ATTACHMENT A

ANALYSIS OF POTENTIAL ENVIRONMENTAL CONSEQUENCES FOLLOWING A STEAM GENERATOR TUBE FAILURE AT R. E. GINNA NUCLEAR POWER PLANT

NOVEMBER 1982

Prepared by:

K. Rubin

E. Volpenhein

Westinghouse Electric Corporation
Nuclear Energy Systems
P.O. Box 355
Pittsburgh, Pennsylvania 15230

Prepared for:

Rochester Gas and Electric
89 East Avenue
Rochester, New York 14649

TABLE OF CONTENTS

ABSTRACT

The potential radiological consequences of a steam generator tube failure event were evaluated for the R. E. Ginna nuclear power plant to demonstrate that standard limitations on initial coolant activity are acceptable. Mass releases following a design basis tube rupture were calculated for both 30 minute and 60 minute operator response times. The site boundary and low population zone exposures were conservatively calculated for these releases. In addition, the standard technical specification limit on initial coolant activity and realistic meteorology were applied to "best estimate" mass release during the January 25, 1982 tube failure event at Ginna. Results show that the conservative assessment of the environmental consequences are within acceptable limits and that the potential exposure from a more realistic event is minimal.

LIST OF TABLES

LIST OF TABLES (Continued)

LIST OF FIGURES

LIST OF FIGURES (Continued)

I. INTRODUCTION

Potential environmental consequences of a steam generator tube rupture event at the R. E. Ginna nuclear power plant have been evaluated to verify that the standard technical specification limit on primary coolant activity is adequate for Ginna. Mass releases were calculated using the computer code LOFTRAN[1] with conservative assumptions of break size, condenser availability, and various operator response times. The effect of steam generator overfill and subsequent water relief through secondary side relief valves was also addressed. Conservative assumptions concerning coolant activity, meteorology, and partitioning between liquid and vapor phases were applied to these mass releases to determine an upper bound on site boundary and low population zone doses. Best estimate mass releases during the January 25, 1982 tube failure event at Ginna were also calculated based on analyses presented in reference 2. These releases were used to estimate potential doses which could have resulted if the accident had occurred with coolant activity limits established in the standard technical specifications.

II. MASS RELEASES

Mass releases during a design basis steam generator tube rupture event were calculated using established FSAR methodology assuming various operator response times. Releases during the Ginna event were also estimated. Contributions from both the intact and faulted steam generators were evaluated as well as flow to the condenser and atmosphere. These mass releases are presented for various time periods during the accident. The assumptions and methodology which were used to generate the results are described in the following sections.

II.1 Design Basis Accident

The accident examined is the complete severance of a single steam generator tube during full power operation. This is considered a condition IV event, a limiting fault, and leads to an increase in the contamination of the secondary system due to leakage of radioactive coolant from the RCS. Discharge of activity to the atmosphere may occur via the steam generator safety and/or power operated relief valves. The concentration of contaminants in the primary system is continuously controlled to limit such releases.

II.1.1 Sequence of Events

If normal operation of the various plant control systems is assumed, the following sequence of events is initiated by a tube rupture:

A. The steam generator blowdown liquid monitor and/or the condenser air ejector radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system.

B. Pressurizer low pressure and low level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side steam flow/feedwater flow mismatch occurs as feedwater flow to the affected steam generator is reduced to compensate for break flow to that unit.

C. The decrease in RCS pressure due to continued loss of reactor coolant inventory leads to a reactor trip signal on low pressurizer pressure or overtemperature delta-T. Plant cooldown following reactor trip leads to a rapid decrease in pressurizer level and a safety injection signal, initiated by low pressurizer pressure, follows soon after reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates auxiliary feedwater addition.

D. The reactor trip automatically trips the turbine and, if offsite power is available, the steam dump valves open permitting steam dump to the condenser. In the event of coincident station blackout, as assumed in the results presented, the steam dump valves automatically close to protect the condenser. The steam generator pressure rapidly increases resulting in steam discharge to the atmosphere through the steam generator safety and/or power operated relief valves.

E. The auxiliary feedwater and borated safety injection flow provide a heat sink which absorbs decay heat and attenuates steaming from the steam generators.

F. Safety injection flow results in increasing pressurizer water volume at a rate dependent upon the amount of auxiliary equipment operating. RCS pressure eventually equilibrates at a pressure greater than the affected steam generator pressure where safety injection flow matches break flow.

The operator is expected to determine that a steam generator tube rupture has occurred and to identify and isolate the faulty steam generator on a restricted time scale in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the faulty unit. Sufficient indications and controls are provided to enable the operator to complete recovery procedures from within the control room. High radiation indications or rapidly increasing water level in any steam generator provide symptoms of the faulted steam generator which ensure identification before the water level increases above the narrow range. For smaller tube failures,

sampling of the steam generators for high radiation may be required for positive identification. However, in that case additional time would be available before water level increases out of narrow range.

Once identified, the faulted steam generator is isolated from the intact steam generators to minimize activity releases and as a necessary step toward establishing a pressure differential between the intact and faulted steam generators. The Main Steamline Isolation Valves (MSIV) provide this capability. In the event of a failure of the MISV for the faulted steam generator, the MSIV for the intact steam generator and the turbine stop valve ensure a redundant means of isolation. Auxiliary feedwater flow is terminated to the faulted unit in an attempt to control steam generator inventory.

The reactor coolant temperature is reduced to establish a minimum of 50°F subcooling margin at the ruptured steam generator pressure by dumping steam from the intact steam generator. This assures that the primary system will remain subcooled following depressurization to the faulted steam generator pressure in subsequent steps. If the condenser is available, the normal steam dump system is used for this cooldown. Isolation of the faulted steam generator ensures that pressure in that unit will not decrease significantly. If the condenser is unavailable or if the MSIV for the faulted steam generator fails, the atmospheric relief valve on the intact steam generator provides an alternative means of cooling the reactor coolant system.

The primary pressure is reduced to a value equal to the faulted steam generator pressure using normal pressurizer spray. This action restores pressurizer level as safety injection flow in excess of break flow replaces condensed steam in the pressurizer, and momentarily stops primary-to-secondary leakage. If normal spray is not available, the pressurizer PORVs and auxiliary spray system provide redundant means of depressurizing the reactor coolant system.

Termination of safety injection flow is required to ensure that break flow is not reinitiated. Previous operator actions are designed to establish sufficient indications of adequate primary coolant inventory and heat removal so that core cooling will not be compromised as a result of SI termination.

This sequence of recovery actions ensures early termination of primary-to-secondary leakage with or without offsite power available. The time required to complete these actions are event specific since smaller breaks may be more difficult to detect. In these analyses, operator action times have been treated parametrically, ranging from 30 minutes to a maximum of 60 minutes to complete the key recovery sequence.

II.1.2 Method of Analysis

Mass and energy balance calculations were performed using LOFTRAN to determine primary-to-secondary mass leakage and the amount of steam vented from each of the steam generators prior to terminating safety injection. In estimating the mass releases during recovery, the following assumptions were made:

A. Reactor trip occurs automatically as a result of low pressurizer pressure or overtemperature delta-T. Loss of offsite power occurs at reactor trip.

B. Following the initiation of the safety injection signal, all safety injection pumps are actuated. Flow from the normal charging pumps is not considered since it is automatically terminated on a safety injection signal.

C. The secondary side pressure is assumed to be controlled at the safety valve pressure following reactor trip. This is consistent with loss of offsite power.

D. Auxiliary feedwater flow is assumed throttled to match steam flow in all steam generators to control steam generator level. Minimum auxiliary feedwater capacity is assumed. This results in increased steaming from the steam generators.

E. Individual operator actions are not explicitly modeled in the analyses presented. However, it is assumed that the operator completes the recovery sequence on a restricted time scale. This time is treated parametrically.

F. For cases where steam generator overfill occurs, water relief from the faulted steam generator to the atmosphere is assumed equal to any additional primary-to-secondary leakage after overfill occurs. Steamline volume is not considered in calculating the time of steam generator overfill.

Prior to reactor trip steam is assumed to be released to the condenser from the faulted and intact steam generators. Steam from all steam generators is dumped to the atmosphere after reactor trip since the condenser is unavailable as a result of station blackout.

Extended steam release calculations, i.e. after break flow has been terminated, reflect expected operator actions as described in the Westinghouse Owners Group's Emergency Response Guidelines(3). Following isolation of the faulted steam generator, it is assumed that steam is dumped from the intact steam generator to reduce the RCS temperature to 50°F below no-load Tavg. From two to eight hours after tube failure, the RCS coolant temperature is reduced to Residual Heat Removal System (RHRS) operating conditions via additional steaming from the intact steam generator. Further plant cooldown to cold shutdown is completed with the RHRS. If steam generator overfill does not occur, the faulted steam generator is depressurized by releasing steam from that steam generator to the atmosphere. An alternate cooldown method, such as backfill into the RCS, is considered if the faulted steam generator fills with water. In that case additional steaming occurs from the intact steam generator. The extended steam and feedwater flows are determined from a mass and energy balance including decay heat, metal heat, energy from one operating reactor coolant pump, and sensible energy of the fluid in the RCS and steam generators.

The sequence of events for the design basis accident are presented in Table II.1.2-1. The primary-to-secondary carryover and steam and feedwater flows associated with each of the steam generators are provided in Tables II.1.2-2 and II.1.2-3 for recovery times of 30 and 60 minutes, respectively. Since individual operator actions were not modelled, the system response is the same for both cases. With 30 minute operator action to terminate break flow,

TABLE II.1.2-1 DESIGN BASIS ACCIDENT SEQUENCE OF EVENTS

Event	Manual (O) Automatic (A)	Time (Sec) 30 Min Recovery	60 Min Recovery
Tube Failure	-	0	0
Reactor Trip	A	27	27
Condenser Lost	-	27	27
SI Signal	A	127	127
Feedwater Isolation	A	134	134
AFW Initiation	A	187	187
AFW Throttled to Faulted SG	O	187	187
Isolation of Faulted SG	O	1800(1)	3600(1)
Steam Dump	O	1800(1)	3600(1)
RCS Depressurization	O	1800(1)	3600(1)
SG Overfill	-	-	2810
SI Terminated	O	1800(1)	3600(1)
Break Flow Terminated	O	1800(1)	3600(1)
RHR Cooling	O	28800	28800

(1) These events are not actually modeled but are assumed to occur within the time indicated.

TABLE II.1.2-2 MASS RELEASES DURING A DESIGN BASIS SGTR: 30 MINUTE RECOVERY

	Time Period			
Flow (lbm)	0-TTRIP	TTRIP-TTBRK	TTBRK-2	2-TRHR
Ruptured SG:				
- Condenser	27820	0.0	0.0	0.0
- Atmosphere	0.0	32640	0.0	21480
- Feedwater	32605	0.0	0.0	21480
Intact SG:				
- Condenser	27380	0.0	0.0	0.0
- Atmosphere	0.0	23050	144650	470000
- Feedwater	37170	13370	206200	487600
Break Flow	3325	100648	0.0	0.0

TTRIP = 27.0 sec = Time of reactor trip
TTBRK = 1800 sec = Time to terminate break flow
TRHR = 28800 sec = Time to establish RHR cooling.

TABLE II.1.2-3 MASS RELEASES DURING A DESIGN BASIS SGTR: 60 MINUTE RECOVERY

Flow (lbm)	Time Period					
	0-TTRIP	TTRIP-TMSEP	TMSEP-TSGOF	TSGOF-TTBRK	TTBRK-2	2-TRHR
Ruptured SG:						
- Condenser	27820	0.0	0.0	0.0	0.0	0.0
- Atmosphere	0.0	33570	4830	43171	0.0	0.0
- Feedwater	32605	0.0	0.0	0.0	0.0	0.0
Intact SG:						
- Condenser	27380	0.0	0.0	0.0	0.0	0.0
- Atmosphere	0.0	23370	1390	390	67970	501100
- Feedwater	37170	13700	1390	380	129600	518700
Break Flow	3325	107742	48070	43171	0.0	0.0

TTRIP = 27.0 sec = Time of reactor trip

TMSEP = 1930 sec = Time to fill SG to moisture separators

TSGOF = 2810 sec = Time to fill SG (w/o steamline volume)

TTBRK = 3600 sec = Time to terminate break flow

TRHR = 28800 sec = Time to establish RHR cooling

liquid level in faulted steam generator remains below the bottom of the moisture separator, Figure II.1.2-1. Hence, for this case, partitioning between the vapor and liquid phases effectively reduces radiological releases for the duration of the accident. For delayed recovery, case 2, the moisture separator begins to flood at 32 minutes. The faulted steam generator is completely filled by 47 minutes. During this time, liquid entrainment within the steam flow would increase so that the effectiveness of partitioning would be reduced. Beyond 47 minutes, i.e. steam generator overfill, water relief from the faulted steam generator is assumed equal to break flow.

