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JOHNSON,A.R. Project Directorate I-3

See Reports

SUBJECT: Forwards application for amend to license DPR-18,increasing allowable RCA to improved TS values (NUREG-1431). Westinghouse proprietary rept WCAP-11668 & nonproprietary rept WCAP-11678 encl.Proprietary rept WCAP-11668 withheld.

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May 23, 1994

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Allen R. Johnson
Project Directorate I-3
Washington, D.C. 20555

Subject: Application for Amendment to Operating License
Reactor Coolant Activity Technical Specifications
Rochester Gas and Electric Corporation
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Ref.(a): NRC Letter, C. Rossi to A. Ladieu (WOG), "Acceptance for Referencing of Licensing Topical Report WCAP-10698...", March 30, 1987.

(b): NUREG-0916, "Safety Evaluation Report Related to the Restart of R.E. Ginna Nuclear Power Plant", May 1982.

(c): RG&E Letter, R. Mecredy to A. Johnson (NRC), "Emergency Response Capability...", October 14, 1992.

(d): NRC Letter, A. Johnson to R. Mecredy (RG&E), Emergency Response Capability - Conformance to Regulatory Guide 1.97, revision 3", February 24, 1993.

(e): WOG Letter, L. Walsh and A. Engel to R. Jones (NRC), "Westinghouse Owners Group Steam Generator Tube Uncovery Issue", March 31, 1992.

(f): NRC Letter, R. Jones to L. Walsh (WOG), "Westinghouse Owners Group, Steam Generator Tube Uncovery Issue", March 10, 1993.

Dear Mr. Johnson:

The purpose of this application for Amendment to Operating License is to amend Appendix A of that License to increase the Technical Specification allowable primary coolant iodine (I-131) activity limit from 0.2 μCi/gm to 1.0μCi/gm; and the total primary coolant activity from 84/E μCi/gm to the Standard Technical Specification value of 100/E μCi/gm, consistent with the Improved Technical Specifications (NUREG-1431).

WCAP-11668 (Proprietary)/WCAP-11678 (Non-Proprietary) evaluate the potential radiological consequences due to a steam generator tube rupture (SGTR) for the R. E. Ginna Nuclear Power Plant. This

A001 1/1 Change NRC PDR ltr Encl 1 w/out prop info

evaluation utilizes the analysis methodology of WCAP-10698-P-A "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill." RG&E has met the requirements for usage of WCAP-10698 as outlined in Nuclear Regulatory Commission letter from C. E. Rossi (NRC) to A. E. Ladieu (WOG), (Reference a). The letter required each utility to provide five plant specific inputs:

1. Demonstration that critical operator action times used in the analysis are realistic and consistent with those observed during simulator exercises.

2. A site specific Steam Generator Tube Rupture radiological offsite consequence analysis.

3. A structural analysis of the main steam lines demonstrating adequacy under water-filled conditions.

4. A list of systems, components, and instrumentation credited for accident mitigation and the specified safety grade for each.

5. A comparison of the plant to the "bounding plant" used in WCAP-10698.

Items 2 and 5 are addressed in the site specific analysis being submitted with this letter, WCAP-11668 (Proprietary). Item 3 was addressed during the review of the Steam Generator Tube Rupture incident at Ginna Station. Its acceptability is documented in NUREG-0916, (Reference b). Item 4 is contained in our post-accident instrumentation submittal (Reference c) (re. Regulatory Guide 1.97). NRC review and acceptance of this information is documented in reference d. Item 1 is discussed below.

During the week of August 19 through 23, 1991 simulator runs of the two most limiting cases identified in WCAP-11668 (Proprietary)/ WCAP-11678 (Non-Proprietary), were made to validate assumed operator response times. These simulator runs demonstrated that not only were the operator response times assumed in the WCAPs conservative with adequate margin, but also that recent refinements in the emergency operating procedures have further reduced the likelihood of overfilling the ruptured steam generator. For the two limiting cases identified operator response prevented overfill with significant margin, and terminated break flow in under 45 minutes. In both cases it was necessary to use the WCAP assumed times for operator action outside the control room. These times were the limiting factors in the scenarios, as control room operators had to wait to proceed with further mitigating actions until these actions were assumed to be completed. Operator critique following the sessions indicated that they felt these times were unrealistically conservative. We, therefore, feel that actual mitigation times would be even less than those demonstrated during the simulator exercise.

WCAP-11668/11678 assumes the tube rupture remains covered through-

out the event. More recently it has been suggested that this assumption may not be conservative. The Westinghouse Owners Group (WOG) has undertaken a program to assess the consequences of tube rupture uncovery. The results indicate that the increase in doses is insignificant. These results were transmitted to the staff (Reference e). The staff has reviewed and approved this transmittal (reference f). Therefore, we feel this issue has been adequately addressed.

It should be noted that a proposal to modify Section 3.1.4.3.a has already been submitted to the NRC via letter May 13, 1994. Page 3.1-21 of Attachments B and C is being submitted in both versions, so that approval is not dependent on the sequence of proposed License Amendment Request reviews.

We request that upon NRC approval, this amendment should be considered effective immediately and implemented within 60 days.

Very truly yours,

Robert C. Mecredy

BJF/190
Attachment

Enclosures:

1. One (1) submittal of an Application for Amendment to Operating License and associated safety evaluation.

2. One (1) copy of an Application for Withholding, CAW-87-123, dated December 3, 1987, accompanying Affidavit, and Proprietary Information Notice

3. One (1) copy of WCAP-11668, "LOFTTR2 Analysis of Potential Radiological Consequences Following a Steam Generator Tube Rupture at the R.E. Ginna Nuclear Plant", November 1987 (Proprietary).
4. One (1) copy of WCAP-11678, "LOFTTR2 Analysis of Potential Radiological Consequences Following a Steam Generator Tube Rupture at the R.E. Ginna Nuclear Plant", November 1987 (Non-Proprietary).

xc: Mr. Allen R. Johnson (Mail Stop 14D1)
Project Directorate I-3
Washington, D.C. 20555

U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Ginna Senior Resident Inspector

WCAP-11668

9405310164

LOFTTR2 ANALYSIS OF POTENTIAL RADIOLOGICAL CONSEQUENCES FOLLOWING A STEAM GENERATOR TUBE RUPTURE AT THE R. E. GINNA NUCLEAR POWER PLANT

O. J. Mendler
K. Rubin

NOVEMBER 1987

Nuclear Safety Department

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

This document contains material that is proprietary to the Westinghouse Electric Corporation. This proprietary information has been marked by means of brackets. The basis for marking the material proprietary is identified by marginal notes referring to the standards in Section 8 of the affidavit of R. A. Wiesemann of record "In the Matter of Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors (Docket No. RM-50-1)" at transcript pages 3706 through 3710 (February 24, 1972).

Due to the proprietary nature of the material contained in this report which was obtained at considerable Westinghouse expense and the release of which would seriously affect our competitive position, we request this information to be withheld from public disclosure in accordance with the Rules of Practice, 10 CFR 2.790, and that the information presented therein be safeguarded in accordance with 10 CFR 2.903. We believe that withholding this information will not adversely affect the public interest.

This information is for your internal use only and should not be released to persons or organizations outside the directorate of Regulation and the ACRS without prior approval of Westinghouse Electric Corporation. Should it become necessary to release this information to such persons as part of the review procedure, please contact Westinghouse Electric Corporation and they will make the necessary arrangements required to protect their proprietary interests.

TABLE OF CONTENTS

I. INTRODUCTION

An evaluation of the potential radiological consequences due to a steam generator tube rupture (SGTR) event has been performed for the R. E. Ginna nuclear power plant to demonstrate that the offsite radiation doses will be less than the allowable guidelines based on the Standard Technical Specification limit on primary coolant activity. The evaluation discussed herein assumes that 15% of the steam generator tubes are plugged.

A design basis steam generator tube rupture was analyzed for Ginna using the methodology developed in WCAP-10698 (reference 1) and the supplement to WCAP-10698 (reference 2). Two single failure cases were considered to determine which is the most limiting single failure for Ginna with respect to radiological consequences. The two cases examined were:

a,c

Case 1 [Intact steam generator power operated relief valve (PORV) fails closed and must be locally opened.

Case 2 Ruptured steam generator power operated relief valve fails open and must be locally isolated.]

