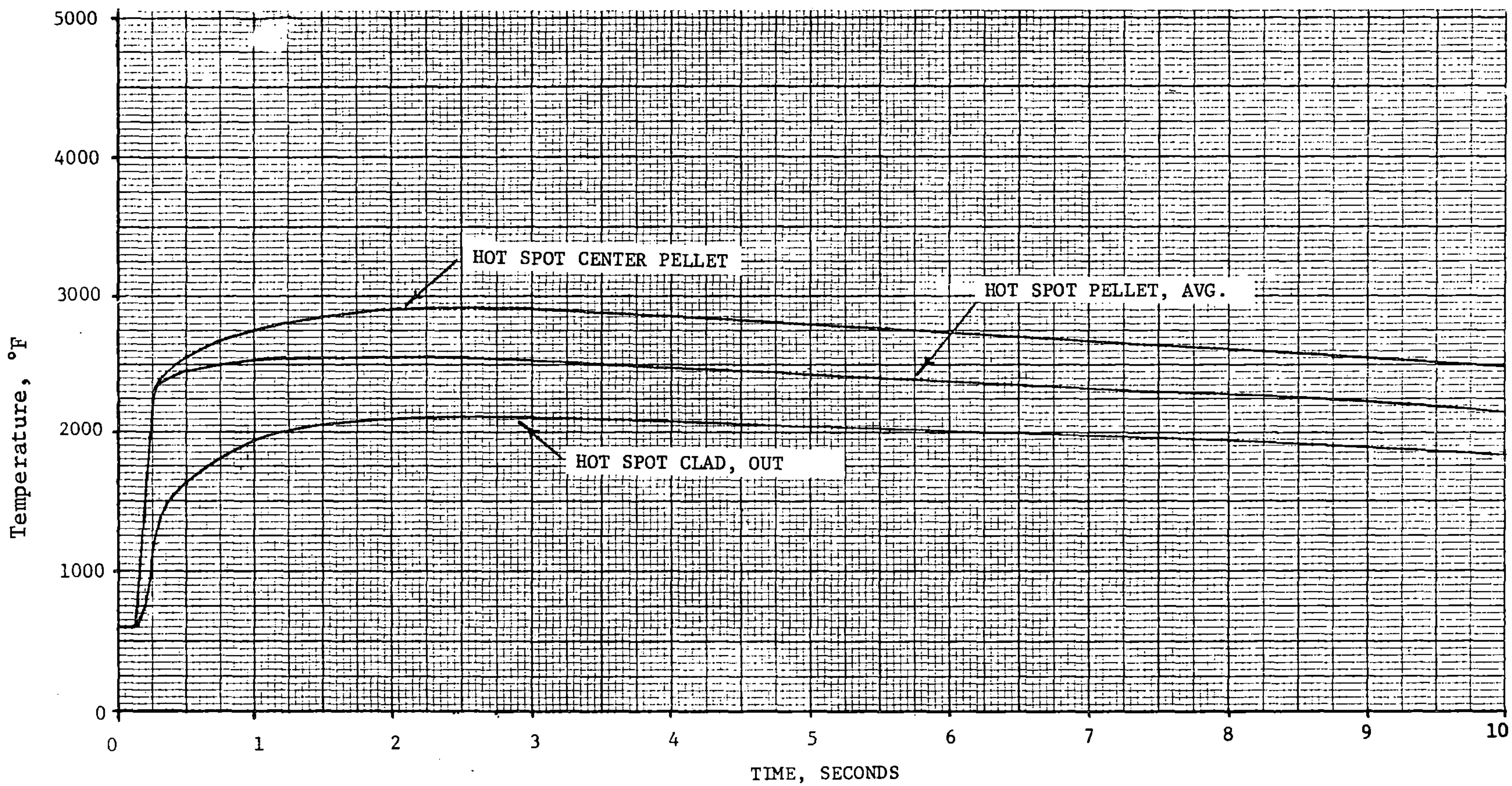


Figure 14.3.3-9
12-1-69



FUEL ROD TEMPERATURES vs TIME
END OF LIFE, ZERO POWER
1 LOOP OPERATION

Figure 14.3.3-10
12-1-69

14.4.1 FUEL HANDLING ACCIDENTS

The following fuel handling accidents are evaluated to ensure that no hazards are created:

a) A fuel assembly becomes stuck inside reactor vessel.

b) A fuel assembly or control rod assembly is dropped onto the floor of the reactor cavity or spent fuel pit.

c) A fuel assembly becomes stuck in the containment penetration valve.

d) A fuel assembly becomes stuck in the transfer carriage or the carriage becomes stuck.

14.4.1.1 Causes and Assumptions

The possibility of a fuel handling accident is remote because of the stringent administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety. Also, before any refueling operations begin, verification of complete control rod assembly insertion is obtained by tripping each control rod assembly individually to obtain indication of rod drop and disengagement from the control rod drive mechanisms. Boron concentration in the reactor coolant is raised to the relatively high refueling concentration and verified by sampling. Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical with all control rod assemblies withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications. As the vessel head is raised, a visual check is made to verify that the control rod assembly drive shafts are free in the mechanism housing.

After the vessel head is removed, the control rod assembly drive shafts are removed from their respective assemblies using the manipulator crane hoist and the shaft unlatching tool. A spring scale is used to indicate that the drive shaft is free of the control rod assembly as the lifting force is applied.

The fuel handling manipulators and hoists are designed so that fuel cannot be raised above a position which provides adequate shield water depth for the safety of operating personnel. This safety feature applies to handling facilities in both the containment and in the spent fuel pit area. In the spent fuel pit, the design of storage racks and manipulation facilities is such that:

Fuel at rest is positioned by positive restraints in an eversafe, always subcritical, geometrical array, with no credit for boric acid in the water.

Fuel can be manipulated only one assembly at a time.

Violation of procedures by placing one fuel assembly in juxtaposition with any group of assemblies in racks will not result in criticality.

Crane facilities do not permit the handling of heavy objects, such as a spent fuel shipping container, above the fuel racks.

Adequate cooling of spent fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel assembly is immersed continuously while in the refueling cavity or spent fuel pit.

Even if a spent fuel assembly becomes stuck in the transfer tube, natural convection maintains adequate cooling. The fuel handling equipment is described in detail in Section 9.

Two Nuclear Instrumentation System source range channels are continuously in operation and provide warning of any approach to criticality.

during refueling operations. This instrumentation provides a continuous audible signal in the containment, and would annunciate a local horn and a horn and light in the Main Control Room if the count rate increased above a preset low level.

Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical by at least 10% $\frac{\Delta k}{k}$ with all control rod assemblies inserted. At this boron concentration the core would also be more than 2% $\frac{\Delta k}{k}$ subcritical with all control rod assemblies withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications.

All these safety features make the probability of a fuel handling accident very low. Nevertheless, it is possible that a fuel assembly could be dropped during the handling operations. Therefore, this accident is analyzed both from the standpoint of radiation exposure and accidental criticality.

Special precautions are taken in all fuel handling operations to minimize the possibility of damage to fuel assemblies during transport to and from the spent fuel pit and during installation in the reactor. All handling operations on irradiated fuel are conducted under water. The handling tools used in the fuel handling operations are conservatively designed and the associated devices are of a fail-safe design.

In the fuel storage area, the fuel assemblies are spaced in a pattern which prevents any possibility of a criticality accident. The motions of the cranes which move the fuel assemblies are limited to a relatively low maximum speed. Caution is exercised during fuel handling to prevent the fuel assembly from striking another fuel assembly or structures in the containment or fuel building.

The fuel handling equipment suspends the fuel assembly in the vertical position during fuel movements, except when the fuel is moved through the transport tube.

The design of the fuel assembly is such that the fuel rods are restrained by grid clips which provide a total restraining force of approximately 60 pounds on each fuel rod. If the fuel rods are in contact with the bottom plate of the fuel assembly, any force transmitted to the fuel rods is limited due to the restraining force of the grid clips. The force transmitted to the fuel rods during fuel handling is not sufficient to breech the fuel rod cladding. If the fuel rods are not in contact with the bottom plate of the assembly, the rods would have to slide against the 60 pound friction force. This would absorb the shock and thus limit the force on the individual fuel rods.

After the reactor is shut down, the fuel rods contract during subsequent cooldown and would not be in contact with the bottom plate of the assembly.

Considerable assembly 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 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 assembly and grid clips and essentially no damage would be expected in any fuel rods.

If the fuel assembly was 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 assumed the extremely remote situation when a fuel assembly is dropped 14 feet and strikes a flat surface, when one assembly is dropped on another, and when one assembly strikes a sharp object.
The analysis of a fuel assembly assumed to be dropped and striking a flat surface considered the stresses the fuel cladding was subjected to and any possible buckling of the fuel rods between the grid clip supports. The results showed that the axial load at the bottom section of the fuel rod,

which would receive the highest loading (approximately 100 lb.) was below the critical buckling load (250 lb.) and the stresses were relatively low and below the yield stress. For the case where one assembly was to be dropped on top of another fuel assembly, the loads will be transmitted through the end plates and the control rod assemblies guide tubes of the struck assembly before any of the loads reach the fuel rods.

The end plates and guide thimbles absorb a large portion of the kinetic energy as a result of bending in the lower plate of the falling assembly. Also, energy is absorbed in the struck assembly top end plate before any load can be transmitted to the fuel rods. The results of this analysis indicated that the buckling load on the fuel rods was below the critical buckling loads and the stresses in the cladding were relatively low and below yield.

The refueling operation experience that has been obtained with Westinghouse reactors has verified the fact that no fuel cladding integrity failures are expected to occur during any fuel handling operations.

Rupture of one complete outer row of fuel rods in a withdrawn spent fuel assembly is assumed as a conservative limit for evaluating the environmental consequences of a fuel handling accident. The remaining fuel assemblies are so protected by the storage rack structure so they are not subjected to lateral bending loads. No damage resulted from an axially applied load of 2200 lb. to a fuel assembly. The maximum load expected to be experienced in service is approximately 1000 lb. This information was used in the fuel handling equipment design to establish the limits for inadvertent axial loads.

14.4.1.2 Activity Release Characteristics

As indicated in the previous sections, the failure of an entire row of fuel rods (15) is considered to be a very conservative assumption. However, for the purpose of this analysis it is assumed that all 204 fuel rods in the assembly rupture and there is a sudden release of the gaseous fission products held in the voids between the pellets and cladding of the fuel rods. The low temperature of the fuel during handling operations precludes further significant release of gases from the pellets themselves after the cladding is breached. Halogen release is also greatly minimized due to their low volatility at these temperatures. The strong tendency for iodine in vapor and particulate form to be scrubbed out of gas bubbles during their ascent to the water surface further reduces the quantity released from the water surface.

Decontamination factors of 1000[1] have been measured with much shallower water depths and much higher gas-to-water ratios. In a Westinghouse laboratory apparatus, elemental iodine (I_2) was passed in an air stream through a solution of 2000 ppm boron as boric acid. This solution is chemically similar to that in the spent fuel storage pool. 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 10. The value decreased with the 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 sodium thiosulfate as an iodine fixing reagent 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 iodined free solution in the spent fuel pool. 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 station, compared with 1.6 cm in the experiment. Assuming the same mass transfer rate in the bubble, the D.F. of 1000 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. Even though this experiment seems to justify the use of a D.F. of 1000, a D.F. of 10 is used in this analysis so as to be consistent with previous D.F. values used in safety analysis reports, as amended, for the Maine Yankee Atomic Power Station and Beaver Valley Power Station

If the fuel assembly is dropped in the spent fuel pit in the fuel building, the increase in radiation level as these radionuclides mix with the fuel building air will be detected by the two radiation monitors located in the ventilation vent or by the fuel pit bridge area monitor. For the duration of fuel handling operations, fuel building exhaust air will be manually diverted through the charcoal filters. For the purpose of site boundary dose calculations, it is assumed that these filters will have a collection efficiency of 90 percent for iodine. It is further assumed that Pasquill "F" meteorology conditions exist with a one meter per second wind speed yielding a dispersion coefficient, χ/Q, equal to 8.14×10^{-4} sec/m^3.

The I-131 equivalent and noble gas activities released to the pool water are listed in Table 14.4.1.2-1. These activities are based on the assumption that 20% of the contained noble gas inventory and 10% of the halogen inventory of the highest power assembly are released to the pool water. This conservatively assumes that all of the iodine in the gaps of 204 rods of the highest power assembly will be present in the gaseous phase. Also assumed is a 100 hour decay period following operation at 2550 MWt for 23,000 hours. This period is the minimum term within which the reactor vessel head and a fuel assembly can be removed following shutdown. Assuming a D.F. of 10, the I-131 equivalent released from the water surface would be 4.51×10^3 curies.

A summary of the assumptions made in the calculation is presented below:

Number of Failed Rods	- 204
Radial Assembly Peaking Factor	- 1.5
Iodine Activity Released to Water	- 10% of total assembly activity after 100 hour decay
Noble Gas Activity Released to Water	- 20% of total assembly activity after 100 hour decay
D.F. of Pool Water for Iodine Removal	- 10
Charcoal Filter System Efficiency	- 90%
χ/Q	- 8.14×10^{-4}

The whole body dose at the site boundary resulting from the release would be about 8.24 rem and the thyroid dose (using TID-14844 method) would be about 188.6 rem. Both values are below those suggested in "10 CFR 100".

The assumptions made in performing this calculation are considered to be very conservative. Decontamination factors of between 100 and 500 would seem to be justified by previous experimental data. The more realistic site boundary doses resulting from the accident would be about 18.86 rem thyroid dose and 8.24 rem whole body dose for a D.F. of 100. If the spent fuel pool water has a D.F. of 500, the thyroid dose would be about 3.77 rem and the whole body dose would be 8.24 rem. These values are well below those suggested in "10 CFR 100."

It is concluded that a dropped assembly would not result in an excessive radiation exposure at the site boundary. If the fuel assembly is dropped in the refueling cavity in the containment, the increase in radiation level, as these radionuclides mix with the containment air will be detected by either the containment gas and particulate monitors or the manipulator crane area monitor. An alarm by any one of these monitors will automatically trip the purge air supply and exhaust fans and close the purge system butterfly valves, thus isolating the system. Because at least one door of the personnel hatch and the equipment hatch are closed during refueling, isolation of the purge system isolates the containment and, consequently, no offsite dose results from a dropped fuel assembly in the containment.

TABLE 14.4.1.2-1

ACTIVITY RELEASE TO POOL WATER
FROM FUEL HANDLING ACCIDENT

Noble Gas	Activity, Curies
Kr-85	1564
Xe-133m	3.54×10^3
Xe-133	1.86×10^5
Halogens	
I-131 (Equivalent)	4.51×10^4

REFERENCE

(1) Diffey, H.R. et. al., " Iodine Clean-up in a Steam Suppression System," International Symposium on Fission Product Release and Transport Under Accident Conditions, Oak Ridge, Tennessee, CONF-65047, Vol. 2, Pg. 776-804, (1965).

14.4.2 RADIOACTIVE GAS RELEASE

The concentration of radioactive waste gases in the primary and auxiliary systems is a function of the rate of fission gas release to the coolant from defective fuel and the rate of removal via the auxiliary systems. The components which retain significant concentrations of radioactive gases are the volume control tank and the waste gas decay tanks. The radioactive gas release analysis considers the rupture of the volume control tank and a waste gas decay tank with the instantaneous release of the radioactive gas inventories of each to the environment.

14.4.2.1 Volume Control Tank Rupture

In this analysis, the volume control tank is assumed to rupture and release to the atmosphere all of the gases which have collected in the vapor space of the tank. Also released are all of the gases and all of the iodines in the liquid inventory of the tank and in the volume of liquid which continues to flow into the tank, until it is isolated. Isolation is assumed to take five minutes, and the flow rate of the entering liquid is assumed to be 60 gpm, which is the normal letdown flow rate.

The maximum activities of the gases in the vapor space with one percent failed fuel are listed in Table 9.1-6. The activities of the gases and iodines in the liquid are based upon the reactor coolant equilibrium activities with one percent failed fuel as listed in Table 9.1-5. For the accident analysis, activities in the liquid have been corrected for density, and a decontamination factor of ten has been applied to the iodines to take credit for letdown flow through the mixed-bed demineralizer prior to the flow entering the volume control tank. This results in a total activity available for release equal to about 15,000 curies of Xe-133 equivalent and 2.2 curies of I-131 equivalent.

Using these sources, a χ/Q from Figure 2.2-9 of 8.0×10^{-4} sec/m^3, and assuming a "puff" ground release, the whole body dose at the exclusion area boundary would be about 0.64 rem, and the thyroid dose about 0.90 rem. These doses are well below those suggested in 10CFR100 and, therefore, this accident is not considered to be an undue hazard.

14.4.2.2 <u>Waste Gas Decay Tank Rupture</u>

The operation of the Waste Gas System is described in Section 11.2.5. The concentrations of radionuclides in the waste gas decay tank are listed in Table 11.2.5-1, which is for a cycle with the recombiner not operating, and Table 11.2.5-2, which is for a cycle with the recombiner operating. As can be seen by comparing these tables, the highest inventory of waste gases occurs at the end of the feed cycle with the recombiner operating. For this accident analysis, it is at this time that the waste gas decay tank is assumed to rupture. Since the tanks are double-walled, include overpressure protection, and are located underground, no mechanism for such rupture is credible. No credit has been taken for absorption or adsorption of particulate matter in the earthen overburden on the tank.

The total activity released would be equivalent to 95,400 curies of Xe-133 and 0.26 curies of I-131.

Since these buried tanks are located in the yard, no credit for building wake is taken. From Figure 2.2-9, this results in a χ/Q of 2.0×10^{-3} sec/m^3. The ensuing whole body dose at the exclusion boundary would be about 10.1 rem and the thyroid dose about 0.27 rem. These doses are below those suggested in 10CFR100 and, therefore, this accident is not considered to be an undue hazard.

14.4.3 RADIOACTIVE LIQUID RELEASE

Accidents in the auxiliary systems which could result in the release of waste liquids must necessarily involve the rupture or leaking of various pipelines, valves, tanks, and pumps.

All liquid processing components are located within the auxiliary building, fuel building, decontamination building, and station yard area. Any liquid leakage or release from these components is collected in sumps and pumped to the Liquid Waste Disposal System (Section 11.2.2.1), or flows directly to the Vent and Drain System (Section 9.7). The auxiliary building and fuel building are of Class I Design.

The boron recovery tanks are located in the station yard area in separately diked enclosures, each of which is of sufficient capacity to retain the total liquid volume resulting from rupture of one boron recovery tank without overflowing to areas outside the enclosure. The collected liquid is pumped either to the unruptured boron recovery tanks or to the Liquid Waste Disposal System. The diked enclosure is of Class I Design.

Piping running between the auxiliary building and the reactor containment, the auxiliary and fuel buildings, and the fuel building and the tanks in the yard area is below grade in concrete trenches or in special piping conduits. Liquids spilled or released from such piping are collected in sumps and pumped into the Liquid Waste Disposal System. Accordingly, any possible release of waste liquids is contained within the station and does not result in uncontrolled release to the environment.

14.5 LOSS-OF-COOLANT ACCIDENT

14.5.1 GENERAL

A loss-of-coolant accident can 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, i.e., equivalent to a 0.75 in. diam hole or less, can cause expulsion of coolant at a rate which can be accommodated by the high head charging pumps, as discussed in Section 6. Should such an accident occur, these pumps can maintain an operational level of water in the pressurizer, permitting the operator to shut down the unit in an orderly manner. 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 system pressure and pressurizer level ultimately causes reactor trip and actuates the Safety Injection System, as discussed in Section 6.2. These countermeasures limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection supplements void formation in the core and causes rapid reduction of nuclear power to a residual level corresponding to delayed fissions and fission product decay.

2. Injection of borated water assures sufficient flooding of the core to prevent excessive fuel temperatures.

Operation of the Safety Injection System prevents melting and/or rupture of fuel cladding, limits the metal-water reaction to a negligible amount (less than 1 percent), and prevents gross core geometry distortion for reactor coolant piping ruptures up to and including the diameter of the largest pipe connected to the reactor vessel. For the case of the rupture of either a hot or cold leg in the Reactor Coolant System, the Design Basis Accident (DBA) is described in Section 14.5.2.

A rupture in the Reactor Coolant System results in the discharge to the containment of reactor coolant and associated heat. The result of this discharge is a decrease in coolant pressure in the Reactor Coolant System and an increase in containment temperature and pressure. Engineered Safeguards then operate, if required, to protect the core and return the containment to subatmospheric pressure.

The initial discharge of coolant is subcooled with respect to Reactor Coolant System pressure and, for discharge rates above charging capability, results in rapid depressurization of the Reactor Coolant System to saturation pressure. During this period of subcooled blowdown, reactor and turbine trips are initiated by low Reactor Coolant System pressure and low pressurizer level. Prior to initiation of reactor trip, heat from full reactor power is added to the coolant, but is removed by forced convection heat transfer in the steam generators. As coolant discharge continues after reaching saturation pressure, a portion of the coolant in the primary system flashes to steam, resulting in the discharge of a steam-water mixture into the containment. The high blowdown flow velocity after complete severance of a large reactor coolant pipe results in rapid steam evolution and the transport of large quantities of entrained liquid or essentially a homogeneous steam-water mixture. A distinct interface does not exist as the core becomes uncovered and water may only remain in the bottom of the reactor vessel at the completion of the rapid blowdown.

During blowdown, core residual heat from delayed fission, fission and capture-product decay, and sensible heat is transferred to the coolant. This sensible heat includes the heat stored in both the core and the core structure and a portion of the thick metal of the Reactor Coolant System. (See Section 14.5.2.1)

During the initial coolant blowdown phase, the Reactor Coolant System low pressure signal initiates reactor trip and, when in conjunction with pressurizer low level signals, initiates safety injection. The safety injection accumulators automatically start to discharge their contents into the Reactor Coolant System when the system pressure is less than accumulator pressure (minimum 600 psig). Safety injection also occurs concurrently depending upon the Reactor Coolant System pressure by means of the combined charging pumps and the low head safety injection pumps which take suction from the refueling water storage tank. Long-term cooling of the core is accomplished by switching to the recirculation mode of core cooling, in which the spilled borated water is drawn from the containment sumps by the low head safety injection pumps and returned to the reactor vessel by the low head safety injection pumps and/or the charging pumps. Residual heat generation and any zirconium-water reaction heat increases the stored energy in the core and reactor vessel for a short period of time until borated safety injection water reaches the vessel. A portion of the safety injection flow flashes to steam as it comes in contact with the hot core, the vessel, and the internal structures, and the steam flows to the containment through the pipe break. The safety injection water from the refueling water storage tank and the containment sumps is subcooled with respect to total reactor pressure; however, some initial flashing may occur.

A rise in containment pressure to approximately 15.7 psia (1.0 psig) initiates containment isolation by isolating nonessential process lines and the Containment Vacuum System (See Section 6.3.2). A further rise to approximately 24.7 psia (10 psig) results in initiation of the Containment Spray Systems (Section 6.3.1). The Containment Spray Subsystems transfer chilled, borated water, plus additional chemicals for fission product removal, from the refueling water storage tank to the containment via the containment spray headers. The spray water droplets absorb heat and airborne fission products from the steam-air atmosphere of the containment by heat and mass transfer, resulting in a decrease in the containment temperature and pressure. The chemical additive (NaOH) in the spray also reduces the airborne, inorganic iodine activity available for leakage. Concurrent with the Containment Spray Subsystems, the Recirculation Spray Subsystems pump the heated water in the containment sumps through the recirculation spray heat exchangers to remove heat from the containment, and then through the recirculation spray headers to remove additional heat and steam from the containment atmosphere. The containment is returned to subatmospheric pressure in less than 30 min, thus terminating all outleakage from the containment. The Recirculation Spray Subsystems provide long-term cooling capability to hold the containment depressurized.

During the early phases of a loss-of-coolant accident, flow reversal, or flow stoppage, and changes in core heat transfer mechanisms occur in the reactor core.

Delayed fissions, fission product decay, and fissioning during power reduction result in heatup of the core. Forces are generated across Reactor Coolant System components due to high flows and pressure waves resulting from the sudden reduction in system pressure during the accident.

Section 14.5.4 presents the evaluation of the effects of a loss-of-coolant accident on the reactor internals and the reactor vessel. The results show that no changes in core geometry occur, nor does the accident cause subsequent rupture of the reactor vessel.

Sections 14.5.2 and 14.5.5 present the methods of evaluation and the effects of a DBA on the containment structure.

Supports for the reactor vessel, steam generators, pressurizer, and reactor coolant pumps are designed to prevent rupture or failure of any safety injection line which is not connected to the pipe assumed to have ruptured. All piping connected to the Reactor Coolant System and penetrating the containment is provided with sufficient anchorage to prevent breaching of the containment at the containment penetration as a result of a reactor coolant pipe break or a seismic disturbance, or both. The design of the steam generator is such that the maximum differential pressure of 1,100 psi acting on the tubes and tube sheet from the secondary side after reactor coolant blowdown will not cause failure of these coolant boundaries.

The steam generators are supported in a manner that prevents rupture of the secondary side of a steam generator and the main steam and feedwater piping as a result of the forces created by the rupture of a reactor coolant pipe or by a seismic disturbance, or both. Section 15.6.1 discusses the Reactor Coolant System supports.

Whenever possible, piping and valves, except root valves and their connections to the coolant piping, have been run in the annular space between the crane wall and the containment wall, outside of the steam generator and pressurized cubicles. Since this space is completely outside the area occupied by the Reactor Coolant System, Engineered Safeguards equipment and piping are completely protected from effects of a loss-of-coolant accident which might damage these systems, except within the cubicle in which the loss-of-coolant accident occurs.

Special precautions have been taken to protect the pressurizer relief valves from mechanical damage which could lead to a loss-of-coolant accident. These valves are required to be located above the operating floor but within a missile shielded cubicle.

Valves and valve operators necessary for operation under loss-of-coolant accident conditions have been specified and selected so that they remain operational at the temperature and pressure conditions existing at the time of a loss-of-coolant accident. Organic material used for electrical insulation and other uses has been selected so that prolonged exposure to high temperatures and radiation (consistent with a TID 14844 type release) will not cause excessive deterioration. The design criteria for, and a description of, the Safety Injection System are presented in Section 6.2.

14.5.2 DESIGN BASIS ACCIDENT

To evaluate the effects of the loss-of-coolant accident (LOCA) on the reactor core, the Reactor Coolant System, and the containment, as described in Section 14.5.1, the Design Basis Accident (DBA), which results in the highest containment pressure and severest conditions to the Reactor Coolant System, is defined as follows:

1. The reactor is assumed to be operating at 102 percent of maximum rated power at the start of the LOCA. For containment analysis, the fission product decay heat rate used during the first hour of the LOCA is equivalent to that from infinite core exposure at rated power. After about 20 hr, the fission product decay heat curve is that equivalent to 3 yr core exposure at rated power.

2. For the reactor evaluation, an instantaneous double-ended displacement rupture is assumed to occur in the cold leg since this is more severe than a break in the hot leg.

   For the containment evaluation, the instantaneous double-ended displacement rupture is assumed to occur in the hot leg since this results in the most severe conditions for the containment.

3. A total loss-of-station power in the site occurs, and one emergency diesel generator has started and is operating to supply emergency power. Another shared diesel is available as a 100 percent spare.

4. Minimum Engineered Safeguards are activated, as defined in Section 6, to limit the consequences of the accident by providing the following:

   a. Two-out-of-three gas pressurized accumulators discharge into the Reactor Coolant System; the third accumulator is assumed to spill out of the pipe break, directly into the containment.

   b. Safety Injection is by one-of-three charging pumps and one-of-two low head safety injection pumps.

   c. Containment spray is by one-of-two Containment Spray Subsystems and two-of-four Recirculation Spray Subsystems.

The emergency diesel generator, operating as described in Item 3, above, provides the necessary power to operate the pumps in 4 (b) and 4 (c); 4 (a) will not require power of any kind to operate. The containment spray pumps have dual steam-electric drives and can be driven by the steam available in the steam generator or by the emergency diesel generator. In addition, service water requirements for the Recirculation Spray Subsystems are served by means of normal gravity flow.

The following sections discuss and analyze the effects on the Reactor Coolant System and the containment resulting from the various loss-of-coolant accidents. Primary emphasis is placed upon the analysis of the effects of the DBA involving a double-ended displacement rupture of the largest size coolant piping. Section 14.5.8 summarizes the consequences of a DBA on the Reactor Coolant Systems, the containment systems, and site dose effects.

14.5.2.1 Methods Used for Loss-Of-Coolant Accident Effects on Containment

Analyses of the effects on the containment of loss-of-coolant accidents were made using the LOCTIC digital computer program described herein. This program calculates the temperature and pressure of the containment as a function of time following a loss-of-coolant accident. The loss-of-coolant accident starts with the break in the coolant line and this is used as zero time for the accident analysis. The program considers the effects of various heat sources and sinks as a function of time in a given containment configuration to calculate temperature and pressure transient history for the containment. The process is a digital integration of the changes taking place. The program assumes that no significant change takes place between each time step and that calculations during each time step are on a steady-state basis. At the end of each time step, the heat inflow and outflow are summed and new containment and Reactor Coolant System conditions established.

The program reads into the computer the input information required to detail the specific containment and Reactor Coolant System under analysis. Table 14.5.2.1-1 gives these data for the Surry Power Station. Also read in are the physical constants for the Containment and Reactor Coolant System materials as given in Table 14.5.2.1-2.

The following is a description of the major points considered by the LOCTIC computer program:

24400 gm/4

(From Table 4.1.3-4)

TABLE 14.5.2.1-1

INPUT DATA TO LOCTIC
VIRGINIA ELECTRIC AND POWER COMPANY - SURRY POWER STATION

Reactor and Coolant System

2652

(a)	(Maximum Rated Power x 1.02) + 8* = (2,546 x 1.02) + 8*	2,605 MWt
(b)	Internal Energy of Reactor Coolant Water	246.9×10^6 Btu
(c)	Sensible Heat in Core	16.35×10^6 Btu
(d)	Total Water in System	423,200 lb
(e)	Temperature (Mass Average, Excluding Pressurizer)	571.8° F
(f)	System Pressure	2,280 psia
(g)	Reactor Coolant System Volume	8,387 ft^3**
(h)	Pressurizer Volume, Total	1,336 ft^3**
	(1) Water Volume	816 ft^3
	(2) Steam Volume	520 ft^3

Heat Transfer Data

(i)	Core Heat Transfer Area	42,461 ft^2
(j)	Core Heat Transfer Coefficients	
	(1) Film	5,420 Btu/hr/sq ft/F
	(2) Boiling	300 Btu/hr/sq ft/F
(k)	Overall Heat Transfer - One Steam Generator	25×10^6 $\frac{Btu}{hr^oF}$

Emergency Core Cooling Systems

(l)	Three Gas Accumulators - Total Volume	4,350 ft^3
	(1) Water Volume	2,775 ft^3
	(2) Pressure	675 psia
	(3) Temperature	120° F

(m) Safety Injection - Charging Pumps and Low Head Safety Injection Pumps - Curves of Flow versus Reactor Pressure are INPUT for Conditions of Normal and Minimum Safeguards.

*8 MWt is the mechanical energy added by the reactor coolant pumps. 1.02 times maximum rated power provides allowance for 2.0 percent for calibration and instrument errors. See Section 14.2.9

**Cold Volume - An additional 3 percent increase in coolant volume is used for containment design.

TABLE 14.5.2.1-1 (CONT'D)

Containment Spray Subsystems

(n) Containment Spray Subsystem - Curves of Flow versus Containment Pressure are INPUT. One or Two Pumps for Minimum or Normal Safeguards, Respectively.

(o) Containment Recirculation Spray Subsystem - (two subsystems)

Minimum and Normal Safeguards

(1) Flow	7,000 gpm
(2) Total Heat Exchanger Transfer Rate to River Water, UA	7,000,000 $\frac{Btu}{hr\,^{o}F}$

Containment

(p) Free Volume	1,800,000 ft^3
(q) Initial Interior Temperature	105^{o} F
(r) Initial Pressure	10 psia
(s) Dew Point in Containment	80^{o} F
(t) Outside Atmospheric Pressure	14.7 psia

TABLE 14.5.2.1-2

PHYSICAL CONSTANTS FOR CONTAINMENT AND COOLANT SYSTEM MATERIALS
VIRGINIA ELECTRIC AND POWER COMPANY - SURRY POWER STATION

	Conductivity, Btu/hr-ft-^{o}F	Specific Heat Btu/lb-^{o}F	Density, lb/cu ft
Carbon steel	26	0.11	490
Stainless steel	10	0.11	490
Concrete (1)	0.8	0.16	145
Paint Film Conductance*	250 Btu/hr-sq ft-^{o}F		

*This paint film conductance corresponds to a 0.006 in. coating of paint on the inside of the containment liner.

Fission Product Decay Heat and Power Decay

The LOCTIC program interpolates a curve representing the decay heat generated versus time after shutdown as a fraction of full operating power. For the purposes of a conservative analysis, it is assumed that during the early stages of the LOCA (up to 1 hr), the decay heat generation rate is equivalent to that resulting from fission products in the core as a result of infinite core exposure. After approximately 20 hr, the fission product concentration is assumed to be that equivalent to 3 yr of core exposure. Fig. 14.5.2.1-1 presents the curve used by LOCTIC for the Surry Power Station.

After receipt of a reactor power reduction signal caused by the LOCA, the reactor power decreases to fission product decay levels over a finite period of time, depending upon the time it takes for control rods to drop, the rate of boron injection, the half-life of the longest-lived delayed neutron precursor, moderator and fuel temperatures, and moderator levels in the reactor vessel during blowdown. Power decay curves have been generated for the case of a 29 in. double-ended rupture and the pressurizer surge line break. Fig. 14.5.2.1-2 presents the curves used by LOCTIC for the Surry Power Station.

The heat from fission product decay, Fig. 14.5.2.1-1, and power decay, Fig. 14.5.2.1-2, during the time interval under consideration is computed and added as sensible heat to the reactor core.

Core Sensible Heat

Since the core contains considerable heat at a temperature above the average reactor coolant temperature, the program computes the transfer of heat from the core to the coolant as a function of the amount of water in the core and whether or not boiling or film heat transfer is occurring. This relationship is conservative from the standpoint of transferring more heat to the containment than would be expected to occur; however, the relationship is invalid for a study of core thermal effects, and codes discussed in Section 14.5.3 are used for this analysis. The program adds fission product decay and power decay heat and all the metal-water reaction heat to sensible heat in the core from which it is transferred to the coolant for transport to the containment.

Reactor Coolant System Hot Metal

Sensible heat is transferred from the Reactor Coolant System hot metal to the coolant. This is a transient heat transfer calculation, and the Dusinberré numerical method, discussed below, is used for solution of the problem. The Reactor Coolant System metal is divided into the following categories for analysis:

Ruptured loop thin metal
Other loops thin metal
Ruptured loop thick metal
Other loops thick metal
Ruptured loop piping
Other loops piping

Reactor vessel head

Reactor vessel shell

Reactor vessel bottom and thermal shield

Pressurizer and surge line

The metal of the valves, pumps, and steam generator heads is included in the two thick metal categories for the ruptured loop and other loops. Table 14.5.2.1-3 presents the equivalent thicknesses and amounts of metal to be treated by the Dusinberré Method[3] for Reactor Coolant System hot metal. When a metal, as listed above, is covered with water, the surface temperature is set equal to the water temperature. When the metal is uncovered, heat outflow is controlled by an input heat transfer coefficient.