The following is a list of figures of pertinent time dependent parameters:

FIGURE II.1.2-1 FAULTED SG WATER VOLUME

FIGURE II.1.2-2 REACTOR COOLANT SYSTEM PRESSURE

FIGURE II.1.2-3 FAULTED SG PRESSURE

FIGURE II.1.2-4 REACTOR COOLANT SYSTEM TEMPERATURE

FIGURE II.1.2-5 PRESSURIZER WATER VOLUME

FIGURE II.1.2-6 FAULTED SG STEAM FLOW

FIGURE II.1.2-7 BREAK FLOW

FIGURE II.1.2-8 BREAK FLOW FLASHING FRACTION

II.2 GINNA EVENT

A detailed thermal-hydraulic analysis of the Ginna event is described in reference 2. The results of that analysis form the basis for the calculation of the potential environmental consequences. The general sequence of events during the Ginna accident, Table II.2-1, was similar to the design basis

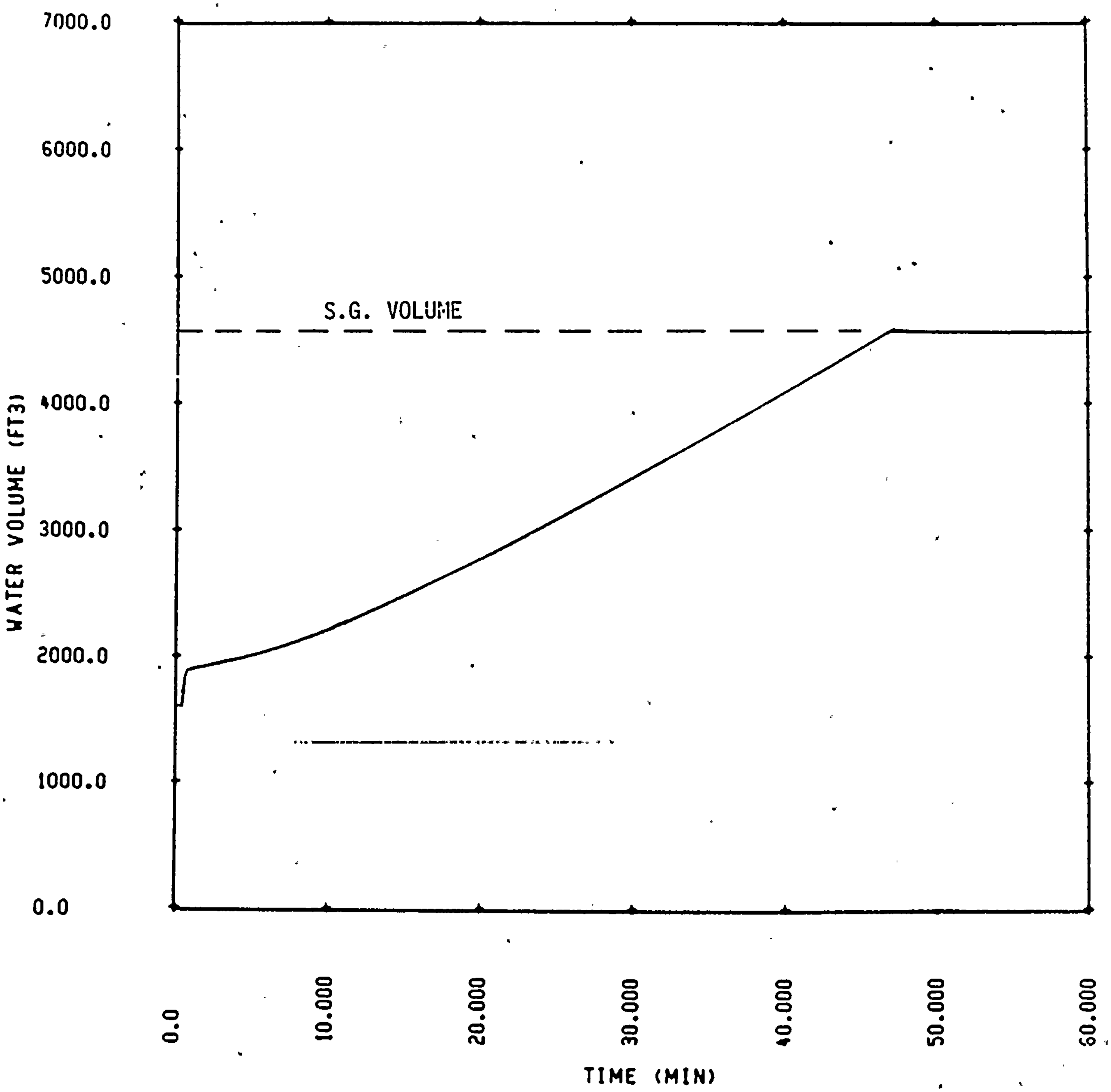


FIGURE II.1.2-1. FAULTED STEAM GENERATOR WATER VOLUME.

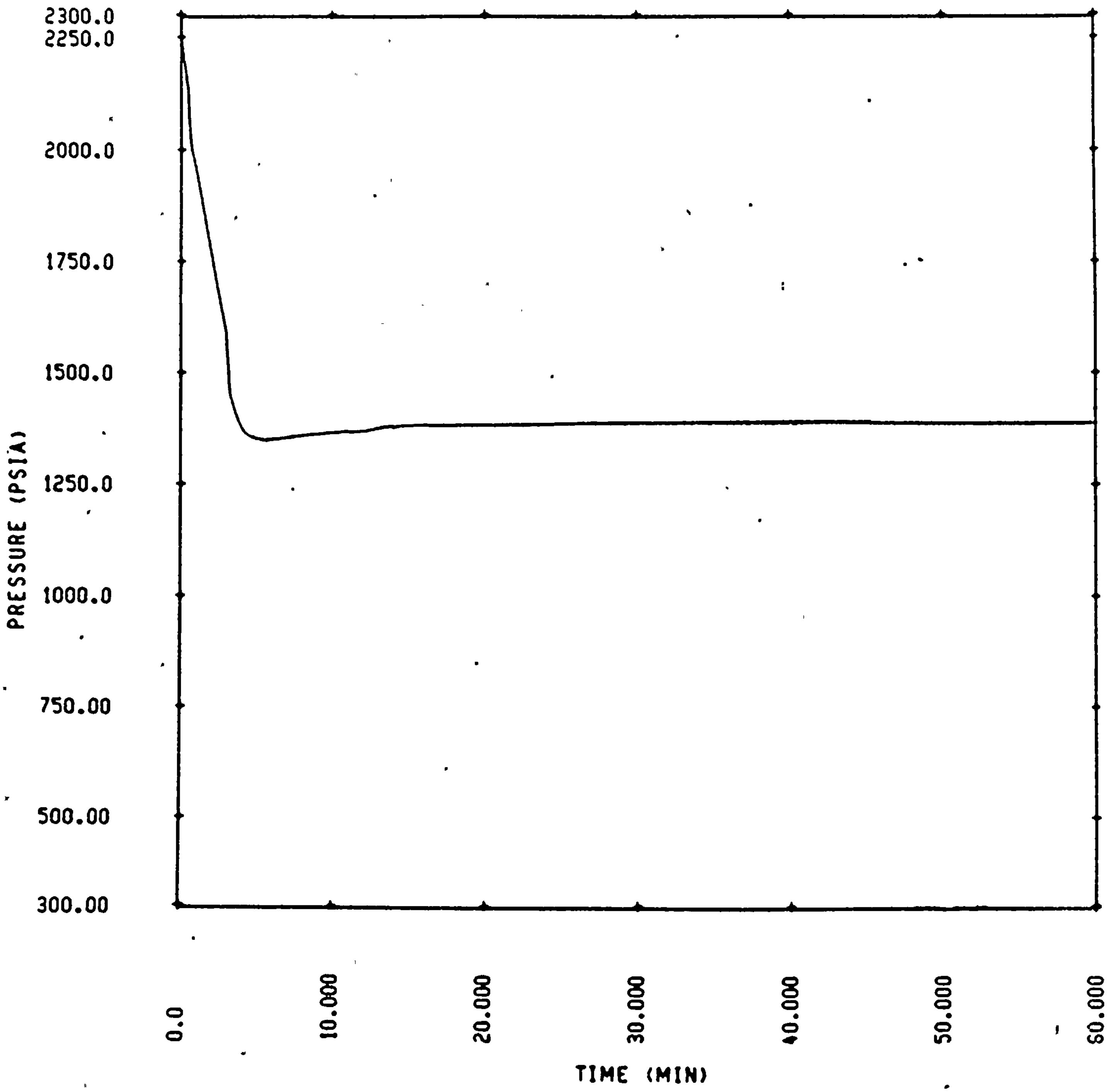


FIGURE II.1.2-2. REACTOR COOLANT SYSTEM PRESSURE.

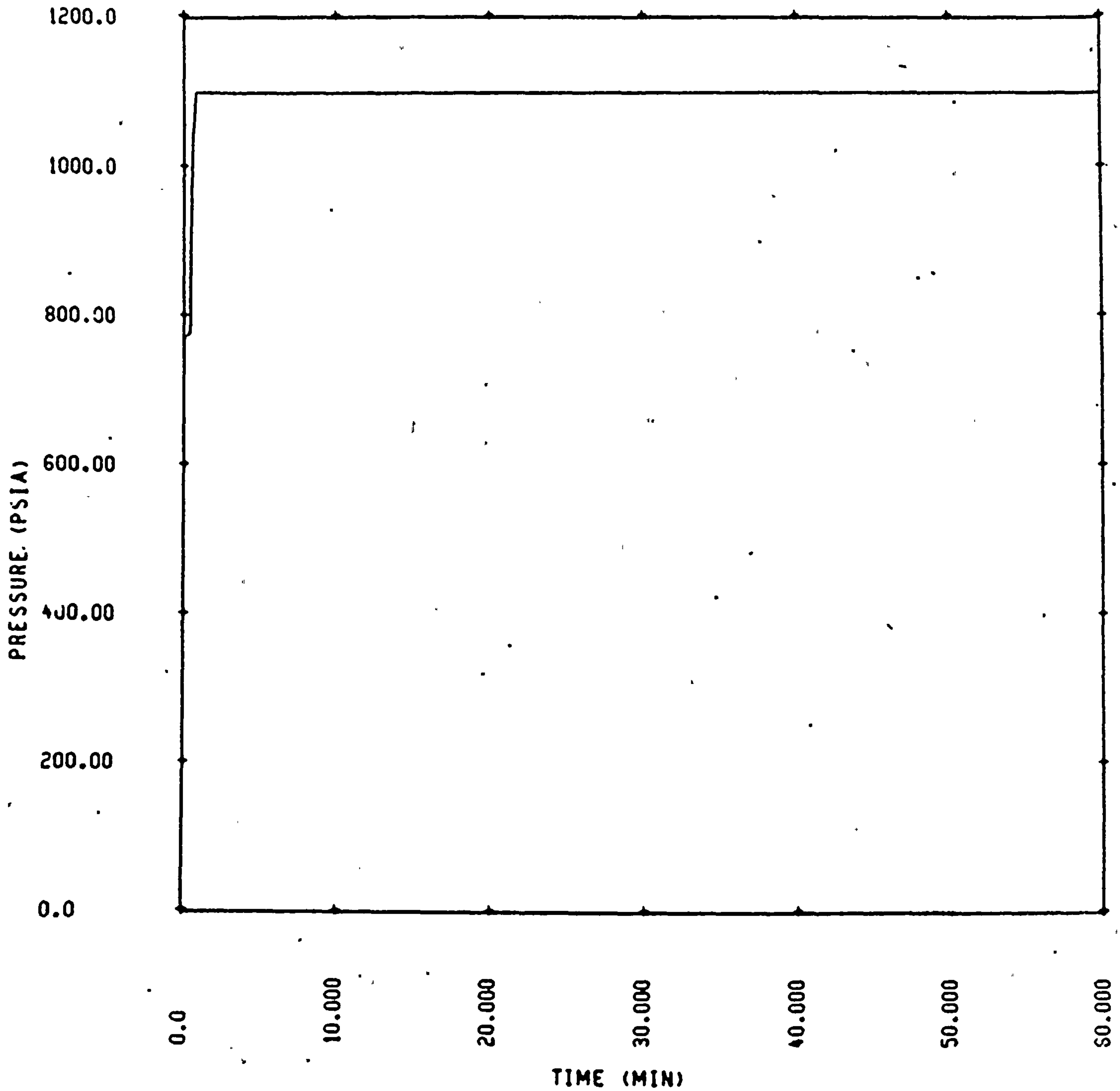


FIGURE II.1.2-3. FAULTED STEAM GENERATOR PRESSURE.