Plant response to the event was modeled using the LOFTTR2 computer code with conservative assumptions of break size and location, condenser availability and initial secondary water mass in the faulted steam generator. The analysis methodology includes the simulation of the operator actions for recovery from a steam generator tube rupture based on the Westinghouse Owners Group Emergency Response Guidelines, which are the basis for the Ginna Emergency Operating Procedures. The mass releases were calculated with the LOFTTR2 program from the initiation of the event until termination of the break flow. For the time period following break flow termination, steam releases and feedwater flows from the intact and faulted steam generators were determined from a mass and energy balance using the calculated RCS and steam generator conditions at the time of leakage termination. The mass releases for both cases were used to determine the radiation doses at the exclusion area boundary and low population zone assuming that the primary coolant activity is at the Standard Technical Specification limit prior to the accident.

II. THERMAL HYDRAULIC ANALYSIS

Integrated mass releases to the atmosphere and condenser during a steam generator tube rupture event were calculated for various time periods during the accident. This section includes the methods and assumptions used to model the SGTR event and calculate the mass releases, as well as the sequence of events for the recovery.

A. Design Basis Accident

The accident modeled is the complete severance of a steam generator tube located at the tube sheet on the cold leg side. It was also assumed that loss of offsite power occurred at the time of reactor trip, and the worst rod was assumed to be stuck at reactor trip.

a,c

[The most limiting single failure with respect to steam generator overfill was determined by reference 1 to be a failed closed PORV on the intact steam generator. Since Ginna is a two loop plant, the intact steam generator PORV must be locally opened before RCS cooldown can begin. This additional time to locally open the intact steam generator PORV will delay RCS depressurization, causing an increase in total primary to secondary leakage. Consequently, more water will accumulate in the faulted steam generator. The case where the intact steam generator PORV fails to open on demand and must be locally opened will be referred to as Case 1.]

a,c

[The most limiting single failure with respect to offsite doses was determined by reference 2 to be a failed open PORV on the faulted steam generator. Failure of this PORV will cause an uncontrolled depressurization of the faulted steam generator which will increase primary to secondary leakage. Pressure in the ruptured steam generator will remain below that in the primary system until the failed PORV can be isolated, and recovery actions completed. The case where the faulted steam generator PORV fails open and must be locally isolated will be referred to as Case 2.]

B. Conservative Assumptions

Plant responses and mass releases from the intact and faulted steam generator prior to break flow termination were calculated using LOFTTR2. While modeling the SGTR event the following assumptions were made:

1. Reactor Trip on Overtemperature Delta-T

a,c

[A turbine runback can be initiated automatically or the operator can manually reduce the turbine load to attempt to prevent a reactor trip on overtemperature delta-T. Although turbine runback is simulated in this analysis, credit is not taken for delaying reactor trip. Reactor trip is assumed to occur on overtemperature delta-T. Due to the assumed loss of offsite power the condenser is not available for steam releases once the reactor is tripped. Consequently, after reactor trip the steam generator PORVs are used for steam releases. Thus an earlier trip time leads to more steam released to the atmosphere from the faulted and intact steam generators.]

2. Power

a,c

[The initial steam generator water mass decreases with increasing power level. Thus, a lower initial power results in a higher initial steam generator secondary water mass which is conservative. On this basis, 100% nominal power was assumed for the analysis rather than considering an overpower factor.]

3. Pressurizer Water Level

a,c

[The RCS depressurization rate increases after the pressurizer empties. A higher pressurizer water level will increase the time required for the pressurizer to empty. This results in maintaining a higher primary to secondary pressure differential and thus a larger break flow rate for a longer time period. Therefore, it is concluded that maximizing the pressurizer water level is conservative.]

4. Steam Generator Secondary Mass

a,c

[A higher initial secondary water mass in the ruptured steam generator was determined by Reference 1 to be conservative for overfill. For the Case 1 analysis turbine runback was assumed initiated and was simulated by artificially increasing initial steam generator water mass. The initial steam generator total fluid mass was assumed to be 10% above nominal value plus the differential mass between 100% power and 70% power to simulate turbine runback. However, if steam generator overfill does not occur, a lower initial mass in the ruptured steam generator results in a conservative prediction of offsite doses. Thus, for Case 2 the initial secondary mass was assumed to correspond to operation at 10% below nominal steam generator water mass.]

5. Break Location

a,c

[The tube rupture analyzed is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet side of the steam generator. The location of the break on the cold side of the steam generator results in higher primary to secondary leakage than a break on the hot side of the steam generator as determined by reference 1. Therefore, it is concluded that the cold leg side break location is conservative.]

6. Reactor Trip Delay

a,c

[As noted previously, the results for cases which produce an earlier reactor trip are more conservative with respect to the radiological consequences from a SGTR. Thus, a minimum delay time from the reactor trip signal to reactor trip is conservative for the analysis. In addition, as noted above, no credit is taken for turbine runback delaying reactor trip.]

7. Turbine Trip Delay

a,c

[Following a reactor trip, the turbine will be automatically tripped after a suitable delay time. Minimizing the delay time between reactor trip and turbine trip reduces the amount of steam flow to the turbine and increases the secondary water inventory. Thus, a minimum delay time between reactor trip and turbine trip is conservative for the SGTR analysis.]

8. Steam Generator Relief Valve Pressure Setpoint

a,c

[With a loss of offsite power, the secondary pressure will increase following reactor and turbine trip, and steam will be relieved through the steam generator PORVs and safety valves. A lower steam generator relief valve pressure setpoint leads to a higher primary to secondary pressure differential and thus higher primary to secondary leakage. Thus, the use of the lowest setpoint for the steam generator relief valves is conservative for the analysis. Since the PORV setpoint is lower than the safety valve setpoints, the use of the PORV setpoint of 1050 psig for steam relief is more conservative.]

9. Pressurizer Pressure for SI Initiation

a,c

[The use of the maximum pressurizer pressure setpoint for SI initiation results in earlier actuation of the SI system. This leads to the maintenance of a higher primary to secondary pressure differential and consequently a higher break flow rate for a longer time period. Thus, the use of the maximum pressurizer pressure setpoint for SI initiation in the analysis is conservative.]

10. Leakage after Overfill

a,c

[Overfill is assumed to occur if the ruptured steam generator becomes water solid. No credit is taken for steam line volume.]

11. Flashing Fraction

a,c

When calculating the amount of break flow that flashes to steam, 100 percent of the break flow is assumed to come from the hot leg side of the break. Since the tube rupture flow actually consists of flow from the hot leg and cold leg sides of the steam generator, the temperature of the combined flow will be less than the hot leg temperature and the flashing fraction will be correspondingly lower. Thus the assumption that 100 percent of the break flow comes from the hot leg is conservative for a SGTR analysis.

C. Operator Action Times

In the event of an SGTR, the operator is required to take actions to stabilize the plant and terminate the primary to secondary leakage. An evaluation has been performed (reference 1) to establish the operator action times for use in the analysis of a design basis SGTR event. The operator actions which are required for recovery from an SGTR and the available data on the times to perform these actions have been reviewed. The available data on operator action times for an SGTR includes information which has been obtained from reactor plant simulator studies as well as plant data from five actual SGTR events. Using this data, operator action times have been established which are considered to be appropriate for a design basis SGTR event. These operator action times were be used as input for the analysis of the design basis SGTR event.

The major operator actions for SGTR recovery which are included in the E-3 guideline of the Westinghouse Owners Group Emergency Response Guidelines were explicitly modeled in the analysis. The operator actions modeled include identification and isolation of the ruptured steam generator, cooldown and depressurization of the RCS to restore inventory, and termination of SI to stop primary to secondary leakage. These operator actions are described below.

1. Identify the ruptured steam generator (step 2).

   High secondary side activity, as indicated by the steam generator blowdown line radiation monitor or air ejector radiation monitor, typically will provide the first indication of an SGTR event. The ruptured steam generator can be identified by high activity in the corresponding steam generator blowdown line, main steamline, or water sample. For an SGTR that results in a high power reactor trip, the steam generator water level will decrease off-scale on the narrow range for both steam generators. The auxiliary feedwater (AFW) flow will begin to refill the steam generators, typically distributing approximately equal flow to both steam generators. Since primary to secondary leakage adds additional inventory which accumulates in the ruptured steam generator, level will return to the narrow range in that steam generator significantly earlier and will continue to increase more rapidly. This response provides confirmation of an SGTR event and also identifies the ruptured steam generator.