Accumulators

The water transfer from the accumulators to the Reactor Coolant System is calculated on the basis of differential pressure and an input flow coefficient. A differential mass transfer is calculated and added to the reactor coolant inventory. The loss of water from an accumulator results in an increase in accumulator gas volume. The driving gas is assumed to be a perfect gas and is expanded to the new volume, with a resulting change in accumulator pressure. When the accumulators are completely discharged, the pressurizing gas exhausts to the containment, and adds to the air partial pressure in the containment.

TABLE 14.5.2.1-3

REACTOR COOLANT SYSTEM HOT METAL CASES - INPUT DATA FOR LOCTIC
VIRGINIA ELECTRIC AND POWER COMPANY - SURRY POWER STATION

Category	Thickness, in.	Weight, lb	Heat Transfer Coefficient Btu/hr/sq ft/^{o}F
Ruptured Loop Thin Metal	2.5	53,950	40.0
Other Loops Thin Metal	2.5	107,900	10.0
Ruptured Loop Thick Metal	6.0	134,645	40.0
Other Loops Thick Metal	6.0	269,290	10.0
Ruptured Loop Piping	4.0	15,000	40.0
Other Loops Piping	4.0	210,000	10.0
Reactor Vessel Head	6.5	230,000	40.0
Reactor Vessel Shell	10.5	337,000	100.0
Reactor Vessel Bottom and Thermal Shield	6.5	118,500	2,000.0
Pressurizer and Surge Line	5.63	210,000	5.0

High Head Safety Injection

Safety injection is assumed to be effective at 30 sec after the LOCA after which water is transferred to the Reactor Coolant System as a function of Reactor Coolant System pressure (See Table 14.5.2.1-1). Differential mass and heat flow are calculated and added to the coolant inventory. High head safety injection is accomplished by the charging pumps. Appropriate delay times for receipt of safeguard signals, valve operation, and pump start are inputs to the LOCTIC program.

Low Head Safety Injection

The treatment of low head safety injection is similar to high head safety injection, with a system discharge curve used to calculate differential mass flow to the Reactor Coolant System (See Table 14.5.2.1-1). This mass and associated heat are added to the coolant inventory.

Reactor Coolant Blowdown

A summation is made at each time interval of all heat and mass input to, and output from, the Reactor Coolant System, and new system conditions are established. A differential discharge of mass and heat from the Reactor Coolant System to the containment is calculated. It is assumed that this discharge reaches equilibrium with the containment steam-air atmosphere, and that any water remaining as liquid after flashing takes place falls to the containment floor where it mixes with water on the containment floor.

Condensing Film Coefficient

A film coefficient, for condensation at surfaces, is calculated for each time interval. This coefficient is a function of the mole ratio of noncondensable gas to steam. The coefficient[2] varies from approximately 2 Btu/sq ft/hr/F at the start of the accident to approximately 150 Btu/sq ft/hr/F when the containment is at maximum pressure. The film coefficient, in addition to being a function of the mole ratio of noncondensable gas to steam, is also a function of the temperature drop across the film and the degree to which turbulent convection flow has developed. For the case of heat flow into large vertical surfaces, the convection flow would be fully developed. These factors tend to increase the film coefficient. During the latter stages of blowdown, the controlling heat transfer mechanism to the static sinks is the ability of the concrete to transfer heat into its interior and not the condensing film heat transfer coefficient.

Static Heat Sinks

The static heat sinks include the containment structure, interior concrete, and miscellaneous metal in the containment. The Dusinberré method is used to calculate the transient flow of heat into these sinks. The containment and interior concrete are divided into the following categories according to thickness:

Walls inside containment (1.0 ft)

Walls inside containment (2.0 ft)

Walls inside containment (3.0 ft)

Walls inside containment (4.0 ft)

Walls inside containment (6.5 ft)

Containment wall below grade

Containment wall above grade

Containment dome

Floor above foundation mat

Foundation mat

Miscellaneous Metals

The model used considers transient heat flow to the containment structure through the composite thermal resistance made up of the paint film, the steel liner, and the concrete. The Dusinberré method allows the face temperature to lag containment temperature as would be expected under actual conditions. The concrete is assumed to be exposed on one or two sides to the containment atmosphere, as appropriate. Heat conducted through the containment cylindrical wall and dome to the outside air is considered, but is minor. Table 14.5.2.1-4 presents the information used by LOCTIC for interior concrete sinks, the containment, and miscellaneous metal.

Containment Spray Systems

The Containment Spray Systems discharge water into the containment via the containment spray and recirculation spray headers, with the system discharge rate being a function of the appropriate driving force. Fig. 14.5.2.1-3 presents the containment spray rate as a function of the pressure in the containment and the height of water in the Refueling Water Storage Tank.

TABLE 14.5.2.1-4

CONCRETE SINKS AND MISCELLANEOUS METAL - INPUT DATA FOR LOCTIC
VIRGINIA ELECTRIC AND POWER COMPANY - SURRY POWER STATION

Category	Thickness, ft	Area, ft^2	Liner Thickness, in.
Walls Inside Containment (2)	1.0	3,320	0.0
Walls Inside Containment (2)	2.0	27,600	0.0
Walls Inside Containment (2)	3.0	19,400	0.0
Walls Inside Containment (2)	4.0	5,000	0.0
Walls Inside Containment (2)	6.5	2,100	0.0
Cont. Wall Below Grade (1)	4.5	20,600	0.38
Cont. Wall Above Grade (1)	4.5	26,147	0.38
Dome (1)	2.5	25,000	0.50
Floor Above Foundation Mat	2.0	11,250	0.0
Foundation Mat	10.0	11,250	0.25

Miscellaneous Metals - 1,200,000 lb

The Recirculation Spray Subsystem rejects heat through heat exchangers to the circulating water discharge canal. This heat transfer is calculated by the standard log mean ΔT calculation method, using an input UA of about 3.5 million Btu/hr/F for each heat exchanger. Table 14.5.2.1-1 gives 7.0 million Btu/hr/F for two exchangers. An exit temperature for the recirculation spray water leaving the heat exchanger is then calculated. The containment recirculation spray headers are located approximately 47 ft above the main operating floor.

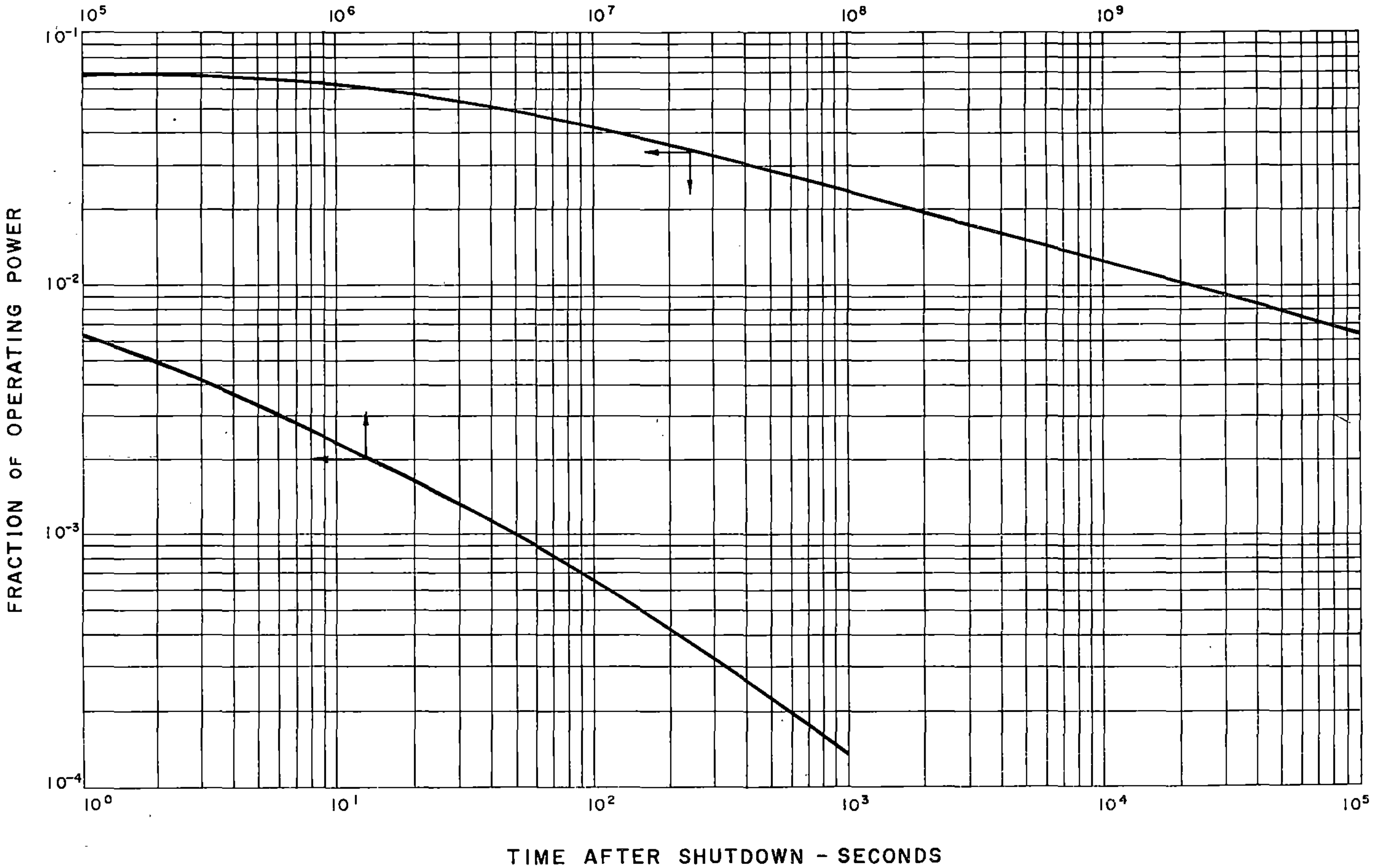
The Containment Spray Subsystems spray chilled water at 45 F from the refueling water storage tank into the containment via the containment spray headers located approximately 96 ft above the main operating floor.

The vertical fall height, on the average, considering the location of the spray headers and probable spray particle trajectories, is about 100 ft. Calculations indicate that the smaller particles approach 100 percent of thermal equilibrium with the containment atmosphere, whereas the larger particles approach 99 percent equilibrium with the containment atmosphere. The LOCTIC computer program conservatively assumes that all of the spray reaches 90 percent of thermal equilibrium with the containment atmosphere. Heat transfer between the containment and recirculation sprays and the containment atmosphere is computed in each time interval.

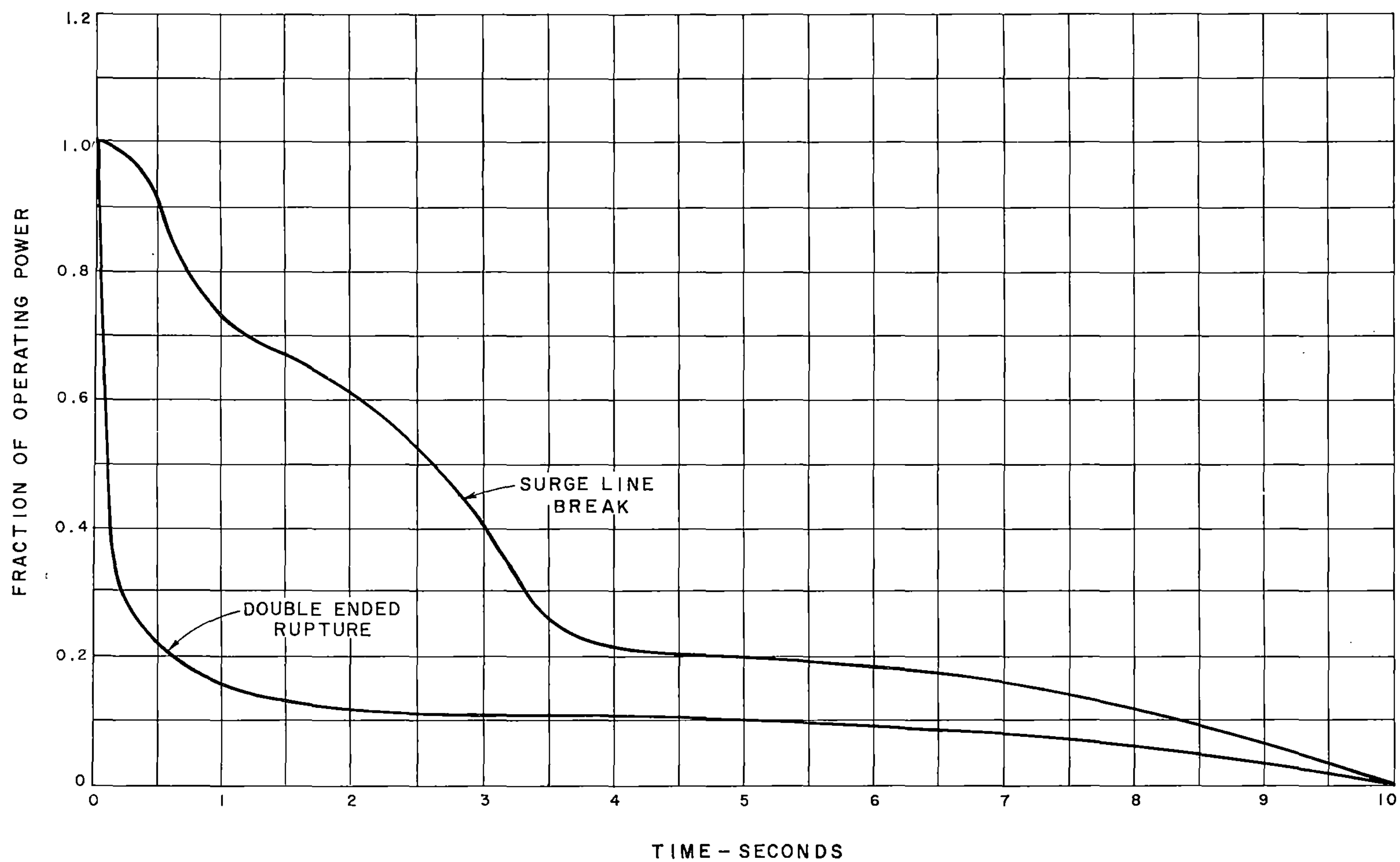
REFERENCES

1. Stone, A., Thermal Insulation of Concrete Homes, Portland Cement Association
2. McAdams, W. H., Heat Transmission, Third Edition, 1954
3. McAdams, W. H., Heat Transmission, Third Edition, 1954, Page 44
4. Perry, J. H., Chemical Engineers Handbook, Second Edition, 1941, Page 543

FIG. 14.5.2.1-1
DEC. 1, 1969

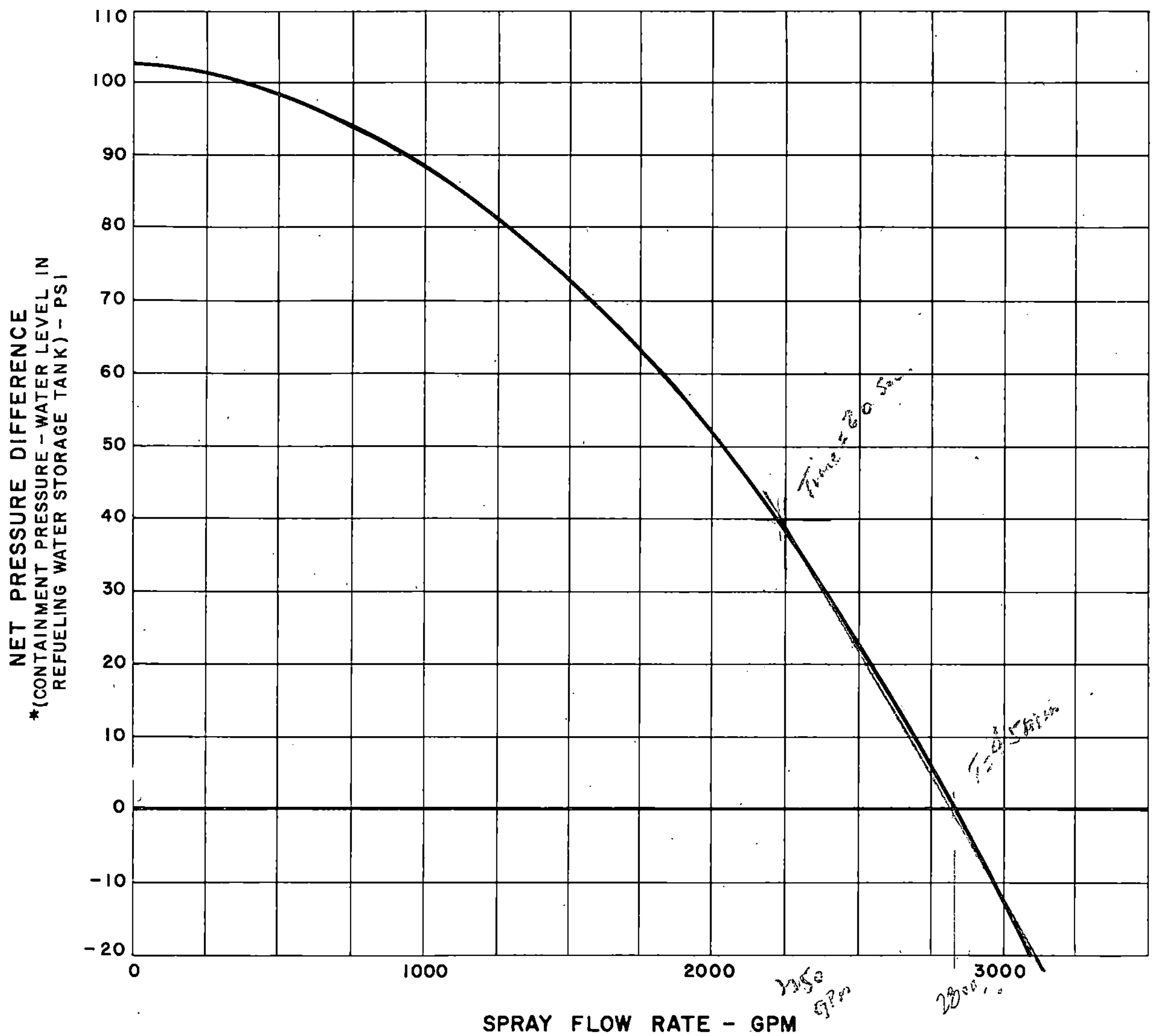


DECAY HEAT GENERATION
AFTER SHUTDOWN
SURRY POWER STATION



NUCLEAR POWER AS A FUNCTION OF TIME AFTER START OF LOSS OF COOLANT ACCIDENT

SURRY POWER STATION



NOTE:
CURVE GIVES FLOW FOR ONE CONTAINMENT SPRAY PUMP-MINIMUM SAFEGUARDS.
FLOW FOR NORMAL SAFEGUARDS IS TWICE FLOW SHOWN.

CONTAINMENT SPRAY DELIVERY RATE AS A FUNCTION OF THE NET PRESSURE DIFFERENCE*

SURRY POWER STATION

14.5.3 Core Thermal Transient

14.5.3.1 Method of Analysis

The discussion which follows has been divided into two major sections: large and intermediate breaks, and small breaks.

The analysis of large and intermediate breaks is performed by considering three separate, though interrelated, aspects of the accident: blowdown hydraulics, reactor kinetics, and core cooling.

1. Blowdown Hydraulics. This calculation provides a description of the thermal and hydraulic response to a rupture of the Reactor Coolant System, through completion of depressurization and the operation of the emergency core 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 kinetic 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.

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.

For small breaks a single code is used to describe both the blowdown hydraulics and core power transient. In these analyses, it is shown that the core does not become completely uncovered during the course of the accident. The core thermal analysis indicates that DNB does not occur in the covered portion of the core.

The following section includes a description of the applicable computer codes used for each aspect of the accident.

Large and Intermediate Break Analysis

Blowdown Analysis

The blowdown analysis is performed using the FLASH-R digital computer code. FLASH-R is based upon a code developed at Bettis called FLASH[1], but it includes modifications that more accurately describe certain features of the commercial Westinghouse PWR systems, including the single pass, rod fuel type core, the location of the reactor coolant pumps, the characteristics of the accumulators charging and low head safety injection pumps.

FLASH-R requires as input a description of the Reactor Coolant System including initial conditions, break size and location, core energy generation rate, and Emergency Core Cooling System characteristics. The code outputs the primary system pressure, coolant enthalpy, core flow, core pressure drop, mass flow through the break, core uncovering and recovery times, and the conditions required to determine the time at which reactor trip occurs.

FLASH-R simulates the Reactor Coolant System by lumping regions of relatively uniform temperature and pressure into three principle regions. Two of these regions are the upper and lower volumes, and correspond to the hot and cold volumes, respectively, in the coolant loop. The upper and lower volumes are connected by an "internal" flow path through the reactor core, and an "external" flow path through the intact loop piping and steam generators. The third region is the pressurizer, which is connected to the upper volume by the surge line. In general, the three regions are at different pressures and temperatures.

In treating the flow between the upper and lower volumes, the entire pressure drop through the "internal" flow path is assigned to the core. In the "external" connecting loop, pressure drops across the coolant pumps, steam generator and an effective length of connecting line are considered. Temperature changes between the upper and lower volumes are assumed to occur only at the core and steam generators. In the determination of flow between regions, inertial effects are considered, as well as the effects of coastdown and cavitation of the coolant pumps.

The code accounts for head addition to the primary coolant which flows through the core. Heat transfer in the core is based on the reactor kinetics studies discussed below, and is input in the form of heat flux as a function of time. Removal of heat from the primary coolant in the steam generators is also accounted for when the primary coolant temperature is greater than the temperature of the secondary fluid.

The flow rate of subcooled water through a break is calculated using Fauske's[2] model for metastable flow in the case of "short" pipes, or Moody's[3] model for equilibrium, homogeneous flow in the case of "long" pipes. Once the leaking region reaches saturation, Moody's correlation is used for both cases. For a double-ended break of a main coolant pipe, two flow paths to the break are considered. Flow from the nearer region is treated as a "short" pipe leak, while flow from the farther region is considered as a "long" pipe leak. For smaller breaks, only one flow path is considered, and it is treated as a "short" pipe leak from the nearer region. These correlations have been shown to be conservative in that they overpredict the mass flow rate through the break.

A conservative bubble separation model, which assumes an upward bubble velocity of two feet per second, is incorporated into the FLASH-R code for use in the determination of froth level in the vessel downcomer annulus, and to account for the effects of vapor entrainment upon flow losses in the core, piping, and downcomer during blowdown.

The conservatism in this bubble correlation results in a higher froth level in the downcomer annulus. This results in an extension of the period of time in which the vessel nozzles are filled with a two phase mixture. Therefore, the onset of dry saturated steam blowdown is delayed and the two phase blowdown with its significantly higher mass flow rate is extended. Thus, FLASH-R overpredicts the mass loss through the break. It should be noted that although vapor entrainment and frothing are considered in the evaluation of mass flow through a break, they are not considered in the determination of the time required to uncover and recover the core. In the determination of core water level for this phase of the analysis, complete separation of steam and water is assumed.

One of the modifications of the original FLASH code was the incorporation into FLASH-R of a calculation which determines the flow rate from the accumulators into the Reactor Coolant System. This calculation is based upon the pressure differential between the accumulators and the Reactor Coolant System, and the flow resistance of the accumulator lines. The accumulator tank nitrogen is conservatively assumed to expand isentropically to replace the injected accumulator water. The accumulator pressure, and the liquid and gas inventories are recalculated at each time step in the program until the tanks are emptied.

Finally, the original FLASH code was extended by the incorporation of a detailed core flooding calculation which considers the effect of a steam bubble in the core formed by steam generation as the core is reflooded.

Core Power Transient - CHIC-KIN Code

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 is divided axially into five sections, with the density in each section as a function of pressure and enthalpy. Subcooled surface boiling is conservatively neglected. After the coolant reaches saturation all of the heat flux is available for the production of voids. Since hot channels of the core have greater than average void fraction, use of an average channel model and neglecting hot channel effects reduces the apparent void, yielding a conservatively high energy input. In addition, coolant bypass around the core is neglected, reducing the calculated void.

Each axial fuel rod section is divided into nine radial regions for the heat transfer calculation. A thermal conductivity of 2.65 Btu/hr-ft-°F for the UO_2, and a fuel-to-gap heat transfer coefficient of 2,000 Btu/hr-ft^2-°F, are chosen for the kinetics calculation. These numbers give a minimum reasonable initial average fuel temperature. This minimizes void formation and decreases the rate of power decay.

For moderator density reactivity feedback, the calculated density coefficient as a function of density for beginning of life is conservatively assumed to be zero at the initial average density. The curve yields a 1% negative reactivity with a density reduction of 20%; whereas in practice there would be much more feedback for the same density change.

Doppler reactivity feedback is simulated as a function of the average fuel temperature during the transient. A conservative weighting factor of 1.6 accounting for spatial dependence is used for the intially unrodded core to reduce the rate of power decrease during shutdown.

Six groups of delayed neutrons are used. For the total effective fraction, a conservative maximum of 0.0072 is used to slow down power decay. Average core pressure is input as a function of time from the FLASH-R output. For the 0.5 ft^2 break, the core inlet flow as shown by the FLASH-R code calculations is used as input to CHIC-KIN. For large cold leg breaks, with violent flow reversal and then near-stagnation, the core pressure drop indicated by FLASH-R is assumed to be a reasonable representation of the forcing action between the two large liquid regions of the system. This pressure drop is used as input to CHIC-KIN, which calculates 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 are very close to those obtained by FLASH-R.

Trip would be activated in all cases by low pressurizer pressure. For the $0.5\ ft^2$ break case, this would occur in approximately 1.25 seconds. Significant rod reactivity insertion is assumed to start in 2.0 seconds after the break, with $-0.025\ \Delta k$ being inserted over the following 2.3 seconds.

For the larger breaks, trip would be similarly actuated, but because void formation is adequate for shutdown, trip was not simulated in these studies.

Core Cooling Analysis - LOCTA Code

The LOCTA-R2 transient digital computer program was developed for evaluating fuel pellet and cladding temperatures during a loss-of-coolant accident. It also 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 mode is used; with three radial nodes in the fuel and one radial node in the cladding.

Internal heat generation can be specified as a function of time. The decay heat is based on the heat generated from:

1. fission products,
2. capture products, and
3. 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 exothermic 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 zirconium oxide film is calculated as a function of time, and its effect on heat transfer is considered. Based on the heat of fusion of Zircaloy, an isothermal clad melt is considered. Once the Zircaloy metal melts, it is retained by the zirconium 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 react with additional water which is available in the vessel.

Information generated by LOCTA -R2 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 the coolant.

The code has been developed to stack axial sections and therefore, describe the behavior of a full length region as a funciton 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, and
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 occurs. The correlation during this period is:[5]

   $$T_w - T_\ell = 1.9e^{-P/900} (q'')^{1/4}$$

2. When DNB occurs, it is assumed that the fuel rods can immediately develop a condition of stable film boiling. No credit is taken for

higher transition boiling coefficients that exist prior to establishing a stable film on the fuel rods. The correlation used during this period is:[6]

$$h = 0.023 \left(\frac{k_v}{D_e}\right)\left[\frac{\rho_v D_e}{\mu_v} \cdot \left(\frac{Q_1+Q_v}{A_c}\right)\right]^{0.8} \cdot \left[\left(\frac{C_p\mu}{k}\right)_v\right]^{0.4}$$

3. During the time the core is uncovered (period of steam flow through the core), laminar or turbulent forced convective coefficients and radiative coefficients are evaluated.

For laminar forced convection to steam:[7,8]

$$\left(\frac{hDe}{k}\right)_{iso} \doteq 3.66$$

$$h/h_{iso} = \left(\frac{T_b}{T_w}\right)^{0.25}$$

For turbulent forced convection to steam:[9,10]

$$\frac{hDe}{k} = 0.020 \left(Re_b\right)^{0.8} \left(Pr_b\right)^{0.4} \left(\frac{T_w}{T_b}\right)^{-0.5}$$

4. The rising clad temperature transient is turned around after the lower portion of the core has been reflooded. For the initial reflooding period, when two phase flow is present due to entrainment, heat transfer coefficients

of approximately 25 Btu/hr-ft^2 are calculated based upon the dispersed flow theory heat transfer correlation[6]. The entrainment process is initiated at a steam velocity of 7 ft/sec leaving the flooded region of the core based upon the work of R. F. Davis[11].

The analytical model used during the core reflooding phase of the accident has been compared to experimental data obtained from the FHUST[12] experimental tests in conjunction with the LOFT program. With the same geometric configuration, flow conditions, etc. as used in the experimental studies, the Westinghouse design model predicts clad temperature turn-around times to be greater than those obtained from the FHUST data. This indicates that the Westinghouse model is conservative.

The nomenclature used in above equations is as follows:

h - Heat transfer coefficient on outer surface of fuel, Btu/hr-ft^2-°F

De - 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_ρ - Specific heat, Btu/lb-°F

k - Thermal conductivity, Btu/hr-ft-°F

T - Temperature (°F)

P - System pressure (psia)

q" - Heat flux (Btu/hr-ft^2)

Subscripts:

v - Evaluation of the property at the saturated vapor condition

l - Evaluation of the property at the saturated liquid condition

b - Evaluation of the property at the bulk fluid condition

w - Evaluation of the property at the clad surfact temperature

iso - Evaluation of the parameter when the temperature differences (T_w - T_b) are small

Small Break Analysis - SLAP Code

Blowdown Analysis

For small breaks up to about 6 in. equivalent diameter (∿ 0.2 ft^2), 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 of the flow through the break. The code considers three regions:

1. The Reactor Coolant System.

2. The pressurizer.

3. The shell side of the steam generators.

In the SLAP Code, fluid can flow between the pressurizer and the Reactor Coolant System, while heat can be transferred between the Reactor Coolant System and the secondary system. The code uses the equations of state, continuity and energy conservation to define the condition in each volume as a funcion of time. Fluid flow between the pressurizer and the Reactor Coolant System is defined by the momentum equation. Heat is transferred between the Reactor Coolant System and the secondary system only in the tube region below the shell side water level. The code accounts for heat transfer in either direction across the tubes, depending upon which fluid is at the higher temperature. The heat transfer rate is assumed to be zero for any portion of the tubes not covered by shell side water.

The initial thermodynamic conditions are designated by:

1. Volume of each region.

2. Reactor Coolant System pressure and temperature.

3. Pressurizer level.

4. Secondary pressure and steam generator shell level.

Under the initial steady operating conditions, fluid flow between the pressurizer and the Reactor Coolant System is zero. The initial heat transfer rate in the steam generators is equal to the reactor operating power.

The initial subcooled flow through the break is treated by the correlation of Fauske[2]. His work concluded that for sharp-edged orifices, test data can be accurately correlated using the incompressible flow equations for a nozzle. Subsequent saturated water flow through the break is treated by the correlation of Moody.[3]

Safety injection water flow into the Reactor Coolant System is described by an input table of injection flow rate as a function of system pressure. A safety injection initiation and start-up delay time of 25 seconds is included in the model. The accumulators automatically discharge their fluid when the Reactor Coolant System pressure drops below the accumulator pressure.

Finally, feedwater flow as a function of time after the break is specified by input tables. Steam flow as a function of safety valve flow area is specified to simulate the operation of the safety valves.

14.5.3.2 Results

The capability of the Emergency Core Cooling System (ECCS) to meet the design criterion was analyzed for the following range of break sizes:

1. Large - Areas of: 9.2 ft^2 (double-ended severance), 6 ft^2 and 3 ft^2
2. Intermediate - Area of: 0.5 ft^2
3. Small - Areas corresponding to pipe diameters of 6 in ($\sim$.2 ft^2), 4 in, 3 in, 2 in, and 1 in

The power level used in the loss-of-coolant evaluations performed for this reactor includes two percent increase above the maximum calculated core thermal rating of 2546 MWt, to account for errors in the steam cycle calorimetric measurements.

Results-Large and Intermediate Area Ruptures

Blowdown and Refill

The large and intermediate area rupture cases which were analyzed are all cold leg breaks since they are more severe than hot leg breaks of comparable area because:

1. Refill of the vessel is retarded with a cold leg break due to spillage of accumulator water through the broken line. In the accident analyses which were performed, the entire contents of one accumulator were assumed to be lost through the break.

2. Generation of steam in the core during the reflooding period tends to retard the reflooding with a cold leg break. This aspect of the accident is discussed below in greater detail.

Figure 14.5.3.2-1 through 14.5.3.2-3 are plots of the water volume in the reactor vessel for the large area ruptures. Figure 14.5.3.2-4 presents the plot of water volume in the reactor vessel for the 0.5 ft^2 break which is representative of the intermediate break ruptures. Figures 14.5.3.2-12 through 14.5.3.2-15 present the pressure and core flow for the double ended, 6 ft^2, 3 ft^2, and .5 ft^2 breaks respectively. Inspection of these figures indicates the sequence of major events during the initial stages of the accident.

1. There is an initial rapid depressurization of the reactor due to the subcooled blowdown, which causes the pressurizer to empty and the Reactor Coolant System to reach saturation conditions. Safety injection actuation would occur during this period due to low pressurizer level and pressure or high containment pressure. Following subcooled blowdown, a period of two-phase blowdown, which is characterized by a slower rate of depressurization, exists.

2. The relatively slow depressurization during the two phase blowdown period continues until the primary system is nearly empty. Actually, there is even a slight increase in pressure during the early stages of two phase blowdown for the intermediate break, due to the delay in

reactor shutdown. As the coolant system begins to run out of water, the core (which is essentially at the low point of the primary system) becomes uncovered. However, the pressure also falls more rapidly near the end of blowdown, and the accumulator setpoint of 675 psia is finally reached. In the analysis, it was conservatively assumed that the accumulators did not begin to inject until the pressure fell below 600 psia.

3. Injection of accumulator water quickly completes the reduction of coolant system pressure to an equilibrium value with the containment. The continued expansion of the nitrogen gas empties the accumulators and causes the core to be reflooded.

4. During blowdown, the volumes plotted represent an equivalent liquid volume which would occur if the liquid and gas phases were completely separated. No credit is taken for an increased froth height due to voids created by boiling in the core. The volume of liquid remaining in the vessel after blowdown is used as a starting point to predict the liquid level during the refilling phase. It should be noted here that the FLASH-R code conservatively predicts less water remaining in the vessel at the end of blowdown, when compared to experimental data (LOFT semiscale tests, etc.), therefore this conservatism is carried throughout the refill phase of the unpredicted water levels. The reason for these conservatisms were discussed previously in the description of the FLASH Code.

These results serve as the principal input for the core kinetic and core thermal calculations.

Several factors have been considered in the analysis that could adversely affect the flow of emergency cooling water to the core. These are as follows:

1. Loss of accumulator water out of the break, or diversion to other parts of the system during blowdown.

2. Steam bubble formation when accumulator water refloods the core.

3. The affect of the nitrogen gas entering the vessel.

Loss of Accumulator Water

All of the analyses assume the complete loss of the volume of one accumulator out of a cold leg break. The question then is to determine how much of the volume of the remaining two accumulators could be spilled. Only a portion of the accumulator injection takes place during blowdown. For the double-ended break, accumulator injection begins at 6.5 seconds and blowdown is completed at 11 seconds. The entire contents of the accumulators are not injected until 29 seconds after the break. During blowdown FLASH-R calculations indicate that only 25,000 lbs.

of the total accumulator mass of 111,000 lbs. is injected (2 of 3 accumulators considered). Therefore, only about 19 percent of the accumulator injection is subject to loss through the break during blowdown. It should be noted that FLASH-R overpredicts the total mass loss from the system; therefore, the potential mass loss of accumulator water reflects this conservatism.