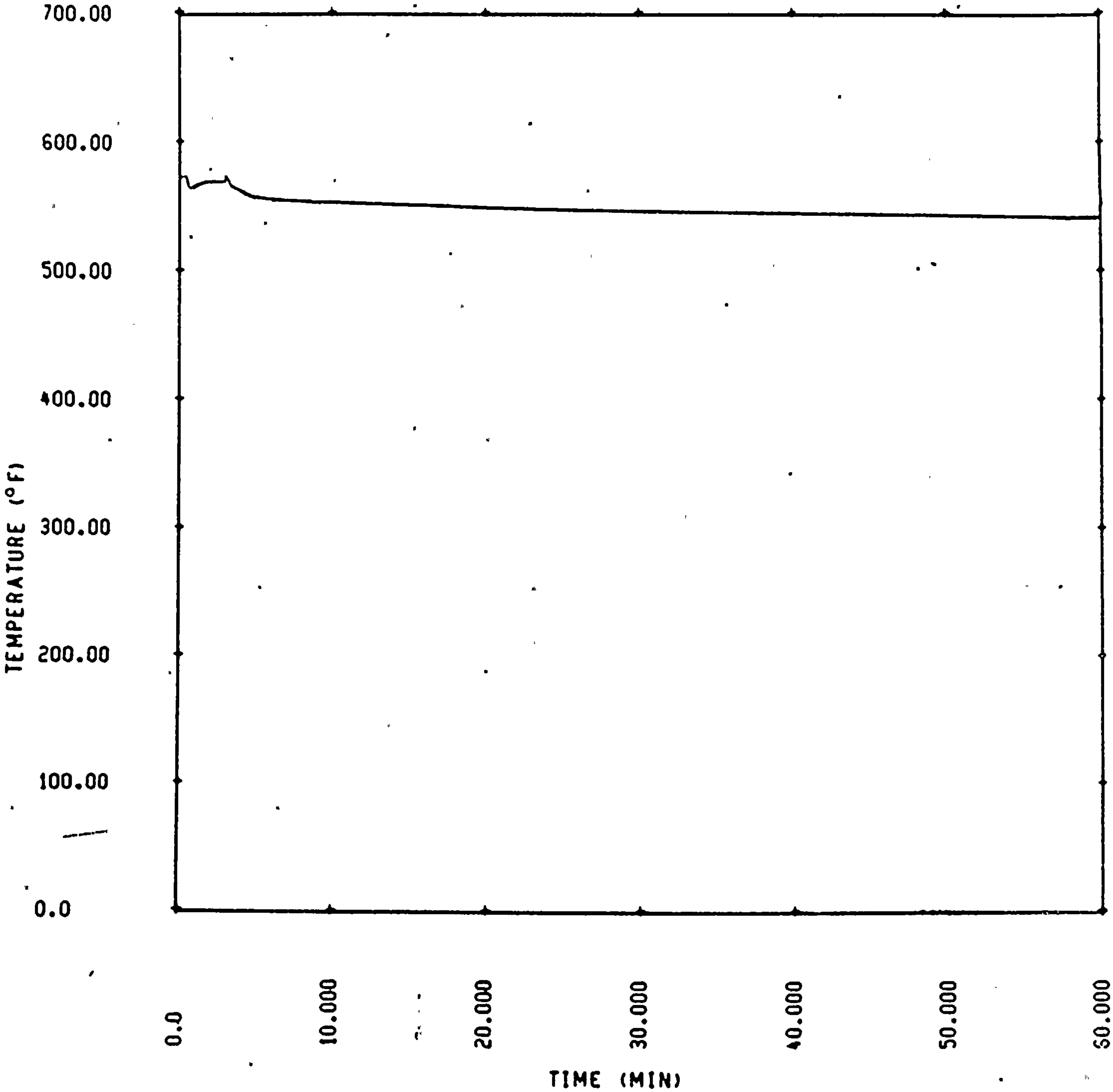


FIGURE II.1.2-4. REACTOR COOLANT AVERAGE TEMPERATURE.

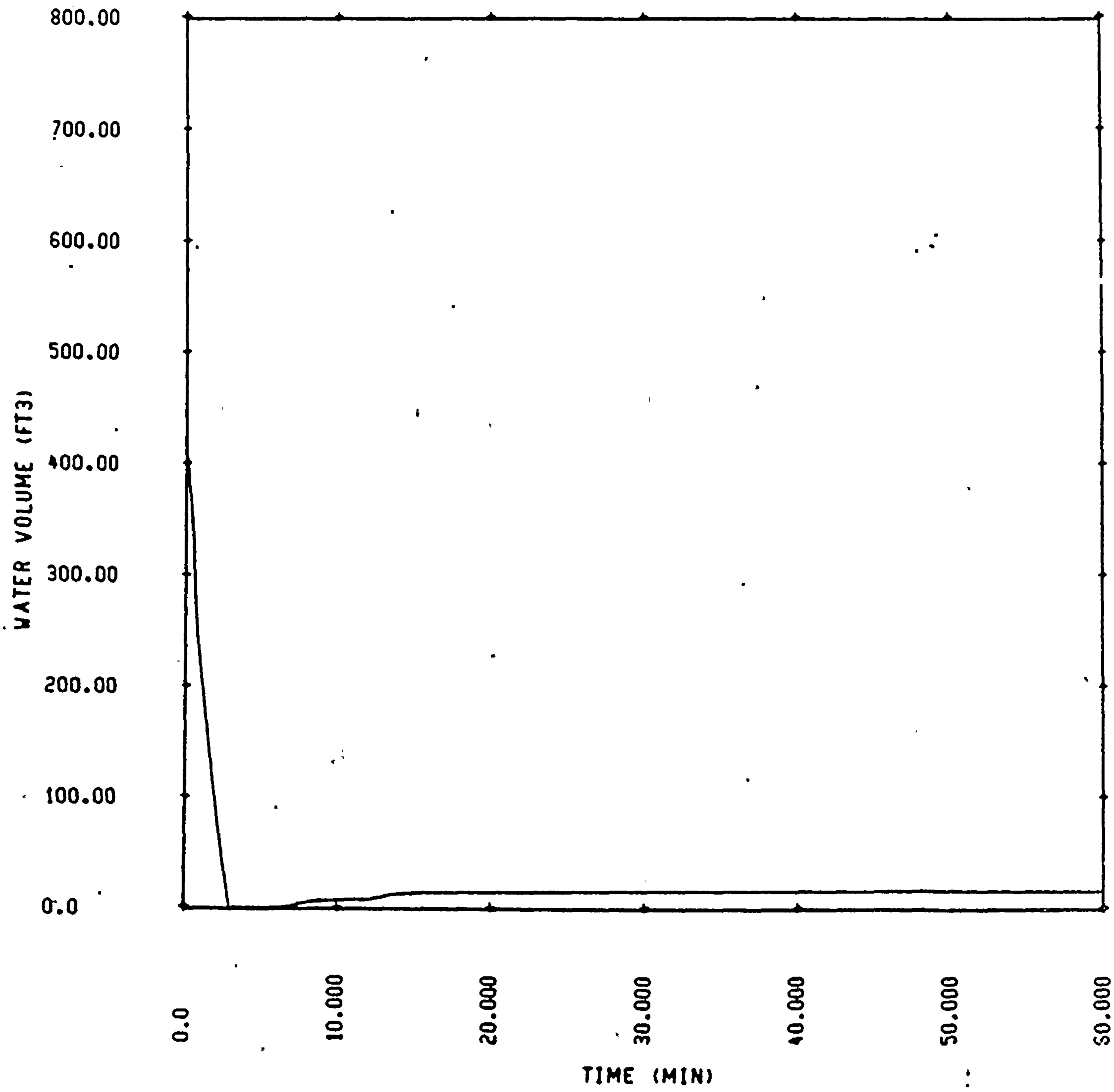


FIGURE II.1.2-5. PRESSURIZER WATER VOLUME.

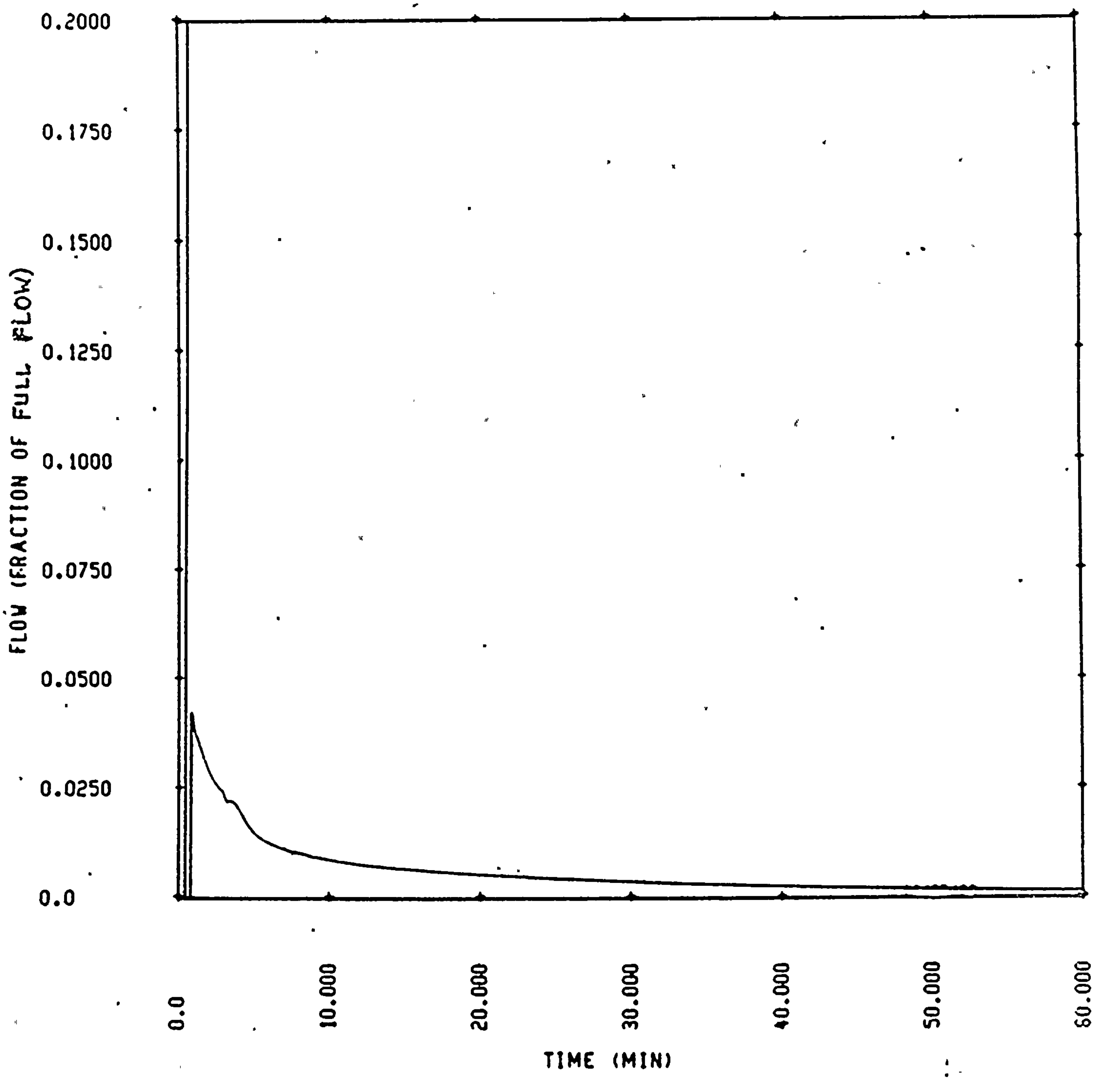


FIGURE II.1.2-6. FAULTED STEAM GENERATOR STEAM FLOW.

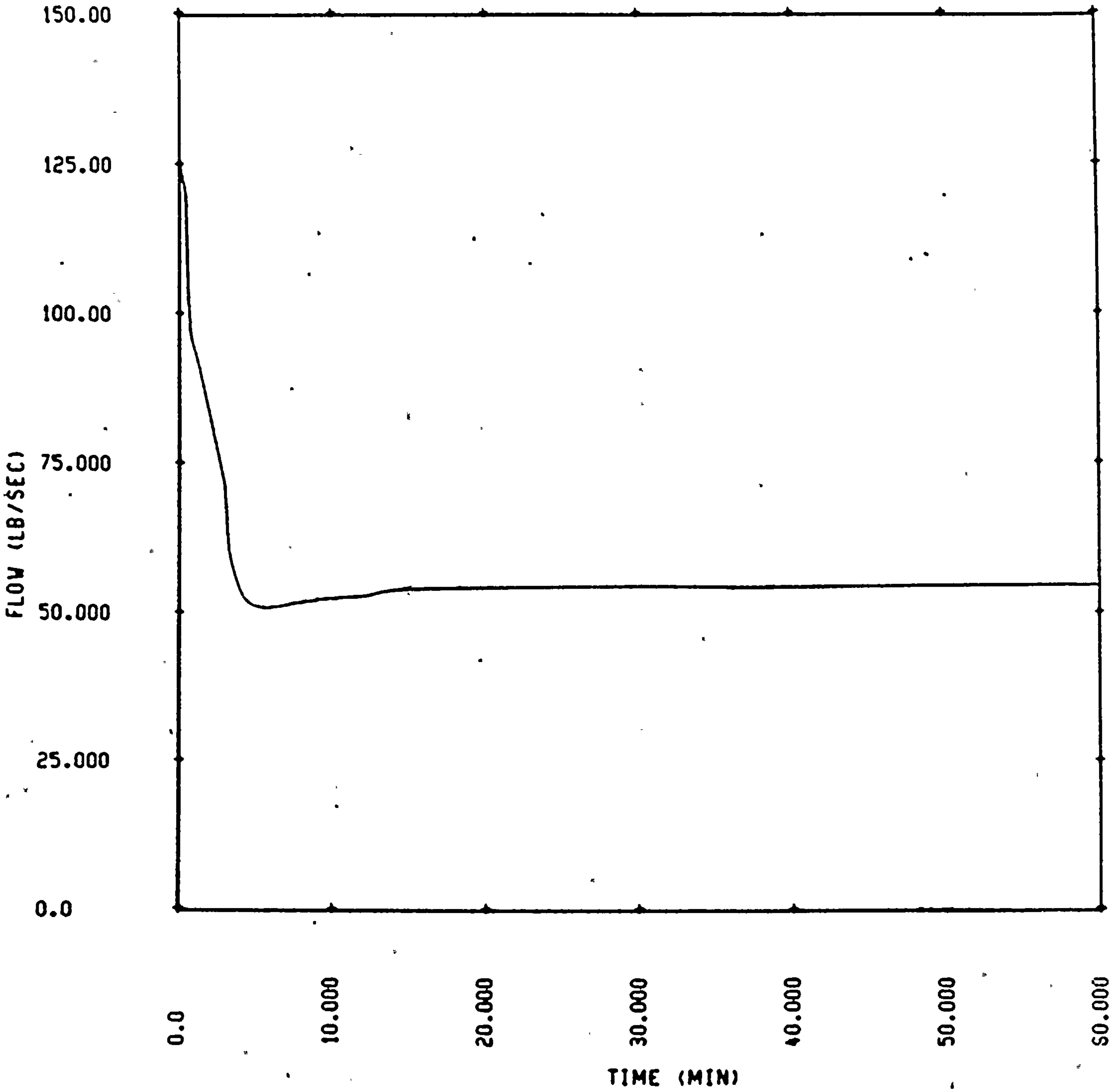


FIGURE II.1.2-7. PRIMARY-TO-SECONDARY LEAKAGE.