2. Isolate the ruptured steam generator from the intact steam generator and isolate feedwater to the ruptured steam generator.(steps 3 and 4).

   Once a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of filling the ruptured steam generator with water by 1) minimizing the accumulation of feedwater flow and 2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary to secondary leakage. In the guideline for steam generator tube rupture recovery in the ERGs, the operator is directed to maintain the level in the ruptured steam generator between just on span and 50% on the narrow range instrument. Reference 1 assumed that the ruptured steam generator would be isolated when level in the steam generator reached [midway between these points][a,c]

[(33% narrow range level).][a,c] For Ginna it was also conservative to use [33 percent][a,c] of level for isolation. The ruptured steam generator was assumed to be isolated at [33 percent narrow range level or at 10 minutes, whichever was longer.][a,c]

3. Cool down the Reactor Coolant System (RCS) using the intact steam generator (step 14).

   After isolation of the ruptured steam generator, the RCS is cooled as rapidly as possible to less than saturation temperature corresponding to the ruptured steam generator pressure by dumping steam from only the intact steam generator. This ensures adequate subcooling in the RCS after depressurization to the ruptured steam generator pressure in subsequent actions. With offsite power available, the normal steam dump system to the condenser will provide sufficient capacity to perform this cooldown rapidly. If offsite power is lost, the RCS is cooled using the power-operated relief valve (PORV) on the intact steam generator since neither the steam dump valves nor the condenser would be available. It is noted that RCS pressure will decrease during the cooldown as shrinkage of the reactor coolant expands the steam bubble in the pressurizer.

4. Depressurize the RCS to restore reactor coolant inventory (steps 17 or 18).

   When the cooldown is completed, SI flow will increase RCS pressure until break flow matches SI flow. Consequently, SI flow must be terminated to stop primary to secondary leakage. However, adequate reactor coolant inventory must first be assured. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after SI flow is stopped. Since leakage from the primary side will continue after SI flow is stopped until RCS and ruptured steam generator pressures equalize, an "excess" amount of inventory is needed to ensure

pressurizer level remains on span. The "excess" amount required depends on RCS pressure and reduces to zero when RCS pressure equals the pressure in the ruptured steam generator. To reduce break flow and establish sufficient pressurizer level, RCS pressure is decreased by opening the pressurizer PORV.

5. Terminate SI to stop primary to secondary leakage (steps 21-23).

   The previous actions will have established adequate RCS subcooling, secondary side heat sink, and reactor coolant inventory following an SGTR to ensure that SI flow is no longer needed. When these actions have been completed, SI flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. Primary to secondary leakage will continue after SI flow is stopped until RCS pressure and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent repressurization of the RCS and reinitiation of leakage into the ruptured steam generator.

Since these major recovery actions will be modelled in the SGTR analysis, it is necessary to establish the times required to perform these actions. Although the intermediate steps between the major actions will not be explicitly modelled, it is also necessary to account for the time required to perform the steps. It is noted that the total time required to complete the recovery operations consists of both operator action time and system, or plant, response time. For instance, the time for each of the major recovery operations (i.e., RCS cooldown, RCS depressurization, etc.) is primarily due to the time required for the system response, whereas the operator action time is reflected by the time required for the operator to perform the intermediate action steps.

The operator action times to initiate RCS cooldown, RCS depressurization and safety injection termination were developed in reference 1 and are listed in Table II.1. In addition to the operator action times developed in reference 1, Rochester Gas and Electric supplied the operator action

times associated with recovering from the single failures (Reference 3). [These operator actions, which would occur outside the control room, include locally opening the intact steam generator PORV, locally closing the intact steam generator PORV block valve and locally closing the faulted steam generator PORV block valve.][a,c] The times associated with performing these operator actions are listed in Table II.2. [It is noted that the 20 minutes to open the intact steam generator PORV consists of 10 minutes to identify and locate the valve and 10 minutes to linearly open the PORV. Due to limitations of LOFTTR2, the operator action to open the intact steam generator PORV was simulated as a step opening after a 15 minute delay, which results in an equivalent integrated steam flow through the PORV at the end of the 20 minute period.][a,c]

TABLE II.1

OPERATOR ACTION TIMES FOR DESIGN BASIS SGTR ANALYSIS

a,c

Action	Time (min)
Identify and isolate ruptured SG	Maximum of 10 min or calculated time to reach 33% narrow range level in the ruptured SG
Operator action time to initiate cooldown	5
Cooldown	Calculated time for RCS cooldown
Operator action time to initiate depressurization	2
Depressurization	Calculated time for RCS depressurization
Operator action time to initiate SI termination	1
SI termination and pressure equalization	Calculated time for SI termination and equalization of RCS and ruptured SG pressures

TABLE II.2

GINNA SPECIFIC OPERATOR ACTION TIMES

a,c

Action	Time (min)
Faulted Steam Generator PORV Block Valve Closing-Local Action	15
Intact Steam Generator PORV Opening-Local Action	20
Intact Steam Generator PORV Block Valve Closing-Local Action	5

D. Transient Description - Case 1

[Case 1 addresses a SGTR in which the single failure assumed is that the intact steam generator PORV fails to open on demand and must be locally opened.][a,c] The sequence of events for Case 1 is presented in Table II.3.

Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator since the primary pressure is greater than the steam generator pressure. In response to this loss of reactor coolant, pressurizer level decreases as shown in Figure II.1. The RCS pressure also decreases as shown in Figure II.2 as the steam bubble in the pressurizer expands. As the RCS pressure decreases due to the continued primary to secondary leakage, automatic reactor trip occurs on an overtemperature delta-T trip signal.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close and steam flow to the turbine is terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remain closed due to the loss of condenser vacuum resulting from the assumed loss of offsite power at the time of reactor trip. Thus, the energy transfer from the primary system causes the secondary side pressure to increase rapidly after reactor trip until the steam generator PORVs lift to dissipate the energy, as shown in Figure II.3.

The RCS pressure decreases more rapidly after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the leak flow continues to deplete primary inventory. The decrease in RCS inventory results in a low pressurizer pressure SI signal. Pressurizer level also decreases more rapidly following reactor trip and the pressurizer eventually empties as shown in Figure II.1. After the pressurizer empties, the RCS pressure rapidly decreases as shown in Figure II.2.

Since offsite power is assumed lost at reactor trip, the RCPs trip and a gradual transition to natural circulation flow occurs. Immediately following reactor trip the temperature rise across the core decreases as core power decays (see Figure II.4), however, the temperature rise subsequently increases as natural circulation flow develops. The cold leg temperatures trend toward the steam generator temperature as the fluid residence time in the tube region increases. The RCS temperatures continue to slowly decrease due to the continued addition of the auxiliary feedwater to the steam generators until operator actions are initiated to cool down the RCS.

Major Operator Actions

1. Identify and Isolate the Ruptured Steam Generator

   Once a tube rupture has been identified, recovery actions begin by isolating steam flow from the ruptured steam generator and throttling the auxiliary feedwater flow to the ruptured steam generator. As indicated previously, the ruptured steam generator is assumed to be identified and isolated when the narrow range level reaches [33%][a,c] on the ruptured steam generator or at [10][a,c] minutes after initiation of the SGTR, whichever is longer. For Ginna, the time to reach [33%][a,c] is less than [10][a,c] minutes, thus the ruptured steam generator is assumed to be isolated at [10][a,c] minutes.

2. Cool down the RCS to Establish Subcooling Margin

   After isolation of the ruptured steam generator, there is a [5][a,c] minute operator action time imposed prior to initiating the cooldown. The actual delay time used in the analysis is 4 seconds longer because of the computer program requirements for simulating the operator actions. After this time, actions are taken to cool the RCS as rapidly as possible by dumping steam from the intact steam generators. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the PORV on the intact steam generator. [For Case 1, as previously noted, the intact steam generator][a,c]

[PORV fails to open. For the analysis, it was assumed that the valve was fully opened after an additional period of 15 minutes (see pg. 10). Thus at 1804 seconds the intact steam generator PORV is opened for RCS cooldown.][a,c] The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance of 17°F for subcooling uncertainty. [When these conditions are satisfied it is assumed that it takes the operator five minutes to close the intact steam generator PORV block valve.][a,c] It is noted that overfill of the ruptured steam generator occurs during this time period at [2372][a,c] seconds, as shown in Figure II.6. This cooldown ensures that there will be adequate subcooling in the RCS after the subsequent depressurization of the RCS to the ruptured steam generator pressure. The reduction in the intact steam generator pressure required to accomplish the cooldown is shown in Figure II.3, and the effect of the cooldown on the RCS temperature is shown in Figure II.4. The RCS pressure also decreases during this cooldown process due to shrinkage of the reactor coolant as shown in Figure II.2.