The flow from each accumulator enters the cold leg pipe between the outlet of the reactor coolant pumps and the cold leg nozzles. The maximum accumulator flow rate is 6800 lbs/sec, and occurs for the double-ended break shortly after the beginning of injection. This flow rate is only approximately 24.3 percent of the steady flow rate 27,972 lbs/sec for normal plant operation, and therefore, there is no possibility of an excessive downcomer pressure drop which could back the flow to other parts of the system. Flow into the inlet of the vessel is also enhanced by the reactor coolant pumps which would coast down during the transient and tend to force coolant in the direction of the reactor.

Steam Bubble

When the core is reflooded by the accumulators, special consideration is given to the possibility of steam generation around the hot fuel rods, resulting in a pressure buildup which could retard the reflooding process. The steam which is generated must be vented from the system through the break. Flow paths for the various potential break locations are illustrated in Figures 14.5.3.2-5 through 14.5.3.2-7

The worst break location is a cold leg break, where the steam must flow through the reactor coolant pipes, steam generator, and reactor coolant pump to escape. There are two paths available for the steam to flow to the break. The first path is directly to the break through the broken loop. The other path is through the two intact loops, back into the inlet annulus, and finally to the break through the inlet nozzle in the broken leg. Because of the back pressure which results from the resistance of the core and coolant loops to steam flow, the liquid level in the downcomer annulus rises at a faster rate than the core level during the initial stage of water injection. The level in the annulus rises until a head is developed between the annulus and the core, equivalent to the pressure drop of the steam flowing through the loops to the break.

The relationship between the downcomer head and corresponding steam flow rate through the loops is shown in Figure 14.5.3.2-8. 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 as discussed below) for saturated steam at a containment pressure of 45 psig during blowdown. It is seen that a head of 1.6 feet produces sufficient steam flow to cool the core, assuming empty loop seals in the coolant piping.

The FLASH-R refill model accounts for the generation of steam in the core. Figure 14.5.3.2-9 shows the amount by which FLASH-R calculates the downcomer level rises above the core level during the reflooding process.

When the first flooding water enters the bottom of the core, the downcomer level is the same as the core level. At this time, there is no developed head, but there is a potential available head equal to the entire downcomer height (16.4 feet). As the core level rises, the potential head is reduced by a corresponding amount. The head potentially available to drive steam through the core as a function of time is shown in Figure 14.5.3.2-9 for the core refill transient, predicted by FLASH-R following a double-ended break. Throughout the reflooding process, the available head far exceeds the 1.6 feet required for minimum steam cooling flow. This provides considerable margin for variations in the pressure drop calculation due to effects such as entrainment and a decrease in the system saturation pressure during containment cooldown. This available head would permit steam flow in excess of the calculated flow, or alternatively, would offset an additional back-pressure build-up due to water filled loop seals (as discussed below) without any loss of safety injection water.

In the unlikely event that all recirculation loop seals (pipe between steam generator and reactor coolant pumps as shown in Figure 14.5.3.2-6) 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 at the start of the reflooding process far exceeds the 9.0 feet of head needed to blow the liquid out of the loop seals. Filled loop seals would, therefore, result in the rapid filling of the downcomer until the head in the downcomer reached 9.0 feet. This would be followed by a back flow of the water from the downcomer into the core in order to equalize downcomer head and core steam pressure. 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. Also, 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 hot leg breaks where the core is reflooded by accumulator water, steam generation in the core does not retard the refill transient because the steam does not have to travel through the loops to reach the break.

Nitrogen Interference

Nitrogen from the accumulators enters the reactor coolant pipes following completion of injection of the accumulator water volume. At this time, the core has been flooded to well above the hot spot, and the clad temperatures have been greatly reduced by the emergency coolant. The peak clad temperature has been reduced to approximately 1200°F before nitrogen can enter the Reactor Coolant System.

After all the accumulator water has been injected, the gas entering the system partly vents through the break and the rest occupies the high volumes in the Reactor Coolant System. The rate of venting depends on the break size and location. The vent paths available are illustrated in Figure 14.5.3.2-10.

Before nitrogen can enter the core region, the water level in the downcomer must be depressed below the level of the core barrel. Therefore, a buildup of nitrogen pressure in the downcomer equal to the static head of water between the bottom of the core barrel and the level of nozzles can be tolerated before the entry of nitrogen into the core region must be considered. The flow balances that develop are very similar to the steam bubble calculation, but in the reverse direction.

It is important to demonstrate the ability to vent nitrogen either through the break (and thereby eliminate the nitrogen in the system) or to the core upper plenum (thereby equalizing the pressure difference between the inlet and outlet of the core, and eliminating the potential for forcing water from the core). Since the downcomer and core outlet plenum communicate through the three reactor coolant loops (flow area greater than 4.0 sq. ft. through each path), and the flow rate from the accumulator is limited by the 12-inch accumulator lines, it is evident that the capability exists for venting the nitrogen through the Reactor Coolant System in order to equalize the pressure between core inlet and outlet plenums. This capability is discussed further in the following paragraphs.

The flow rate of nitrogen into the system is discussed first. The accumulator process is assumed to be isothermal whereas in fact, the polytropic exponent would be expected to vary between 1 (isothermal) and 1.4 (adiabatic) depending on the amount of wall heating. The assumption of constant temperature provides the highest pressure at the end of accumulator water injection and therefore, the highest mass flow of nitrogen into the coolant system. Note here the inconsistent, but conservative, comparison with the water injection rate calculations, in which an isentropic expansion was assumed to minimize the stored energy driving force for injection. If the isentropic case were assumed for the gas flow study, lower gas flow rates would result. It has been calculated that the nitrogen pressure has reached 217 psia when all the water has been injected (vs. 145 psia for the isentropic case). At this pressure, the nitrogen flow into the reactor coolant cold leg pipe from each acculator is 232 lbs/sec. This is based on choked nitrogen flow with friction through the length of the accumulator line.

The venting rates are discussed next. Consider first a double-ended cold leg break (refer to Figure 14.5.3.2-10). The gas entering the system, depresses the downcomer annulus and expands to fill the entire annulus which communicates with the inlet nozzle of the broken loop. Neglecting for the moment the reverse venting flow through intact loops, it has been calculated that a nitrogen flow rate of 600 lbs/sec could be accommodated through the broken cold leg nozzle, based upon a containment back pressure of 45 psig and an annulus pressure of 45 psig plus the head of water in the core above the bottom of the core barrel (prior to the nitrogen entering the core). Forty-five psig was selected for this study because it is representative of the containment pressure at the end of accumulator injection. Therefore, since the nitrogen flow into the two intact reactor coolant loops is only two times 232 lbs/sec, the nitrogen pressure in the downcomer does not build to a level sufficient to allow nitrogen to enter the core region.

As the postulated cold leg break size is decreased, credit must be taken for the venting capability in the reverse direction to the core outlet plenum. In the limit, the break size would be reduced to zero, thereby requiring all flow to be vented to the outlet plenum in order to maintain an equilibrium between core inlet and outlet pressure. However, since accumulators inject for only relatively large breaks, some venting through the cold leg break is always available. Therefore, the limiting case for reverse venting is the hot leg break where all flow from three accumulators must be vented to the core outlet plenum.

For the hot leg break and based on the conservative isothermal case, the nitrogen pressure is 217 psia when all the accumulator water has been injected. Now, using the flow resistance through the loop (pipes, pump, steam generator) and a driving head equivalent to the head of water above the bottom of the downcomer, approximately 145 lbs/sec/loop can be vented before any N_2 would get around the downcomer to enter the core. The injection flow of nitrogen at this time is 190 lbs/sec/loop. Therefore, a brief imbalance in pressure could result, and a small amount of nitrogen could bubble around the downcomer into the core region. This imbalance exists only briefly until the downcomer, loops, steam generator, and outlet plenum fill with nitrogen and imposes an equal pressure on both the inlet and outlet of the core, thereby equalizing the water level in the core and downcomer. This terminates the flow of nitrogen into the core region. It is therefore impossible to impose, even for a short time, nitrogen pressure gradients much larger than 10 psi across the core.

Note that the small amount of nitrogen entering the core region during the pressure equalization transient mixes with the core water inventory. The lower core support casting is made with many small holes, thereby separating the nitrogen into many flow paths. During this time the core is cooled by a bubbly mixture of nitrogen and water, and the clad temperature continues to decrease.

Therefore, even if the conservative isothermal case is considered, it is concluded that the introduction of nitrogen does not interfere with the core cooling process.

As stated above, the accumulator nitrogen expansion process would be expected to vary between isothermal and isentropic. Based on the more realistic isentropic case, the nitrogen pressure is 145 psia when all the accumulator water has been injected. The peak injection flow is 135 lbs/sec/loop. Since 145 lbs/sec/loop can be vented, no nitrogen enters the core region. Therefore, this case reaffirms the conclusion that the introduction of nitrogen does not interfere with the core cooling process.

Core Power Transient During Blowdown

For large breaks, the high rate of subcooled blowdown, and resulting rapid depressurization quickly produce extensive voids in the core. Void formation is also aided by backflow, which forces a saturated steam-water mixture from the reactor outlet plenum down into the core. The result is, that for the large breaks which are studied, the reactor shuts down almost immediately without benefit from control rod insertion. No rod insertion is considered in the CHIC-KIN analyses of the large break transients. The power calculated by CHIC-KIN for these cases is shown in Figure. 14.5.3.2-11.

A "standardized" decay heat plus delayed neutron curve is used as a minimum power level in the thermal analysis, until the bottom of the core becomes uncovered. The above procedure results in the generation of more power than that which is actually calculated using the conservative assumptions mentioned above. After the bottom of the core becomes uncovered, the delayed neutron effect is no longer considered, since the direct fission contribution is negligible after this time.

For the intermediate break, the initial depressurization does not form enough voids to shut down the core. However, FLASH-R calculations indicate a trip due to low pressurizer pressure would occur in approximately 1.25 seconds. At this point, the depressurization has reduced reactor power to approximately 50% of full power. In the CHIC-KIN analyses, significant reactivity insertion was assumed to start at 2.0 seconds, with $-0.025\ \Delta k$ being inserted over the following 2.3 seconds. This causes reactor power to decrease rapidly.

Core Thermal Analysis Results

The core thermal analysis was performed using the blowdown and recovery data and the core power transients which were described in the previous sections.

Figures 14.5.3.2-12 through 14.5.3.2-15 present a plot of core pressure and core flow and the calculated heat transfer coefficient used for all breaks. Figures 14.5.3.2-16 through 14.5.3.2-18 present the maximum clad temperature transient, both for the design case and for the case where adiabatic conditions exist after blowdown for the double-end, six ft^2 and three ft^2 cold leg break sizes.

For the 0.5 ft^2 break, blowdown is not complete until after reflooding has resulted in effective cooling of the core and a continuously decreasing cladding temperature. This occurs at approximately 102 seconds. The case with adiabatic conditions after this time along with the design transient, is presented in Figure 14.5.3.2-19. The zirconium-water reaction was computed to be less than one percent in all cases and is an insignificant factor in the containment pressure transient. Table 14.5.3.2-1 summarizes the important results of the transients.

Results - Small Breaks

The analyses discussed in the previous section demonstrated the adequacy of the accumulators to reflood the core and limit the temperature rise of the core for large and intermediate size breaks. In this section, the discussion deals with breaks of 6 inch equivalent diameter or less, where the charging pumps must play an increased role in the initial reflooding because of the slower depressurization of the Reactor Coolant System. As indicated previously, this analysis was performed for a core thermal rating of 102% of the maximum calculated power 2546 MWt.

Ruptures of a very small cross section (up to approximately the equivalent of a 0.75 inch ID connecting pipe) cause expulsion of coolant at a rate which can be accomodated by the charging pumps, thus enabling the operational level in the pressurizer to be maintained and permitting the operator to execute an orderly shutdown. The top of the core remains covered throughout the accident, and no clad damage is expected. Since instrument taps and sample connections are less than 0.75 inch diameter, protection from rupture of these lines is afforded by the charging pumps.

Should a larger break occur, the pressure decrease in the Reactor Coolant System causes fluid to flow to it 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 points are reached. Before the reactor trip signal occurs, it is assumed that the plant is in an equilibrium condition, with the heat generated in the core being removed by the secondary system. Following the trip, heat from decay, hot internals, and the vessel is transferred to the Reactor Coolant System fluid, and then to the secondary system. The secondary pressure increases until the safety valves open to relieve steam. Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The secondary flow aids in the reduction of primary system pressure. When the primary system pressure reaches 600 psig, the accumulators begin to inject.

When the core becomes uncovered, the stored energy and decay heat in the core that is transferred to the coolant becomes a function of water level in the core (assumed to be directly proportional, plus 10% to account for froth).

To analyze the effects of a small primary system break, three cases of safety injection capability were studied:

Case A: Three charging pumps delivering through three cold leg lines for five minutes, then delivering through all six lines (valves in hot leg injection lines are now open) for the remainder of the transient with no spilling. This represents the full safety injection capability.

Case B: Two charging pumps delivering through three cold leg lines for five minutes, then delivering through all six lines (valves in hot leg injection lines are now open) for the remainder of the transient (full spill through the break from one line).

Case C: Single failure; one charging pump with the same delivery assumptions as Case B. This case represents the minimum engineered safeguards which are available automatically for small break protection. It should be noted that since a single diesel generator has the capability of supplying power to two charging pumps, and since switching facilities exist in the main Control Room to enable the operator to align two pumps to a single diesel, the operator would take action in relatively long time scale transients (4" or smaller diameter breaks) to carry out this action. Thus in most cases operator action would convert Case C to Case B.

It is of interest, however, to examine Case C to determine the severity of the small break loss-of-coolant accident in the event of mal-operator during maintenance or failure of the operator action to supply two pumps from a single running diesel generator.

The spilling assumption for Cases B and C corresponds to an accident initiating break located downstream of the check valve in the 6 in. stub which enters the reactor coolant pipe. Calculations indicate that for the smaller breaks, the reactor pressure at the location of the break maintains a relatively uniform back pressure in all injection lines so that no significant flow imbalance occurs. However, the full flow delivered through the safety injection line connected to the ruptured reactor coolant pipe was assumed to spill through the break in addition to the break flow.

Complete severance of the 6 in. stub would be a large enough break to cause reactor coolant depressurization to the 600 psig accumulator set point in time for the accumulators to be effective in limiting the maximum cladding temperature to a value consistent with the design criteria. A break in a safety injection line upstream of the check valve would not constitute a loss-of-coolant accident since the check valve would remain closed due to reactor coolant pressure.

The delivery curves for Cases A, B, and C are shown in Figures 14.5.3.2-20 and 14.5.3.2-21. The Reactor Coolant System pressures and volumes for each of the cases analyzed are presented in Figures 14.5.3.2-22 through 14.5.3.2-25. The 1 inch transients are less severe than the 2 inch transients. Therefore, the 1 inch transients are not presented.

It should be noted that the volume plots are quiet water level plots, assuming complete separation of steam and water in the reactor vessel. The existence of a water filled loop seal was considered in all cases. The plots of quiet water level in the core take into account the depression of the core water level necessary to maintain the full downcomer required to clear the loop seal and subsequently allow steam blowdown. This depicts a break for the worst break location, i.e., a cold leg break between the pump outlet and the reactor vessel inlet. Saturated water blowdown is allowed to continue until the loop seal is uncovered.

As indicated by the curves of Figure 14.5.3.2-25, the core mid-plane remains flooded for all breaks up to and including a 6 in. diameter hole for the fully operational Safety Injection System (Case A).

Protection for a 6 in. break is afforded by the accumulators. The mid-plane of the core is uncovered for approximately the same length of time as a 0.5 ft^2

break. Since the core is uncovered at a later time after shutdown, the residual heat generation is less. Therefore, the peak clad temperature for break sizes from 4 to 6 inches is expected to be less than that for the 0.5 ft^2 break.

The curves of Figure 14.5.3.2-24 indicate that core mid-plane flooding is maintained for breaks up to and including 3 inches for the minimum safety injection case (Case C) and breaks up to and including 4 inches for the partial case (Case B). The core mid-plane is uncovered briefly for the 4 inch break for Case C.

Core Thermal Analysis - Small Breaks

The core thermal analysis for ruptures from 0.5 ft^2 to a double ended rupture was previously presented. The conservative analysis presented for these cases indicated that the core was uncovered. It was also assumed that DNB occurred at 0.5 seconds. For small breaks the clad temperature transient is influenced by two important factors.

1. The ability to keep the hot spot covered with at least a two phase (frothing) mixture of water and steam through the entire transient.

2. DNB does not occur during blowdown.

The ability to evaluate whether or not DNB occurs requires a detailed knowledge of the core conditions throughout the initial phase of the accident. For this purpose, Westinghouse has recently developed multi-dimensional design models; the SATAN-R code for the blowdown analysis and the THINC code for the core thermal transient. Core parameters obtained from SATAN-R, such as pressure, power and flow, are used as input to the THINC code. The THINC computer program is used to calculate coolant density, mass velocity, enthaply, vapor voids, and static pressure distribution along parallel flow channels.

Knowing the local conditions at any axial elevation, it then evaluates the DNBR at that core location. Once the confirmation of no DNB is determined, typical loss of coolant analyses are made using the SLAP code for predicting the blow-down characteristics and the LOCTA-R code for evaluating the core thermal analysis.

The 4 in. equivalent diameter break with minimum safety injection (Case C) was analyzed to illustrate the core thermal transient characteristics of small breaks.

Figure 14.5.3.2-26 presents the minimum DNBR versus time after rupture for the 4 in. break analysis. The minimum DNBR for this case is 1.48. No portion of any rod which does not become uncovered exhibits DNB, and thus the mechanism of heat transfer for this region is nucleate boiling.

Figure 14.5.3.2-27 presents the transient core volume for the 4 in. break. As stated previously, the core mid-plane is uncovered briefly if the quiet volume only is considered. However, these curves demonstrate the capability of the minimum safety injection to maintain a two phase mixture of steam and water well above the core mid-plane for the entire transient. The core thermal analysis presented uses the benefit of the froth level shown.

Since no DNB occurs and the hot spot of the core never becomes uncovered, the hot spot clad temperature gradually decreases from its initial steady state value of approximately 700°F. However, the upper portion of the hot rod obtains a higher maximum clad temperature of 1325°F due to this portion of rod being uncovered for approximately 150 seconds.

During this uncovered period, the upper part of the hot rod is partially cooled by the steam generated in the covered portion of the core.

Conclusions

For breaks up to and including the double-ended severance of a reactor coolant pipe, the Emergency Core Cooling System with minimum safeguards will limit the clad temperature to below the melting temperature of Zircaloy-4 and assure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved. The Emergency Core Cooling System design meets the core cooling criteria with substantial margin for all cases.

The peak clad temperature calculated was 2080°F which occurred for the double-ended break. The clad temperature is limited to 1324°F for a 4" equivalent diameter break. This protection for the small break is provided by only one of the three charging safety injection pumps.

REFERENCES

1. "FLASH; a Program for Digital Simulation of the Loss of Coolant Accident" S. G. Margolis, and J. A. Redfield, Bettis Atomic Power Laboratory, Report WAPD-TM-534.

2. "The Discharge of Saturated Water Through Tubes", by H. K. Fauske, AICHE, Reprint 30, Seventh National Heat Transfer Conference, AICHE and ASME, Cleveland, Ohio, August 9 to 12, 1964.

3. "Maximum Flow Rate of Single Component, Two-Phase Mixture" by F. H. Moody, Paper No. 64-HT-35, and ASME publication.

4. "CHIC-KIN . . . A Fortran Program of Intermediate and Fast Transients in a Water Moderated Reactor", V. A. Redfield, WAPD-TM-479, January 1, 1965.

5. W. H. Jens and P. A. Lottes "Analyses of Heat Transfer, Burnout Pressure Drop, and Density Data for High Pressure Water," USAEC Report ANL-4627. 1951.

6. R. S. Dougall, and W. M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes and Upward Flow of the Fluid at Low Qualities," MIT Report No. 9079, September, 1963.

7. H. Hausen, "Darstellung des Warmeuberganges in Rohren durch verall gemeinerte Potenzbezienhungen," VDI Zeit., No. 4, P. 91, 1943.

8. W. M. Kays, "Numerical Solutions for Laminar-Flow Heat Transfer in Circular Tubes," Trans ASME, Vol. 77, 1955, pp. 1265-2374.

9. D. M. McEligot, P. M. Magee, and G. Leppert, "Effect of Large Temperature Gradients on Convective Heat Transfer: The Downstream Region," J. of Heat Transfer, vol. 88, 1966 pp. 67-76.

10. D. M. McEligot, L. W. Ormand, and H. C. Perkins, "Internal Low Reynolds Number Turbulent and Transitional Gas Flow with Heat Transfer," J. of Heat Transfer, vol. 88, 1966, pp. 239-245.

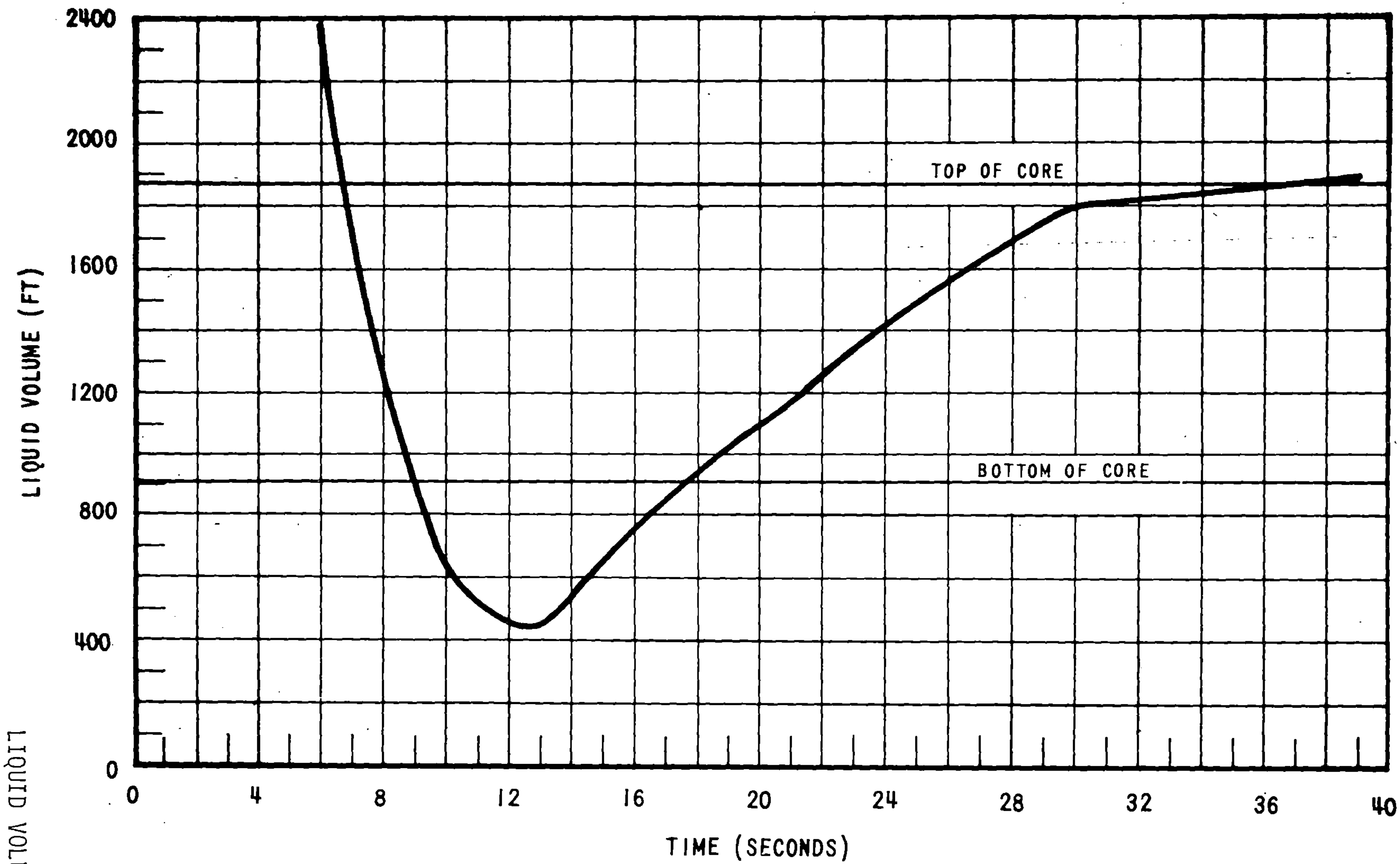
11. Davis, R. F., "The Physical Aspect of Steam Generation at High Pressure and the Problem of Steam Contamination," I. Mech E., (1940).

12. "Fuel Heatup Simulation Tests", K. A. Dietz (ed.) Quarterly Technical Report, Engineering and Test Branch, October 1967 - December 1967, IDO-17242 (May 1968).

TABLE 14.5.3.2-1

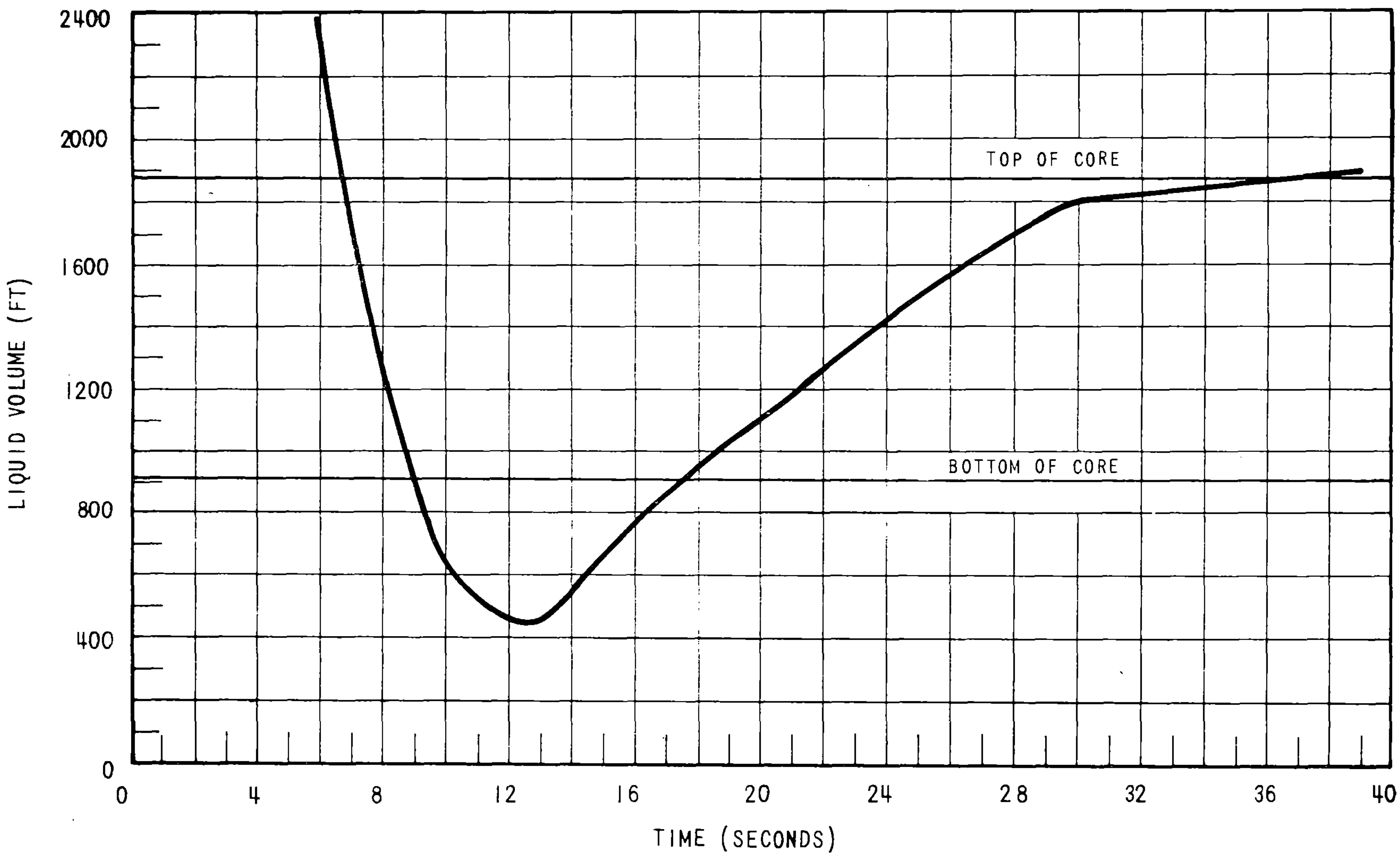
MAXIMUM CLAD TEMPERATURE AND PERCENT CLAD
BURST VERSUS BREAK SIZE FOR MINIMUM ENGINEERED SAFEGUARDS

Break Size, Cold Leg	Maximum Clad Temperature, °F	Total Percent Clad Burst
Double-ended	2080	79
6 ft^2	1910	76
3 ft^2	1480	48
0.5 ft^2	1840	78



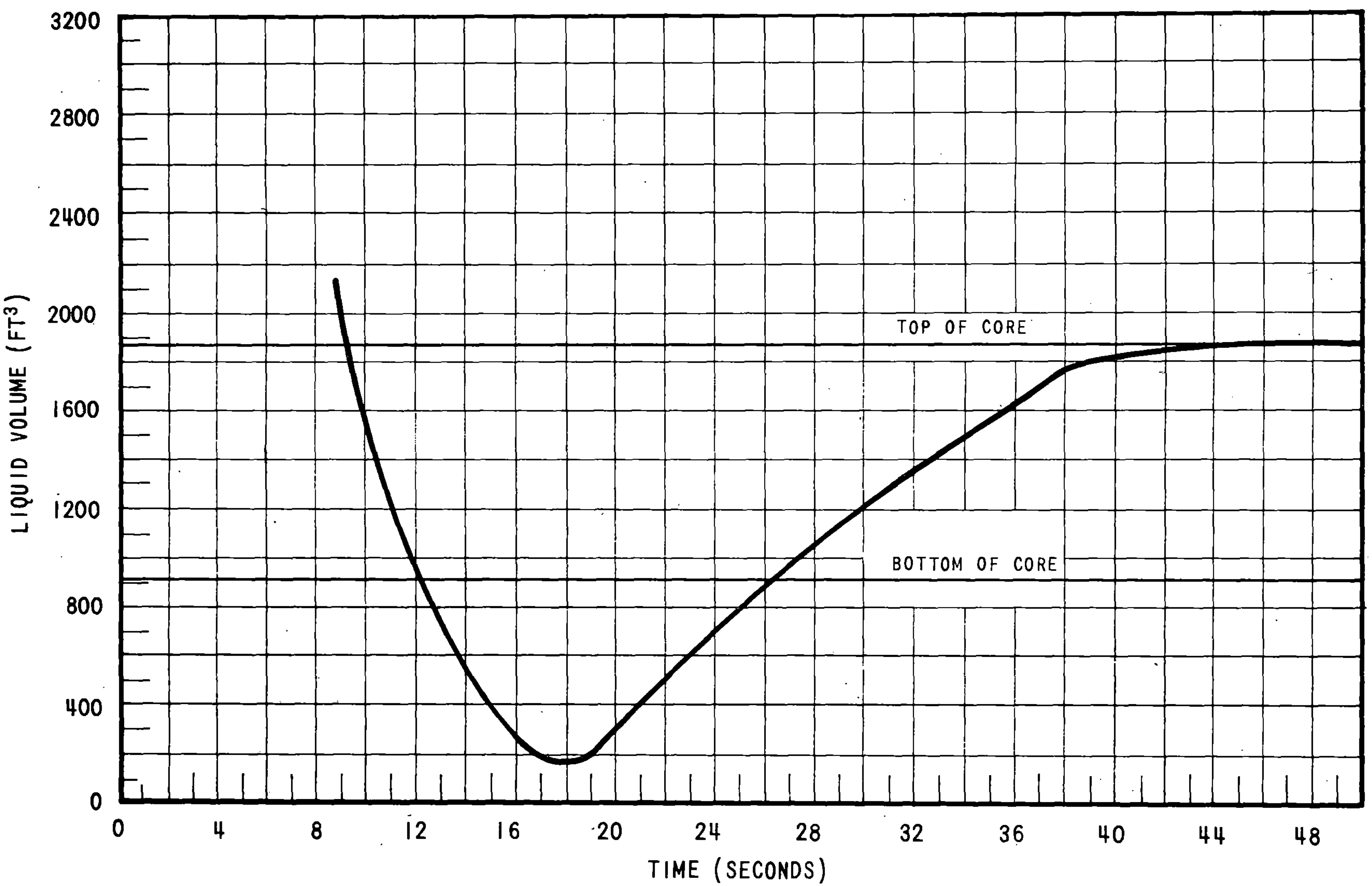
LIQUID VOLUME VERSUS
TIME DOUBLE-ENDED
COLD LEG BREAK

FIGURE 14.5.3.2-1
2-13-70



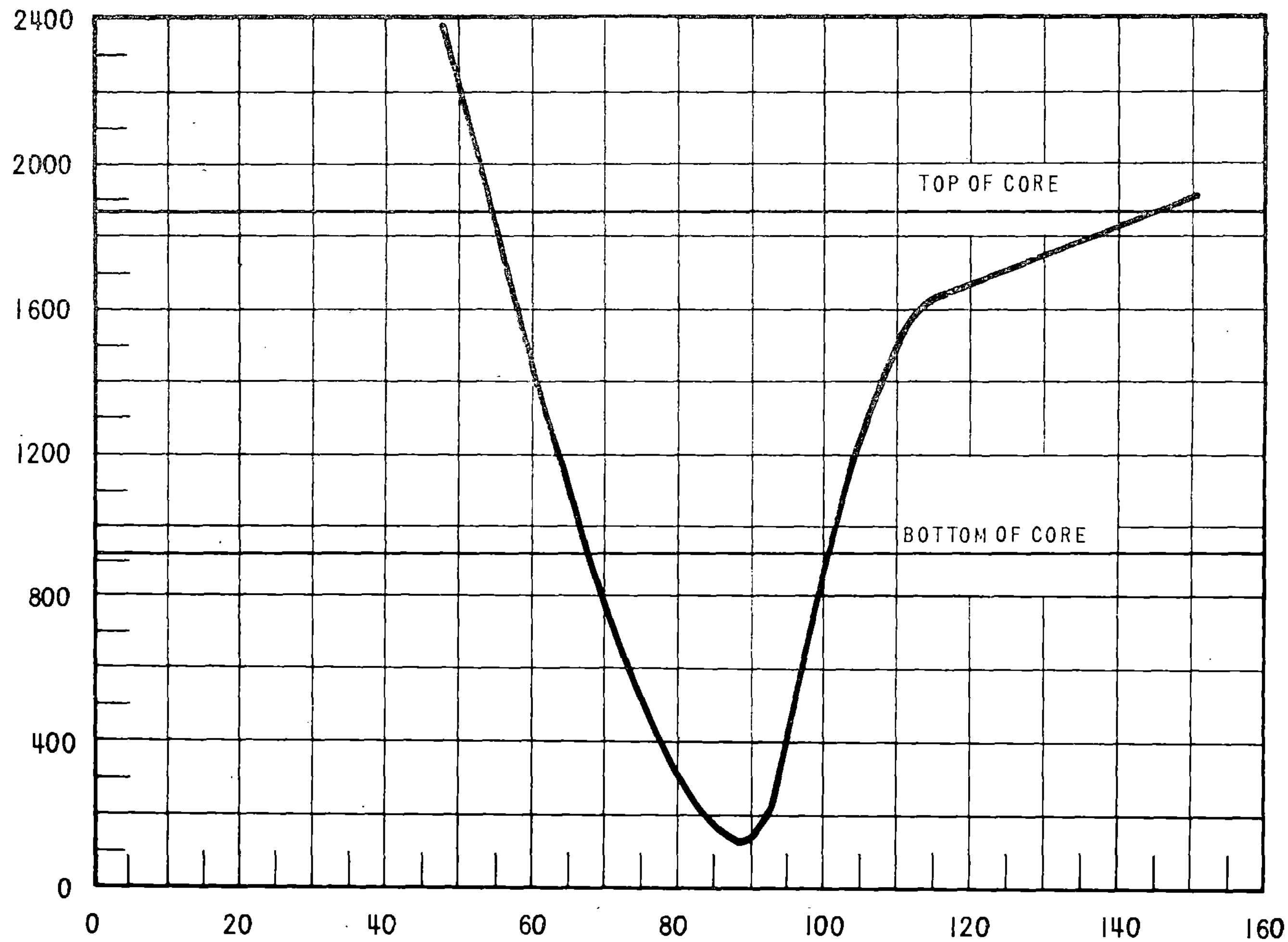
LIQUID VOLUME VERSUS
TIME 6 FT^2 - COLD LEG BREAK