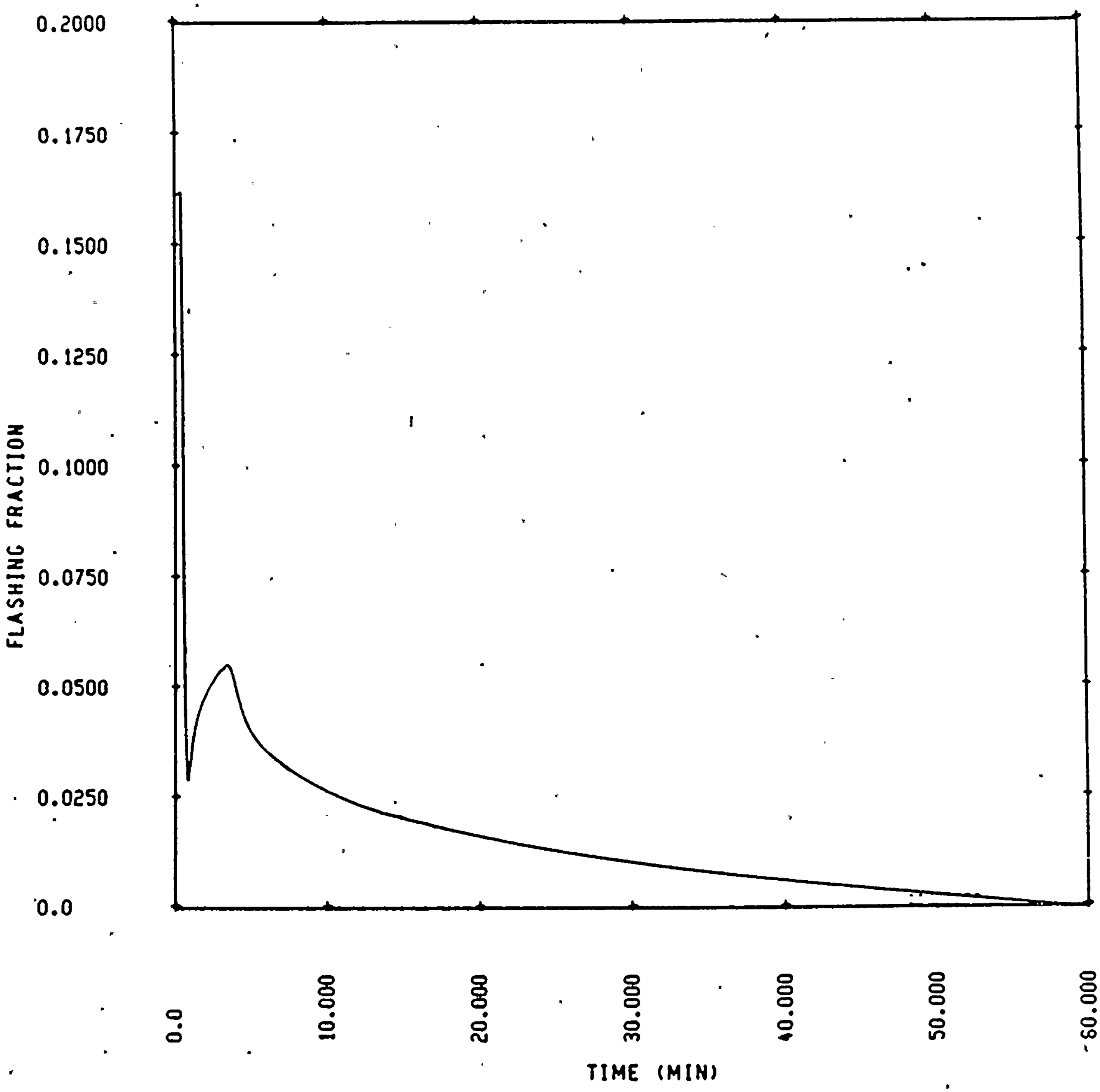


FIGURE II.1-2-8. BREAK FLOW FLASHING FRACTION.

TABLE II.2-1 GINNA SEQUENCE OF EVENTS

Event	Manual (0) Automatic (A)	Time (sec) Actual	Simulated
Tube Failure	-	0	0
Reactor Trip	A	182	182
Condenser Lost	-	4500	4500
SI Signal	A	190	198
Feedwater Isolation	A	192	198
AFW Initiated	A	220	239
AFW Throttled to Faulted SG	0	410	410
Isolation of Faulted SG	0	890	530
Steam Dump	0	770	530
RCS Depressurization	0	2700	2700
SG Overfill (1)	-	-	3130
SI Terminated	0	4310	4310
Break Flow Terminated	0	10800	10800
RHR Cooling	0	77580	77580

(1) includes steamline volume

event described in section II.1.1. Break flow in excess of normal charging flow depleted reactor coolant inventory and eventually resulted in reactor trip on low pressurizer pressure. A safety injection signal followed soon after trip. Normal feedwater flow was automatically terminated on the safety injection signal and auxiliary feedwater flow was initiated. The steam dump system operated to control steam gene- rator pressure below the safety valve setpoint and establish no-load reactor coolant temperature. Auxiliary feedwater and safety injection flows absorbed decay heat and temporarily stopped steam releases from the steam generators.

Emergency recovery actions were quickly initiated to mitigate the consequences of the accident. Pre-trip symptoms of the faulted steam generator, including steam flow/feed flow mismatch and steam generator level deviation alarms, provided tentative indications of the faulted steam generator which were confirmed soon after reactor trip by rapidly increasing steam generator level and high radiation indications. Auxiliary feedwater flow was reduced to the faulted unit in an attempt to control inventory. Isolation of the faulted steam generator was completed within 15 minutes of tube failure by closing the associated MSIV. Continued auxiliary feedwater flow to the intact steam generator effectively reduced the primary system temperature to establish 50°F subcooling margin. Normal spray was unavailable since reactor coolant pumps were manually tripped soon after reactor trip as directed by emergency procedures. Consequently, one pressurizer PORV was used as an alternative means of depressurizing the primary system to restore pressurizer level and reduce break flow. This was completed within 45 minutes. Safety injection flow was subsequently terminated after 72 minutes. Continued charging flow and reinitiation of safety injection flow resulted in additional primary-to-secondary leakage until approximately 3 hrs after tube failure.

Mass releases during the Ginna event are presented in Table II.2-2. LOFTRAN results indicate that the faulted steam generator and steamline filled with water after approximately 52 minutes, Figure II.2-1. Beyond this time water relief from the faulted steam generator was assumed equal to any additional primary-to-secondary leakage. The measured primary and faulted steam generator pressures and calculated break flow flashing fraction during the accident

TABLE II.2-2 BEST ESTIMATE MASS RELEASES DURING GINNA SGTR EVENT

	Time Period					
Flow (lbm)	0-TTRIP	TTRIP-TMSEP	TMSEP-TSGOF*	TSGOF*-2	2-TTBRK	TTBRK-TRHR
Faulted SG:						
- Condenser	162100	16900	0	0	0	0
- Atmosphere	0	0	0	130442	105684	0
- Feedwater	163400	46800	0	0	0	0
Intact SG:						
- Condenser	160100	288000	25200	14500	0	0
- Atmosphere	0	0	0	23870	54743	978387
- Feedwater	171700	52300	0	89700	53008	983292
Break Flow	10300	54330	99170	130442	105684	0

TTRIP = 182.0 sec = Time of reactor trip

TMSEP = 1335 sec = Time to fill SG to moisture separator

TSGOF = 2192 sec = Time to fill SG

TSGOF* = 3131 sec = Time to fill SG and steamline

TTBRK = 10200 sec = Time to terminate break flow

TRHR = 77580 sec = Time to establish RHR cooling

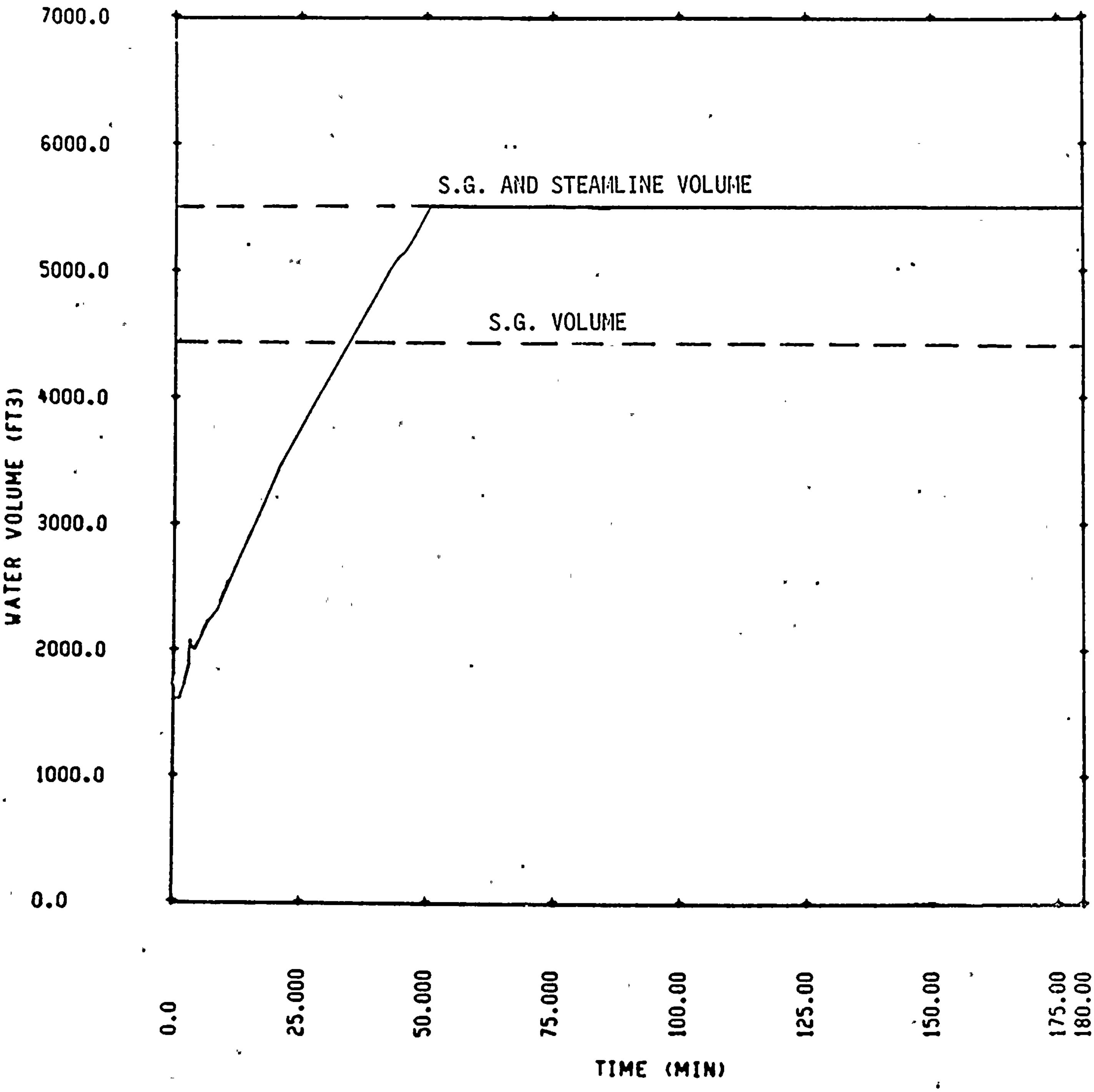


FIGURE II.2-1. CALCULATED FAULTED STEAM GENERATOR WATER VOLUME DURING THE GINNA EVENT.