3. Depressurize RCS to Restore Inventory

After the RCS cooldown, a [2][a,c] minute operator action time is included prior to depressurization. The RCS is depressurized at 2688 seconds to assure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by opening a pressurizer PORV. The depressurization is continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than 0% plus an allowance of 3% for pressurizer level uncertainty, or pressurizer level is greater than 80% minus an allowance of 3% for pressurizer level uncertainty, or RCS subcooling is less than the 17°F allowance for subcooling uncertainty. The RCS depressurization reduces the break flow as shown in Figure II.5 and increases SI flow to refill the pressurizer, as shown in Figure II.1.

4. Terminate SI to Stop Primary to Secondary Leakage

The previous actions should have established adequate RCS subcooling, verified a secondary side heat sink, and restored the reactor coolant inventory following a SGTR to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated when the RCS pressure increases, minimum AFW flow is available and at least one intact steam generator level is in the narrow range, RCS subcooling is greater than the 17°F allowance for subcooling uncertainty, and the pressurizer level is greater than the 3% allowance for pressurizer level uncertainty. To assure that the RCS pressure is increasing, SI was not terminated in the analysis until the RCS pressure increased to 50 psi above the ruptured steam generator pressure.

After depressurization is completed, an operator action time of [1]^{a,c} minute is imposed prior to SI termination. The primary to secondary leakage continues after the SI flow is terminated until the RCS and ruptured steam generators equalize. This occurs when the intact steam generator PORV is locally opened to cooldown the RCS so that subcooling may be maintained. When the PORV is opened the increased energy transfer from primary to secondary depressurizes the RCS to the ruptured steam generator pressure.

TABLE II.3

SEQUENCE OF EVENTS

CASE 1

a,c

EVENT	Time (sec)
Reactor Trip	50
Ruptured SG Isolated	600
Intact SG PORV Opened	1804
Overfill Ruptured SG	2372
Intact SG PORV Isolated	2568
PRZR PORV Opened	2688
PRZR PORV Closed	2738
SI Terminated	2798
Intact PORV Opened	3288
Break Flow Terminated	3428