FIGURE 14.5.3.2-2
2-13-70



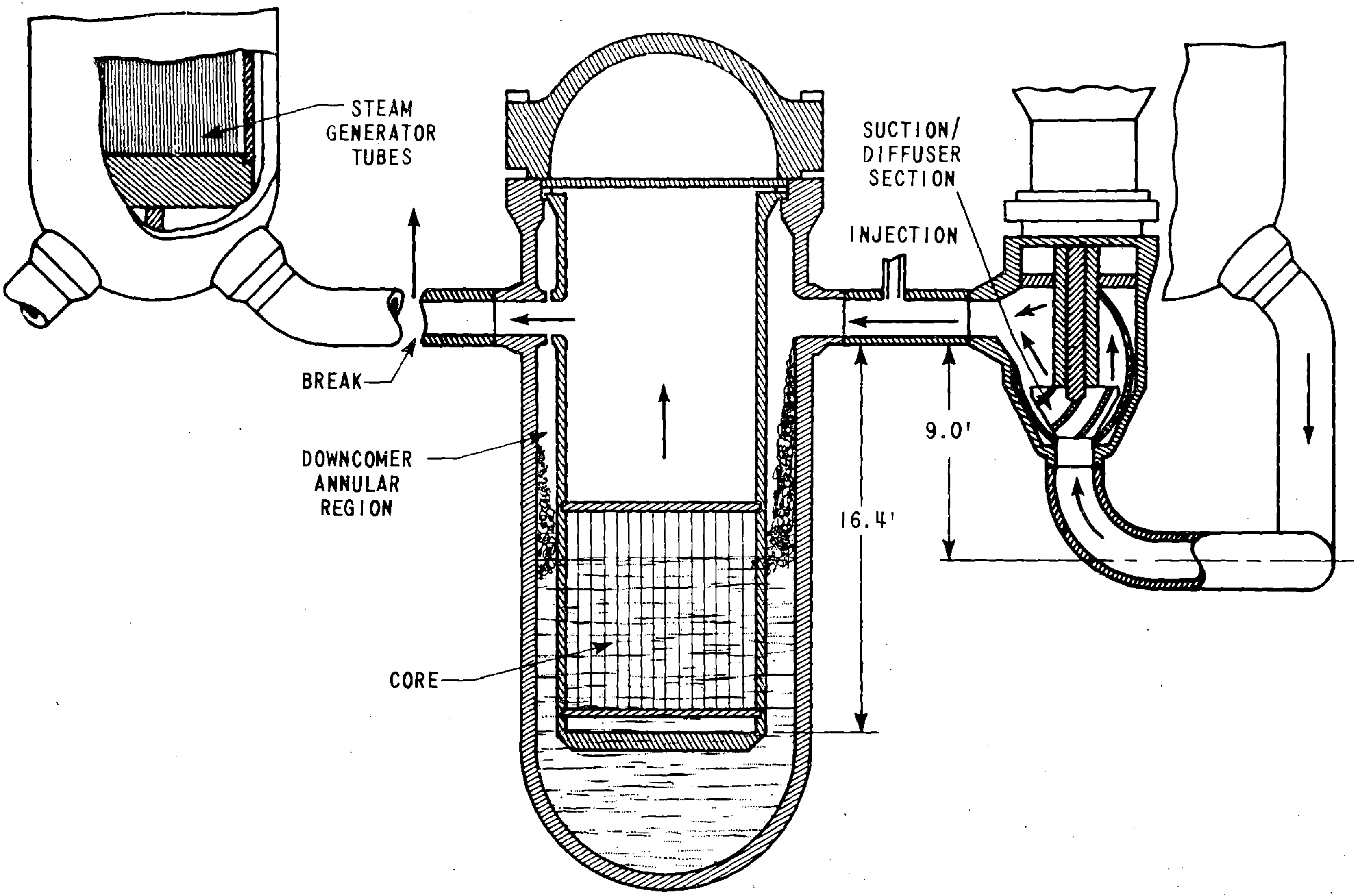
LIQUID VOLUME VERSUS
TIME 3.0 FT^2 - COLD LEG BREAK

FIGURE 14.5.3.2-3
2-13-70



LIQUID VOLUME VERSUS
TIME 0.5 FT^2 - COLD LEG BREAK

FIGURE 14.5.3.2-4
2-13-70



HOT LEG BREAK STEAM
FLOW PATH SCHEMATIC

FIGURE 14.5.3.2-5
2-13-70

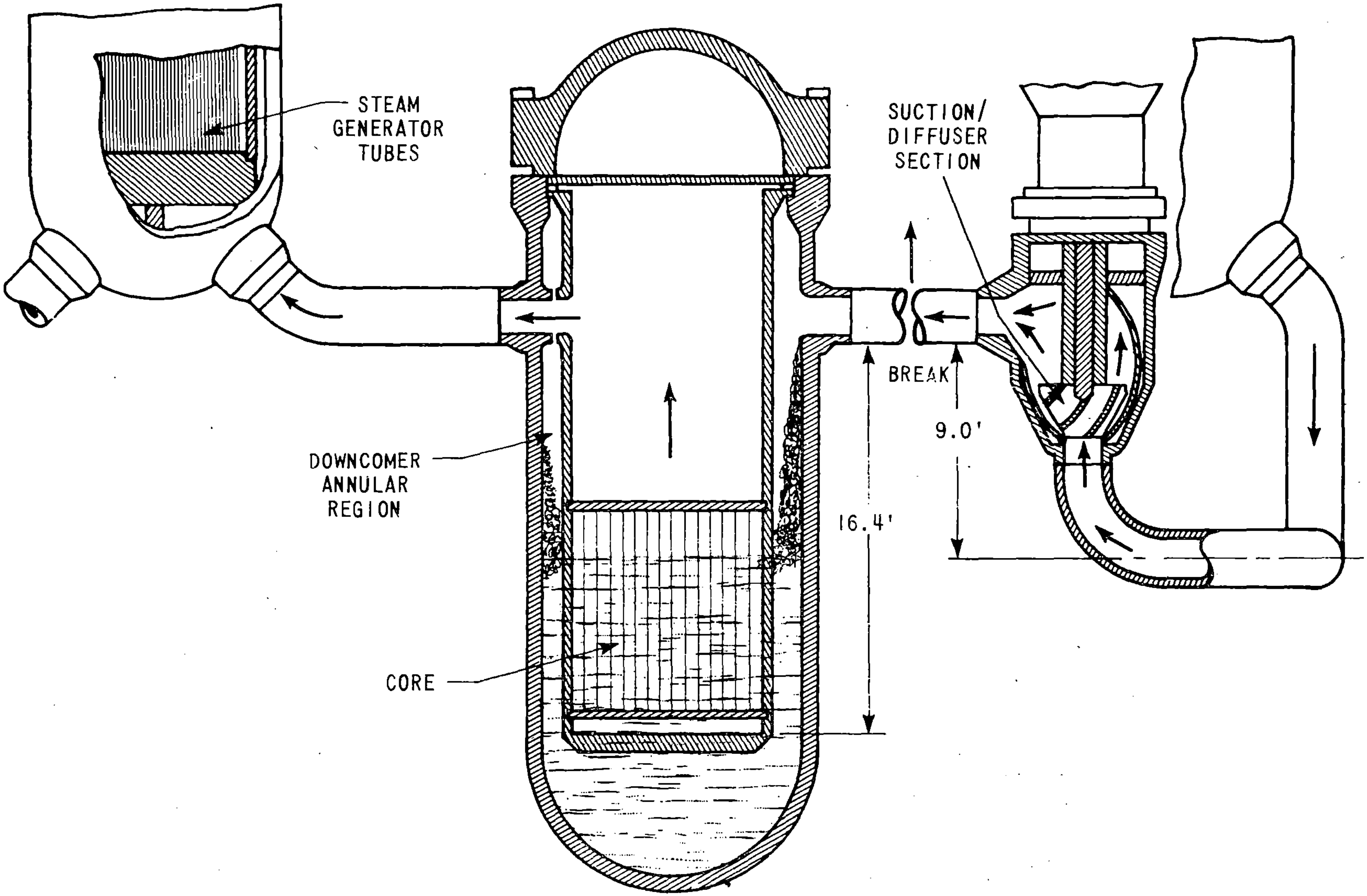
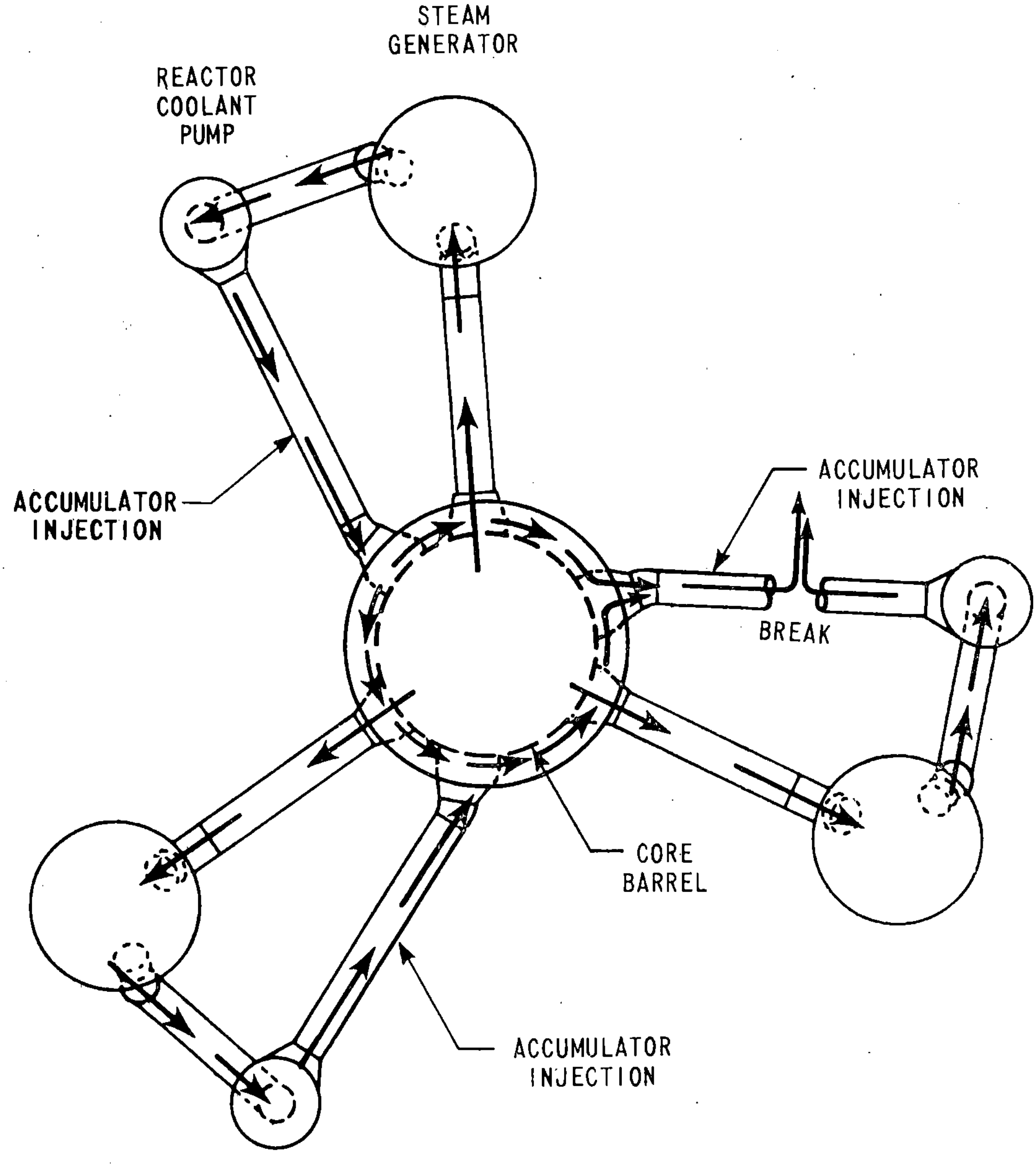


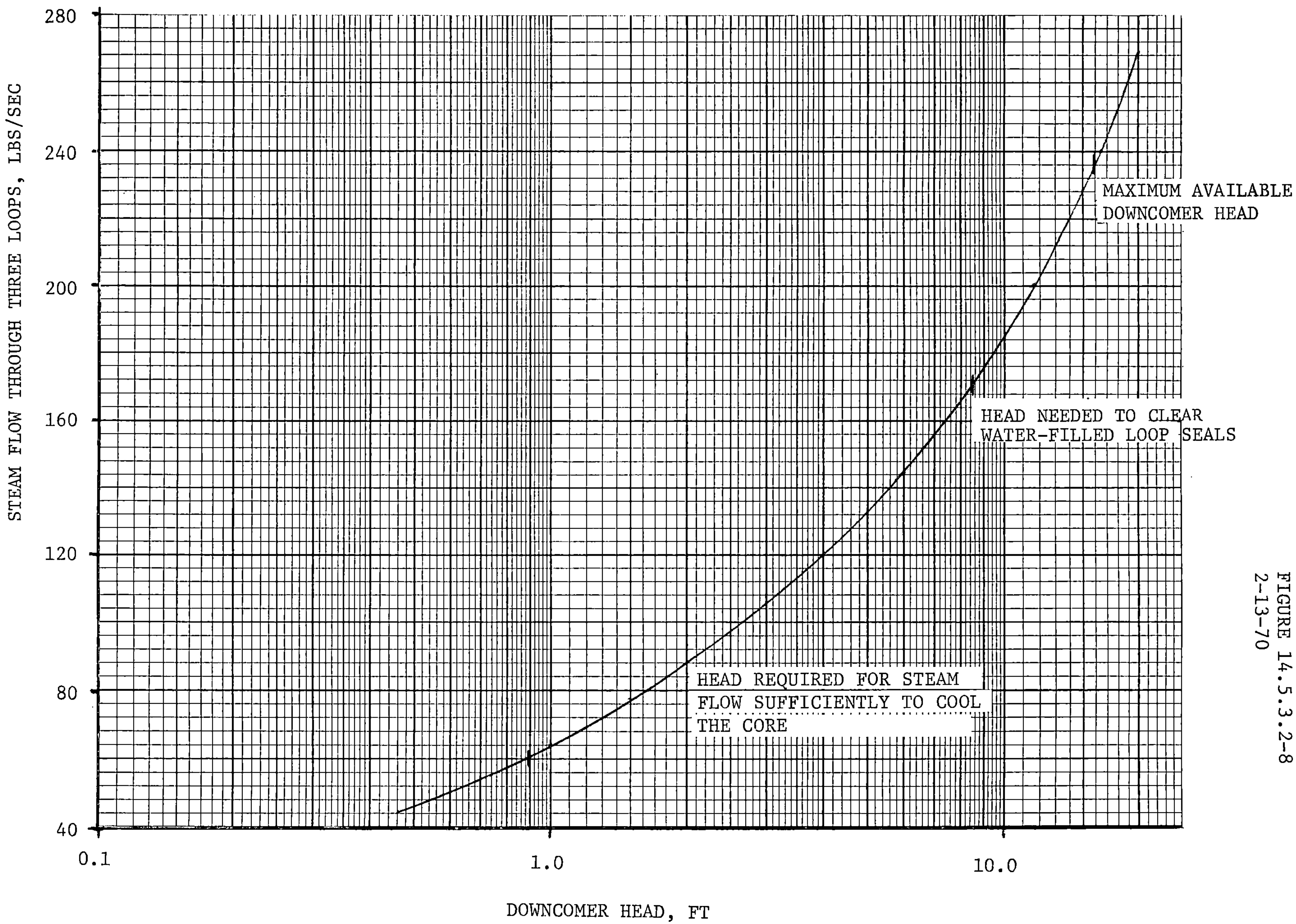
FIGURE T4.5.3.2-6
2-13-70

COLD LEG BREAK STEAM
FLOW PATH SCHEMATIC

FIGURE 14.5.3.2-7
2-13-70



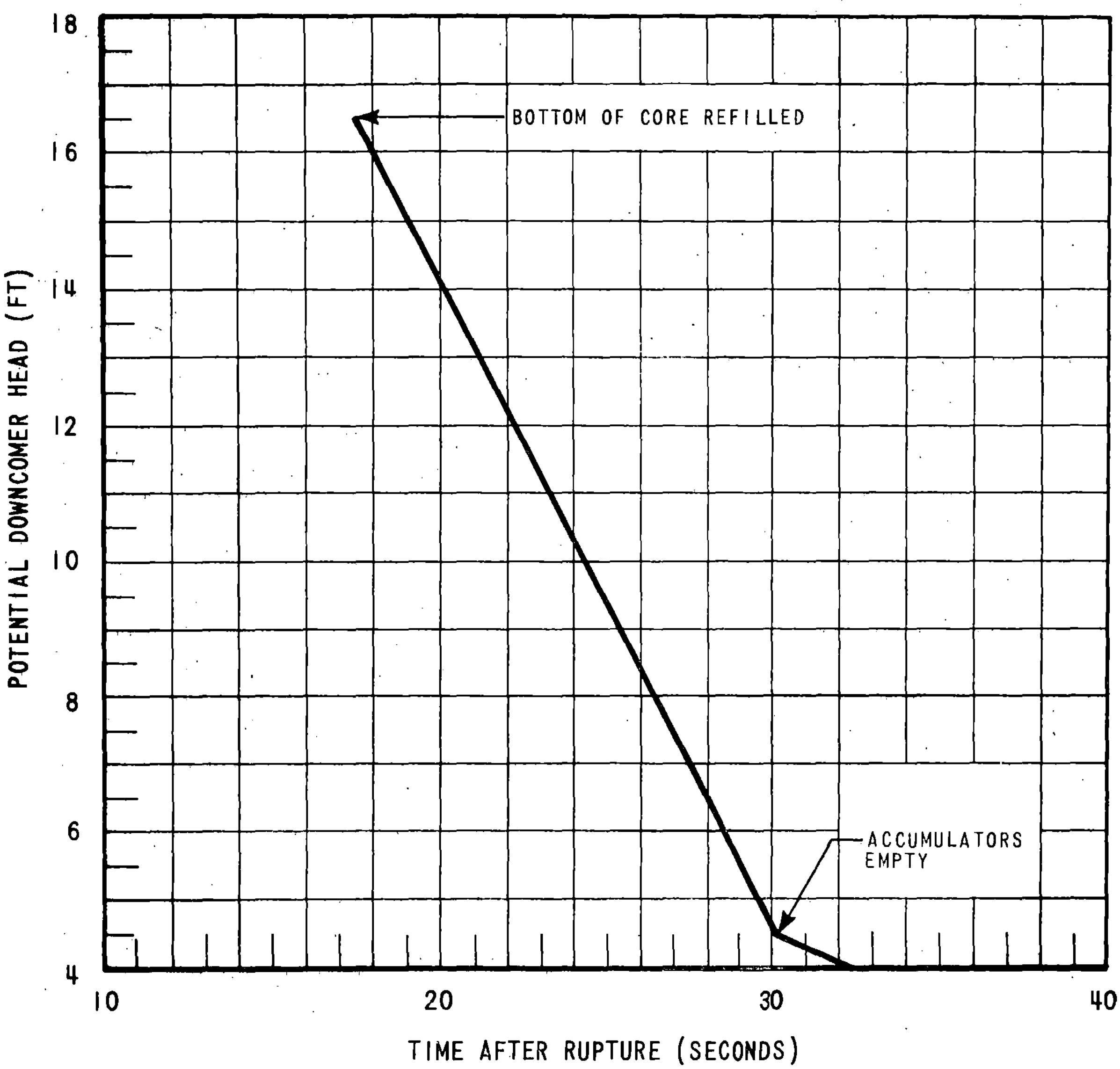
COLD LEG BREAK STEAM
FLOW PATH



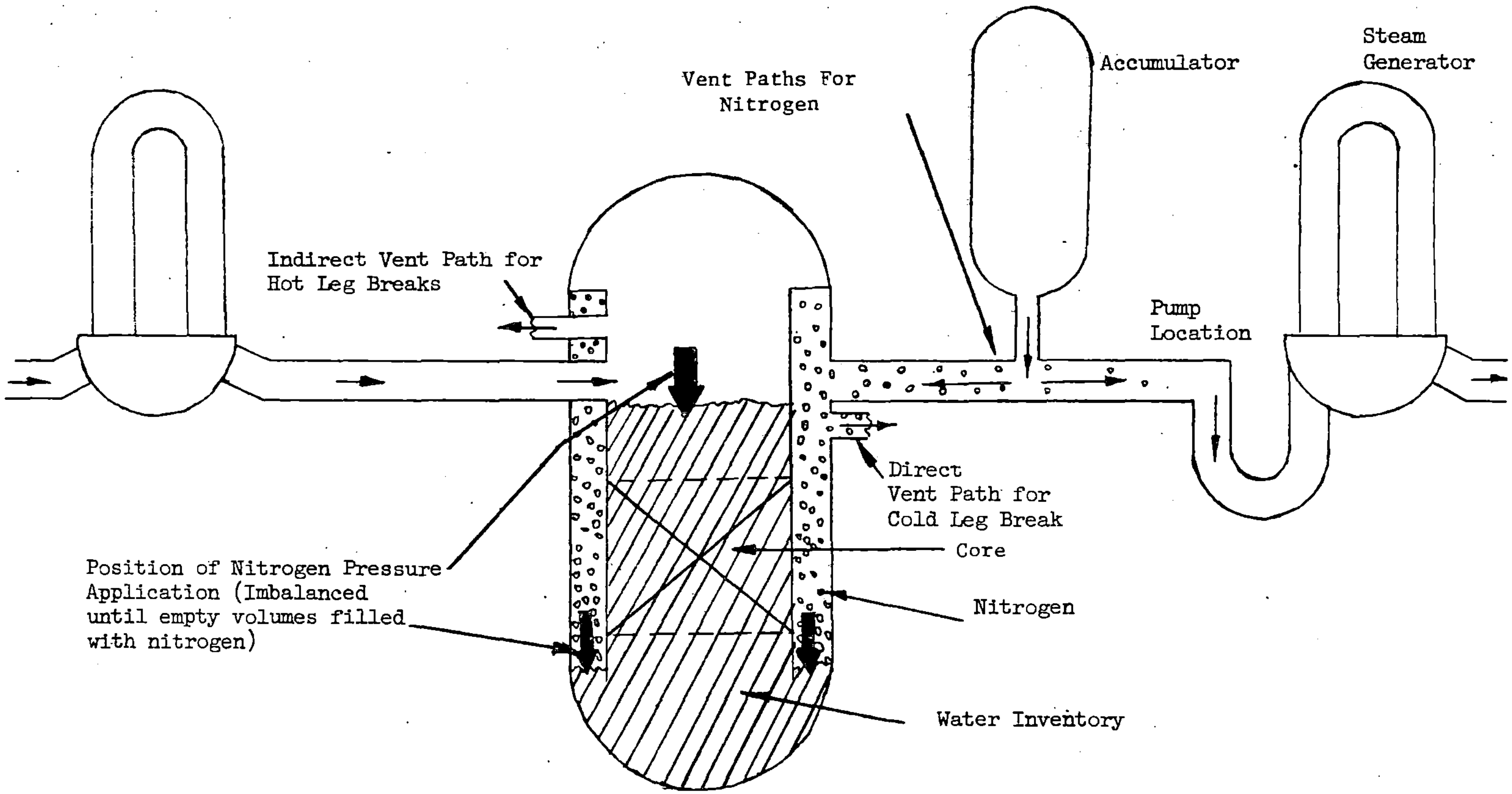
WATER HEAD IN DOWNCOMER VS
STEAM FLOW (SATURATED CONDITIONS
AT 60 PSIA)