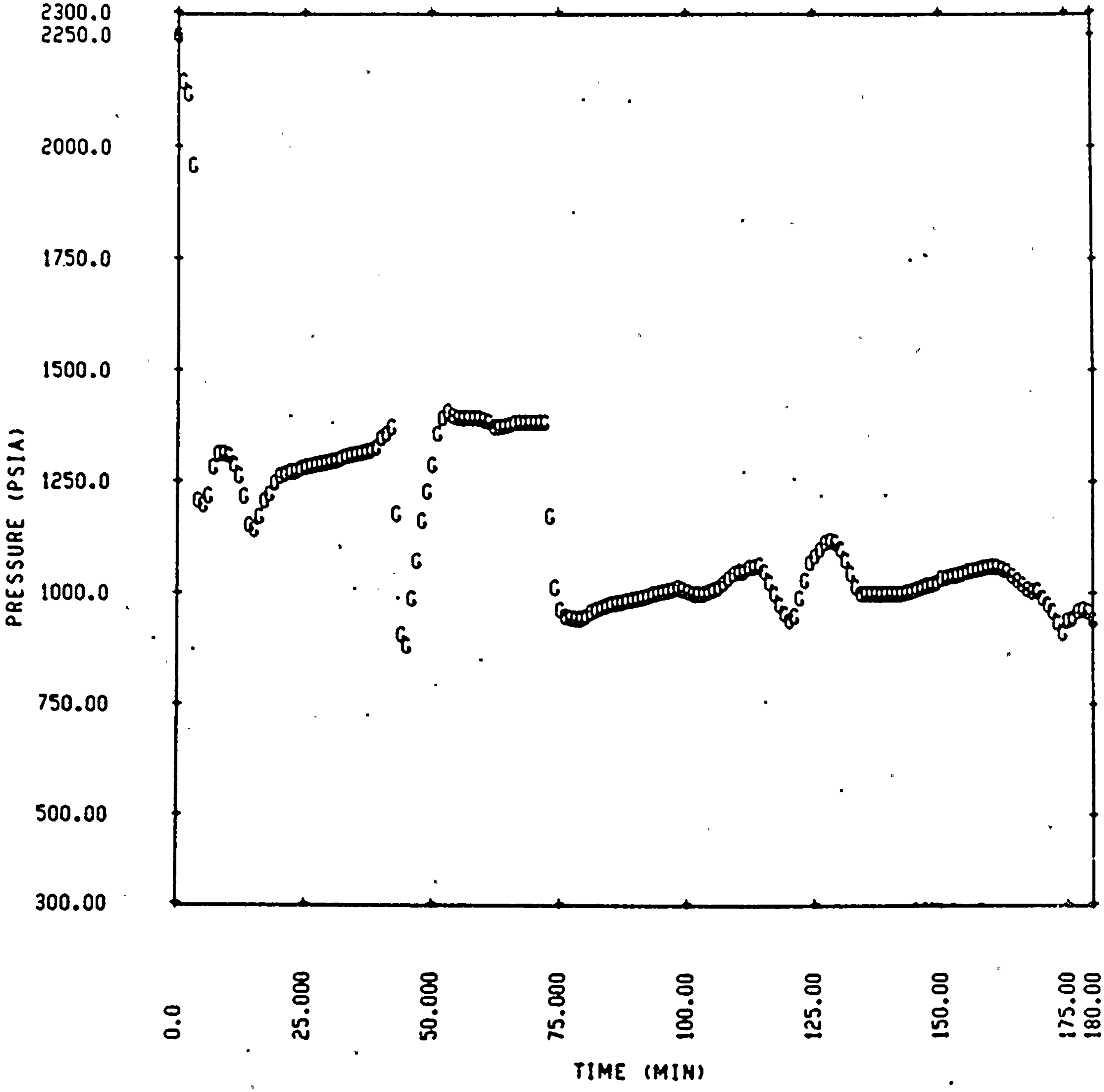


FIGURE II.2-2. REACTOR COOLANT SYSTEM PRESSURE DURING THE GINNA EVENT.

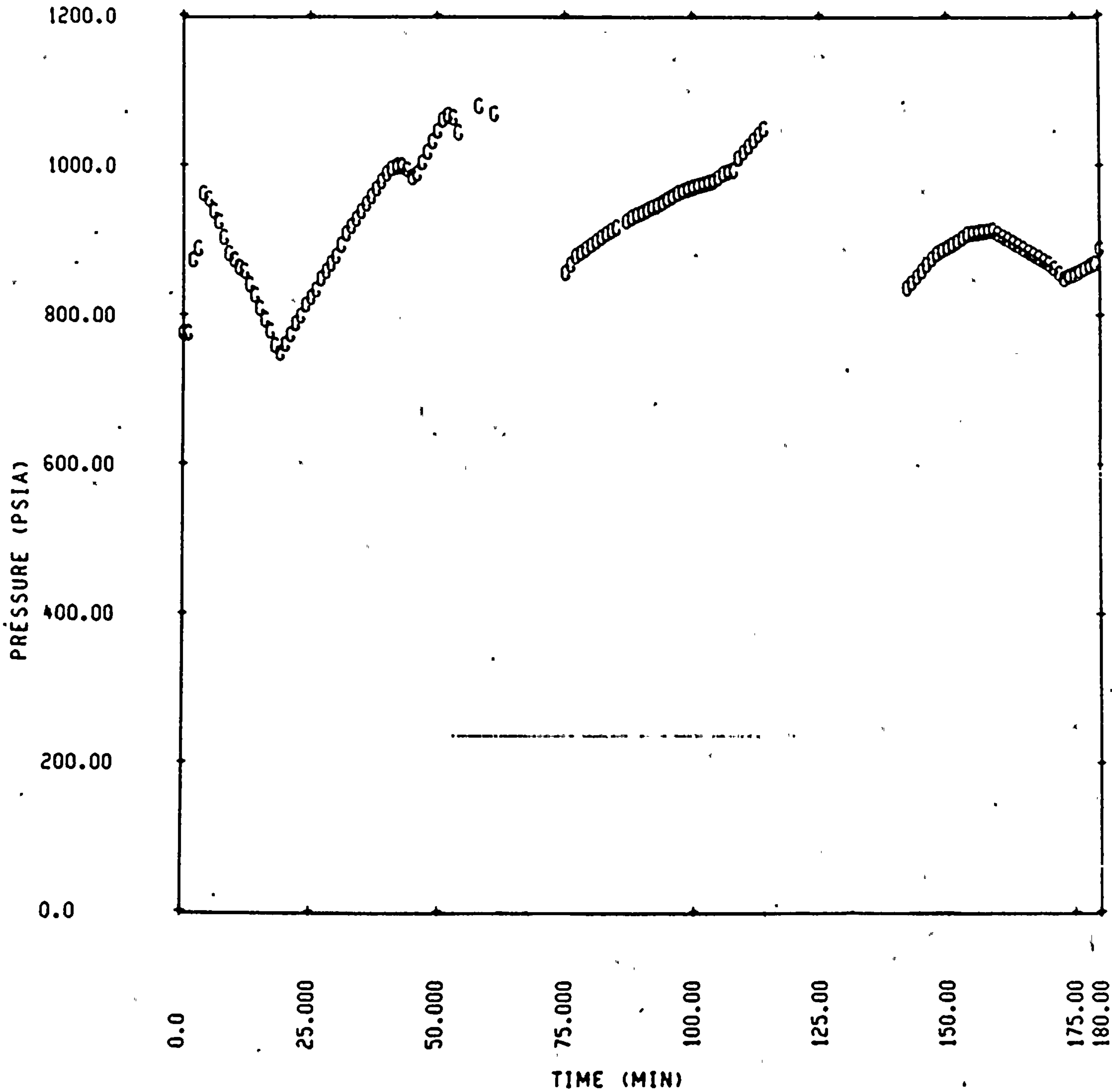


FIGURE II.2-3. FAULTED STEAM GENERATOR PRESSURE DURING THE GINNA EVENT.

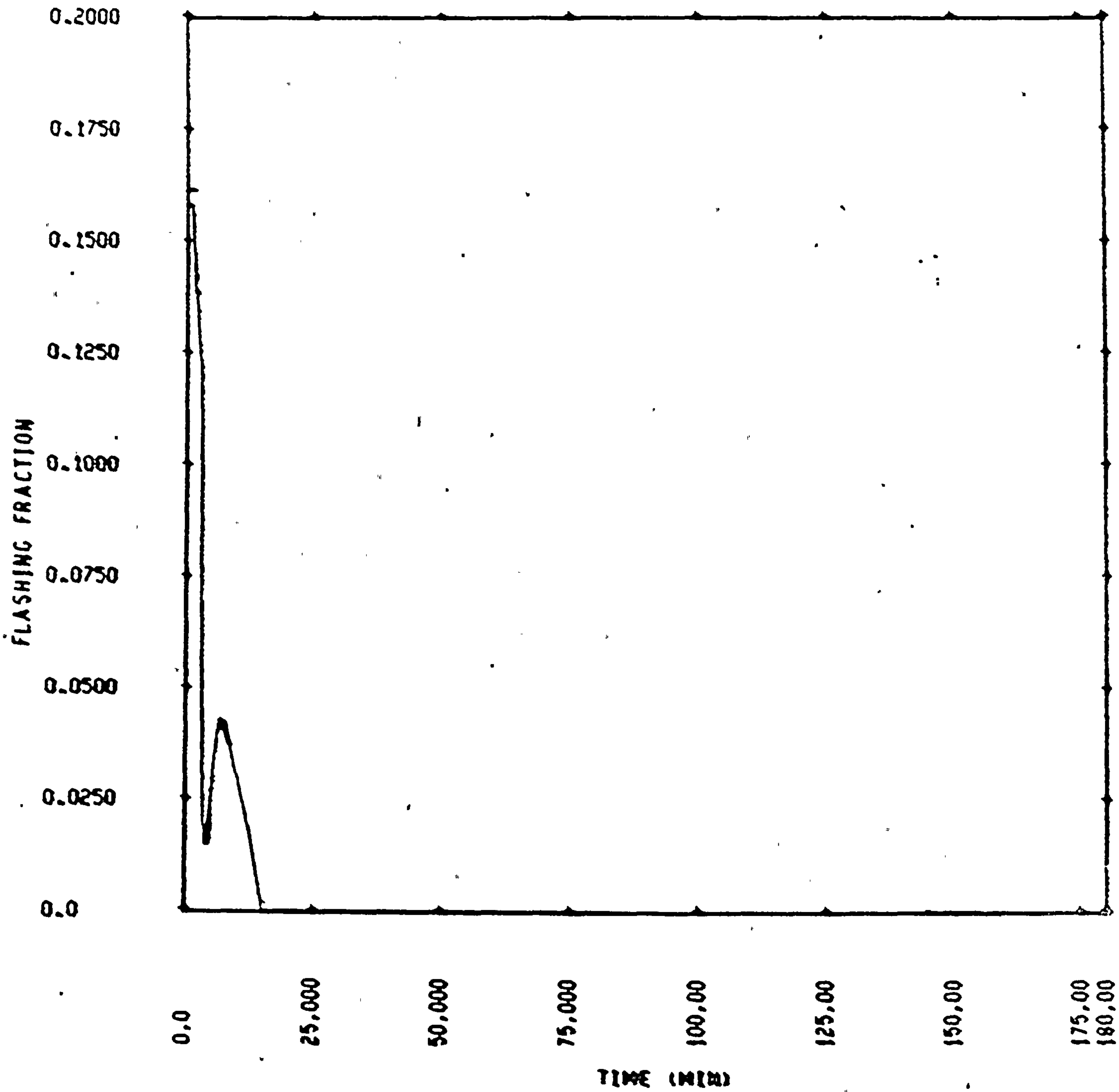


FIGURE II.2-4. CALCULATED BREAK FLOW FLASHING FRACTION DURING THE GINNA EVENT.

are presented in Figures II.2-2 thru II.2-4. These results show that approximately 236,000 lbm of mass were released after the faulted steam generator and steamline was calculated to fill with water. Approximately 130,000 lbm of this were released in the first 2 hrs. Steam flow to condenser was terminated at approximately 75 minutes. Mass releases were terminated when the RHRS was placed in service after 21.5 hrs.

III. ENVIRONMENTAL CONSEQUENCES ANALYSIS

Introduction

For the evaluation of the radiological consequences of a steam generator tube rupture, it is assumed that the reactor has been operting with a small percent of defective fuel for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant. Hence, radionuclides from the primary coolant enter the steam generator, via the ruptured tube, and are released to the atmosphere through the steam generator safety or power operated relief valves.

The radioactivity released to the environment, due to a SGTR, depends upon primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow, time dependent break flow flashing fractions, time dependent scrubbing of flashed activity, partitioning of the activity from the non flashed fraction of the break flow between the steam generator liquid and steam and the mass of fluid discharged to the environment. All of these parameters were conservatively evaluated for a design basis tube failure, i.e. double ended rupture of a single tube, as described in Section II.1. The mass releases during the Ginna event were also estimated in Section II.2. The environmental consequences at these events were calculated and are discussed in the following sections.

III.1 DESIGN BASES ANALYTICAL ASSUMPTIONS

The major assumptions and parameters used in the analysis are itemized in Table III.1-1 and are summarized below.

Source Term Calculations

The concentrations of nuclides in the primary and secondary system, prior to the accident are determined as follows:

a. The iodine concentrations in the reactor coolant will be based upon preaccident and accident initiated iodine spikes.[4]

   i. Preaccident Spike - A reactor transient has occured prior to the SGTR and has raised the primary coolant iodine concentration to 60 μCi/gram of Dose Equivalent I-131.

   ii. Accident Initiated Spike - The reactor trip or primary system depressurization associated with the SGTR creates an iodine spike in the primary system which increases the iodine release rate from the fuel to the primary coolant to a value 500 times greater than the release rate corresponding to the maximum equilibrium primary system iodine concentration of 1μCi/gram of Dose Equivalent (D.E.) I-131. The duration of the spike is assumed to be 4 hours. Iodine appearance rates in the reactor coolant are presented in Table III.1-2. Doses are calculated for both cases of spiking.

b. The noble gas activity in the reactor coolant is based on 1 percent fuel defects, as provided in Table III.1-3.

   The assumption of 1 percent fuel defects for the calculation of noble gas activity is conservative, since 1μCi/gram D.E. I-131 and 1 percent defects cannot exist simultaneously. Iodine activity based on 1 percent defects would be greater than twice the Standard Technical Specification limit.

c. The secondary coolant activity is based on the D.E. of 0.1 μCi/gram of I-131.

d. Iodine at the rupture point is assumed to consist of 99.9 percent elemental and 0.1 percent organic iodine.

Dose Calculations

The following assumptions and parameters are used to calculate the activity released and the offsite doses following a SGTR.

a. The mass of reactor coolant discharged into the secondary system through the rupture and the mass of steam and/or water released from the intact and faulted steam generators, to the environment is presented in Tables II.1.2-2 and 3.

b. The time dependent fraction of rupture flow that flashes to steam and is immediately released to the environment is shown in Figure III-1-1.

c. The time dependent elemental iodine attenuation factor [5] for retention of atomized primary droplets by the moisture separators and dryers and for scrubbing of steam bubbles as they rise from the leak site to the water surface is presented in Figure III.1-2.