a,c

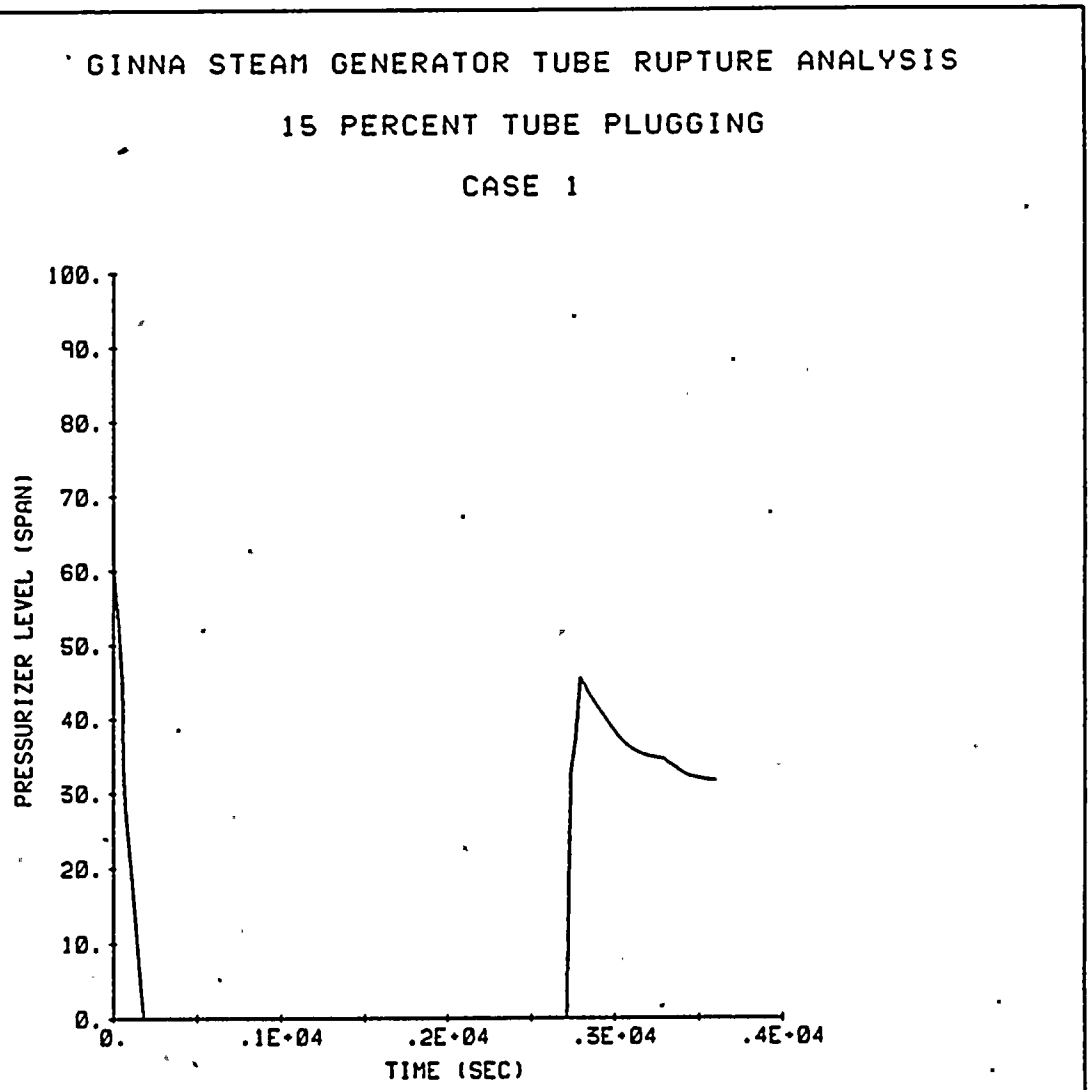


Figure II.1 Pressurizer Level - Case 1

a,c

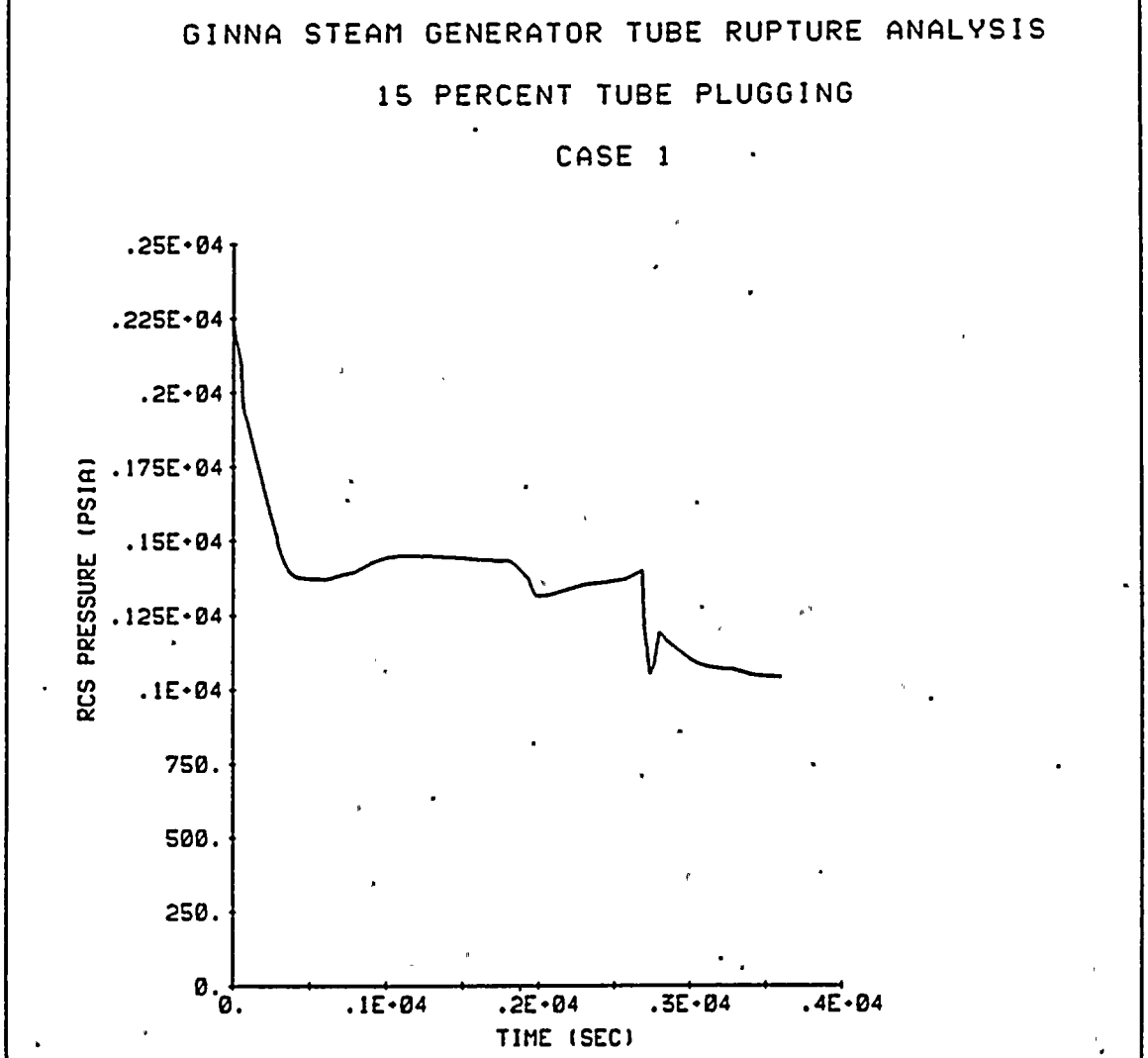


Figure II.2 RCS Pressure - Case 1

a,c

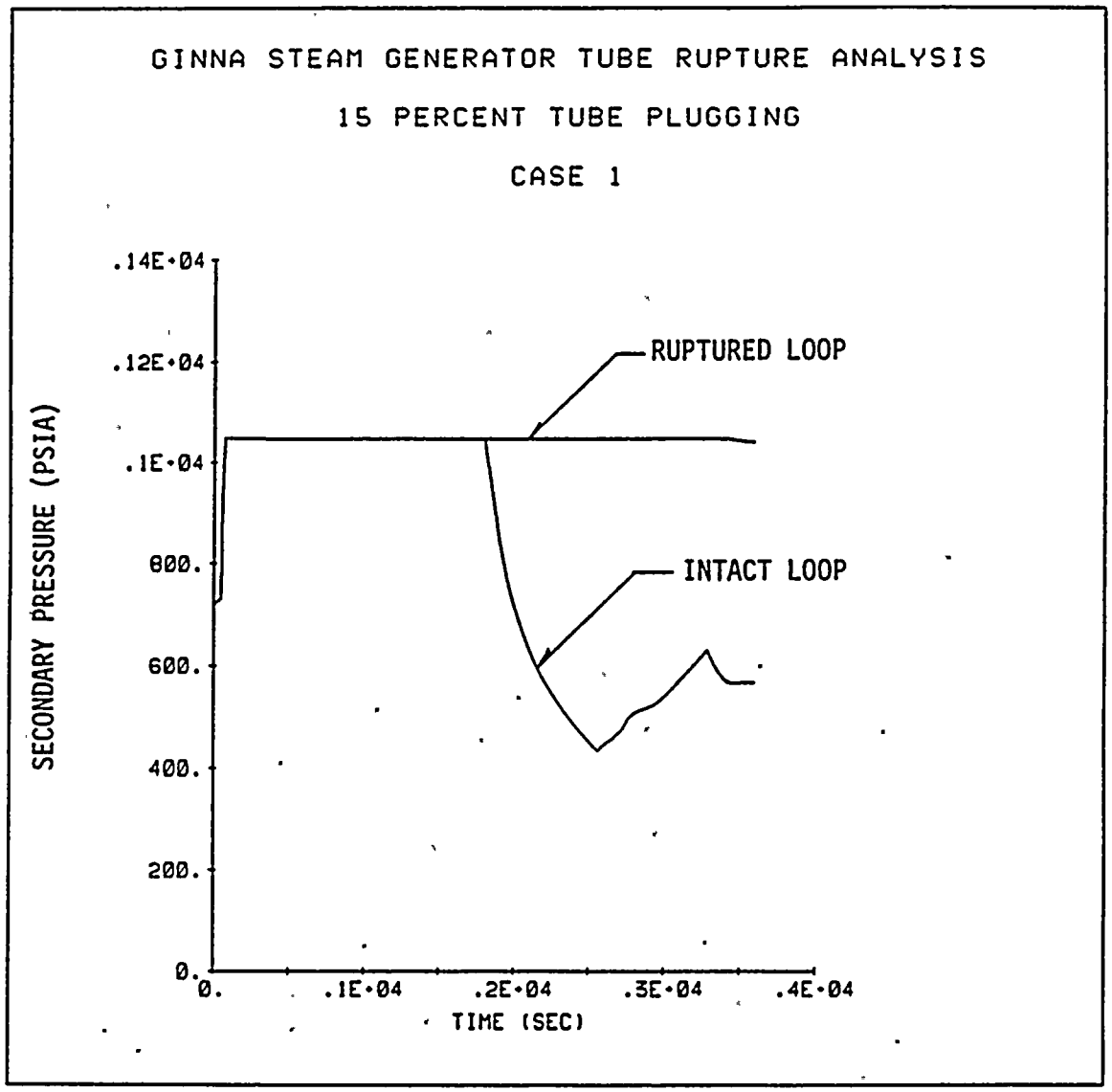


Figure II.3 Secondary Pressure - Case 1

a,c

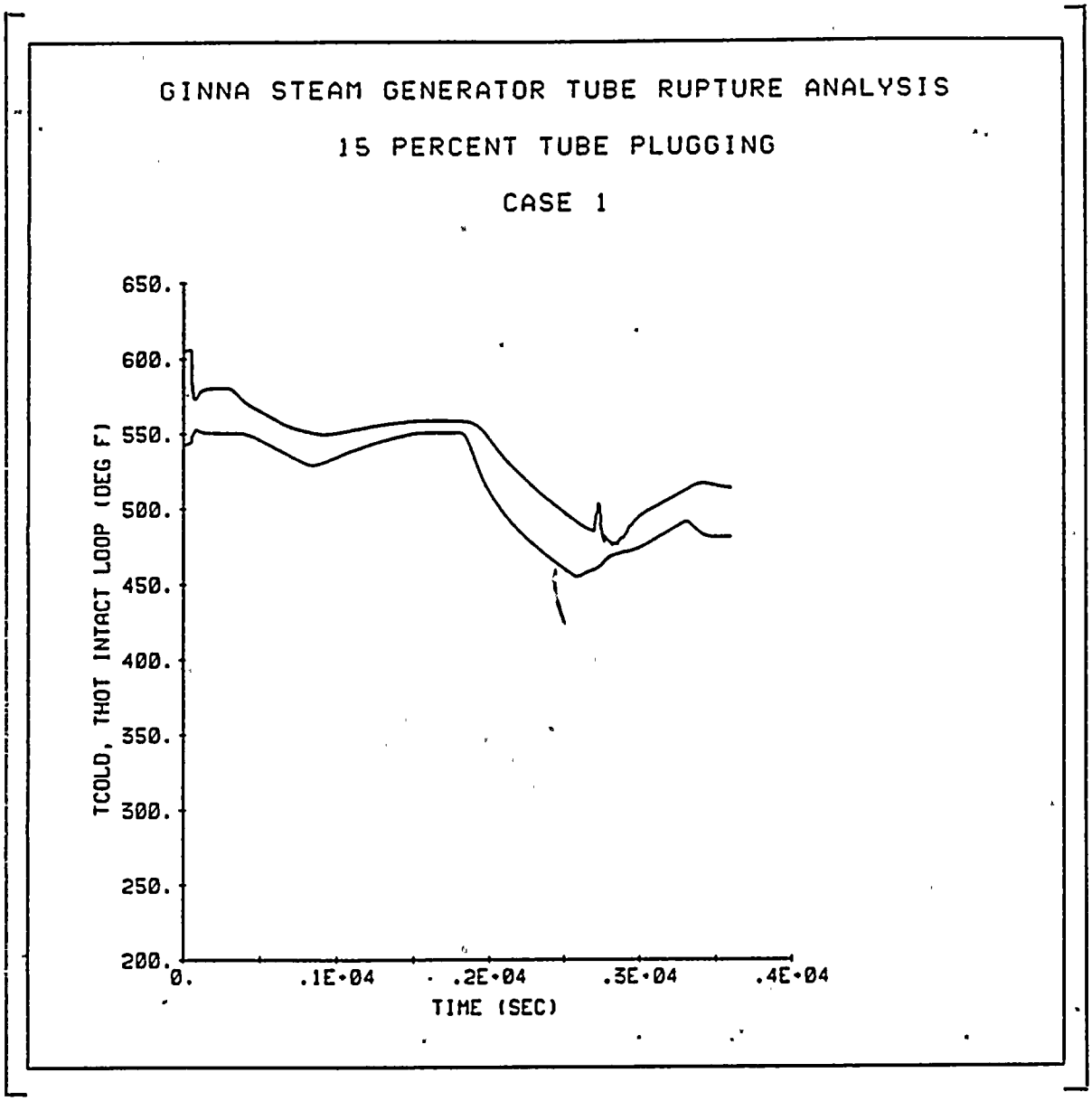


Figure II.4 Intact Loop Hot and Cold Leg RCS Temperatures - Case 1

a,c

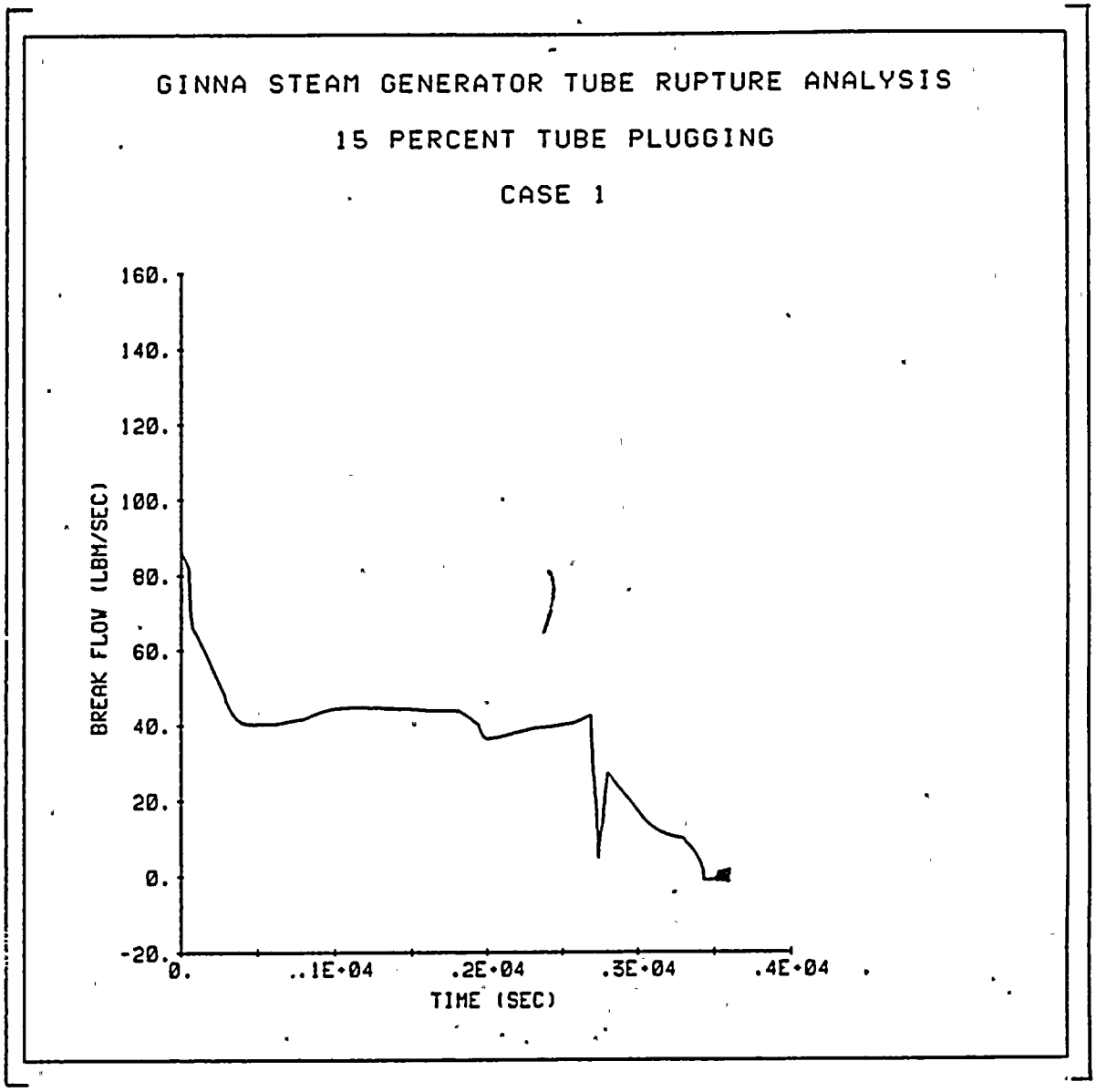


Figure II.5 Primary to Secondary Leakage - Case 1

a,c

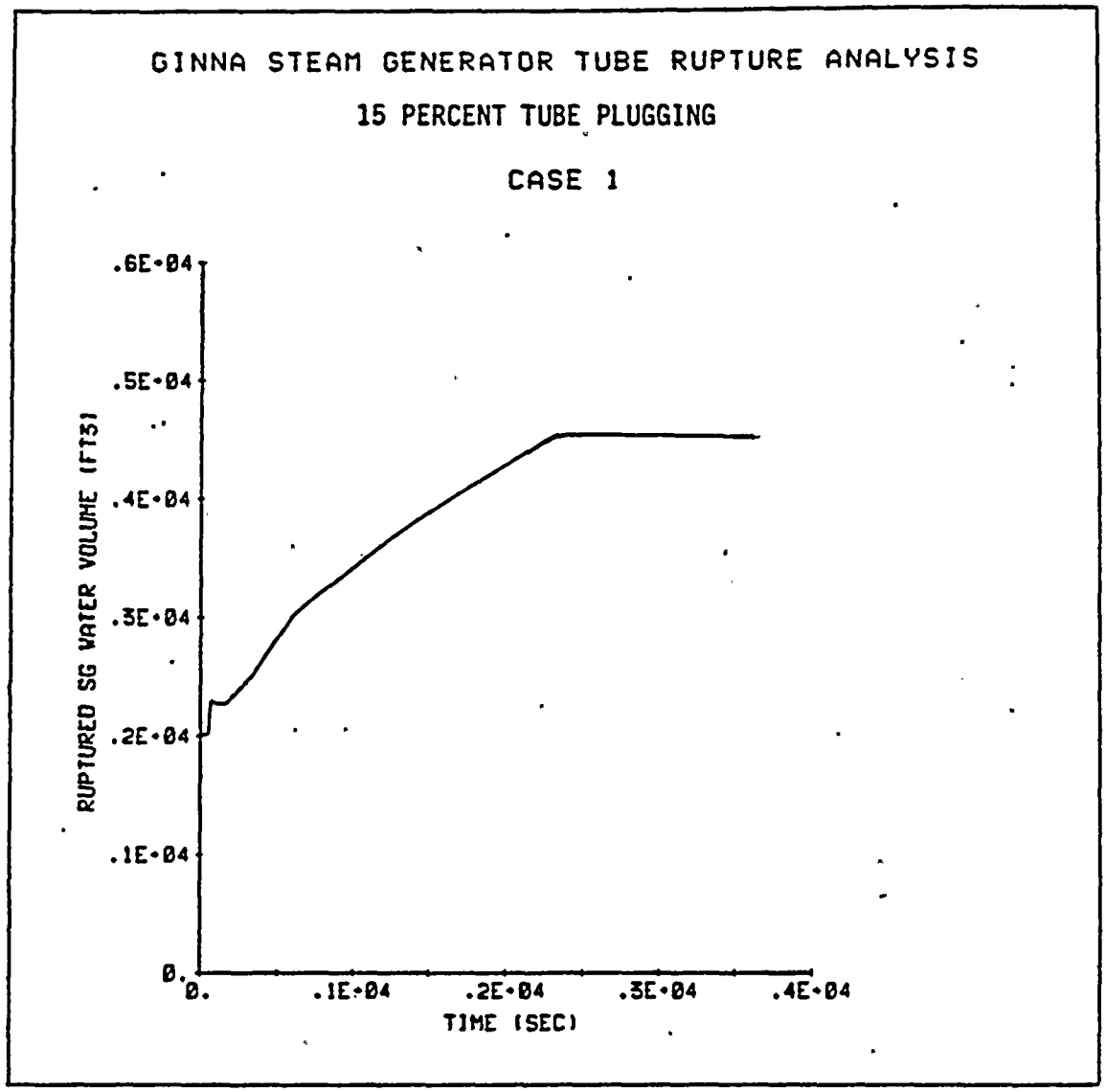


Figure II.6 Ruptured SG Water Volume - Case 1

E. Transient Description - Case 2

[Case 2 addresses a SGTR in which the single failure assumed is that the faulted steam generator PORV fails open at the time the faulted steam generator is isolated.][a,c] Thus, the Case 2 transient is similar to the Case 1 transient until that time. The sequence of events for Case 2 is presented in Table II.4.

Following the tube rupture the RCS pressure decreases as shown in Figure II.7 due to the primary to secondary leakage. In response to this depressurization, the reactor trips on overtemperature delta-T. After reactor trip, core power rapidly decreases to decay heat levels and the RCS depressurization becomes more rapid. The steam dump system is inoperable due to the assumed loss of offsite power, which results in the secondary pressure rising to the steam generator PORV setpoint as shown in Figure II.8. The decreasing pressurizer pressure leads to an automatic SI signal on low pressurizer pressure. Pressurizer level also decreases more rapidly following reactor trip until it eventually empties, as shown in Figure II.9.

Major Operator Actions

1. Identify and Isolate the Ruptured Steam Generator

   The ruptured steam generator is assumed to be identified and isolated at [10 minutes after the initiation of the SGTR or when the narrow range level reaches 33%, whichever time is greater.][a,c] For this case, the time to reach [33%][a,c] narrow range level is [652][a,c] seconds, and thus, it was assumed that the ruptured steam generator is isolated at that time. [The ruptured steam generator PORV is also assumed to fail open at this