FIGURE 14.5.3.2-8
2-13-70

FIGURE 14.5.3.2-9
2-13-70



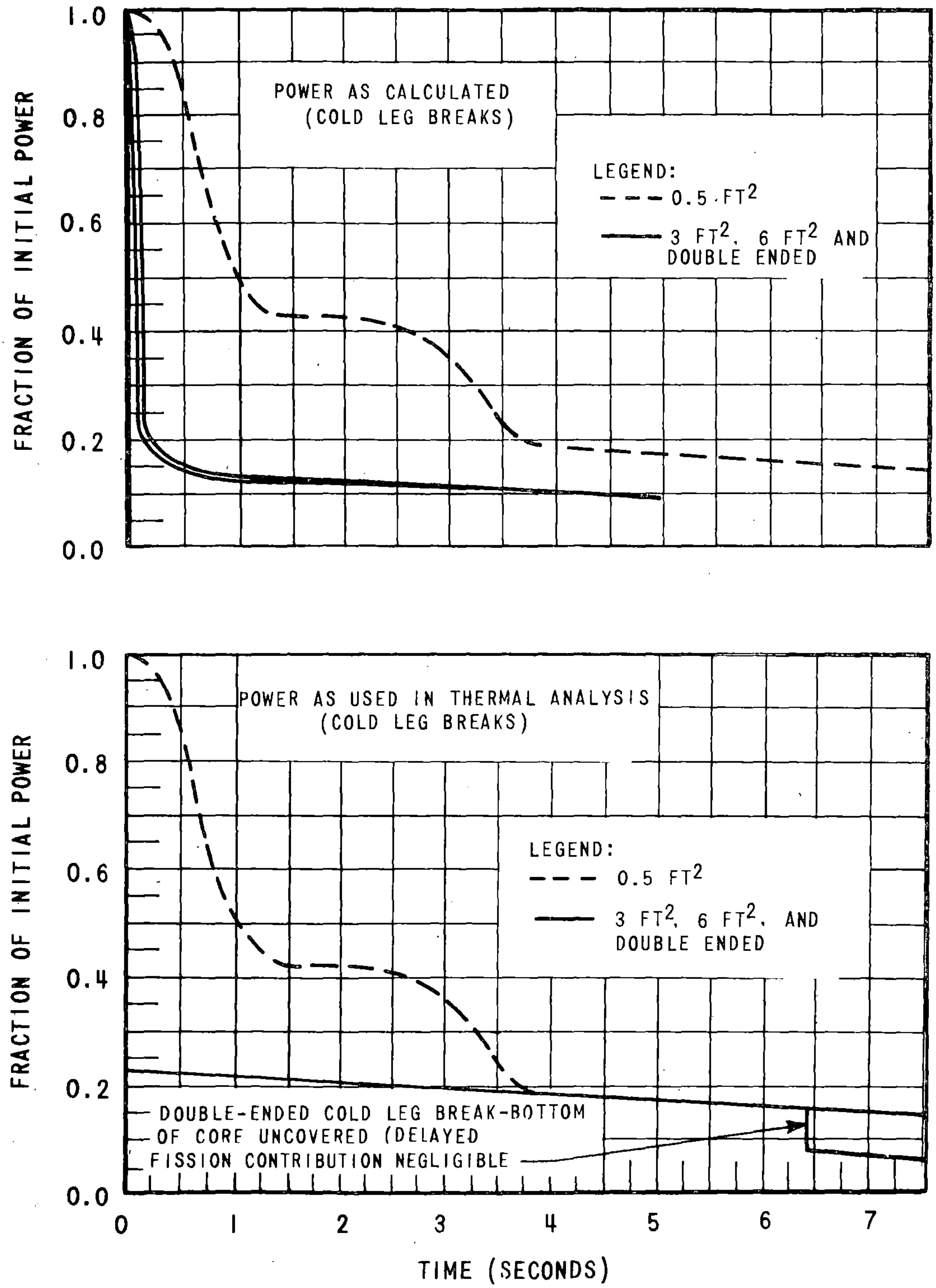
AVAILABLE DOWNCOMER HEAD VERSUS TIME
DOUBLE-ENDED COLD LEG BREAK



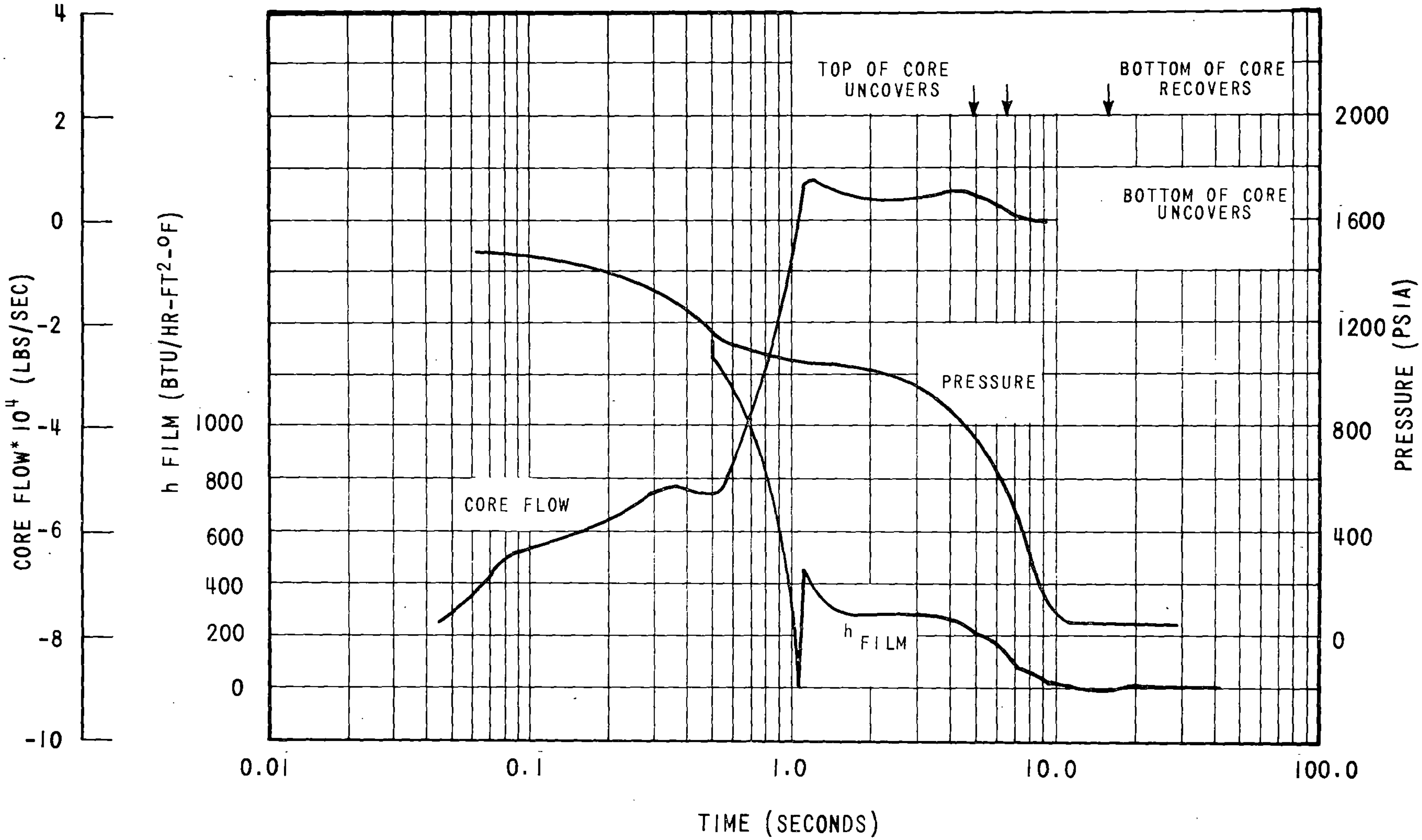
NITROGEN VENT PATHS

FIGURE 14.5.3.2-10
2-13-70

FIGURE 14.5.3.2-11
2-13-70

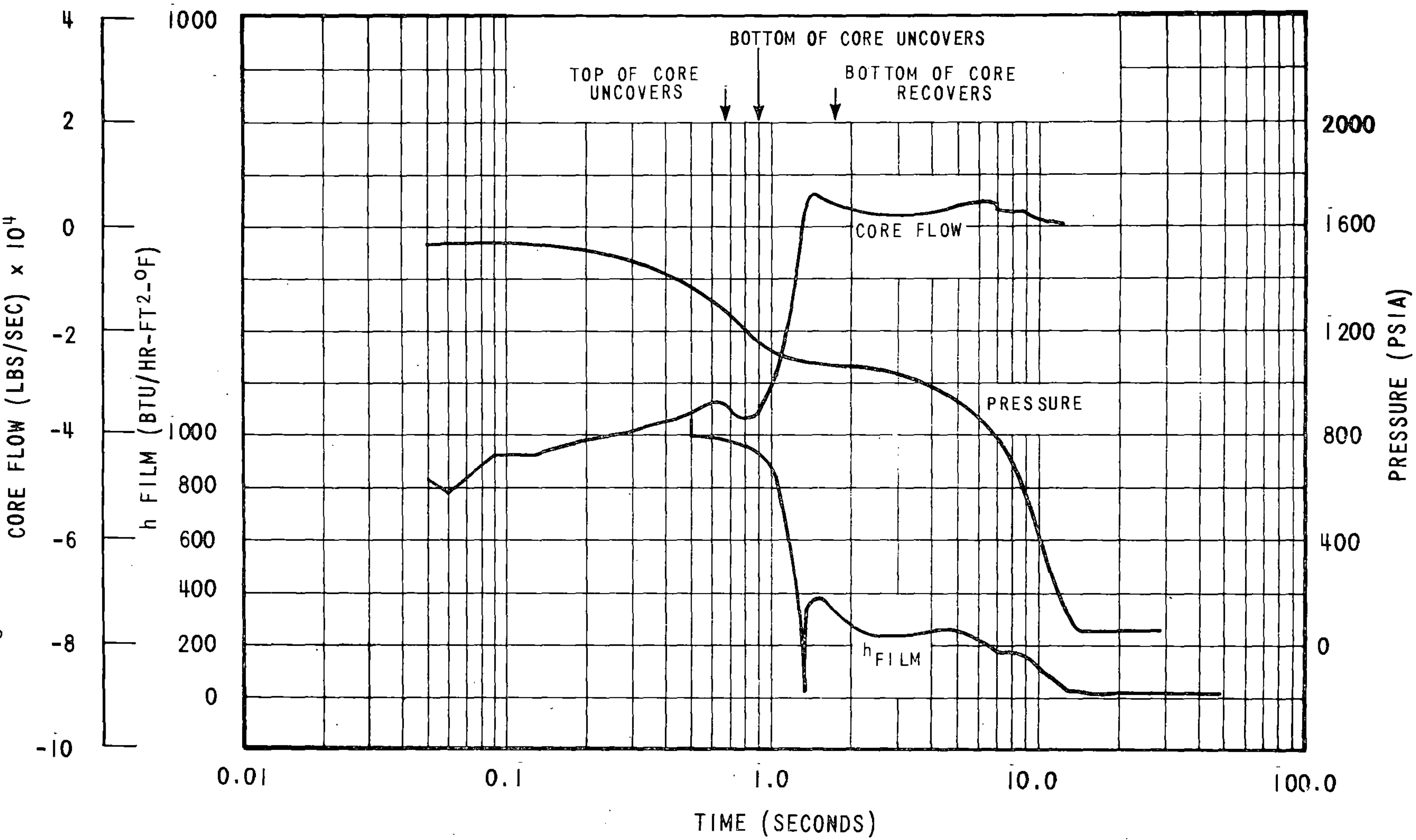


POWER TRANSIENT
DURING BLOWDOWN



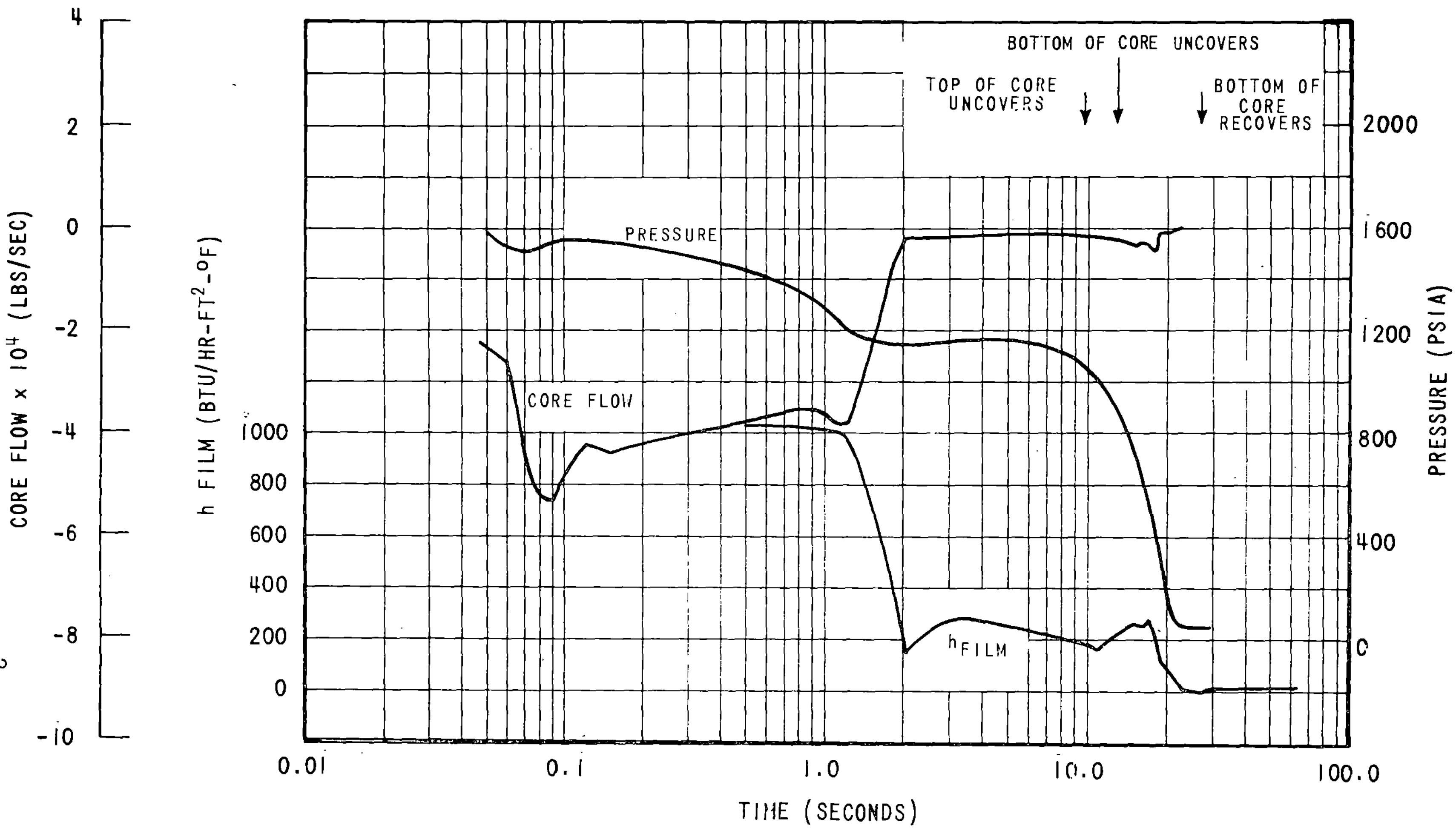
DOUBLE-ENDED COLD LEG BREAK

FIGURE 14.5.3.2-12
2-13-70



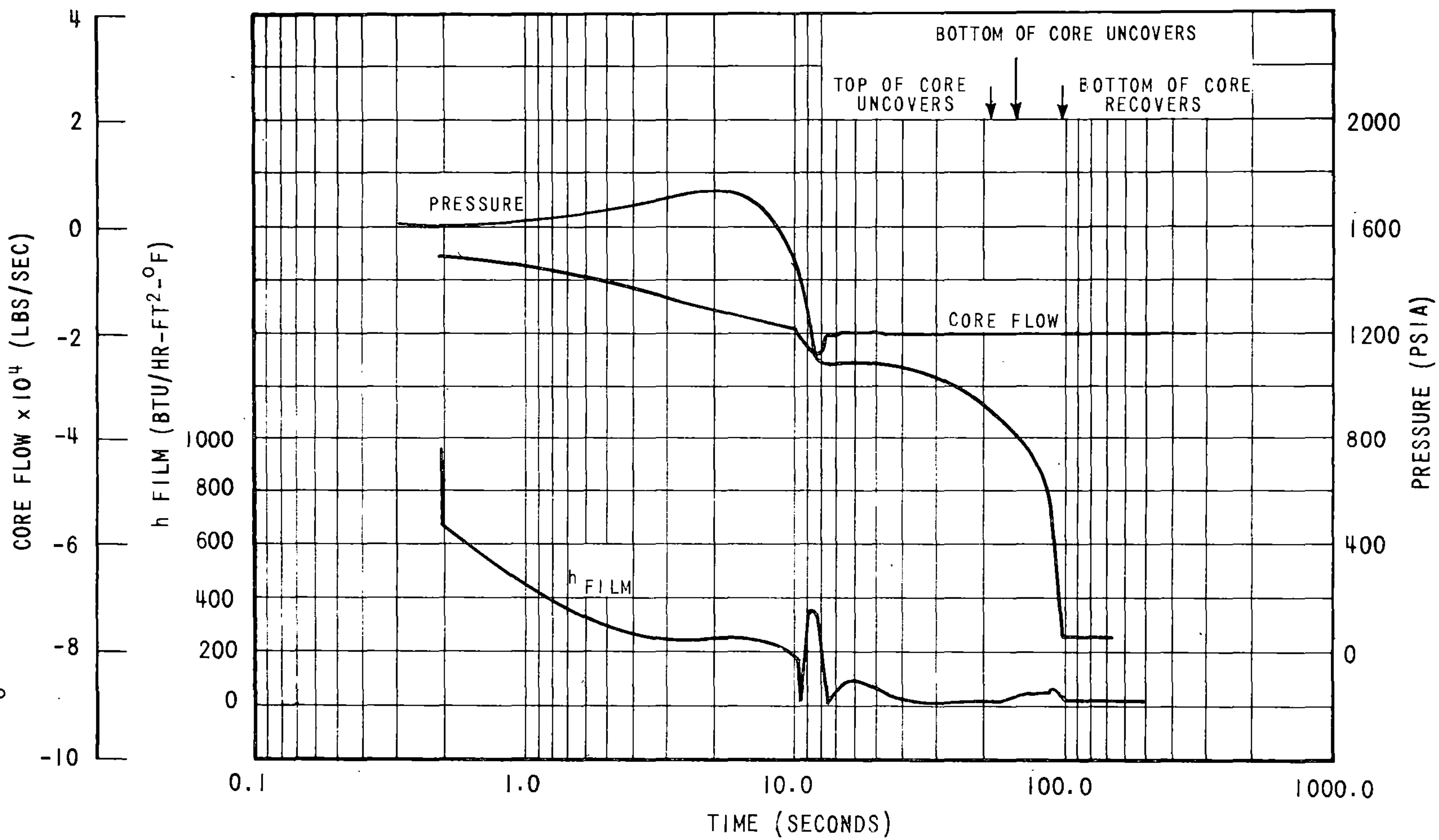
6 FT^2 COLD LEG BREAK

FIGURE 14.5.3.2-13
2-13-70



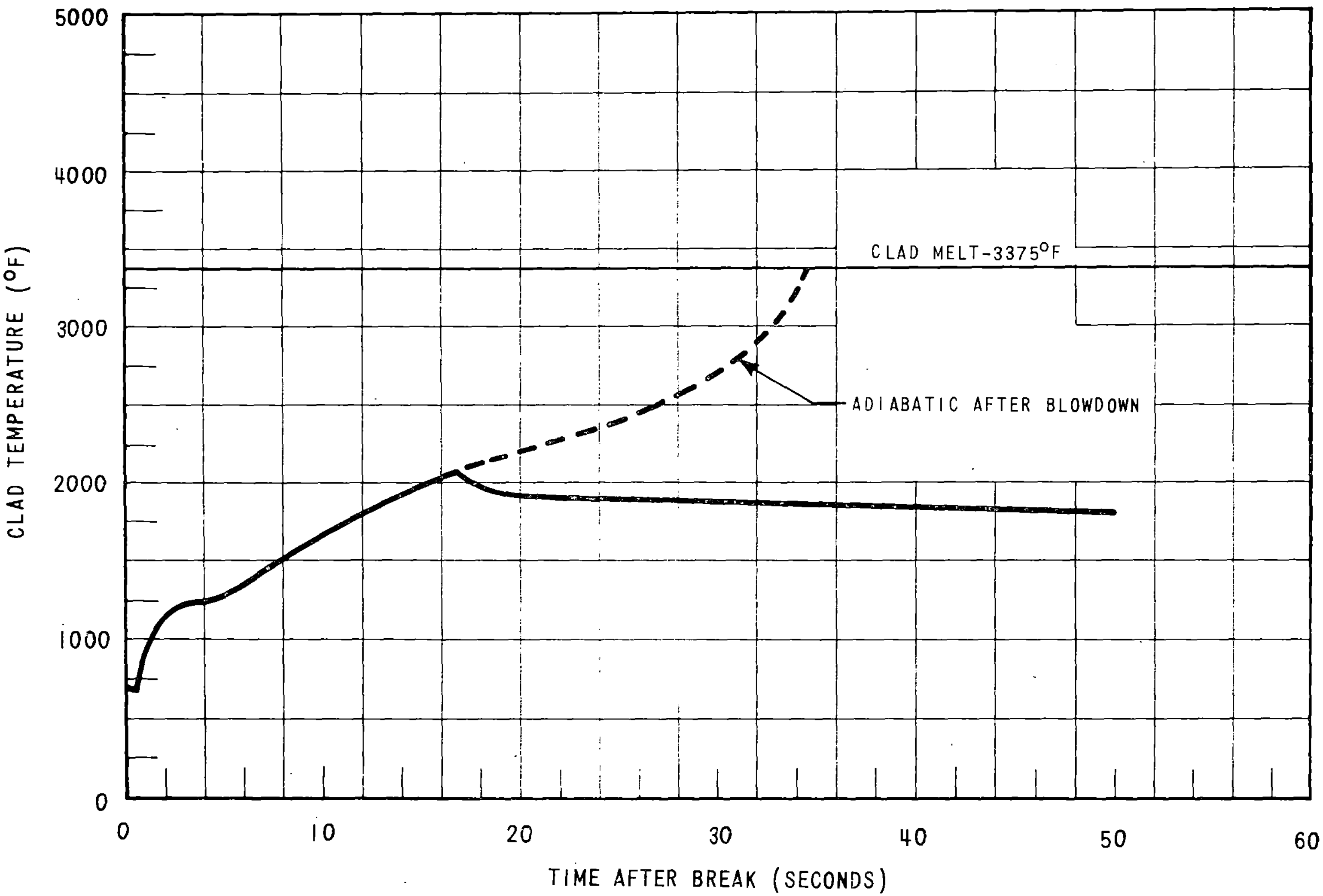
3 FT^2 COLD LEG BREAK

FIGURE 14.5.3.2-14
2-13-70



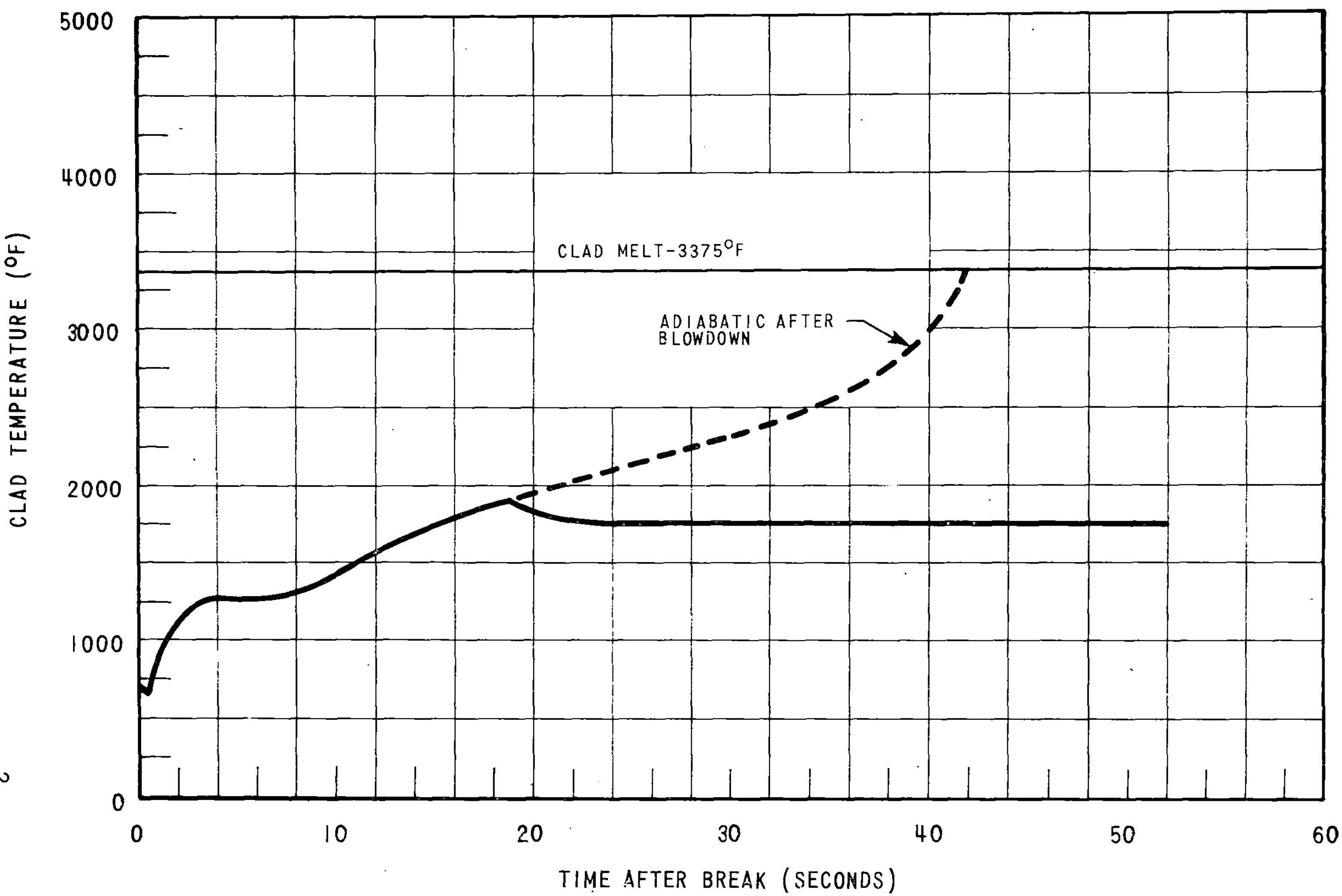
5 FT^2 COLD LEG BREAK

FIGURE 14.5.3.2-15
2-13-70



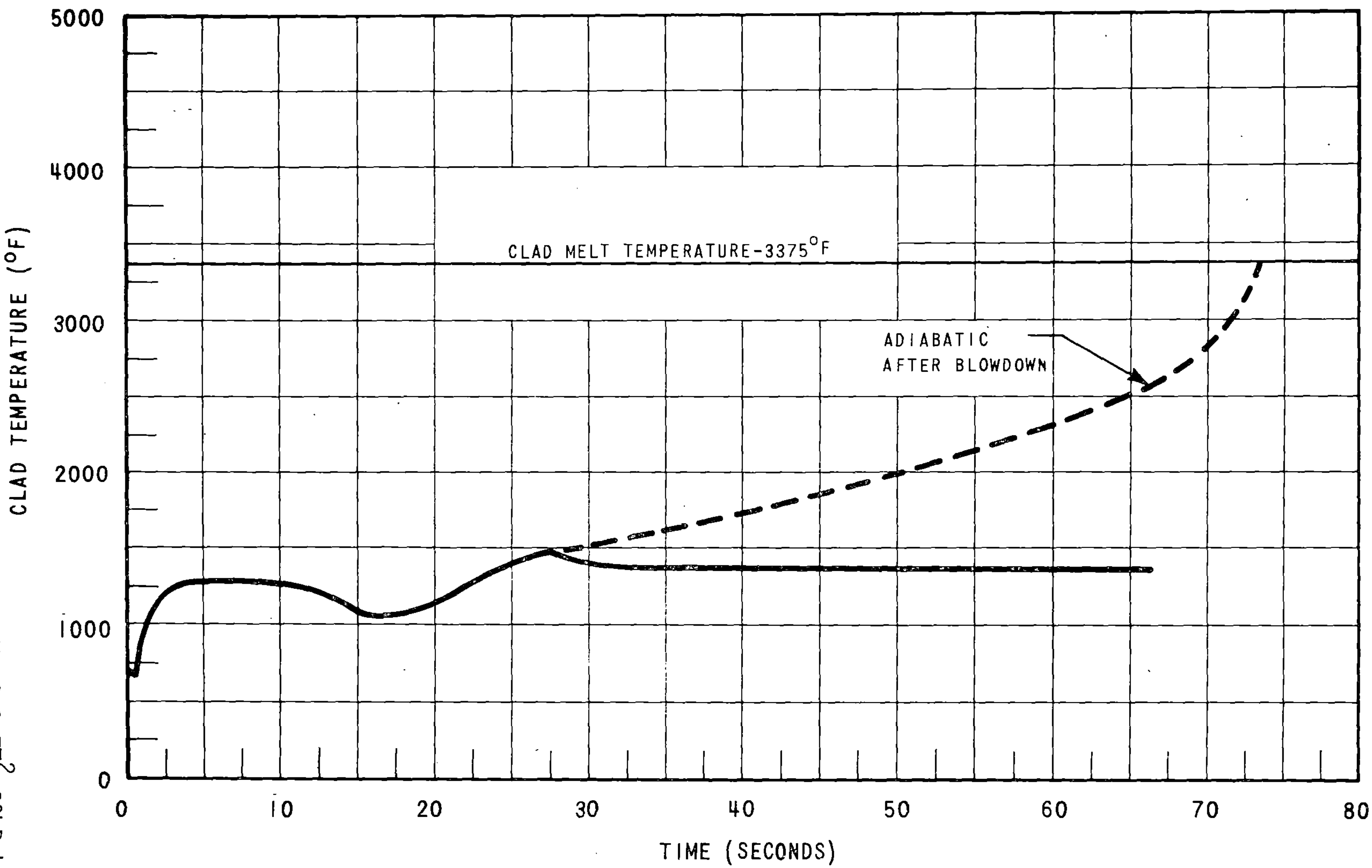
2546 MWt DOUBLE-ENDED COLD LEG BREAK HOT SPOT CLAD TEMPERATURE VERSUS TIME

FIGURE 14.5.3.2-16



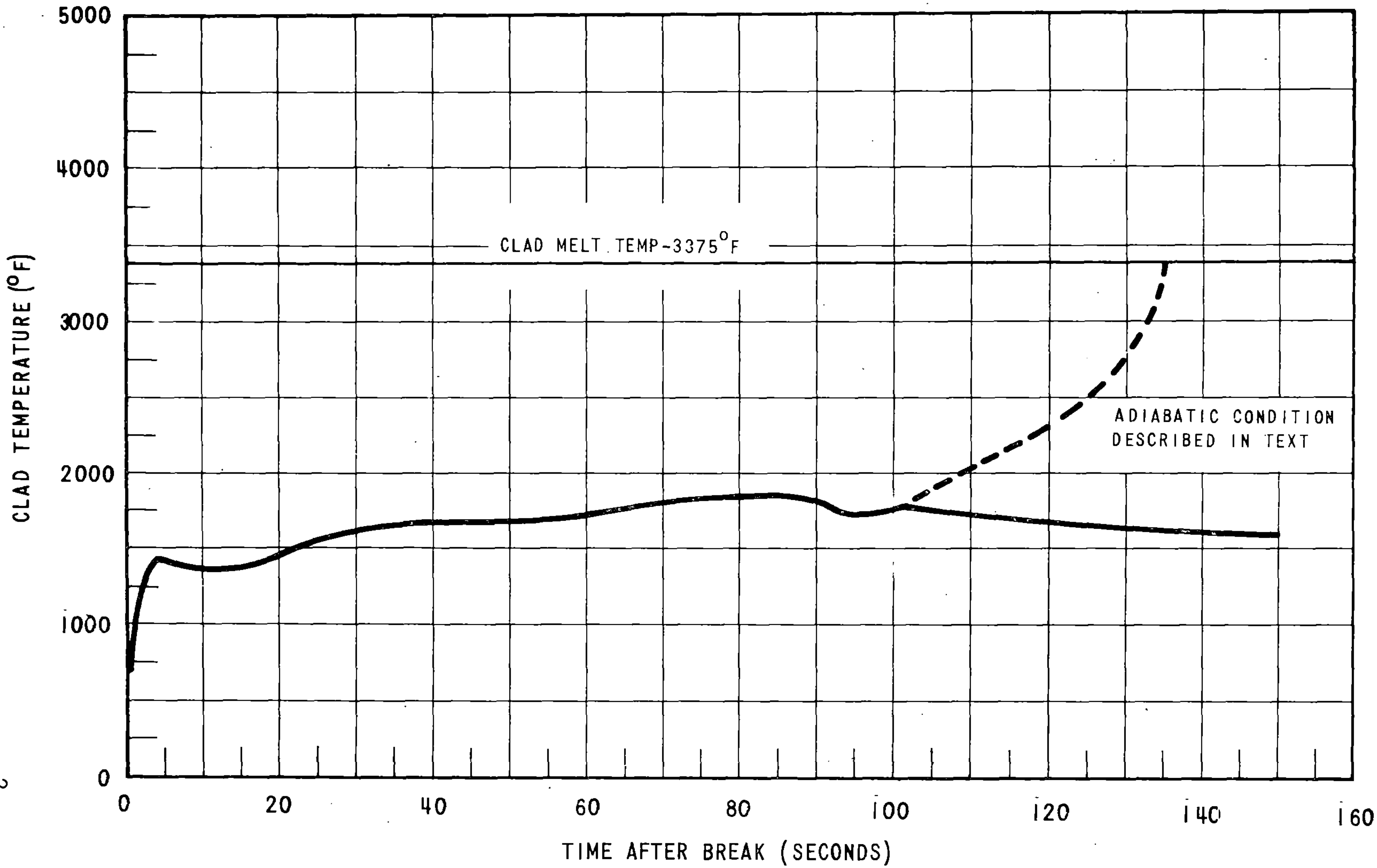
2546 MWt 6.0 FT^2 COLD LEG BREAK
HOT SPOT CLAD TEMPERATURE
VERSUS TIME

FIGURE 14.5.3.2-17
2-13-70



2546 MWt 3.0 FT^2 COLD LEG BREAK
HOT SPOT CLAD TEMPERATURE
VERSUS TIME

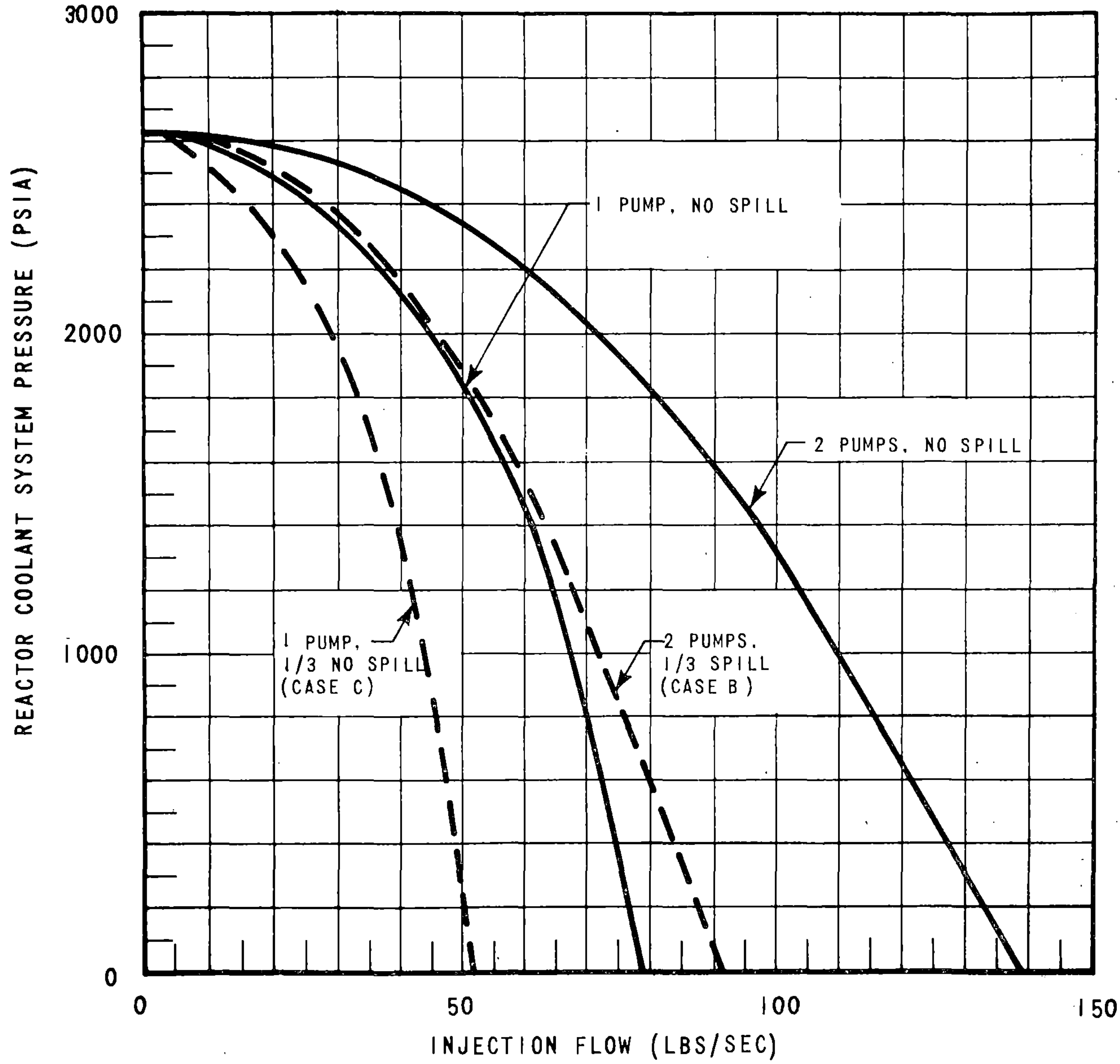
FIGURE 14.5.3.2-18
2-13-70



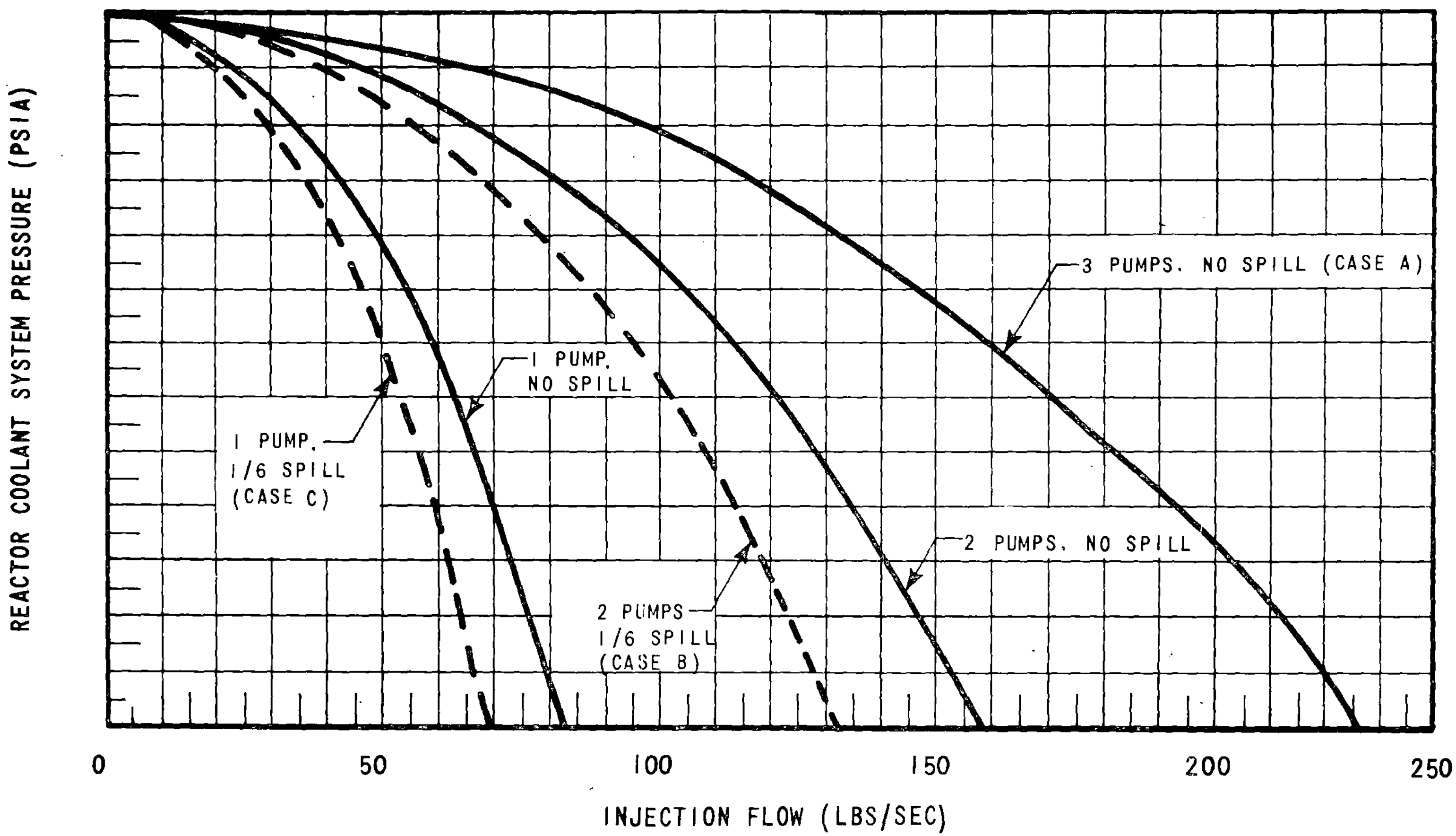
2546 MWt 0.5 FT^2 COLD LEG BREAK
HOT SPOT CLAD TEMPERATURE
VERSUS TIME

FIGURE 14.5.3.2-19

FIGURE 14.5.3.2-20
2-13-70

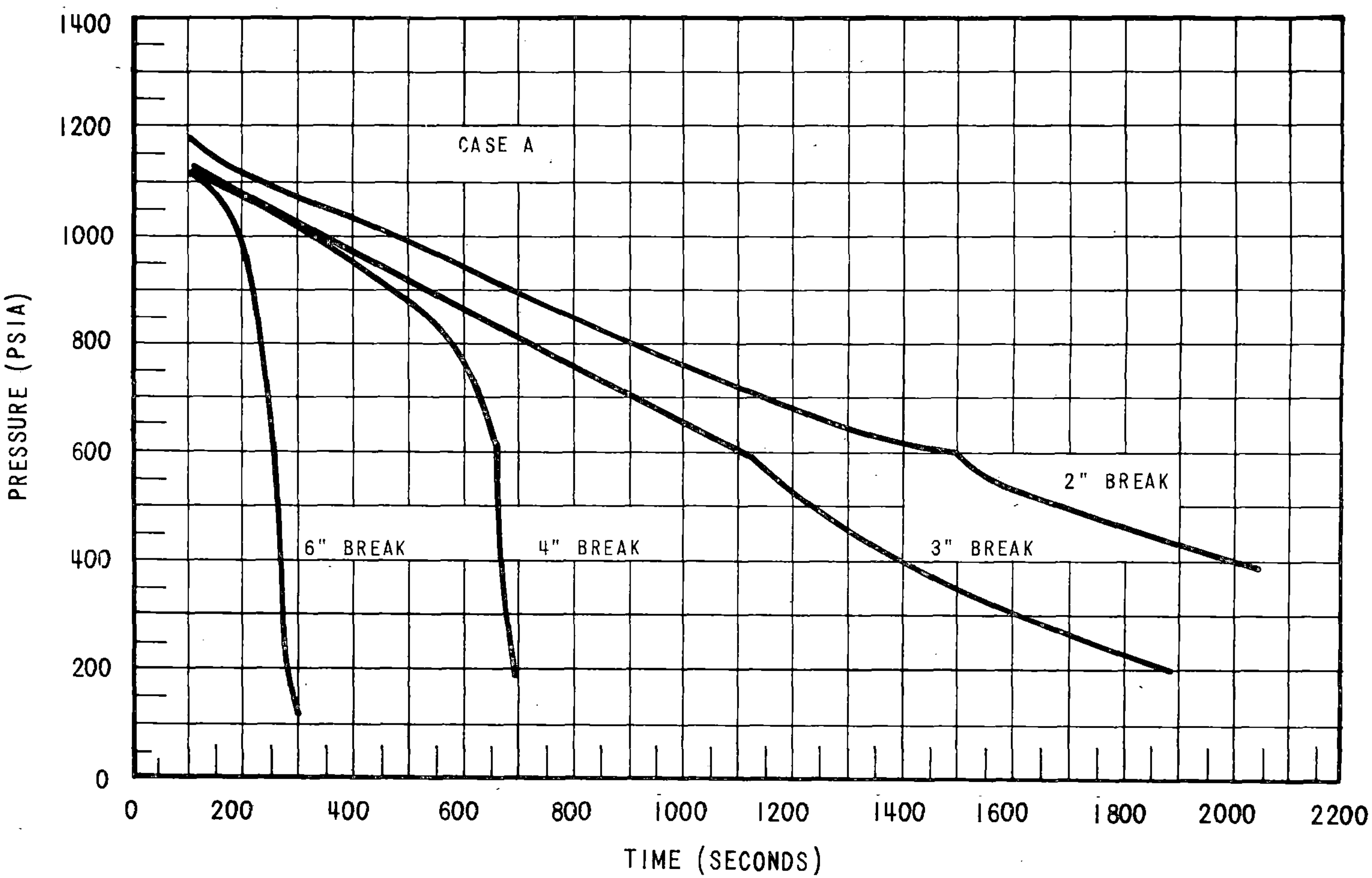


SAFETY INJECTION
3 LINES INJECTING



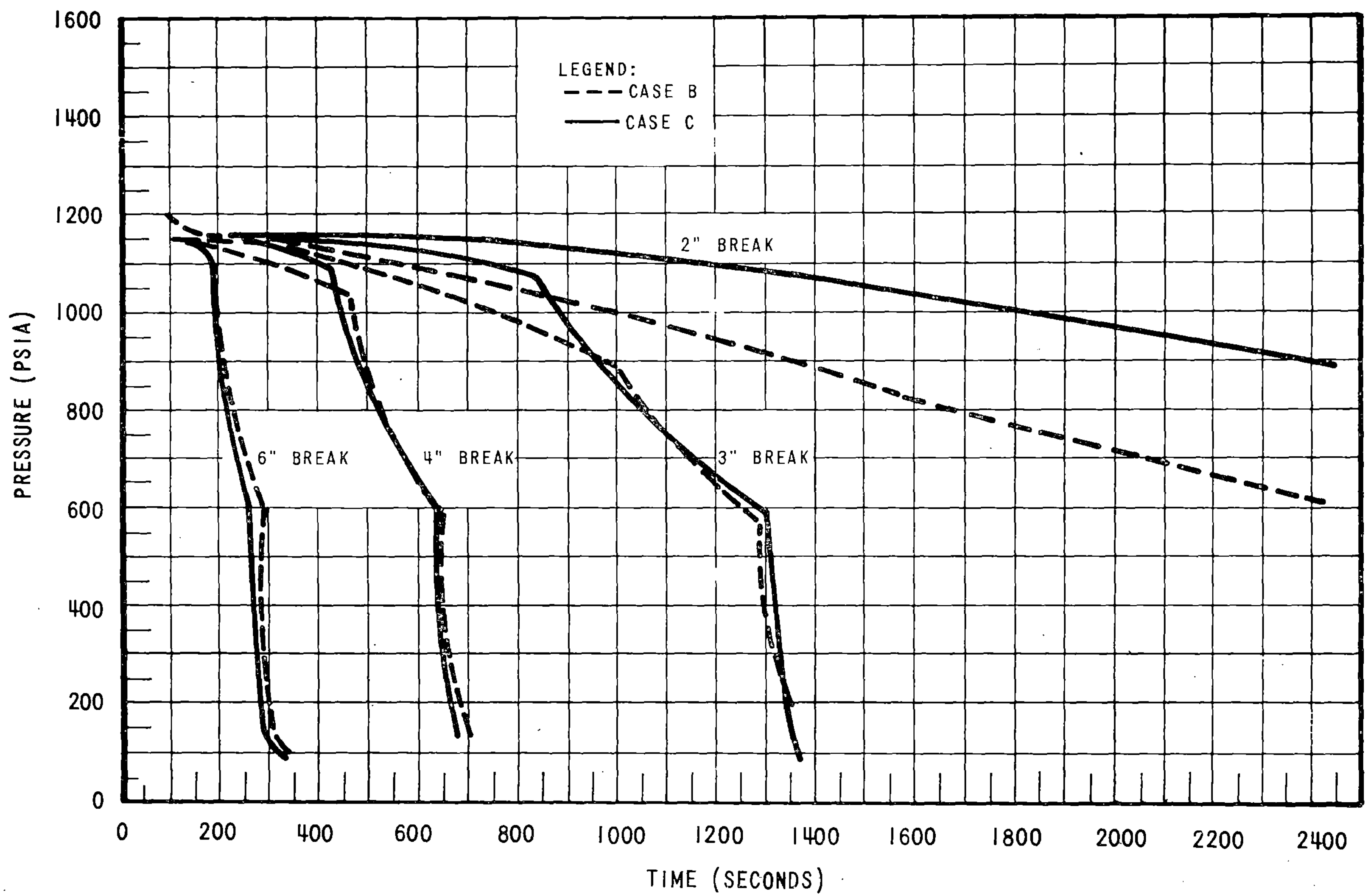
SAFETY INJECTION
6 LINES INJECTING

FIGURE 14.5.3-2-21



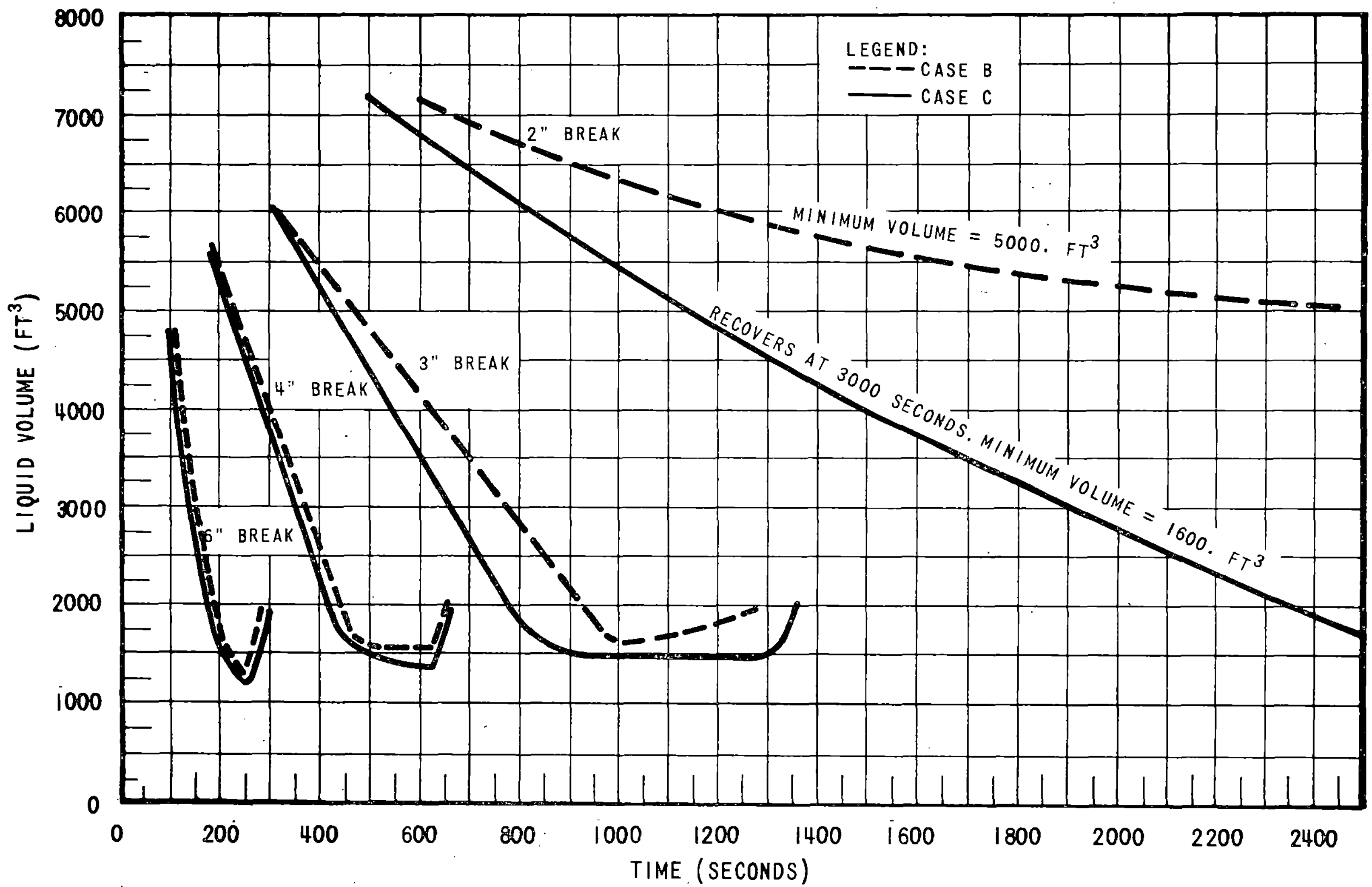
PRESSURE VERSUS TIME

FIGURE 14.5.3.2-22
2-13-70



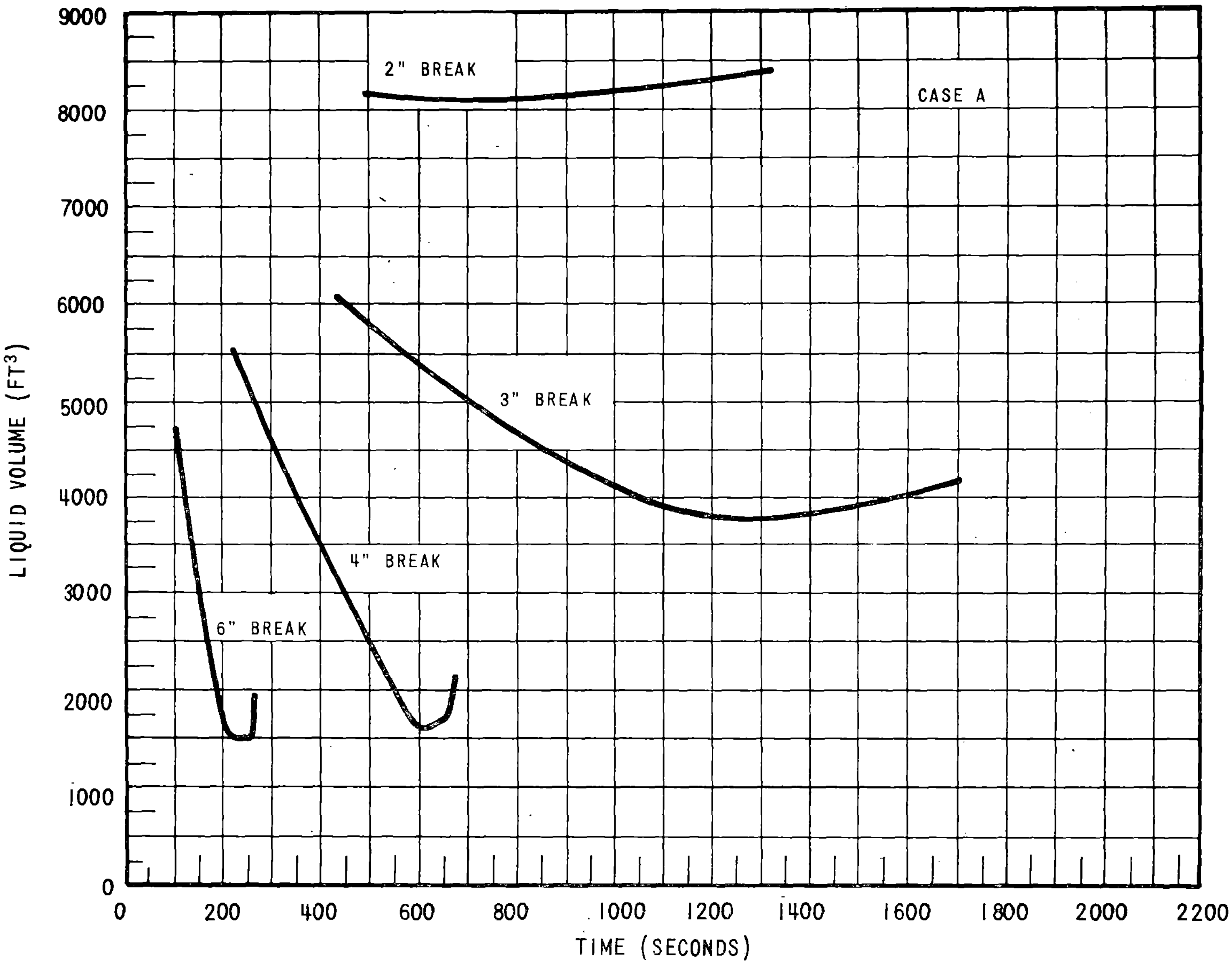
PRESSURE VERSUS TIME

FIGURE 14.5.3.2-23
2-13-70



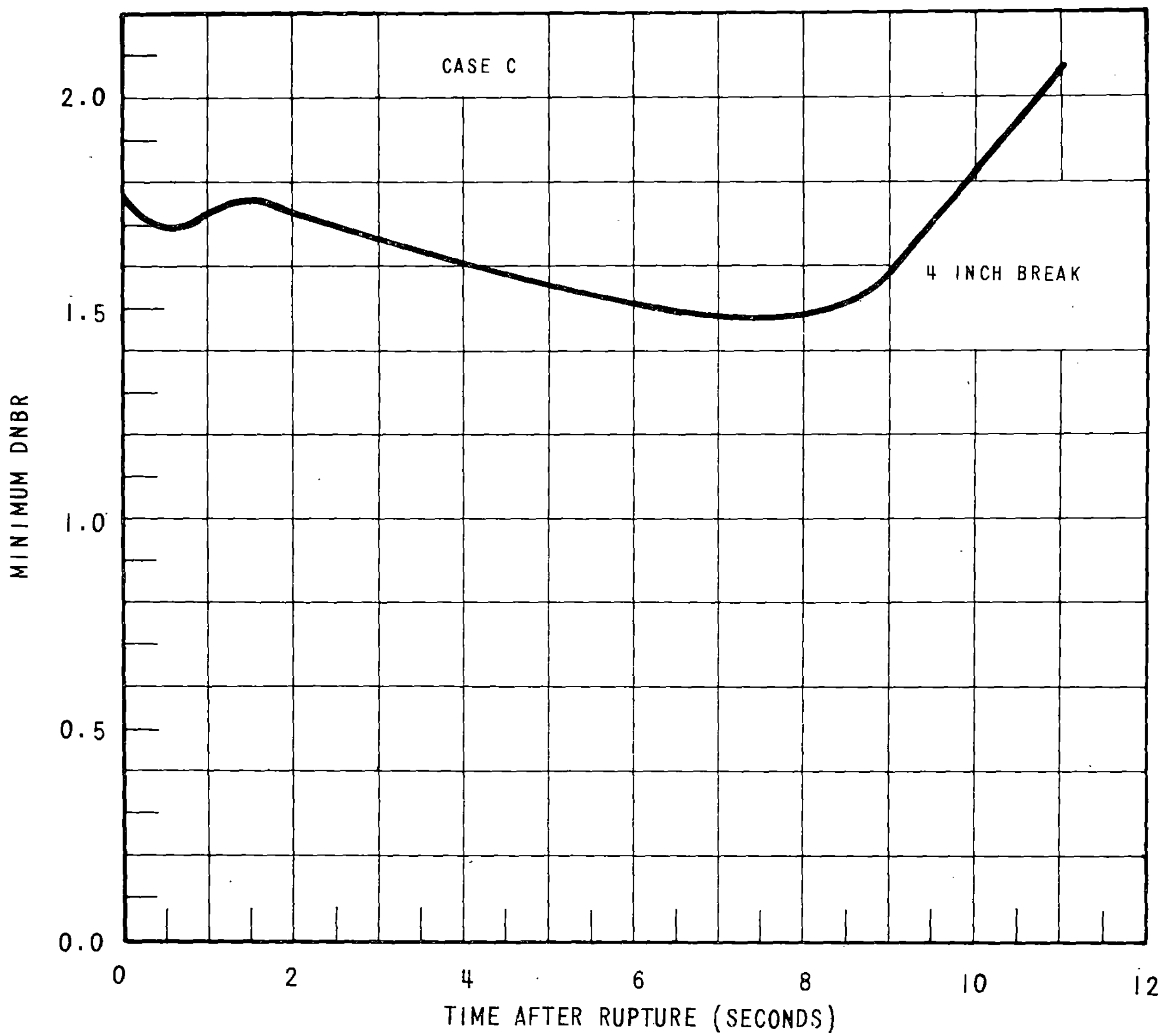
LIQUID VOLUME VERSUS TIME

FIGURE 14.5.3.2-24
2-13-70



LIQUID VOLUME
VERSUS TIME

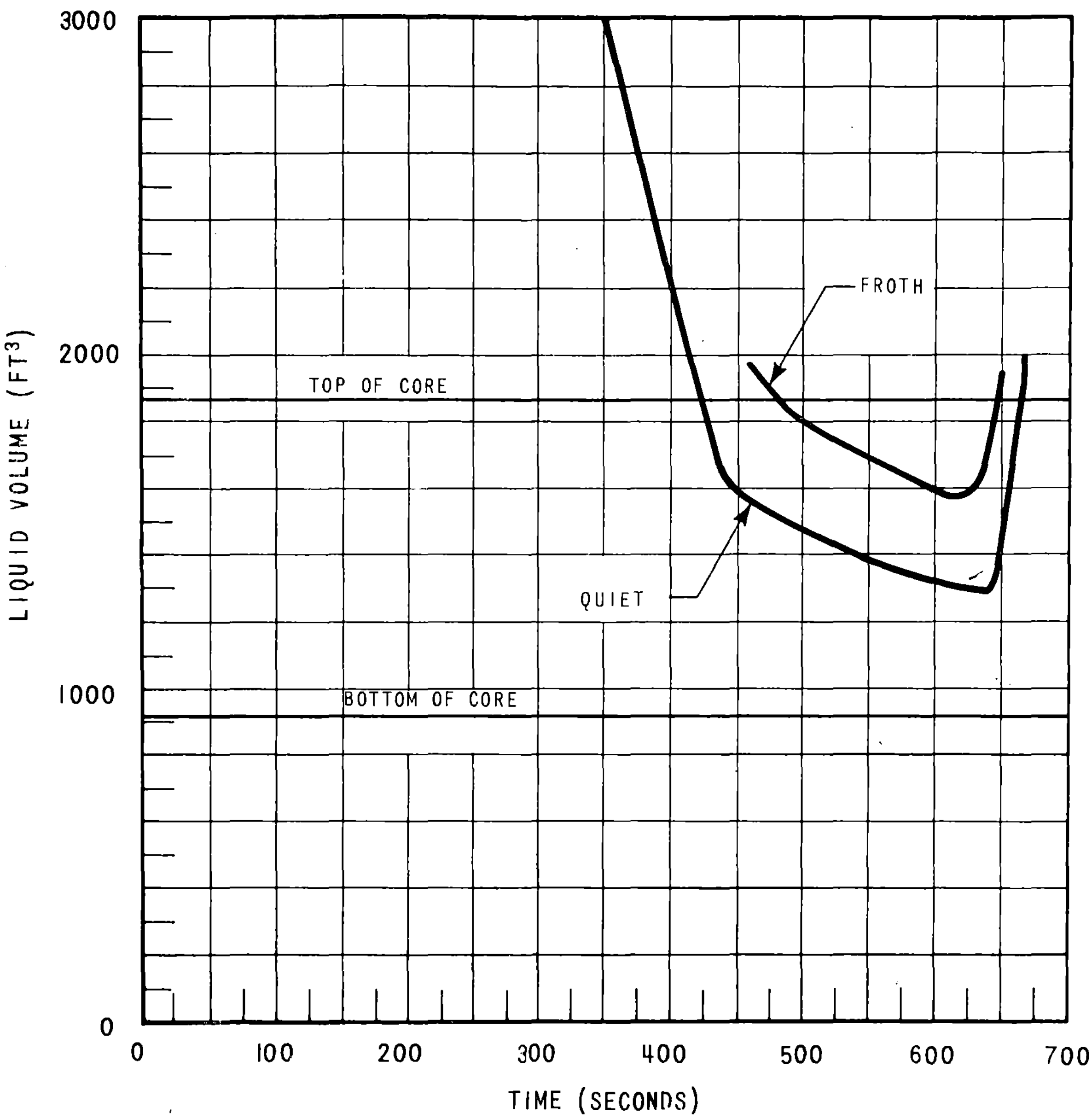
FIGURE 14.5.3.2-25
2-13-70



MINIMUM DNBR
VERSUS TIME

FIGURE 14.5.3.2-26
2-13-70

FIGURE 14.5.3.2-27
2-13-70



VOLUME VERSUS TIME
4-INCH COLD LEG BREAK
SAFETY INJECTION-CASE C

14.5.4 CORE AND INTERNALS INTEGRITY ANALYSIS

14.5.4.1 Internals Evaluation

The forces exerted on the reactor internals and the core, following a loss-of-coolant accident, are computed by employing the BLODWN-2 digital computer program developed for the space-time-dependent analysis of multi-loop PWR plants. The detailed results of the analysis are submitted as a supplement to this report.

14.5.4.2 Design Criteria

Following a loss-of-coolant accident, the basic requirement is that the plant shall be shut down and cooled down in an orderly manner so that fuel cladding temperature is kept within the specified limits. This implies that the deformation of the reactor internals must be kept sufficiently small so that the core geometry remains substantially intact to allow core cooling and insertion of a sufficient number of control rod assemblies.

After the break, the reduction in water density greatly reduces the reactivity of the core, thus shutting down the core independent of the control rod assemblies. In other words, the core is shutdown whether or not the control rod assemblies are tripped. (The subsequent refilling of the core by the Emergency Core Cooling System uses borated water to maintain the core in a subcritical state.) Therefore insertion of most of the control rod assemblies gives further assurance of the ability to shut the unit down and keep it in a safe shutdown condition.

Maximum allowable deflection limitations are established for those regions of the internals that are critical for unit shutdown. Allowable stress limits are adopted to assure physical integrity of the components.

In the event of a sudden double-ended Reactor Coolant System pipe rupture (complete severance in a few milliseconds), pressure waves are produced in the reactor causing vertical and horizontal excitation of the components. A study has been made to analyze the response of the reactor vessel internal structures under these conditions.

14.5.4.3 Blowdown and Force Models

Blowdown Model

BLODWN-2 is a digital computer program used for calculation of local fluid pressure, flow, and density transients that occur in Reactor Coolant System during a loss-of-coolant accident. This program applies to the subcooled, transition and saturated two-phase blowdown regimes. This is in contrast to programs such as WHAM[1] which are applicable only to the subcooled region and which, due to their method of solution, could not be extended into the region in which large changes in the sonic velocities and fluid densities take place.

BLODWN-2 is based on the method of characteristics, wherein the resulting set of ordinary differential equations obtained from the laws of conservation of mass, momentum and energy, are solved numerically using a fixed mesh in both space and time.

Although one-dimensional conservation laws are employed, the code can be applied to describe 3-dimensional system geometries through the use of the equivalent piping networks. Such piping networks may contain any number of pipes or channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansions, orifices, pumps and free surfaces (such as in a pressurizer). System losses such as friction, contraction, expansion, etc. are considered.

Comparison with Experimental Data

BLODWN-2 predictions have been compared with data obtained by Phillips Petroleum Company from their LOFT semi-scale and 1/4 scale blowdown experiments.

An example of these comparisons is shown in Figure 14.5.4-2 which illustrates the pressure history in the blowdown pipe for the semi-scale test #522. This was a bottom blowdown test for the "Bettis Flask No. 1" geometry with initial uniform fluid conditions of 1268 psia and 445°F. It is seen, that the BLODWN-2 digital computer program gives good agreement in both the subcooled and the saturated regimes.

Force Model

BLODWN-2 evaluates the pressure and velocity transients for a maximum of 2400 locations throughout the system. These pressure and velocity transients are stored as a permanent tape file and are made available to the program FORCE which uses a detailed geometric description in evaluating the loading on the reactor internals.

Each reactor component for which force calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of:

1. The pressure differential across the element.

2. Flow stagnation on, and unrecovered orifice losses across the element.

3. Friction losses along the element.

Input to the code, in addition to the BLODWN-2 pressure and velocity transients, includes the effective area of each element on which the force acts due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

14.5.4.4 Responses of Reactor Internals to Blowdown Forces

Vertical Excitation

Structural Model and Method of Analysis

The internal structure is simulated by a multi-mass system connected with springs and dashpots representing the viscous damping due to structural and impact losses. The gaps between various components, as well as Coulomb type of friction, is also incorporated into the overall model. Since the fuel elements in the fuel assemblies are kept in position by friction forces originating from the preloaded fuel assembly grid fingers, any sliding that occurs between the fuel rods and assembly is considered as a type of Coulomb friction. A series of mechanical models of local structures were developed and analyzed so that certain basic nonlinear phenomena previously mentioned could be understood. Using the results of these models, a final eleven-mass model is adopted to represent the internals structure under vertical excitation. The modeling is conducted in such a way that uniform masses are lumped into easily identifiable discrete masses while elastic elements are represented by springs as shown in Figure 14.5.4-1. A legend for the different masses is given in Table 14.5.4-1. The masses are readily recognized as Items W1 through W11. The core barrel and the lower package are easily discernable. The fuel assemblies have been segregated into two groups. The majority of the fuel mass, W4, is indirectly connected to the deep beam structure

TABLE 14.5.4-1

MULTIMASS VIBRATIONAL MODEL-DEFINITION OF SYMBOLS

W1 - Core Barrel
W2 - Lower Package
W3 - Fuel Assemblies, Major
W4 - Fuel Rods, Major
W5 - Fuel Assemblies, Minor
W6 - Fuel Rods, Minor
W7 - Core Plate & Short Column
W8 - Deep Beam
W9 - Core Plate & Long Columns
W10 - Top Plate (Ctr.)
W11 - Core Barrel

K1 - Hold down Spring
K2 - Lower Package, Major
K3 - Top Nozzle Springs, Major
K5 - Top Nozzle Springs, Minor
K7 - Short Columns
K8 - Upper Core Plate
K9 - Long Columns
K10 - Top Plate
K11 - Core Barrel

Snubbers

S1 - Core Barrel Flange
S2 - Hold down Spring
S3 - Top Nozzles Bars, Major
S4 - Pedestal Bars, Major
S5 - Top Nozzles, Bars, Minor
S6 - Pedestal Bars, Minor
S7 - Top Nozzle Bumpers, Major
S8 - Top Nozzle Bumpers, Minor
S9 - Pedestals, Major
S10 - Pedestals, Minor
S11 - Deep Beam Flange

Impact Dampers

D1 - Barrel Flange
D2 - Hold down Spring
D3 - Top Nozzle Bars, Major
D4 - Pedestal Bars, Major
D5 - Top Nozzle Bars, Minor
D6 - Pedestal Bars, Minor
D7 - Top Nozzles, Major
D8 - Top Nozzles, Minor
D9 - Pedestal, Major
D10 - Pedestal, Minor
D11 - Deep Beam Flange

Structural Dampers

C1 - Hold down Spring
C2 - Lower Package
C3 - Top Nozzle, Major
C5 - Top Nozzle, Minor
C7 - Short Columns
C8 - Upper Core Plate
C9 - Long Columns
C10 - Top Plate
C11 - Core Barrel

Clearances

G1 - Hold down Spring
G3 - Fuel Rod Top, Major
G4 - Fuel Rod Bottom, Major
G5 - Fuel Rod Top, Minor
G6 - Fuel Rod Bottom, Minor
G7 - Fuel Assembly, Major
G8 - Fuel Assembly, Minor

Preloads

P1 - Hold down Spring
P3 - Top Nozzle Springs, Major
P5 - Top Nozzle Springs, Minor