Retention by moisture separators and scrubbing are effected by differential pressure (ΔP) across the ruptured tube and water level. Specifically for the first 4 minutes ΔP is assumed to be high (> 1000 psi) and water level low (just above top of tube bundle). For this period, neither retention nor scrubbing is assumed and the overall factor is 1.0. For times greater than 4 minutes, the ΔP decreases to approximately 300 psi and remains constant. For times greater than 4 but less than 32 minutes, retention by the separators is constant and at a maximum. At 32 minutes the separators begin to flood and at 47 minutes the generator is filled. Retention by the separators decreases from the maximum at 32 minutes to zero at 47 minutes. Scrubbing increases with rising water level.

d. The 1 gpm primary to secondary leak is assumed to be split evenly between the steam generators.

e. All noble gas activity in the reactor coolant which is transported to the secondary system via the tube rupture and the primary-to-secondary leakage is assumed to be immediately released to the environment.

f. Case 1 assumes 30 minute operator action to terminate break flow. The liquid level in the faulted SG remains below the moisture separator. Case 2 assumes 60 minute operator action. The moisture separator begins to flood at 32 minutes and the generator is filled at 47 minutes.

g. The elemental iodine partition factor between the liquid and steam of the intact SG is assumed to be 100. The time dependent partition factor for the faulted SG is presented in Figure III.1-3.

h. Offsite power is lost following reactor trip.

i. Eight hours after the accident, the RHR system is assumed to be in operation to cool down the plant. Thus, no additional steam release is assumed.

j. Neither radioactive decay, during release and transport, nor ground deposition of activity was considered.

k. Short-term atmospheric dispersion factors (χ/Q's) for accident analysis and breathing rates are provided in Table III.1-4.

l. Decay constants, average beta and gamma energies and thyroid dose conversion factors are presented in Table III.1-5.

OFFSITE THYROID DOSE CALCULATION MODEL

Offsite thyroid doses are calculated using the equation:

$$D_{Th} = \sum_i DCF_i \sum_j (IAR)_{ij} \quad (BR)_j \quad (\chi/Q)_j$$

where

$(IAR)_{ij}$ = integrated activity of isotope i released* during the time interval j in Ci

and $(BR)_j$ = breathing rate during time interval j in meter^3/second

$(\chi/Q)_j$ = offsite atmospheric dispersion factor during time interval j in second/meter^3

$(DCF)_i$ = thyroid dose conversion factor via inhalation for isotope i in rem/Ci

D_{Th} = thyroid dose via inhalation in rems

OFFSITE TOTAL-BODY DOSE CALCULATIONAL MODEL

Assuming a semi-infinite cloud of beta and gamma emitters, offsite total-body doses are calculated using the equation:

$$D_{TB} = 0.25 \sum_i \bar{E}_i \sum_j (IAR)_{ij} \quad (\chi/Q)_j$$

where

$(IAR)_{ij}$ = Integrated activity of isotope i released* during the j^{th} time interval in Ci

and $(\chi/Q)_j$ = offsite atmospheric dispersion factor during time interval j in second/meter^3

$\overline{E}_i$ = conservatively assumed to be the sum of the beta and gamma energy for the i^{th} isotope in mev/dis.

D_{TB} = total-body dose in rems

* No credit is taken for cloud depletion by ground deposition and radioactive decay during transport to the exclusion area boundary or to the outer boundary of the low-population zone.

Results

Thyroid and Total-Body doses at the Site Boundary and Low Population Zone are presented in Table III.1-6. All doses are within the guidelines of 10CFR100.

TABLE III.1-1

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCES OF
A STEAM GENERATOR TUBE RUPTURE (SGTR)

I. Source Data

a. Core power level, MWt	1520
b. Steam generator tube leakage, gpm	1
c. Reactor coolant iodine activity:	
1. Accident Initiated Spike	Initial activity equal to the dose equivalent of 1.0 μCi/gm of I-131 with an assumed iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500. See Tables III.1-2 and 3.
2. Pre-Accident Spike	An assumed pre-accident iodine spike, which has resulted in the dose equivalent of 60 μCi/gm of I-131 in the reactor coolant.
d. Reactor coolant noble gas activity, both cases	Based on 1-percent failed fuel as provided in Table III.1-3.

TABLE III.1-1 (Sheet 2)

	e.	Secondary system initial activity	Dose equivalent of 0.1 μCi/gm of I-131
	f.	Reactor coolant mass, grams	1.27×10^8
	g.	Steam generator mass (each), grams	3.39×10^7
	h.	Offsite power	Lost
	i.	Primary-to-secondary leakage duration	Case 1 - 30 min Case 2 - 60 min
	j.	Species of iodine 99.9 percent elemental 0.1 percent organic	
II.		Atmospheric Dispersion Factors	See Table III.1-4
III.		Activity Release Data	
	a.	Faulted steam generator	
		1. Reactor coolant discharged to steam generator, lbs.	See Table III.1.2-2 or 3
		2. Flashed reactor coolant, fraction	See Figure III.1-1
		3. Iodine attenuation factor for flashed fraction of reactor coolant	See Figure III.1-2

TABLE III.1-1 (Sheet 3)

4. Total steam release, lbs	See Table III.1.2-2 or 3
5. Iodine partition factor for the nonflashed fraction of reactor coolant that mixes with the initial iodine activity in the steam generator	See Figure III.1-3
6. Location of tube rupture	Top of Bundle
b. Intact steam generator	
1. Primary-to-secondary leakage, lbs/hr	180
2. Flashed reactor coolant, fraction	0
3. Total steam release, lbs	See Table III.1.2-2 or 3
4. Iodine partition factor	100
5. Isolation time, hrs	8

TABLE III.1-2

IODINE APPEARANCE RATES IN THE
REACTOR COOLANT (CURIES/SECOND)
FOR A DESIGN BASIS SGTR

	I-131	I-132	I-133	I-134	I-135
Equilibrium Appearance Rates due to Technical Specification Fuel defects	1.88×10^{-3}	4.44×10^{-3}	3.48×10^{-3}	6.14×10^{-3}	4.68×10^{-3}
Appearance Rates due to an Iodine Spike-500X equilibrium rates	0.94	2.22	1.74	3.07	2.34

TABLE III.1-3

REACTOR COOLANT IODINE AND NOBLE GAS ACTIVITY

Nuclide	*Iodine Activity based on 1 μCi/gram of Dose Equiv. I-131
I-131	0.785 μCi/gram
I-132	0.344
I-133	1.01
I-134	0.204
I-135	0.787
	Noble Gas Activity Based on 1 percent Fuel Defects
Xe-131m	1.8 μCi/gram
Xe-133m	15
Xe-133	240
Xe-135m	0.41
Xe-135	7.98
Xe-138	0.454
Kr-85m	2.04
Kr-85	6.9
Kr-87	1.18
Kr-88	3.58

*Secondary coolant iodine activity is based on 0.1 μCi/gram of Dose Equivalent I-131 and is therefore 10 percent of these values.

TABLE III.1-4

SHORT-TERM ATMOSPHERIC DISPERSION FACTORS AND BREATHING RATES FOR ACCIDENT ANALYSIS

Time (hours)	Site Boundary [6] χ/Q(Sec/m^3)	Low Population [6] Zone χ/Q(Sec/m^3)	Breathing [7] Rate (m^3/Sec)
0-2	4.8×10^{-4}	-	3.47×10^{-4}
0-8	-	3×10^{-5}	3.47×10^{-4}

TABLE III.1-5

ISOTOPIC DATA

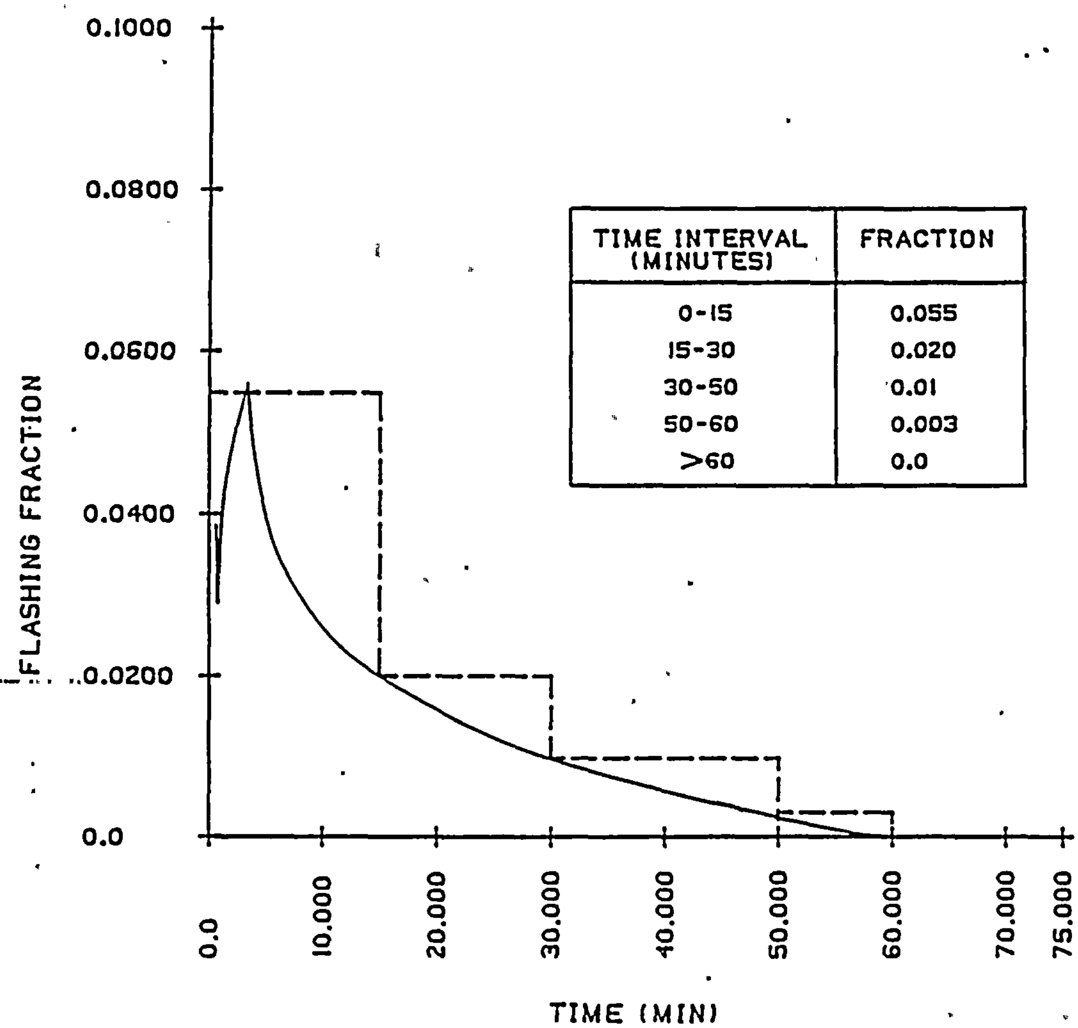
Isotope	Decay Constant (1/Hr)	E_γ (Mev/dis)	E_β (Mev/dis)	DCF[8] (R/Ci)
I-131	0.00359	-	-	1.49(6)
I-132	0.301	-	-	1.43(4)
I-133	0.033	-	-	2.69(5)
I-134	0.800	-	-	3.73(3)
I-135	0.103	-	-	5.60(4)
Xe-131m	0.00245	0.0029	0.165	-
Xe-133m	0.0128	0.020	0.212	-
Xe-133	0.00548	0.03	0.153	-
Xe-135m	2.67	0.43	0.099	-
Xe-135	0.0753	0.25	0.32	-
Xe-138	2.45	1.2	0.66	-
Kr-85m	0.158	0.16	0.25	-
Kr-85	0.00000735	0.0023	0.251	-
Kr-87	0.547	0.793	1.33	-
Kr-88	0.248	2.21	0.25	-

TABLE III.1-6

RESULTS OF DESIGN BASIS ANALYSIS

	Doses (Rem) Case 1	Case 2
1. Accident Initiated Iodine Spike		
Site boundary 0-2 hr.)		
Thyroid	2.9	91.5
Total-body	0.31	0.5
Low Population Zone (0-8 hr)		
Thyroid	0.19	5.7
Total-body	0.02	0.03
2. Pre-Accident Iodine Spike		
Site boundary (0-2 hr)		
Thyroid	22.3	273
Total-body	0.31	0.5
Low Population Zone (0-8 hr)		
Thyroid	1.4	17.1
Total-body	0.02	0.03