time.][a,c] The failure causes the steam generator to rapidly depressurize, which results in an increase in primary to secondary leakage. The depressurization of the ruptured steam generator increases the break flow and energy transfer from primary to secondary which results in RCS pressure and temperature decreasing more rapidly than in Case 1. The ruptured steam generator

depressurization causes a cooldown in the intact steam generator loop. As the intact steam generator hot leg temperature decreases below the steam generator water temperature reverse heat transfer takes place as shown in Figure II.10. [It is assumed that the time required for the operator to identify that the ruptured steam generator PORV is open and to close the associated block valve is 15 minutes. Thus, at 1558 seconds the depressurization of ruptured steam generator is terminated.][a,c]

2. Cool Down the RCS to establish Subcooling Margin

[After the ruptured steam generator PORV block valve is closed, there is a 5 minute operator action time imposed prior to initiation of cooldown.][a,c] The depressurization of the ruptured steam generator affects the RCS cooldown target temperature since the temperature is dependent upon the pressure in the ruptured steam generator. Since offsite power is lost the RCS is cooled by dumping steam to the atmosphere using the intact steam generator PORV. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance of 17°F for instrument uncertainty. Because of the lower pressure in the ruptured steam generator the associated temperature the RCS must be cooled to is also lower, which has the net effect of extending the time for cooldown. For Case 2 cooldown begins at [1858][a,c] seconds and is completed at [2852][a,c] seconds.

The reduction in the intact steam generator pressure required to accomplish the cooldown is shown in Figure II.8, and the effect of the cooldown on the RCS temperature is shown in Figure II.10. The RCS pressure also decreases during this cooldown process due to shrinkage of the reactor coolant as shown in Figure II.7.

3. Depressurize to Restore Inventory