represented by mass W8. There is also a portion of the fuel mass, W6, which connects through the long columns to the top plate. The stiffness of the top plate panels is represented by K8. The hold down spring, K1, is bolted-up between the flange of the deep beam structure and the core barrel flange with the preload, P1. After preloading the hold down spring, a clearance, G1, exists between the core barrel flange and the solid height of the hold down spring. Within the fuel assemblies, the fuel elements W4 and W6 are held in place by frictional contact with the grid spring fingers. Coulomb damping is provided in the analysis to represent this frictional restraint.

The analytical model is also provided with viscous terms to represent the structural damping of the elastic elements. The viscous dampers are represented by C1 through C11.

Restrictions are placed on the displacement amplitudes by specifying the free travel available to the dynamic masses. Available displacements are designed by symbols G1 through G8.

The displacements are tested during the solution of the problem to see if the available travel has been achieved. When the limit of travel has been attained, stops are engaged to arrest further motion of the dynamic masses. The stops or snubbers are designed by the symbols S1 through S11.

Contact with the snubbers results in some damping of the motion of the model. The impact damping of the snubbers is represented by the devices D1 through D11.

During the assembly of the reactor, bolt-up of the closure head presets the spring loading of the core barrel and the spring loading on the fuel assemblies. Since the fuel assemblies in the model have been segregated into two groups, two preload values are provided in the analysis. Preload values P1, P3, and P5 represent the hold down spring preload on the core barrel and the top nozzle springs preload values on the fuel assemblies. The formulation of the transient motion response problem and digital computer programming were performed. The effects of an earthquake vertical excitation are also incorporated into the program.

In order to program the multi-mass system, the appropriate spring rates, weights, and forcing function for the various masses were determined. The spring rates and weights of the reactor components are calculated. The forcing functions for the masses are obtained from the FORCE program described in the previous section. It calculates the transient forces on reactor internals during blowdown using transient pressures and fluid velocities.

For the blowdown analysis the forcing functions are applied directly to the various internal masses.

For the earthquake analysis of the reactor internals, the forcing function, which is simulated earthquake response, is applied to the multi-mass system at the ground connections. Therefore, the external excitation is transmitted to the internals through the springs at the ground connections.

Analysis is being performed for variations in rupture opening time, and for hot leg and cold leg breaks. The response of the structure to these excitations indicates that the vertical motion is irregular with peaks of very short duration. The deflections and motion of some of the reactor components are limited by the solid height of springs as is also the hold down spring located above the barrel flange.

The internals behave as a nonlinear system during the vertical oscillations produced by the blowdown forces. The nonlinearities are due to the Coulomb frictional forces between grids and rods, and to gaps between components causing discontinuities in force transmission. The frequency response is consequently a function not only of the exciting frequencies in the system, but also of the amplitude. Different break conditions excite different frequencies in the system. This situation can be understood when the response under blowdown forces is compared with the response due to vertical seismic acceleration. Under seismic excitation, the system behaves almost linearly because component motion is not sufficient to cause closing of the various gaps in the structure or slippage in the fuel rods.

Under hot leg blowdown excitation conditions the core moves upward, touches the core plate, and falls down on the lower structure causing oscillations in all the components. The response shows that the case could be represented as two large vibrating masses (the core and the barrel), with the rest of the system oscillating with respect to the barrel and the core.

Damping effects have also been considered; it appears that the higher frequencies disappear rapidly after each impact or slippage.

The results of the computer program used for solving the transient motion response problem give not only the frequency response of the components, but also the maximum impact force and deflections. From these results, the stresses are computed using the standard "Strength of Material" formulas. The impact stresses are obtained in an analogous manner using the maximum forces seen by the various structures during impact.

Transverse Excitation

The loading from the hydraulic pressure transient on the upper core barrel is represented by a dynamic pressure wave.

The dynamic stability and the maximum distortion of the upper core barrel is analyzed. The response to the initial peak of the pressure wave is obtained neglecting the effect of the water and solid-water interaction in limiting the response of the core barrel.

The upper barrel does not collapse during a hot leg break, and it has an allowable stress distribution during a cold leg break.

The guide tubes are studied applying the blowdown forces to the structures and calculating the resulting deflections. The guide tubes are considered as being elastically supported at the upper plate and simply supported at the lower end with variable cross section. Consideration is given to the frequencies and amplitudes of the forcing function and the response is computed to assure that the deflections do not prevent control rod assembly insertion.

Results of analyses show that the deformation of the guide tubes is within the limits established experimentally to assure control rod assembly insertion.

Allowable Deflection and Stability Criteria

Fuel Assemblies

The limitations for this case are related to the stability of the thimbles in the upper end. The upper end of the thimbles can not experience stresses above the buckling compressive stresses because any buckling of the upper end of the thimbles distorts the guide line and could affect the free fall of the control rod assembly. The buckling stress for the thimbles is 62,300 psi, and the yield stress is 62,500 psi.

Upper Package

The local deformation of the upper core plate where a guide tube is located shall be less than 0.100 in. This deformation causes the plate to contact the guide tube since the clearance between plate and guide tube is 0.1 in. This limit prevents the guide tubes from being put in compression.

For a plate local deformation of 0.150 in. the guide tube is compressed and deformed transversely to the established upper limit and consequently the value of 0.150 in. is adopted as the maximum core plate local deformation, with an allowable of 0.100 in.

Upper Barrel

The upper barrel deformation has the following limits:

a. To assure reactor trip and to avoid disturbing the control rod assembly guide structure, the barrel can not interfere with any guide tubes. This condition requires a stability check to assure that the barrel does not buckle under the accident loads. The minimum distance between guide tube and barrel is 9 in. This value is adopted as the limit above which "no loss of function" can no longer be guaranteed. An allowable deflection of 4.5 in. has been selected.

b. To assure core cooling the outward movement of the upper barrel must be such that the inlet flow from the unbroken cold legs is not impaired. From this condition an outward barrel deflection of 6 in. in front of the inlet nozzle has been established as the "no loss of function" value. An allowable deflection of 3 in. has been selected.

Control Rod Assembly Guide Tubes

The guide tubes in the upper core support package housing control rod assembly required for unit shutdown have the following deflection limit:

> The maximum horizontal deflection of a beam should not exceed 1.75 inch over the length of the guide tube. An allowable distortion of 1.0 in. has been selected.

Allowable Stress Criteria

The allowable stress criteria falls into two categories dependent upon the nature of the stress state; membrane or bending. A direct or membrane state of stress has a uniform stress distribution over the cross section. The allowable (maximum) membrane or direct stress is taken to be equal to the stress corresponding to 20% of the uniform material strain or the yield strength, whichever is higher. For unirradiated type 304 stainless steel at operating temperature the stress corresponding to 20% of the uniform strain, is 39,500 psi. For irradiated type 304 stainless steel, the stress limit is higher.

For a bending state of stress, the strain is linearly distributed over a cross-section. The average strain value is one half of the outer fiber strain where the stress is a maximum. Thus, by requiring the average bending stress to satisfy the allowable criteria for the direct state of stress, the average absolute strain may be 20% of the uniform strain. Consequently, the outer fiber strain may be 40% of the uniform strain. The maximum allowable outer fiber bending stress is then taken to be equal to the stress corresponding to 40% of the uniform strain or the yield strength, whichever is higher. For unirradiated type 304 stainless steel operating temperature, the stress-strain curve gives the maximum stress intensity as 50,000 psi. For irradiated type 304 stainless steel, the stress limit is higher; therefore, it is conservative to use the unirradiated value.