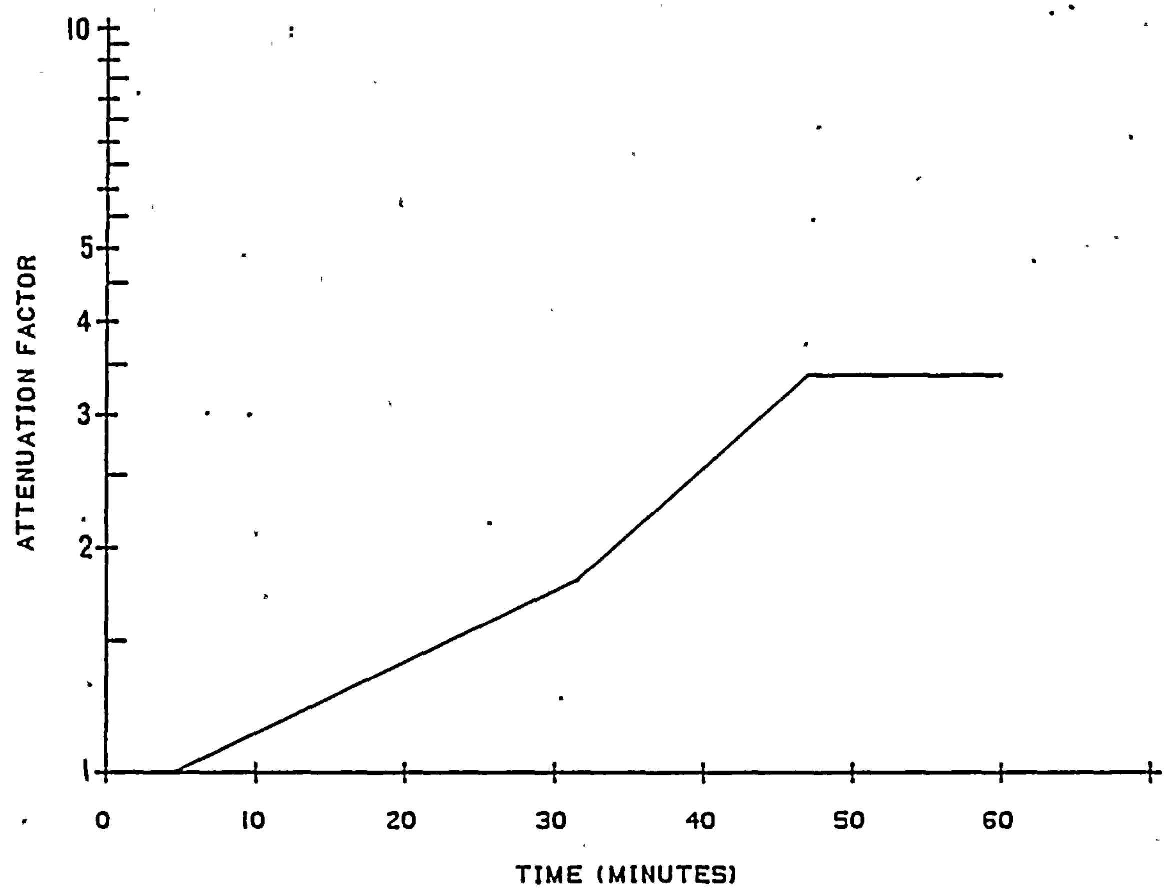
FIGURE: III.1-1



TIME INTERVAL (MINUTES)	FRACTION
0-15	0.055
15-30	0.020
30-50	0.01
50-60	0.003
>60	0.0

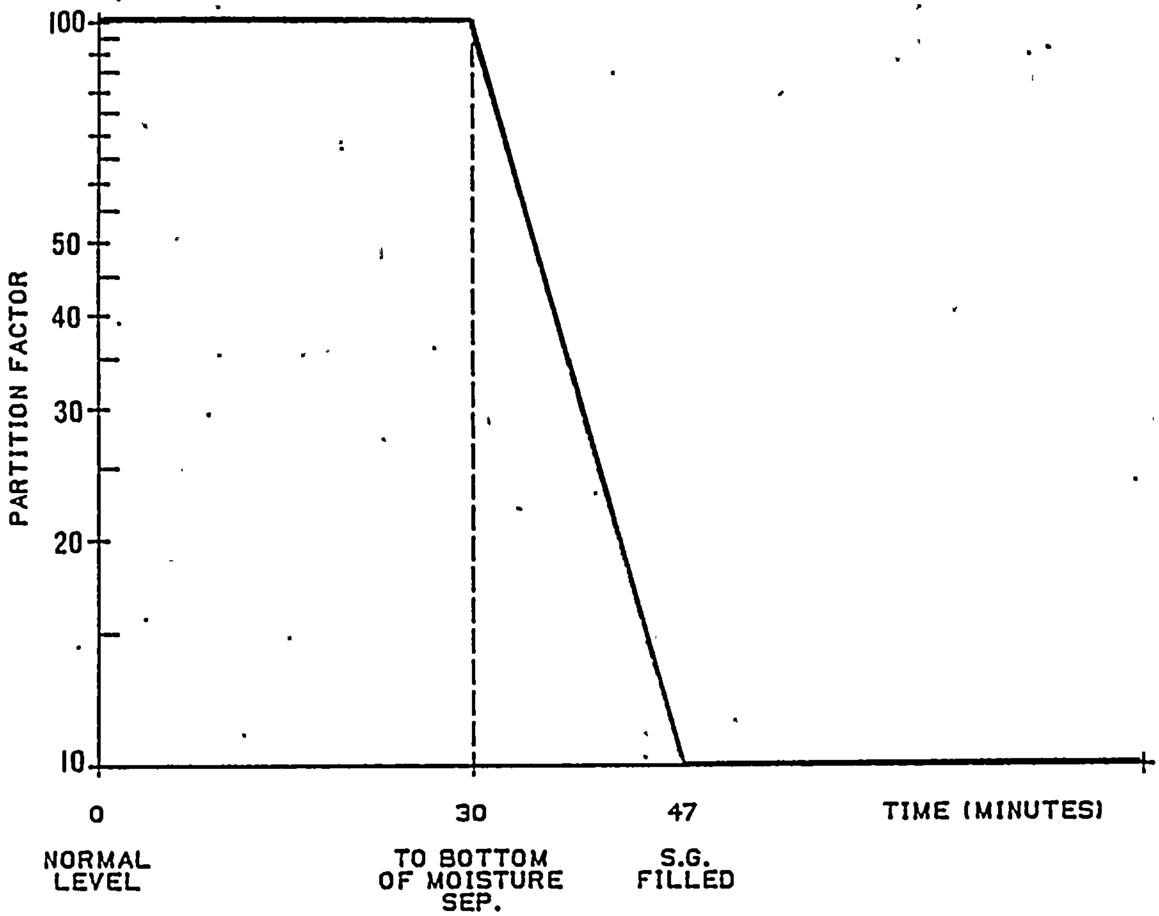
BREAK FLOW FLASHING FRACTION

FIGURE : III.1-2



ATTENUATION FACTOR FOR FLASHED REACTOR COOLANT

FIGURE : III.1-3



FAULTED S.G. PARTITION FACTOR
FOR NON FLASHED REACTOR COOLANT

III.2 Best Estimate Analytical Assumptions

The major assumptions and parameters used in the analysis are itemized in Table III.2-1 and are summarized below.

Source Term Calculations

The concentrations of nuclides in the primary and secondary system, prior to the accident are determined as follows:

a. The iodine concentrations in the reactor coolant will be based upon preaccident and accident initiated iodine spikes.[4]

   i. Preaccident Spike - A reactor transient has occurred prior to the SGTR and has raised the primary coolant iodine concentration to 8 μCi/gram of Dose Equivalent I-131. (The basis for the spiking factors is presented in Ref. 9.)

   ii. Accident Initiated Spike - The reactor trip or primary system depressurization associated with the SGTR creates an iodine spike in the primary system which increases the iodine release rate from the fuel to the primary coolant to a value 30[9] times greater than the release rate corresponding to the maximum equilibrium primary system iodine concentration of 1μCi/gram of Dose Equivalent (D.E.) I-131. The duration of the spike is assumed to be 4 hours. Iodine appearance rates in the reactor coolant are presented in Table 2. Doses are calculated for both cases of spiking.

b. The noble gas activity in the reactor coolant is based on 1-percent fuel defects, as provided in Table 3 of Part III.1.

c. The secondary coolant activity is based on the D.E. of 0.1μ Ci/gram of I-131.

d. Iodine at the rupture point is assumed to consist of 100 percent elemental iodine.

The assumption of 1-percent fuel defects for the calculation of noble gas activity is conservative since 1μCi/gram D.E. I-131 and 1 percent defects cannot exist simultaneously. Iodine activity based on 1 percent defects would be greater than twice the Technical Specification limit.

Dose Calculations

The following assumptions and parameters are used to calculate the activity released and the offsite doses following a SGTR.

a. The mass of reactor coolant discharged into the secondary system through the rupture and the mass of steam and/or water released from the intact and faulted steam generators, to the environment is presented in Table III.2-2.

b. The time dependent fraction of rupture flow that flashes to steam and is immediately released to the environment is shown in Figure III.2-1.

c. The time dependent elemental iodine attenuation factor for retention of atomized primary droplets by the moisture separators and dryers and for scrubbing of steam bubbles as they rise from the leak site to the water surface is presented in Figure III.2-2.

Retention by moisture separators and scrubbung are effected by differential pressure (ΔP) across the ruptured tube and water level. Specifically for the first 5 minutes ΔP is assumed to be high (550 psi) and water level low (top of tube bundle). For this period, retention and scrubbing are assumed and the overall factor is 1.45. For times greater than 5 minutes the ΔP decreases to approximately 450 psi and is assumed constant for the duration of the flashing period. For times greater than 5 but less than 22 minutes, retention by the separators is assumed constant and at a maximum. At 22 minutes the separators begin to flood and at 52 minutes the generator and steam line are filled. Retention by the separators decreases from the maximum at 5 minutes to zero at 36 minutes. Scrubbing increases with rising water level.

d. The 1 gpm primary to secondary leak is assumed to be split evenly between the steam generators.

e. All noble gas activity in the reactor coolant which is transported to the secondary system via the tube rupture and the primary-to-secondary leakage is assumed to be immediately released to the environment.

f. The moisture separator begins to flood at 22 minutes and the generator and steam line are filled at 52 minutes.

g. The elemental iodine partition factor between the liquid and steam of the intact SG is assumed to be 5000. The time dependent partition factor for the faulted SG is presented in Figure III.2-3.

h. Offsite power is available.

i. 21.5 hours after the accident, the RHR system is assumed to be in operation to cool down the plant. Thus, no additional steam release is assumed.

j. Neither radioactive decay, during release and transport, nor ground deposition of activity was considered.

k. Short-term atmospheric dispersion factors (X/Q's) for accident analysis and breathing rates are provided in Table III.2-3.

l. Decay constants, average beta and gamma energies and thyroid dose conversion factors are presented in Table 5 of Part III.1.

Offsite Thyroid and Total-Body Dose Calculational Models

See Part III.1

Results

Thyroid and total-body doses at the site boundary and low population zone are presented in Table III.2-4. All doses are within the guidelines of 10CFR100.

TABLE III.2-1

PARAMETERS USED IN THE BEST ESTIMATE EVALUATION
THE RADIOLOGICAL CONSEQUENCES OF
THE GINNA EVENT

I. Source Data

a.	Core power level, MWt	1520
b.	Steam generator tube leakage, gpm	1
c.	Reactor coolant iodine activity:	
	1. Accident Initiated Spike	Initial activity equal to the dose equivalent of 1.0 μCi/gm of I-131 with an assumed iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 30. See Tables III.2-2, III.1-3.
	2. Pre-Accident Spike	An assumed pre-accident iodine spike, which has resulted in the dose equivalent of 8 μCi/gm of I-131 in the reactor coolant.
d.	Reactor coolant noble gas activity	Based on 1-percent failed fuel As provided in Table III.1-3 of Section III.1
e.	Secondary system initial activity	Dose equivalent of 0.1 μCi/gm of I-131.
f.	Reactor coolant mass, grams	1.27×10^8
g.	Steam generator mass (each) grams	3.39×10^7
h.	Offsite power	Available

TABLE III.2-1 (Continued)

i. Primary-to-secondary leakage duration	185 min
j. Species of iodine	100 percent elemental
II. Atmospheric Dispersion Factors	See Table III.2-3
III. Activity Release Data	
a. Faulted steam generator	
1. Reactor coolant discharged to steam generator, lbs.	See Table II.2-2
2. Flashed reactor coolant, fraction	See Figure III.2-1
3. Iodine attenuation factor for flashed fraction of reactor coolant	See Figure III.2-2
4. Steam and water releases, lbs	See Table II.2-2
5. Iodine partition factor for the nonflashed fraction of reactor coolant that mixes with the initial iodine activity in the steam generator	See Figure III.2-3
6. Location of tube rupture	4 inches above tube sheet
b. Intact steam generator	
1. Primary-to-secondary leakage, lbs/hr	180

TABLE III.2-1 (Continued)

2.	Flashed reactor coolant fraction	0
3.	Total steam release, lbs	See Table II.2-2
4.	Iodine partition factor	5000
5.	Isolation time, hrs	21.55

c. Condenser

1.	Iodine partition factor	5000

TABLE III.2-2

IODINE APPEARANCE RATES IN THE
REACTOR COOLANT (CURIES/SECOND)

	I-131	I-132	I-133	I-134	I-135
Equilibrium Appearance Rates due to Technical Specification Fuel Defects	1.88×10^{-3}	4.44×10^{-3}	3.48×10^{-3}	6.14×10^{-3}	4.68×10^{-3}
Appearance Rates due to an Iodine Spike-30X equilibrium rates	5.64×10^{-2}	1.33×10^{-1}	1.04×10^{-1}	1.84×10^{-1}	1.4×10^{-1}

TABLE III.2-3

SHORT-TERM ATMOSPHERIC DISPERSION FACTORS AND BREATHING RATES FOR ACCIDENT ANALYSES

Time (hours)	Site Boundary[6] χ/Q (Sec/m^3)	Low Population[6] Zone χ/Q (Sec/m^3)	Breathing[7] Rate (m^3/sec)
0-2	4.8×10^{-5}	--	3.47×10^{-4}
0-8	--	3×10^{-6}	3.47×10^{-4}
8-24	--	3×10^{-6}	1.75×10^{-4}