After the RCS cooldown, a [2][a,c] minute operator action time is included prior to depressurization. The RCS is depressurized at [2974][a,c] seconds to assure adequate coolant inventory prior to

terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by opening a pressurizer PORV. The depressurization is continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than 0% plus an allowance of 3% for pressurizer level uncertainty, or pressurizer level is greater than 80% minus an allowance of 3% for pressurizer level uncertainty, or RCS subcooling is less than the 17°F allowance for subcooling uncertainty. The RCS depressurization reduces the break flow as shown in Figure II.11 and increases SI flow to refill the pressurizer, as shown in Figure II.9.

4. Terminate SI to Stop Primary to Secondary Leakage

The previous actions should have established adequate RCS subcooling, verified a secondary side heat sink, and restored the reactor coolant inventory following an SGTR to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated when the RCS pressure increases, minimum AFW flow is available and at least one intact steam generator level is in the narrow range, RCS subcooling is greater than the 17°F allowance for subcooling uncertainty, and the pressurizer level is greater than the 3% allowance for pressurizer level uncertainty. To assure that the RCS pressure is increasing, SI was not terminated until the RCS pressure increased to 50 psi above the ruptured steam generator pressure.

After depressurization is completed, an operator action time of [1]^{a,c} minute is imposed prior to SI termination. Figure II.11 shows that the primary to secondary leakage continues after the SI flow is stopped until the RCS and ruptured steam generator pressure equalize. The ruptured steam water volume is shown in Figure II.12. For Case 2, the ruptured steam generator does not overfill.

TABLE II.4

SEQUENCE OF EVENTS

CASE 2

a,c

EVENT	TIME (sec)
Reactor Trip	49.4
Ruptured SG Isolated	652
Ruptured SG PORV Fails Open	656
Ruptured SG Block Valve Closed	1558
Intact SG PORV Opened	1858
Intact SG PORV Closed	2852
PRZR PORV Opened	2974
PRZR PORV Closed	3006
SI Terminated	3066
Break Flow Terminated	3438