For combinations of membrane and bending stresses, the maximum allowable stress is taken to be equal to the maximum stress corresponding to the strain distribution having the maximum outer fiber strain not in excess of 40% uniform strain and average strain not in excess of 20% uniform strain. Analagous to the uniaxial case, the maximum allowable membrane and total stress intensities for multiaxial stress distributions are 39,500 psi and 50,000 psi.

14.5.4.5 Effects of Loss-of-Coolant and Safety Injection On the Reactor Vessel

The effects of injecting safety injection water into the Reactor Coolant System following a postulated LOCA has been analyzed. WCAP 7304L (W Proprietary) which has been submitted to the AEC, gives a description of the program associated with this analysis. Below is a summary of the conditions that were considered.

For the reactor vessel, three modes of failure are considered including the ductile mode, brittle mode, and fatigue mode.

a) Ductile Mode

The failure criterion used for this evaluation is that there shall be no gross yielding across the vessel wall using the material yield stress specified in Section III of the ASME Boiler and Pressure Vessel Code. The combined pressure and thermal stresses during safety injection through the vessel thickness as a function of time have been calculated and compared to the material yield stress at various times during the safety injection transient.

The results of the analyses showed that local yielding may occur in approximately the inner 12 percent of the base metal and in the cladding.

b) Brittle Mode

The possibility of a brittle fracture of the irradiated core region has been considered from both a transition temperature approach and a fracture mechanics approach.

The failure criterion used for the transition temperature evaluation is that a local flaw cannot propagate beyond any given point where the applied stress remains below the critical propagation stress at the applicable temperature at that point.

The results of the transition temperature analysis showed that the stress-temperature condition in the outer 65 percent of the base metal wall thickness remains in the crack arrest region at all times during the safety injection transient. Therefore, if a defect were present in the most detrimental location and orientation (i.e., a crack on the inside surface and circumferentially directed), it could not propagate any farther than approximately 35 percent of the wall thickness, even considering the worst case assumptions used in this analysis.

Both a local crack effect and a continuous crack effect have been considered with the latter requiring the use of a rigorous finite element axisymmetric code. The results of the fracture mechanics analysis, considering the effects of water temperature, heat transfer coefficients and fracture toughness of the material as a function of time, temperature and irradiation show that the integrity of the reactor vessel is maintained throughout the life of the unit.

c) Fatigue Mode

The failure criterion used for the failure analysis is the one presented in Section III of the ASME Boiler and Pressure Vessel Code. In this method the piece is assumed to fail once the combined usage factor at the most critical location for all transients applied to the vessel exceed the code allowance usage factor of one.

The results of this analysis show that the combined usage factor never exceeds 0.2, even after assuming that the safety injection transient occurs at the end of unit life.

In order to cause a fatigue failure during the safety injection transient at the end of unit life, it has been estimated that a wall temperature of approximately 1100°F is needed at the most critical area of the vessel (instrumentation tube welds in the bottom head).

The design basis of the Emergency Core Cooling System ensures that the maximum Zircaloy cladding temperature does not exceed the Zircaloy-4 melt temperature. This is achieved by prompt recovery of the core through flooding, with the passive accumulator and the active injection systems. Under these conditions, a vessel temperature of 1100°F is not considered a credible possibility and the evaluation of the vessel under such-elevated temperatures is a hypothetical case.

For the ductile failure mode, such hypothetical rise in the wall temperature would increase the depth of local yielding in the vessel wall.

The results of these analyses show that the integrity of the reactor vessel is never violated.

The safety injection nozzles have been designed to withstand ten postulated safety injection transients without failure. This design and associated analytical evaluation were made in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code.

The maximum calculated pressure plus thermal stress in the safety injection nozzle during the safety injection transient was calculated to be approximately 50,900 psi. This value compares favorably with the code allowable stress of 80,000 psi.

These ten safety injection transients are considered along with all the other design transients for the vessel in the fatigue analyses of the nozzles. This analysis shows the estimated usage factor for the safety injection nozzles to be 0.47 which is well below the code allowable value of 1.0.

The safety injection nozzles are not in the highly irradiated region of the vessel and thus they are considered ductile during the safety injection transient.

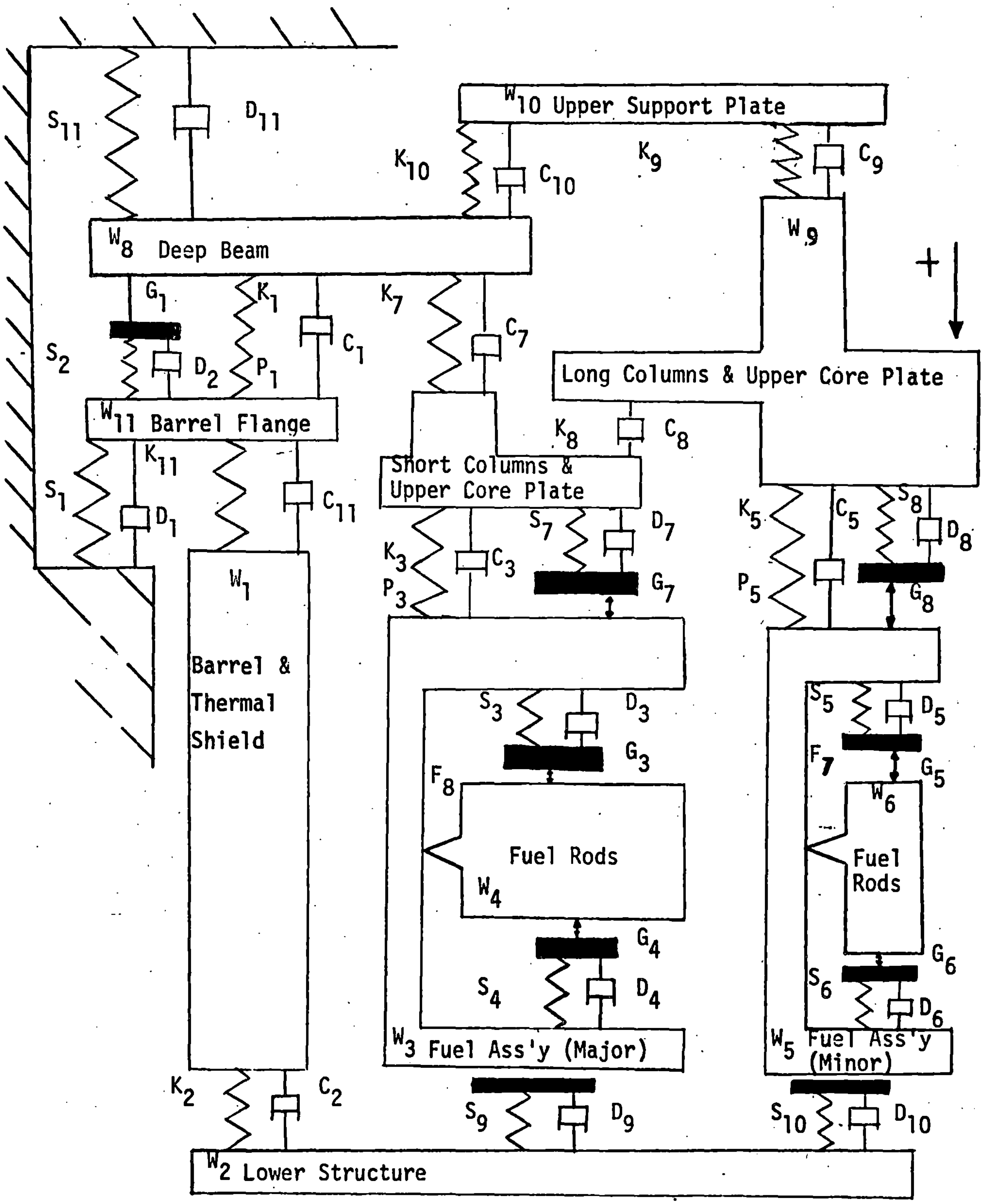
The effect of the safety injection water on the fuel assembly grid springs has been evaluated and due to the fact that the springs have a large surface area to volume ratio, and are in the form of thin strips, they are expected to follow the coolant temperature transient with very little lag; hence, no thermal shock is expected and the core cooling is not compromised.

Evaluations of the core barrel and thermal shield have also shown that core cooling is not jeopardized under the postulated accident conditions.

REFERENCES

(1) S. Fabic: "Computer Program WHAM for Calculation of Pressure, Velocity, and Force Transients in Liquid Filled Piping Networks," Kaiser Engineers Report No. 67-49-R (November 1967).

Figure 14.5.4-1
12-1-69



MULTI-MASS VIBRATIONAL MODEL

LOFT SEMISCALE VESSEL BLOWDOWN RUN #522

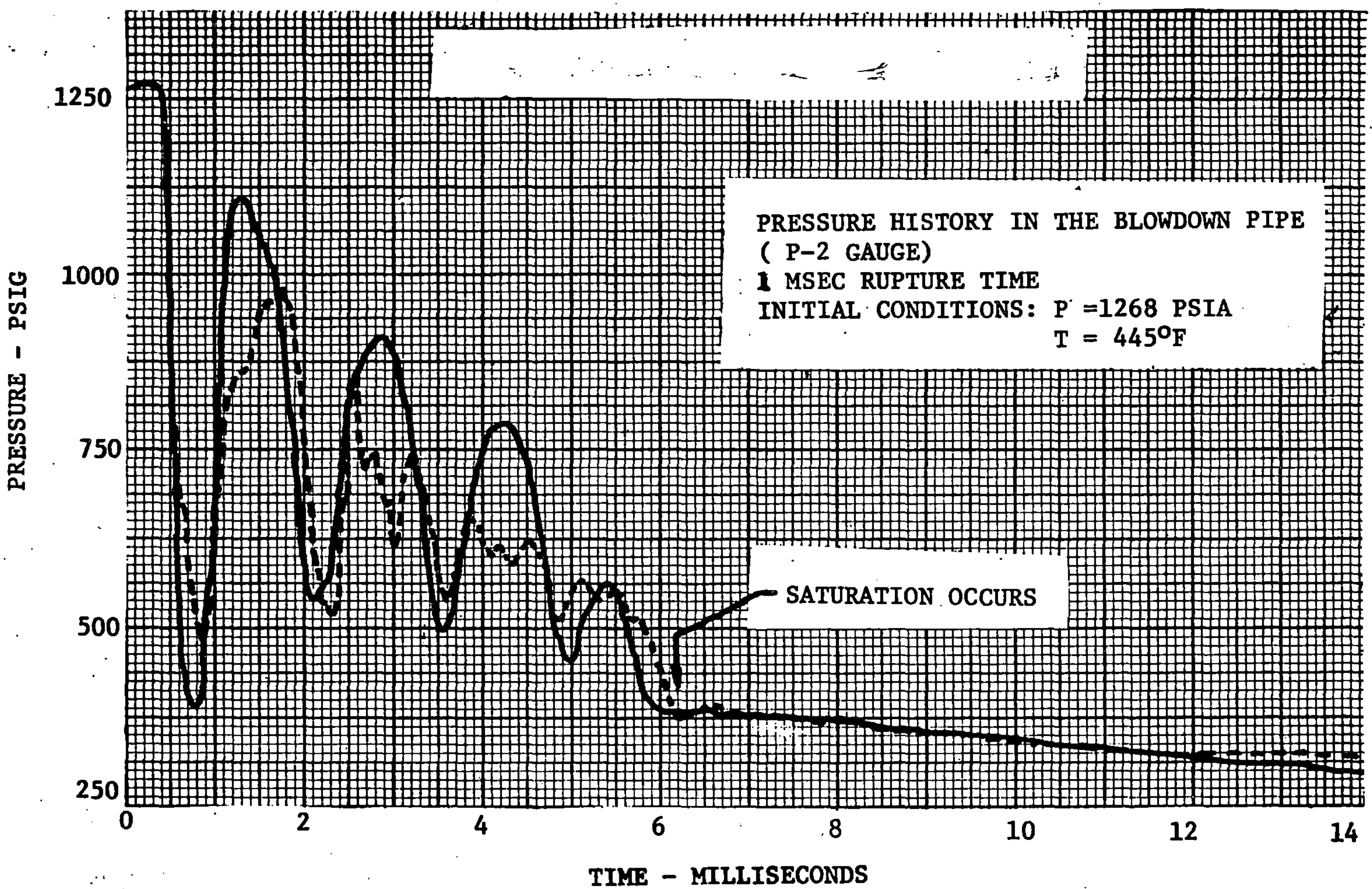


Figure 14.5.4-2
12-1-69

14.5.5 CONTAINMENT TRANSIENT ANALYSIS

Fig. 14.5.5-1 illustrates the containment pressure transient for a 29 in. double-ended rupture of a reactor coolant pipe (hot leg) with normal safeguards in operation as follows:

1. Two charging pumps become effective 30 sec after the start of the accident.

2. Two low head safety injection pumps become effective 30 sec after the start of the accident.

3. Two nitrogen-pressurized accumulators discharge to the Reactor Coolant System when Reactor Coolant System pressure drops below accumulator pressure. The third accumulator discharges to the floor.

4. Two containment spray pumps become effective 30 sec after receipt of the containment spray signal. The containment spray signal will be initiated when the containment pressure reaches approximately 24.7 psia.

5. Two recirculation spray pumps become effective 2 min after the containment spray pumps become effective. The remaining two recirculation spray pumps become effective 5 min after the containment spray pumps become effective.

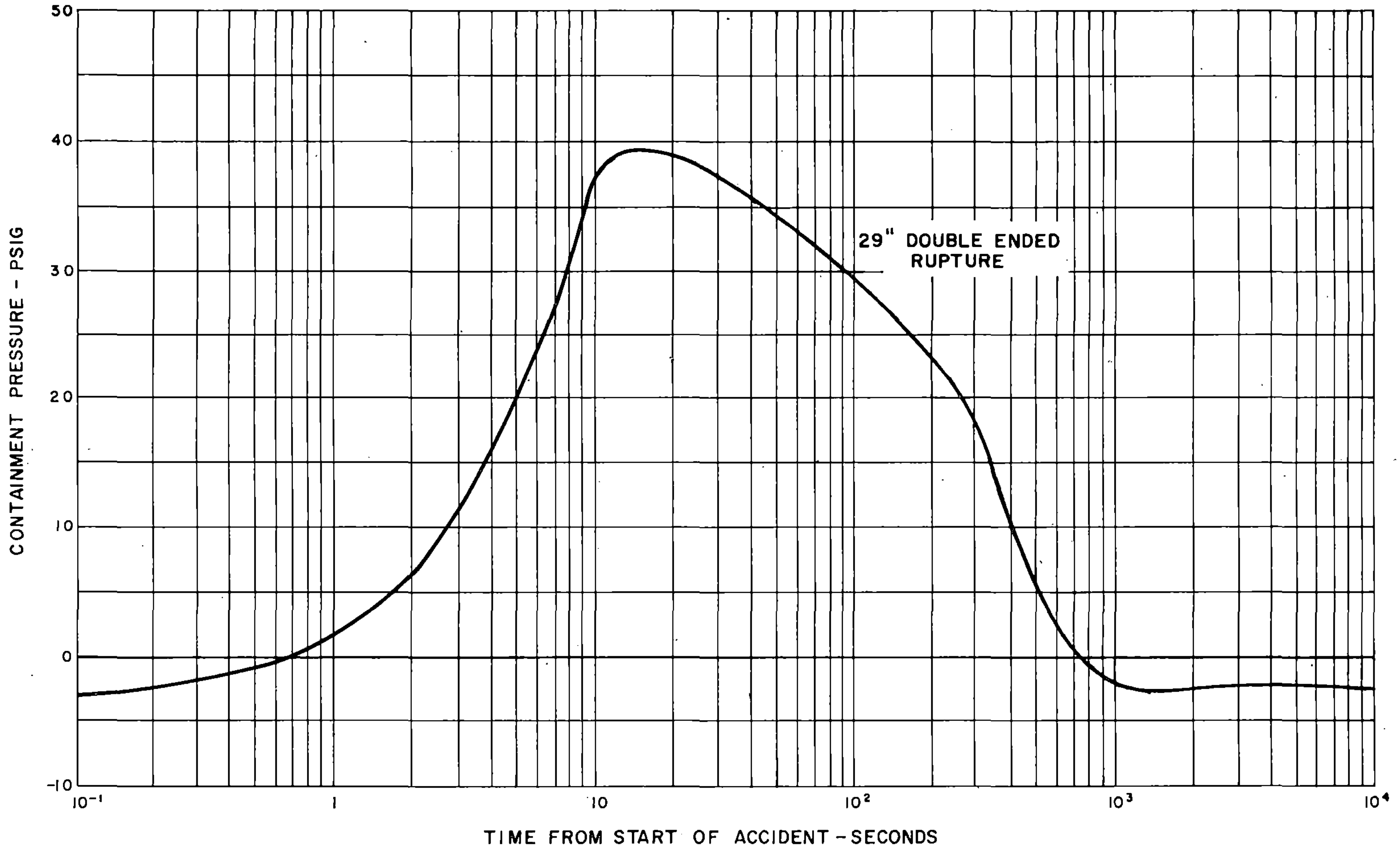
The 29 in. reactor coolant pipe double-ended rupture results in the highest containment pressure. For this break, the primary system blowdown is complete and the initial containment peak pressure is reached within 15 sec. For break sizes other than the 29 in. double-ended rupture, blowdown time is extended, with more heat absorbed by the steel and concrete in the containment and by the Consequence Limiting Safeguards before completion of blowdown.

The 29 in. double-ended rupture results in a maximum peak containment pressure of approximately 39.2 psig, which is well below the containment design pressure of 45 psig. Following onset of this accident, the containment is depressurized to subatmospheric condition in 13 min, which is significantly less than the design criterion of 30 min.

Annual variation of ambient conditions changes the containment initial conditions. However, postaccident containment pressure transients still remain within the above criterion.

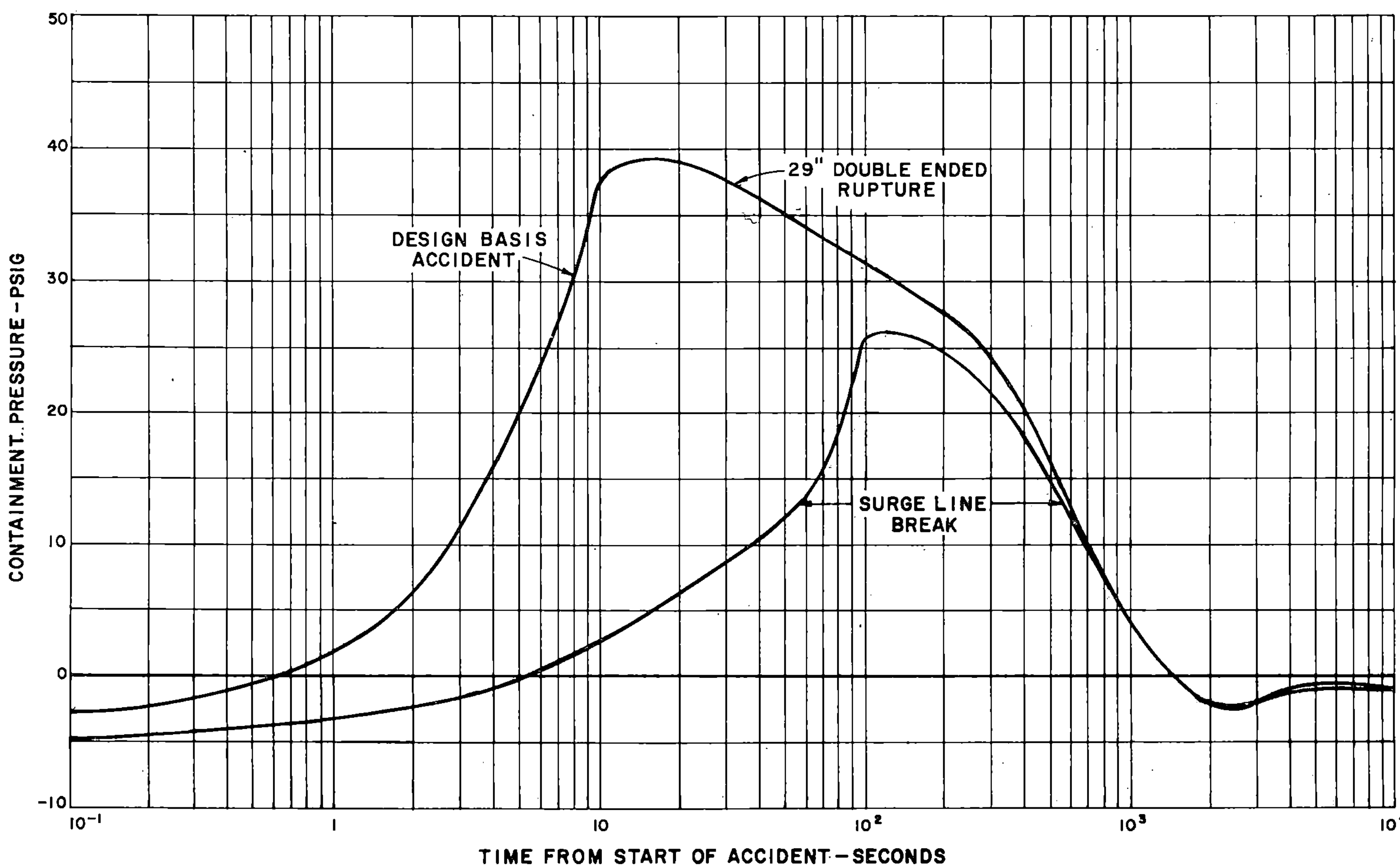
Fig. 14.5.5-2 shows the containment pressure transients resulting from a 29 in. double-ended rupture or a pressurizer surge line rupture with the operation of "Minimum Safeguards," as defined in Section 6.

The 29 in. double-ended rupture with "minimum safeguards," shown in Fig. 14.5.5-2, is the DBA described in Section 14.5.2. The peak containment pressure after the accident is essentially the same as that shown in Fig. 14.5.5-1. The operation of "Minimum Safeguards," Fig. 14.5.5-2, results in higher subsequent containment pressures and temperatures; however, the maximum calculated pressure of 39.2 psig in both cases is still within the design criteria of 45 psig. For the double-ended rupture with minimum safeguards, containment depressurization time is extended to 24 min, but is still within the design criterion of 30 min.



LOSS OF COOLANT ACCIDENT
NORMAL SAFEGUARDS
SURRY POWER STATION

FIG. 14.5.5-1
2-16-70



LOSS OF COOLANT ACCIDENT
MINIMUM SAFEGUARDS
SURRY POWER STATION

FIG. 14.5.5-2
2-16-70

14.5.6 CONTAINMENT IODINE REMOVAL BY SPRAY SYSTEM

The Containment Depressurization System (Section 6.3.1) is designed to reduce post accident containment pressure by condensing steam and to absorb iodine present in the containment atmosphere in an inorganic vapor form (I_2 and HI) with chemical spray. The rate at which elemental iodine can be removed from the containment atmosphere by a reactive chemical spray may be calculated using Griffith's method (1) or by using Parsly's computerized method (2). Both methods are based on the experimental work of Taylor (3) who showed that the overall mass transfer rate at which elemental iodine is transferred into reactive solutions is controlled by the gas film resistance of the vapors surrounding the spray drops. The methods are also based on the experimental work of Ranz and Marshall (4) (5) who developed a correlation for calculating the mass transfer coefficient when the rate of transfer is controlled by the gas film resistance.

The fundamental assumption in using an alkaline solution as a reactive spray to remove iodine vapor from the containment is illustrated in the following equations:

$$OH^- + I_2 \rightleftarrows I^- + HOI$$

$$3\ HOI \rightleftarrows 2\ I^- + 3\ H^+ + IO_3^-$$

We consider two different cases. In the first, we assume that the mass transfer is gas-film limited, that there is no resistence to mass transfer of iodine within the liquid drop because chemical reactions prevent the existence of elemental iodine in the drop. As soon as I_2 has undergone reaction it can no longer exert a partial pressure, thereby changing the partition very much in favor of the liquid phase and increasing the mass transfer rate. Experimental evidence from ORNL and BNWL tests (6)(7)(9)(10) confirm these theoretical assumptions for NaOH-borated water solutions.

In the second case considered, we assume that there is resistance to mass transfer both in the gas film immediately outside the drop and in the liquid immediately beneath the surface of the drop. In the gas-film-limited case (Case I), the mass flux is expressed in terms of a gas deposition velocity, whereas in Case II an overall deposition velocity is used. All other factors in the expression (liquid flow-rate, header height, containment volume, drop size, and terminal velocity) are independent of whether the process is gas-film or gas-and-liquid-film controlled.

For the analysis of the offsite dose effects presented in Section 14.5.7, HI is conservatively assumed to be absorbed at the same rate as elemental iodine. It is also assumed in Section 14.5.7 that 10 percent of the iodine in the containment atmosphere is in the form of CH_3I and/or aerosols and that the sprays are completely ineffective in removing these organic iodides and/or aerosols.

The derivations which follow are indicative of the correlations used in the stipulated references and are presented here to show the methods used for computing the iodine removal coefficients.

In the gas-film-controlled case, the dependence of mass transfer rate on film conditions is expressed by:

$$V_g = \frac{D_v}{d} \left(2 + 0.6\ Re^{1/2} Sc^{1/3}\right), \text{ cm/sec} \qquad (1)$$

(see Table 14.5.6-2, for nomenclature)

Equation (1) is well substantiated by experiments in a variety of systems in which the gas film is controlling, as reported by Ranz and Marshall (4)(5).

The surface area of drops (S_d) available for iodine absorption can be calculated from equation (2) which is based on the conservative assumption that all the drops are spherical and can be expressed by the same diameter:

$$S_d = \frac{A_d\, F\bar{t}}{V_d} = \frac{\pi d^2}{\frac{\pi d^3}{6}} \frac{Fh}{v} = \frac{6}{d} \frac{Fh}{v}, \quad cm^2 \qquad (2)$$

(see Table 14.5.6-2 for nomenclature)

The containment atmosphere is assumed to be well mixed and all the drops are assumed to contain an excess of chemical reagent to react with the iodine and convert it to a nonvolatile form. Based on the foregoing, the rate of removal of elemental iodine from the containment atmosphere can be calculated on the basis of an exponential removal as the spray passes through the containment following the relationship:

$$C_t = C_o \exp(-\lambda t) \qquad (3)$$

(see Table 14.5.6-2 for nomenclature)

The iodine removal coefficient, λ, is calculated by the relationship:

$$\lambda = \frac{V_g\, S_d}{V_c} \quad sec^{-1} \qquad (4)$$

for the gas-film-controlled case.

For the gas-and-liquid-film-controlled case, the gas film deposition velocity V_g is replaced by the overall deposition velocity V_D defined by the relationship:

$$\frac{1}{V_D} = \frac{1}{V_g} - \frac{1}{K_L\, H} \qquad (5)$$

where

$$K_L = \frac{2\pi^2}{3} \quad \frac{D_L}{d} \qquad (6)$$

Combining equation (4) and (2) :

$$\lambda = \frac{6\ Fh\ V_g}{v\ d\ V_c} \quad sec^{-1} \qquad (7)$$

For the conservative case, V_g is replaced by V_D and Eqn 7 becomes:

$$\lambda = \frac{6\ Fh\ V_D}{v\ d\ V_c} \qquad (8)$$

(see Table 14.5.6-2 for above nomenclature)

The iodine removal coefficient (λ) can be evaluated for parameters which represent the design conditions for the Containment Spray System. In this system, caustic is metered from a tank into the flow to the containment spray pumps in proportion to the total flow from the refueling water storage tank so that the pH of the borated water in the containment is at 8.2 when the refueling water storage tank and caustic storage tank are emptied.

The refueling water storage tank initially has a capacity of 350,000 gal of borated water at a concentration of 2,500 ppm boron (1.428 percent H_3BO_3, 0.23 N, pH 4.75). The flow from one or two containment spray pumps is discharged through the top spray headers, which are 96 ft above the operating floor, into the containment at an initial pH of 9 to 12, depending on the combination of pumps operating, at a rate of approximately 2,600 or 5,200 gpm. In this analysis, only one pump discharging at about 2,600 gpm is considered to be in operation. The atomized spray droplets from the containment spray header have a mean surface diameter of 858 microns.

In addition to the containment sprays, the recirculation spray headers are located an average of 47 ft above the operating floor. One or more recirculation spray pumps, rated at 3,500 gpm each, take suction from the containment sump and pump water through the recirculation spray headers. The borated water in the sump rises from a pH of approximately 7.6, 5 min after flow is initiated, to a pH of 8.2 when the refueling water storage tank is empty. The atomized spray droplets from the recirculation spray headers have a mean surface diameter of 1,082 microns.

For purposes of averaging conditions and of calculating iodine removal coefficients, we divide the time post DBA into three segments:

1. 32-150 seconds, when one containment spray (caustic) is active. No sprays are operating at times less than 32 seconds.
2. 150-330 seconds, one containment spray (caustic) and one recirculation spray (acid) are operating during this period.
3. 330 seconds until the end of the time considered for leakage (40 min), one containment spray (caustic) and two recirculation sprays (caustic) are operating during this period.

For each time period, properties are evaluated at a temperature slightly above the average temperature of that time period. This is conservative because an increase in temperature decreases V_g/v.

For caustic sprays, i.e., the containment spray and the recirculation sprays when $t>330$ sec, $H = 3,000$ (caustic)
is used for the liquid-to-gas partition factor, as used in the Diablo Canyon 2 analysis (B). For an acid spray, i.e., the recirculation spray $150<t\leq 330$ sec, it is conservatively assumed that $H = 1,500$ (acid),

even though recent experiments at ORNL (9) and Rattelle-Northwest (10) indicate that both low pH and high pH sprays are effective in iodine removal.

For each type of spray header (containment and recirculation) we calculate the mean surface diameter of the drop size distribution of the entire header, which in turn is the flow-weighted sum of the drop distributions of the individual nozzles.

Case I calculations make use of equations (1) and (7). Case II calculations make use of equations (1), (5), (6), and (8). These calculations are summarized by Table 14.5.6-1. λ_t is the sum of the coefficients due to the containment spray and the recirculation spray; λ_s is λ_t divided by 3.8 (λ_t is used to conservatively counteract uncertainties as detailed in reference (8)).

In actual tests run to date at ORNL and BNWL (7, 9, 10), the experimental half-lives for iodine removal were seldom found to exceed 1 min for the variety of spray solutions tested. The spray solutions tested included NaOH - H_3BO_3 spray solutions.

The experience of chemical processors who routinely employ caustic solutions provides guidance in the design of the caustic storage and handling equipment. This experience indicated no foreseeable problems in solution stability and material compatibility for the time and conditions of exposure following a loss-of-coolant accident.

TABLE 14.5.6-1

Summary of Calculations for

Iodine Removal Coefficients

SURRY POWER STATION

Parameter	Units	First Period		Second Period		Third Period	
Time	sec	32-150		150-330		330-end	
T	°C	130		120		90	
μ_v	gm/cm-sec	0.000160		0.000161		0.000169	
Iw	cm^2/sec	0.04186		0.05051		0.08511	
Se		1.7315		1.7384		1.746	
ρ_v	gm/cm^3	0.00221		0.001835		0.001136	
ρ_1	gm/cm^3	0.9345		0.9433		0.9665	
		Cont.	Recirc.	Cont.	Recirc.	Cont.	Recirc.
d	microns	858		858	1082	858	1082
F	cm^3/sec	1.50×10^5	0	1.55×10^5	2.21×10^5	1.68×10^5	4.41×10^5
h	cm	2930		2930	1433	2930	1423
V_g	cm/sec	7.21		8.77	7.89	1206	11.49
v	cm/sec	265		285	347	342	419
Ve	cm^3	5.1×10^{10}		5.1×10^{10}	5.1×10^5	5.1×10^{10}	5.1×10^{10}
Case) λ	sec^{-1}	0.0164	0	0.0181	0.0133	0.0238	0.0188
I) λ_t	sec^{-1}	0.0164		0.0259		0.0628	
) λ_s*	hr^{-1}	15.53		24.55		59.49	
D_L	cm^2/sec	7.414×10^{-5}		6.606×10^{-5}		4.472×10^{-5}	
		Cont.	Recirc.	Cont.	Recirc.	Cont.	Recirc.
K_L	cm/sec	0.00569		0.00507	0.00402	0.00343	0.00272
H		3000		3000	1500	3000	3000
V_D	cm/sec	5.069		5.36	3.42	5.55	4.71
Case) λ	sec^{-1}	0.0115	0	0.0117	0.0034	0.0110	0.0078
II) λ_t	sec^{-1}	0.0115		0.0151		0.0188	
) λ_s*	sec^{-1}	10.92		14.31		17.81	

* $\lambda_s = (\lambda_t)$ (3600 sec/hr)/3.8

TABLE 14.5.6-2 NOMENCLETURE

A_d	surface area per drop, cm^2
C_o	amount of iodine in the initial containment atmosphere, curies
C_t	amount of iodine in the containment at time t, curies
d	mean surface diameter, microns or cm
D_L	diffusivity of iodine in water, cm^2/sec
D_v	diffusivity of iodine in steam-air mixture, cm^2/sec
F	spray flow rate, cm^3/ sec
h	header height, cm
H	liquid-to-gas iodine partition factor
K_L	liquid film mass transfer coefficient, cm/sec
λ	iodine removal coefficient, sec^{-1}
λ_s	conservative total iodine removal coefficient, $hr^{-1} = \frac{\lambda_T x3600}{3.8}$
λ_T	total iodine removal coefficient, sec
μ_v	viscosity of steam-air mixture, gm/cm/sec
R_e	Reynolds number $\left(\frac{\rho_v V d}{\mu_v}\right)$
ρ_L	Density of water at temperature T, gm/cm^3
ρ_v	density of steam-air mixture, gm/cm^3
S_c	Schmidt number $\left(\frac{\mu_v}{\rho_v D_v}\right)$
S_d	total surface area of the spray drops in the containment atmosphere, cm^2
t	time, sec
$\bar{t}$	average residence time of the drop in the atmosphere, sec
T	average temperature of steam-air mixture during time period, ^oC
T	terminal velocity of drop, cm/sec
V_c	containment volume, cm^3
V_d	volume per drop, cm^3

V_0	overall deposition velocity, cm/sec
V_g	gas film deposition velocity, cm/sec
v	terminal velocity of the drop, cm/sec

BIBLIOGRAPHY

1. V. Griffiths, "The Removal of Iodine from the Atmosphere by Sprays," UKAEA Report AHSB (S) R 45 (1963)

2. L. F. Parsly, Jr., "Removal of Elemental Iodine from Steam-Air Atmospheres by Reactive Sprays," USAEC Report ORNL-TM-1911 (1967)

3. R. F. Taylor, "Absorption of Iodine Vapors by Aqueous Solutions," Chem. Engng. Sci.10 (1959) (68-79)

4. W. E. Ranz, W. R. Marshall, Jr., "Evaporation from Drops, Part I," Chem. Engng. Prog. 48 3 (1952) (141-146)

5. W. E. Ranz, W. R. Marshall, Jr., "Evaporation from Drops, Part II, "Chem. Engng. Prog. 48 4 (1952) (173-180)

6. L. F. Parsly, Jr., J. K. Franzreb, "Removal of Iodine Vapor from Air and Steam-Air Atmospheres in the Nuclear Safety Pilot Plant by Use of Sprays," USAEC Report ORNL-4253 (1968)

7. T. H. Row, "Spray and Pool Absorption Technology Program," USAEC Report ORNL-4360, (April 1969) (29)

8. Effectiveness of Spray System for Iodine Cleanup, AEC/IRL Staff Evaluation for Diablo Canyon Unit 2

9. Wm. B. Cottrell, "ORNL Nuclear Safety Research and Development Program Bimonthly Report for January - February 1970", USAEC Report ORNL-TM-2919, (May, 1970) (73)

10. "Nuclear Safety Quarterly Report, February, March, April, 1970," BNWL-1315-2, May, 1970

14.5.7 OFFSITE EFFECTS

The Design Basis Accident is a hypothetical accident assumption developed to provide a design basis for a nuclear power station. The DBA is in all probability an incredible event. For analytical purposes, a large number of substantially conservative assumptions are made to produce results not readily subject to contest. The exaggerated results, therefore, constitute conditions far in excess of those which could credibly be expected to occur. Therefore, the results of a DBA should not be associated with situations of credible probability.