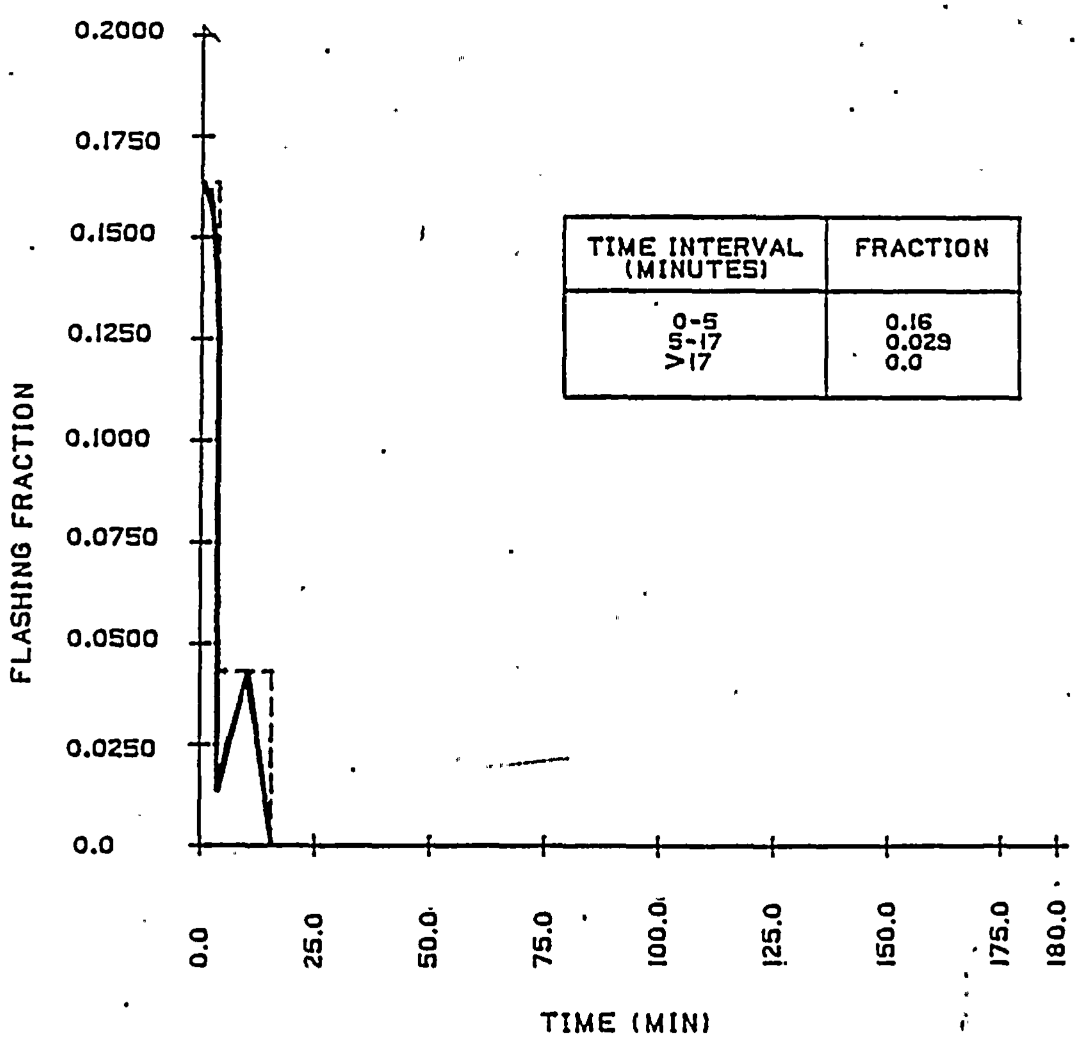
Note: χ/Q's are 10 percent of the R.G. 1.145 values.

TABLE III.2-4

RESULTS OF GINNA EVENT ANALYSES

	Doses (Rem)
1. Accident Initiated Iodine Spike	
Site boundary (0-2 hr)	
Thyroid	2.9
Total-body	0.5
Low Population Zone (0-8 hr)	
Thyroid	1.4
Total-body	0.048
2. Pre Accident Spike	
Site boundary (0-2 hr)	
Thyroid	8.5
Total-body	0.5
Low Population Zone (0-8 hr)	
Thyroid	1.5
Total-body	0.048

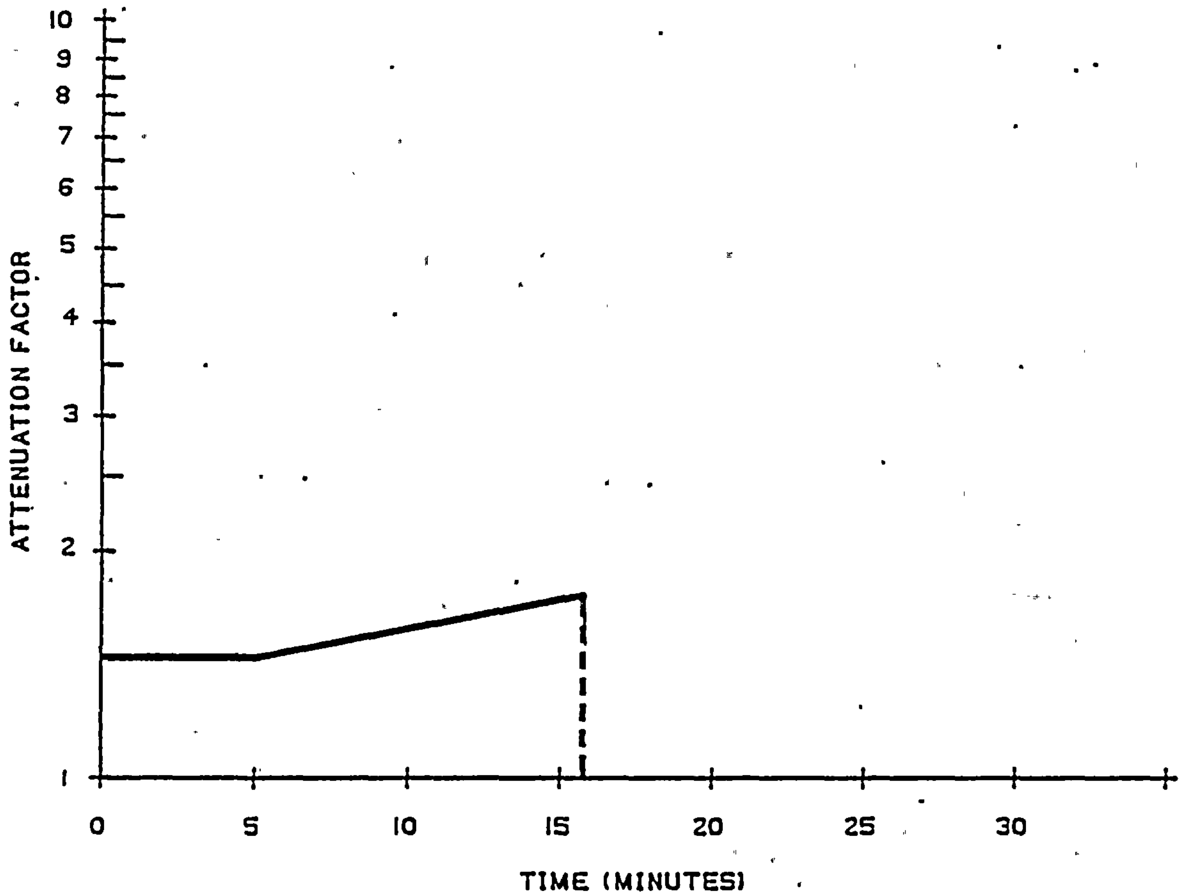
FIGURE : III.2-1



TIME INTERVAL (MINUTES)	FRACTION
0-5	0.16
5-17	0.029
>17	0.0

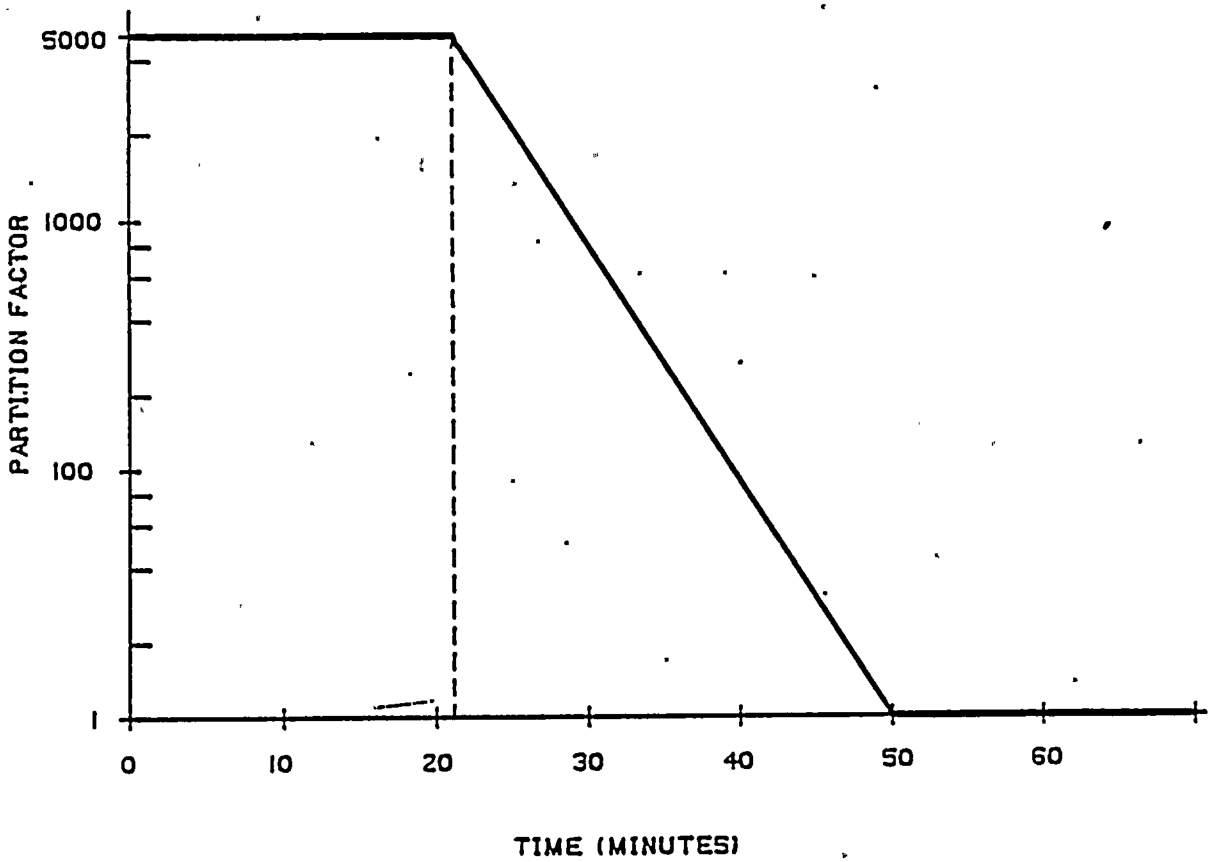
BREAK FLOW FLASHING FRACTION FOR THE GINNA EVENT

FIGURE : III.2-2



ATTENUATION FACTOR FOR FLASHED
REACTOR COOLANT
FOR THE GINNA EVENT

FIGURE : III.2-3



FAULTED S.G. PARTITION FACTOR
FOR THE GINNA EVENT

IV. SUMMARY AND CONCLUSIONS

The potential environmental consequences of a steam generator tube failure at the R. E. Ginna nuclear power plant were evaluated in order to demonstrate that the Standard Technical Specifications limit on primary coolant activity is acceptable. The mass releases during a design basis event, i.e. a double ended rupture of a single tube, were conservatively calculated using the computer code LOFTRAN. For these analyses, the sequence of recovery actions initiated by the tube failure were assumed to be completed on a restricted time scale. Two cases were considered: a) 30 minute recovery, and b) 60 minute recovery. The effect of steam generator overfill on radiological releases was also considered. Mass releases during the design basis event were used with conservative assumptions of coolant activity, meteorology, and attenuation to estimate an upper bound of site boundary and low population zone exposures.

The mass releases from the January 25, 1982 steam generator tube failure at Ginna were also calculated from results presented in reference 2. These releases were used with the Standard Technical Specification limit on initial coolant activity and a more realistic meteorology to evaluate potential doses on a more realistic basis.

Results of the design basis analyses indicate that the conservative site boundary and low population zone exposures from a steam generator tube failure are within 10CFR100 limitations with the Standard Technical Specification limit on initial coolant activity. Estimates of the potential radiological releases from a more realistic event with the same initial coolant activity demonstrate that the design basis analysis is very conservative. Consequently, the Standard Technical Specification limit on coolant activity are sufficient to ensure that the environmental consequences of a steam generator tube failure at the R. E. Ginna plant will be within acceptable limits.

REFERENCES

1. L. A. Campbell, "LOFTRAN CODE DESCRIPTION", WCAP-7878 Rev. 3, January (1977).

2. E. C. Volpenhein, "ANALYSIS OF PLANT RESPONSE DURING JANUARY 26, 1982 STEAM GENERATOR TUBE FAILURE AT THE R. E. GINNA NUCLEAR POWER PLANT", Westinghouse Electric Co., October (1982).

3. WESTINGHOUSE OWNERS GROUP EMERGENCY RESPONSE GUIDELINES SEMINAR, September 1981.

4. NRC Standard Review Plan 15.6-3, Rev. 2, "Radiological Consequences of a Steam Generator Tube Failure", July, 1981.

5. NRC NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident", Postma, A.K., Tam, P.S., Jan. 1978.

6. NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants", August, 1979.

7. NRC Regulatory Guide 1.4, Rev. 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for Pressurized Water Reactors", June 1974.

8. NRC Regulatory Guide 1.109, Rev. 1, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I", Oct. 1977.

9. Lutz, R. J., "Iodine and Cesion Spiking Source Terms for Accident Analysis," WCAP-9964, Rev. 1, July 1981.