The Design Basis Accident is defined as the circumferential double-ended failure of a reactor coolant pipe, the total loss-of-coolant through such double-ended failure, a total loss of station power, the availability of only "minimum safeguards", and the release of the following core fission product inventory to the reactor containment:

100 percent of the noble gases
50 percent of the halogens
1 percent of the remaining fission products

The evaluations of the clad thermal transient in Section 14.5.3 show that some of the clad would not be ruptured as a result of the Design Basis Accident. Thus, a conservative evaluation could be performed using the fuel rod gap activities listed in Appendix 14A. However, the calculations presented in this section are for the even more conservative core meltdown case suggested by TID-14844. The estimated actual potential doses resulting from a gap release would be approximately 2 percent of those reported herein.

The temperature and pressure transient aspects of this accident are discussed and analyzed under Loss-of-Coolant Accident, Section 14.5.5. This section describes the method and results of radiological analyses for this accident. These analyses are divided into two parts: thyroid dose from inhaling the iodines in the containment leakage plume, and whole body exposure as a result of direct radiation from both the containment and immersion in the resultant pulse of gas leakage.

In each analysis, doses are calculated at the exclusion boundary, at the nearest occupied private residence, and at the low population zone boundary.

14.5.7.1 Thyroid Dose

Integrated thyroid doses were calculated using methods and assumptions similar to those given in TID-14844, except for atmospheric dispersion. The assumptions used include:

1. Prior to the incident, the reactor was operating at its maximum normal rating of 2,546 MWt, with the core fission product inventory at equilibrium.

2. 50 percent of the iodines released from the Reactor Coolant System into the containment atmosphere plate-out, or are otherwise adsorbed on surfaces and structures inside the containment.

3. Ten percent of the iodine is assumed to be present in the form of organic iodines and/or aerosols. No credit is taken for removal by the sprays for this fraction. Containment spray and iodine interaction is as detailed in Case II, FSAR section 14.5.6.

4. During the phase of the accident in which the containment is above atmospheric pressure, the leak rate from the containment is constant at a value of 0.1 percent of the contained volume per day.

5. Upon reduction of the containment to subatmospheric pressure (the lowest barometric pressure of record), all leakage from the containment ceases.

6. All containment leakage occurs at ground level.

7. Dispersion of the radioactive material occurs under Pasquill Type "F" meteorology conditions with a one meter per second average wind speed.

8. The volume source correction for building wake effects, "C" equals 0.5.

Using aforementioned assumptions (7) and (8), and the equation from Table 2.2-10, FSAR section 2.2.3, values of χ/Q were calculated at the locations specified above.

The following points regarding conservatism of assumptions (3), (4), (5), and (6) should be noted:

Assumption (3): An analysis of the behavior of a chemical spray as a function of the pertinent parameters which are expected in the containment atmosphere following the DBA is described in Section 14.5.6. Since it is unlikely that as much as 10 percent of the iodine could be converted to the organic form in the 40 min depressurization time and there is considerable question as to the amount of aerosols that would be present and unaffected by sprays, the assumption of 10 percent nonremovable radioiodine is conservative.

Assumption (4): Since the dose is directly proportional to the containment leak rate and since the driving force for containment leakage is the differential pressure across the containment shell, it is conservative to assume that the leakage rate remains constant throughout the accident at its maximum design value.

Assumption (5): In this analysis, it is assumed for conservatism, that it takes 40 min to reduce the containment pressure to subatmospheric. Calculations show that with minimum Engineered Safeguards Systems operating, it would take less than 30 min to accomplish this depresurization.

Assumption (6): This assumption results in maximum ground level concentrations and maximum integrated thyroid doses at all points downwind of the containment.

The results of the analysis for the thyroid dose are presented in Table 14.5.7-1 and Figs. 14.5.7-1 and 14.5.7-2. Fig. 14.5.7-1 shows integrated thyroid dose at the exclusion boundary, or the point of highest integrated dose to the public, as a function of time after initiation of the accident. Fig. 14.5.7-2 shows integrated thyroid dose as a function of downwind distance beyond the exclusion boundary. The assumption that containment leakage stops after 40 min results in the integrated thyroid dose at the exclusion boundary reaching a maximum value of 71.7 rem at 40 min and remaining constant for the duration of the accident, which is assumed to be 31 days in the calculation of direct dose from the containment in the next subsection.

Inspection of Table 14.5.7-1 shows that the integrated thyroid doses to the public are well within the 300 rem criterion suggested in 10CFR100[2]. For example, in the context of a maximum theoretical postulated incident the integrated thyroid dose at the nearest private occupied residence, 8,980 ft from the nearest containment structure, would be 11.2 rem, while the dose at the low population boundary, 3 miles from the nearest containment structure, would be 5.2 rem.

14.5.7.2 Whole Body Dose

The whole body dose consists of both the external radiation contribution from the radioactive pulse plume leaking from the containment and the 31-day direct radiation dose from activity inside the containment.

Integrated whole body doses caused by external radiation from passage of the leakage pulse plume were calculated, using the basic TID-14844 method for source data. A conservative model was used, i.e., the plume was assumed to be a semi-infinite medium with an activity concentration equal to that calculated for the plume center line at the particular distance involved. The eight assumptions listed on pages 14.5.7-2 and 14.5.7-3, which were used in the preceding calculation of the thyroid dose, were also used in this calculation.

The integrated whole body doses from the radioactivity contained within the containment structure were calculated, based upon the radiation source data plotted in Fig. 14.5.7-4, which are conservative, since no allowance was made for clean-up.

Except at the exclusion boundary, the whole body dose from direct radiation within the containment is negligible at the other distances of interest in comparison with the whole body dose from the plume. At the exclusion boundary the maximum theoretical postulated total whole body dose 2 hr after the accident is 3.76 rem, of which 3.61 rem is from the plume and 0.15 rem is from the containment; and 31 days after the accident, is 4.04 rem, of which 3.61 rem is from the plume and 0.43 rem is from the containment.

Fig. 14.5.7-3 shows the integrated whole body dose from the radioactive pulse plume and the containment as a function of distance beyond the exclusion boundary, and Table 14.5.7-1 gives the integrated whole body doses at the three distances of interest. As can be seen from the table, all of the doses are well below the 25 rem criterion suggested in 10CFR100.

14.5.7.3 Expected Exclusion Boundary Dose

At each location of interest, a comparison of the integrated whole body dose, which is caused by direct radiation from the containment plus exposure to the leakage plume, with integrated thyroid dose indicates that the thyroid dose is the limiting consideration. Furthermore, an examination of Table 14.5.7-1 shows that the highest thyroid dose occurs at the exclusion boundary. With 40 min of constant leakage this value is 71.7 rem, which is 24 percent of the 300 rem criterion suggested in 10CFR100. The design basis for the Engineered Safeguards System is to depressurize the containment in 30 min. Assuming constant leakage, this would give a thyroid dose at the exclusion boundary of 63.2 rem.

Subatmospheric containment differs from other dry reactor containments in that there is considerable time during the depressurization phase of the accident when the pressure difference between the containment atmosphere and the outside atmosphere is so small that little leakage occurs. Considering this, a variable leak rate analysis was made.

The variable leak rate is based upon a model in which it is assumed that leakage occurs by orifice flow and is constant at 0.1 percent of the contained volume per day, until the containment pressure falls below the critical pressure, 10.08 psig; at this time, the leak rate is assumed to decrease with decreasing differential pressure across the containment shell. Also, a small correction is made for the density variations of the containment atmosphere.

The pressure transient used in this analysis was generated by the LOCTIC computer code assuming that minimum Engineered Safeguards operate, and is shown in

Fig. 14.5.5-2. Leakage terminates when the containment pressure reaches a value equal to the lowest barometric pressure anticipated at the site. For this analysis, the atmospheric pressure was assumed to be 28.35 in. Hg.

Fig. 14.5.7-5 compares the integrated thyroid dose at the exclusion boundary for the variable vs. constant leakage assumptions. As can be seen from the figure, with variable leak rate the integrated thyroid dose at the exclusion boundary will be 55.6 rem. This is 21 percent less than the dose calculated by assuming constant leak rate and duration of the accident equal to 40 min.

REFERENCES

1. T.I.D. 14844, J. J. DiNunno, et al, Calculation of Distance Factors for Power and Test Reactor Sites, United States Atomic Energy Commission, March, 1962

2. Chapter 10, Code of Federal Regulations, Part 100 - Reactor Site Criteria

TABLE 14.5.7-1

INTEGRATED THYROID AND EXTERNAL RADIATION DOSES
FOLLOWING DESIGN BASIS ACCIDENT, BASED ON
LEAKAGE RATE OF 0.1% OF CONTAINED VOLUME PER DAY

Location	Distance	Exposure Time	Dose rem
	Integrated Thyroid Doses		
Exclusion Boundary	1,650 ft	40 min	71.7
Nearest Occupied Residence	1.7 mi	40 min	11.7
Low Population Zone Boundary	3.0 mi	40 min	5.2
	Integrated Whole Body Doses		
Exclusion Boundary	1,650 ft	2 hr(1)	3.76
Nearest Occupied Residence	1.7 mi	31 days(1)	0.53
Low Population Zone Boundary	3.0 mi	31 days(1)	0.27

(1) Exposure consists of exposure to integrated leakage plume plus time stated for exposure to radioactivity contained within the containment. Exposure times are stated this way to permit comparison with 10CFR100.

14.5.8 SUMMARY

For breaks up to and including the double-ended severance of a reactor coolant pipe, the Emergency Core Cooling System with partial effectiveness ("Minimum Safeguards") will limit the clad temperature to below the melting temperature of Zircaloy-4 and ensure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved. The Emergency Core Cooling System design meets the core cooling criteria with substantial margin for all cases. It was also concluded from this study that two high head safety injection pumps are capable of maintaining core mid-plane flooding for all break sizes up to approximately a 4 in. equivalent diameter break. For larger breaks the required protection is supplied by the accumulators.

The design of the fuel assemblies and the core support structures has been such that the pressure oscillations and flow transients resulting from any loss-of-coolant accident can be accommodated without changes which would affect the capability of the Safety Injection System to perform its required function.

The containment structure will be capable of containing, without loss of integrity, any equipment failure in the Reactor Coolant System which could result in an undue hazard to the public. The Containment Spray System will remove heat and airborne fission products from the containment atmosphere and will return the containment to a subatmospheric condition, thus terminating leakage to the environment. The Recirculation Spray Subsystem will transfer the heat from the containment to the Service Water System, thereby removing residual heat and will maintain the containment in a subatmospheric condition during the subsequent recovery period.

The radiation dose at the exclusion boundary for the DBA was evaluated based on a constant leakage of 0.1 percent of the containment volume per day, a leakage time of 40 min, the assumptions and methods as stipulated in TID-14844, and other conservative bases given in Section 14.5.7. The integrated thyroid dose at the exclusion boundary is 71.7 rem which is 24 percent of the 300 rem criterion suggested by 10CFR100.

The integrated whole body dose for both the leakage plume and direct radiation from the containment is at the exclusion boundary 3.76 rem after 2 hr and 4.04 rem after 31 days. Since the containment depressurizes in less than 30 min and the assumed constant containment leak rate of 0.1 percent of containment volume per day is equivalent to a constant containment pressure of 45 psig during leakage, the dose stipulated above are considered to be higher than would reasonably be expected.

The maximum offsite dose resulting from a fuel handling accident in the spent fuel pit results in an integrated whole body dose from the "puff" release of about 0.20 rem and an integrated thyroid does (using TID 14884 method) of 4.2 rem at the site boundary. No offsite dose results if the fuel accident occurs in the refueling cavity inside the containment.

The maximum offsite dose resulting from the failure of a steam generator tube results in an integrated whole body dose of about 0.30 rem and a thyroid dose of about 0.28 rem at the site boundary.

It is concluded that neither a fuel handling accident nor a steam generator tube rupture result in excessive radiation exposure at the site boundary.

The offsite dose resulting from the rupture of a main steam pipe is dependent upon the iodine activity in the reactor coolant, the primary to secondary leak rate, the steam generator blowdown rate, and the dilution available for blowdown in the discharge canal and the restrictions of 10CFR20. Parametric equations were derived to permit setting operation limits in the Technical Specification which will ensure that the thyroid dose at the boundary will be less than approximately 13.6 rem if a main steam line ruptures. (Refer to FSAR 14.3.2.5).

APPENDIX 14A

RADIATION SOURCES

This appendix presents the quantities of radioactive isotopes present in the core, and the fuel rod gap. A general discussion of the derivations is also provided.

Total Activity in the Core

The total core activity calculation is consistent with TID 14844 and data from ORNL-2127.* Numerical values for certain significant isotopes are given in Table 14A-1.

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.

---

* J. O. Blomeke and Mary F. Todd, "Uranium-235 Fission-Product Production as a Function of Thermal Neutron Flux, Irradiation Time and Decay Time", ORNL-2127, August 19, 1957.

TABLE 14A-1

CORE AND GAP ACTIVITY

Assumptions: Operation at 2546 MWt for 500 days

Temperature Distribution Specified in Table 14A-2

Isotope	Curies In the Core ($X\ 10^7$)	Curies in the Gap ($X\ 10^5$)
I-131	6.27	16.9
I-132	9.57	3.1
I-133	14.4	14.0
I-134	17.3	3.53
I-135	12.8	7.08
Kr-85	.092	1.46
Xe-133	14.3	32.2
Xe-133m	.388	.602
Xe-135	5.43	.437

The diffusion coefficient, D', for Xe and Kr in UO_2, varies with temperature in accordance with 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 = $1 \times 10^{-11}\ \mathrm{sec}^{-1}$

T = temperature in degrees Kelvin

R = gas constant

The above expression is valid for temperatures above 1473°K. Below 1473°K fission gas release occurs mainly by two temperature independent phenomena, recoil and knock-out, and is predicted by using D' at 1473°K. 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 2546 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 is about 2.5% of the core inventory.

The percentage of the total core activity present in the gap for each isotope is also listed in Table 14A-1.

The core temperature distribution used in this analysis, is presented in Table 14A-2.

TABLE 14A-2

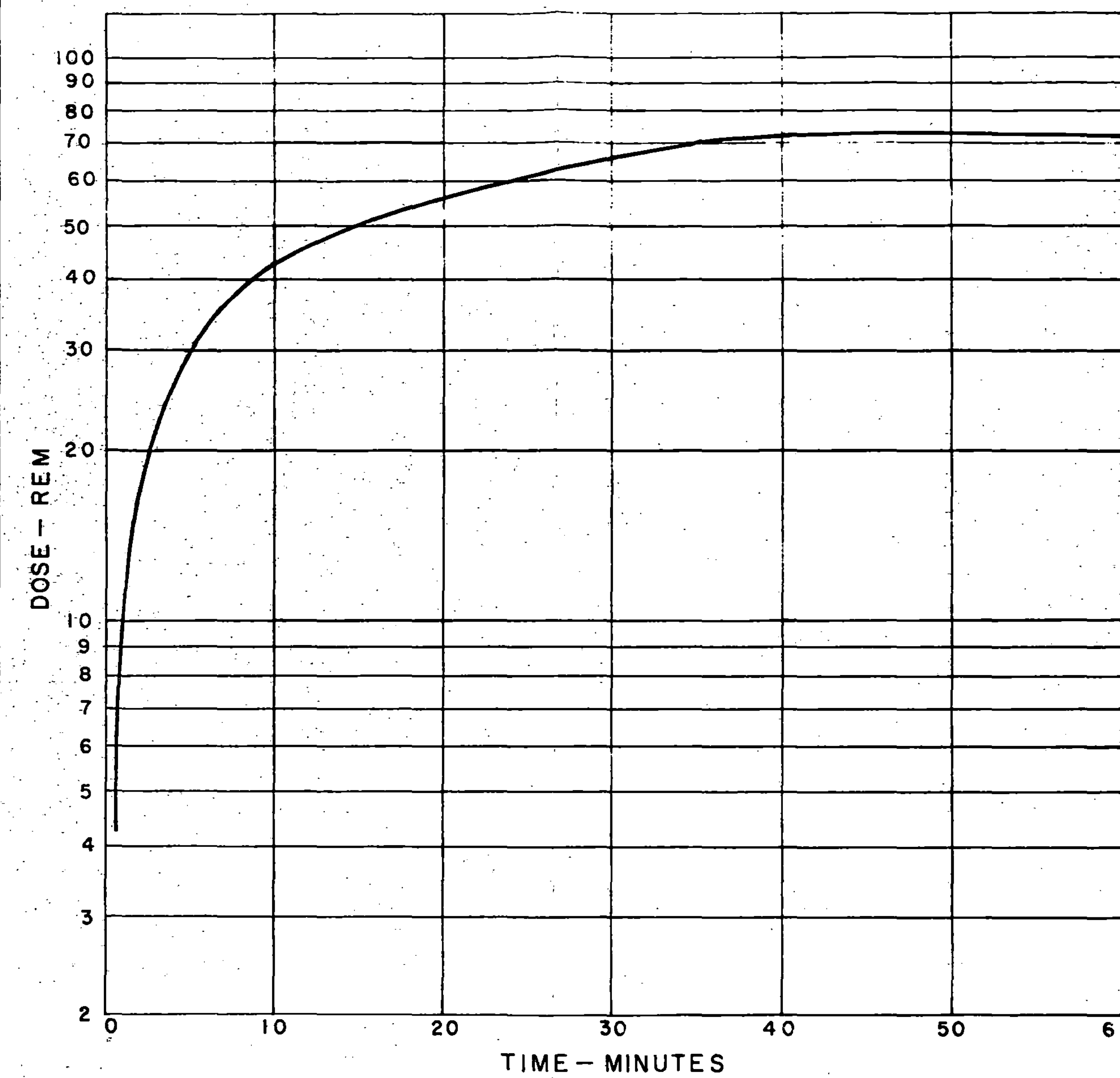
CORE TEMPERATURE DISTRIBUTION

Percent of Core Fuel Volume Above the Given Temperature	Local Temperature, °F
0.01	4100
0.40	3700
2.20	3300
5.90	2900
11.30	2500

REFERENCES

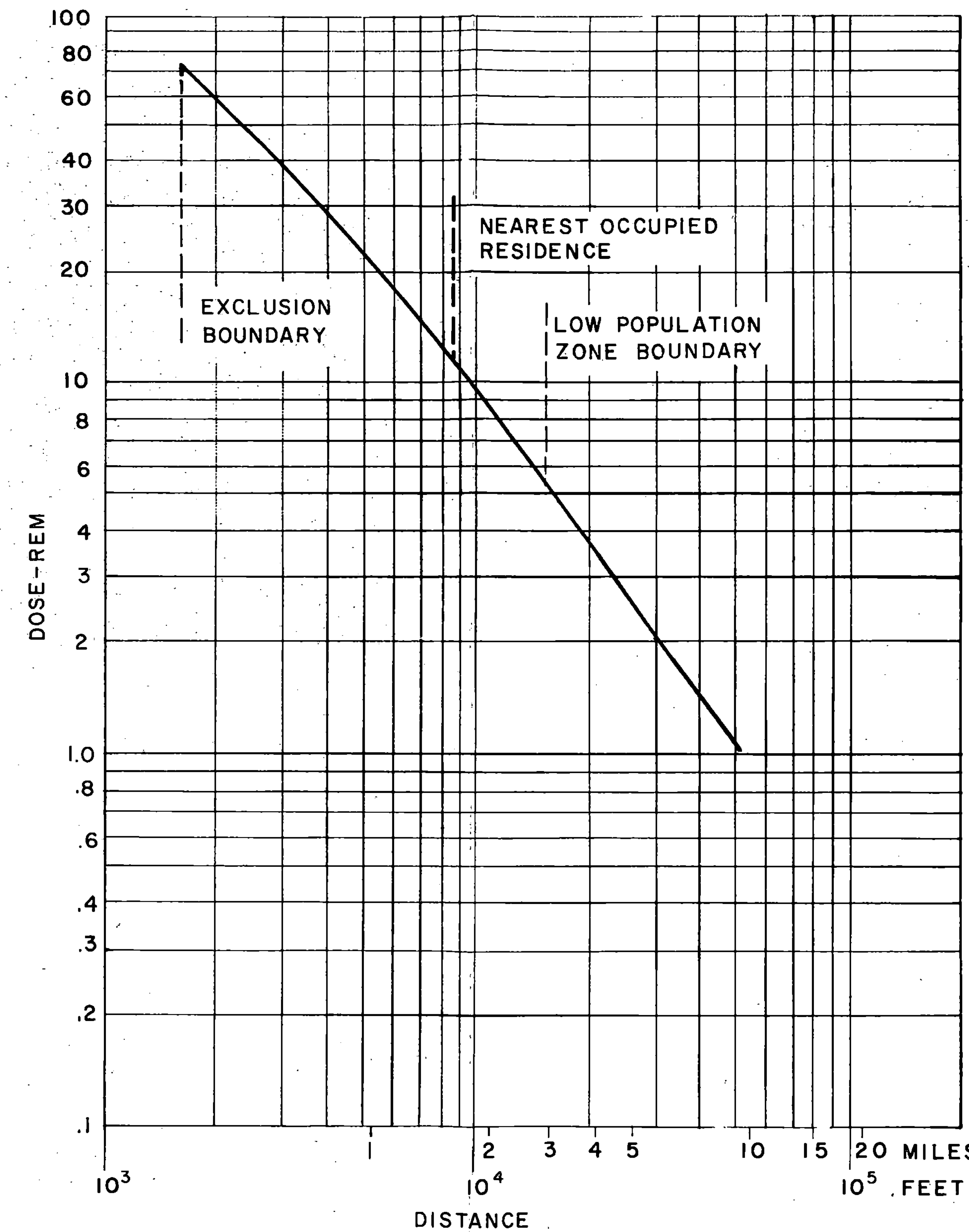
1. Toner, D. F., and Scott, J. S., "Fission Product Release from UO_2," Nuclear Safety, Vol. 3, No. 2, December 1961.

2. Belle, J., Uranium Dioxide: Properites and Nuclear Applications, Naval Reactors, Division of Reactor Development United States Atomic Energy Commission, 1961.



CONDITIONS AS ESTABLISHED
IN SECTION 14.5.7

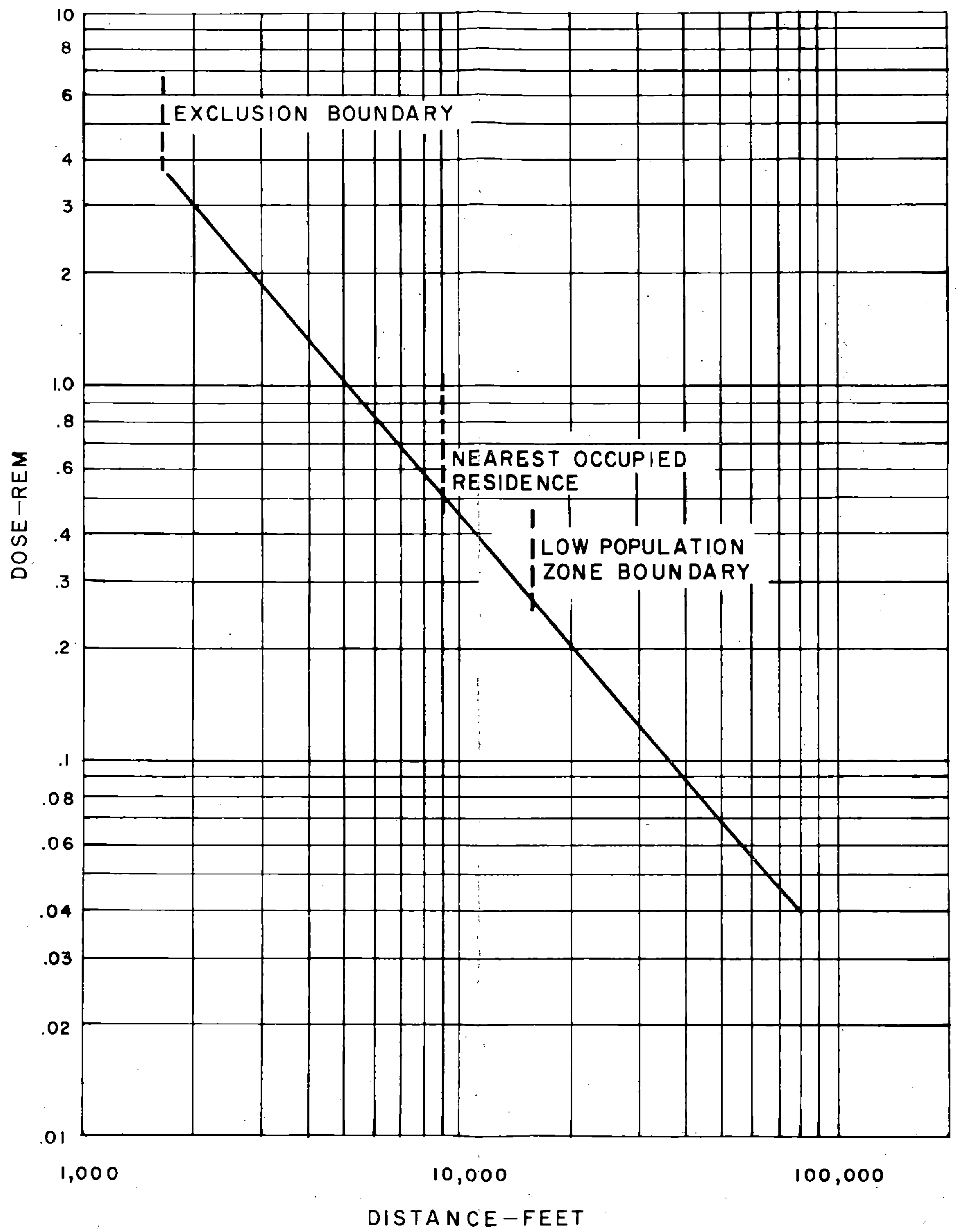
INTEGRATED THYROID DOSE
AT EXCLUSION BOUNDARY
DESIGN BASIS ACCIDENT
SURRY POWER STATION



CONDITION AS ESTABLISHED IN SECTION 14.5.7

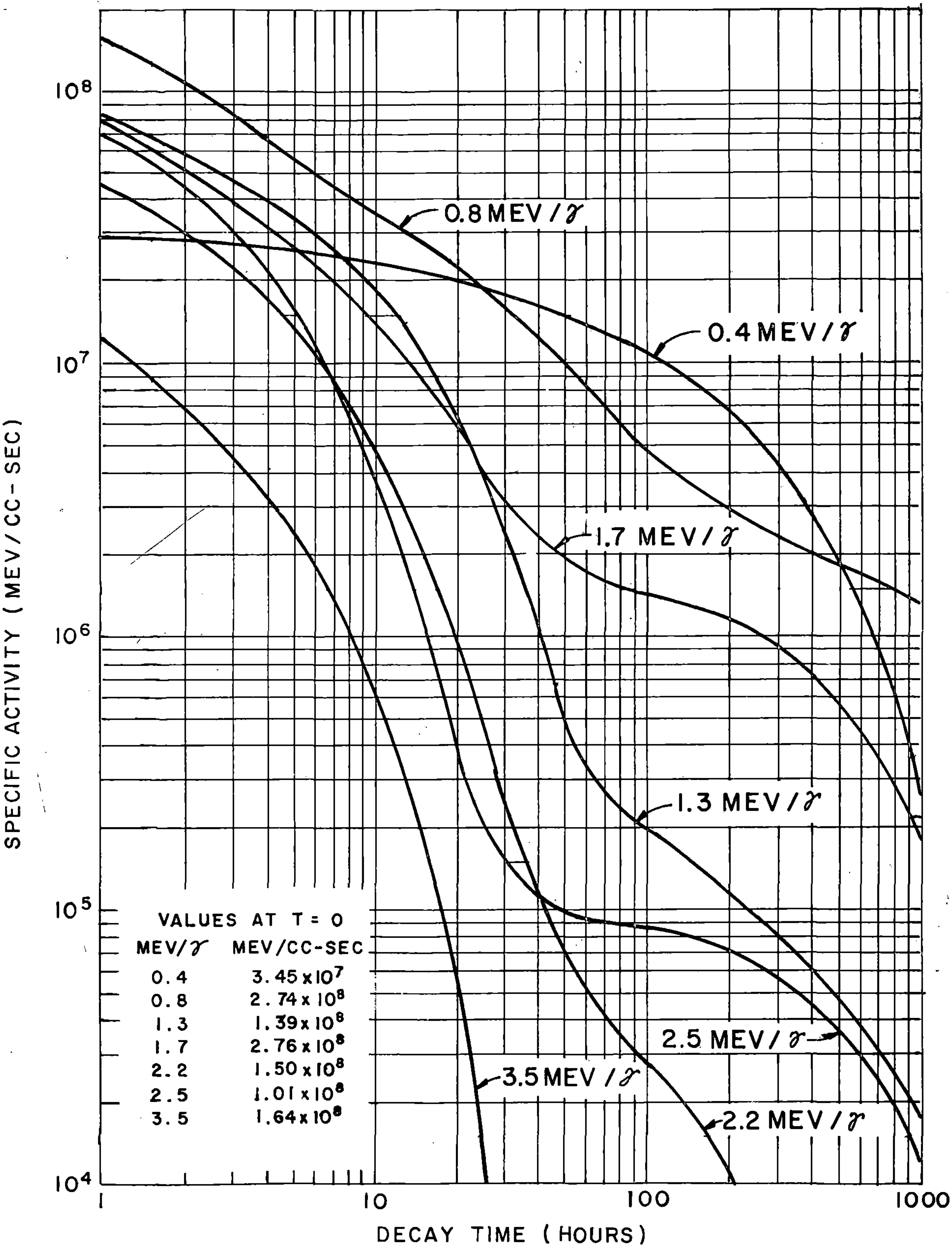
INTEGRATED THYROID DOSE
DESIGN BASIS ACCIDENT
SURRY POWER STATION

SF FIG 14.5.7-3
OCT. 15, 1970



CONDITION AS ESTABLISHED IN SECTION 14.5.7

INTEGRATED WHOLE BODY DOSE
DESIGN BASIS ACCIDENT
SURRY POWER STATION

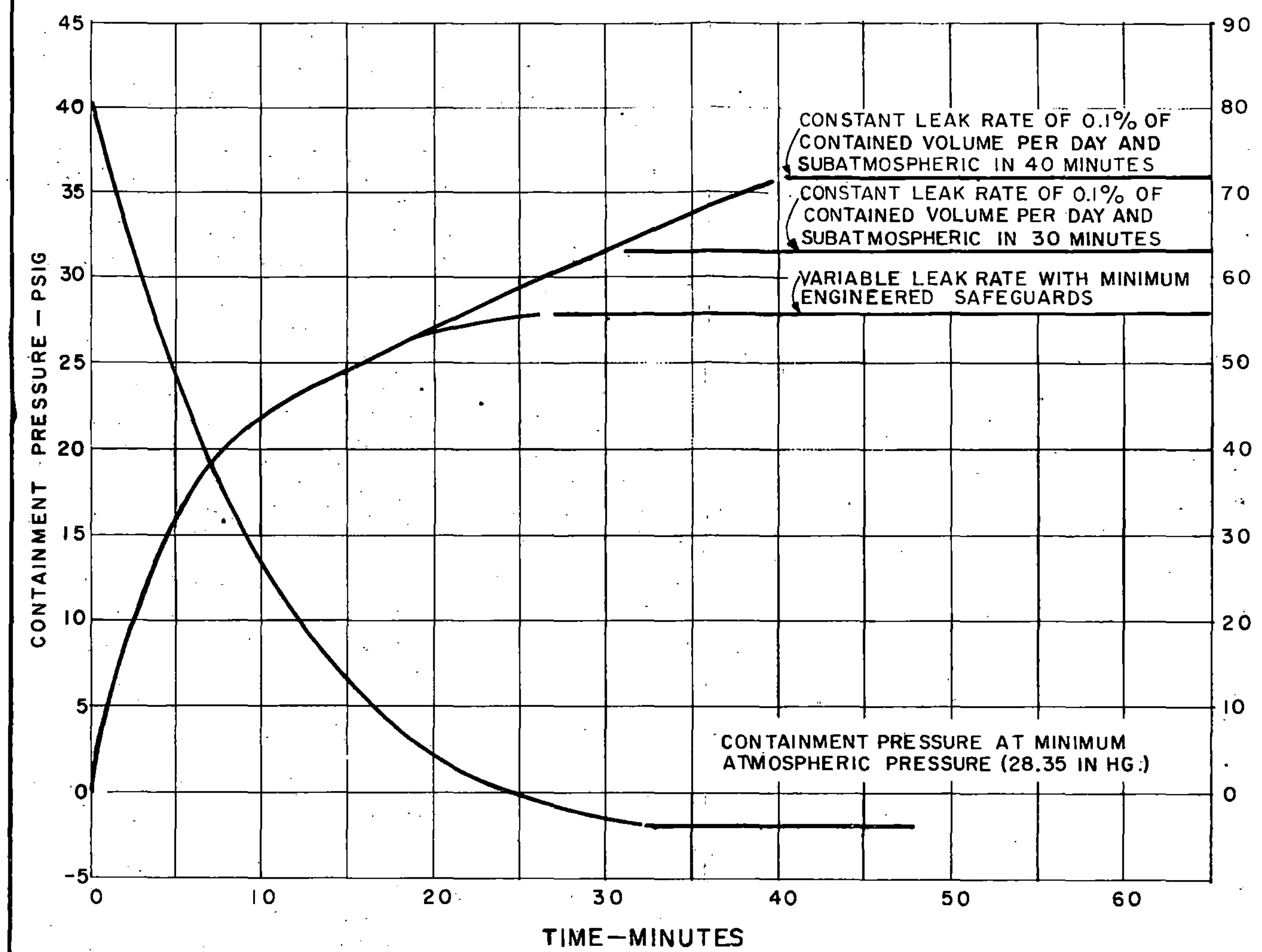


MEV/γ	MEV/CC-SEC
0.4	3.45×10^7
0.8	2.74×10^8
1.3	1.39×10^8
1.7	2.76×10^8
2.2	1.50×10^8
2.5	1.01×10^8
3.5	1.64×10^8

CONTAINMENT RADIATION SOURCES

DESIGN BASIS ACCIDENT

SURRY POWER STATION



INTEGRATED THYROID DOSE AT EXCLUSION BOUNDARY vs TIME
DESIGN BASIS ACCIDENT
SURRY POWER STATION