a,c

a,c

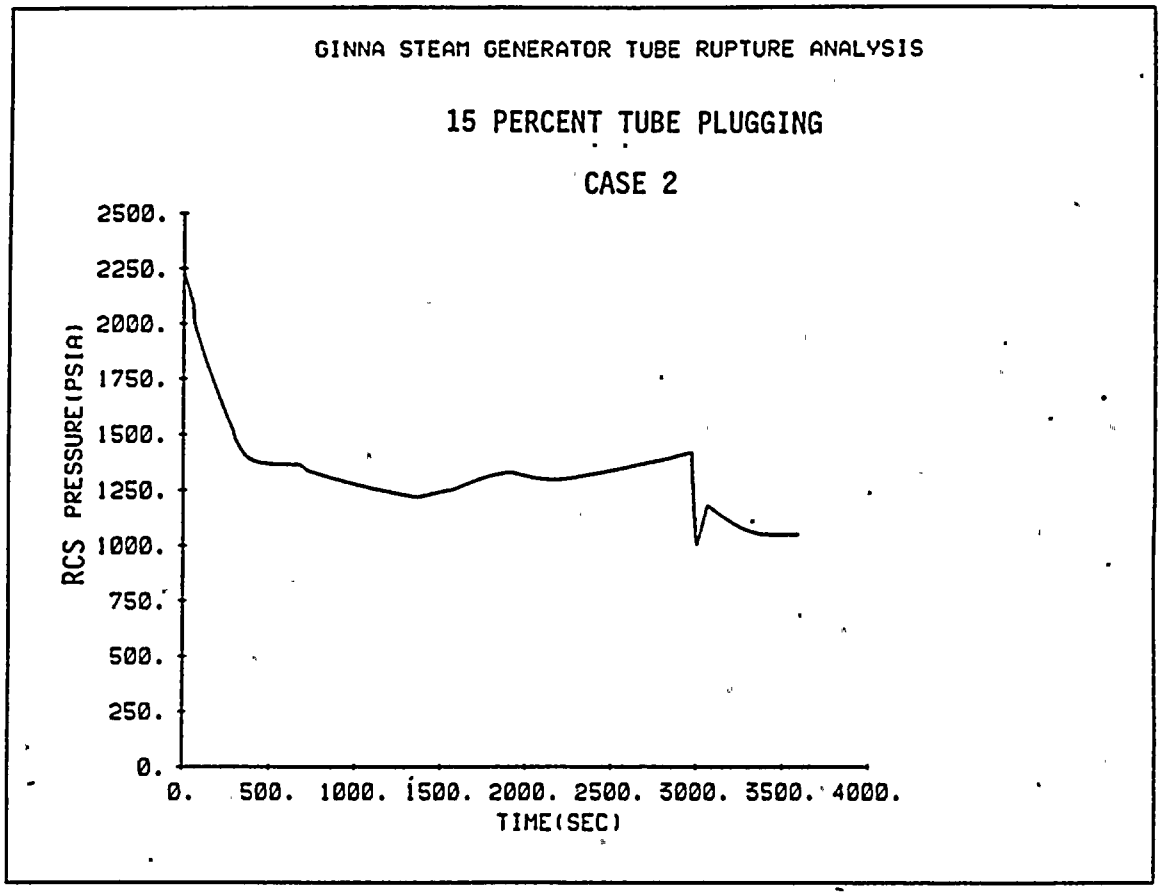


Figure II.7 RCS Pressure - Case 2

a,c

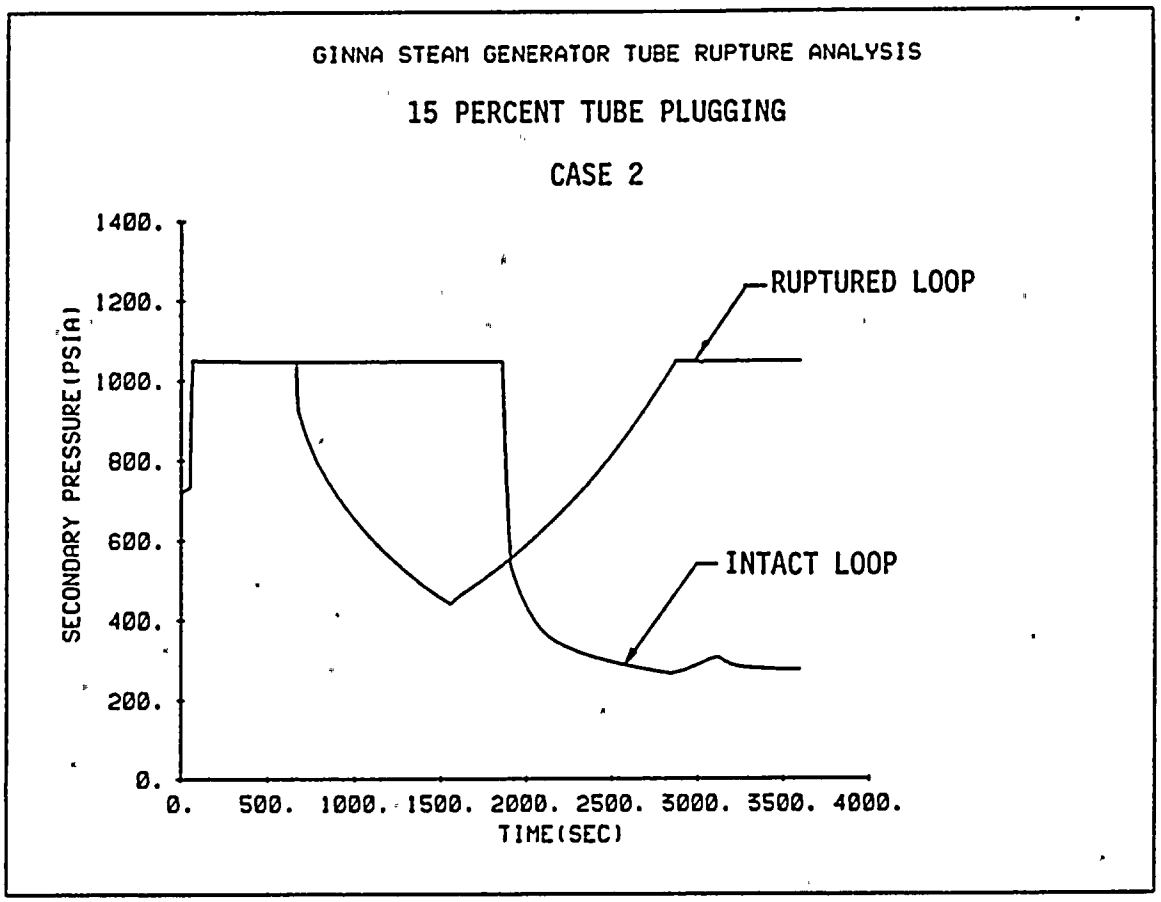


Figure II.8 Secondary Pressure - Case 2

a,c

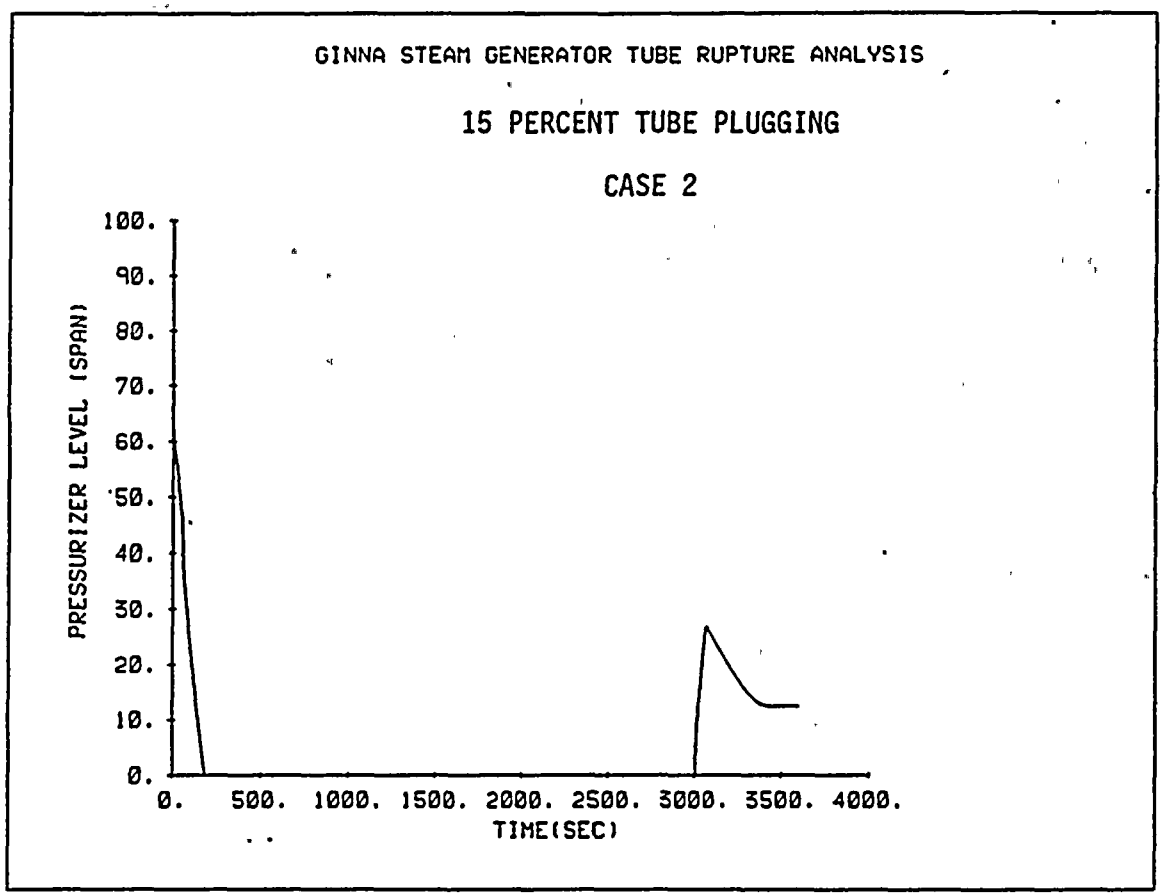


Figure II.9 Pressurizer Level - Case 2

a,c

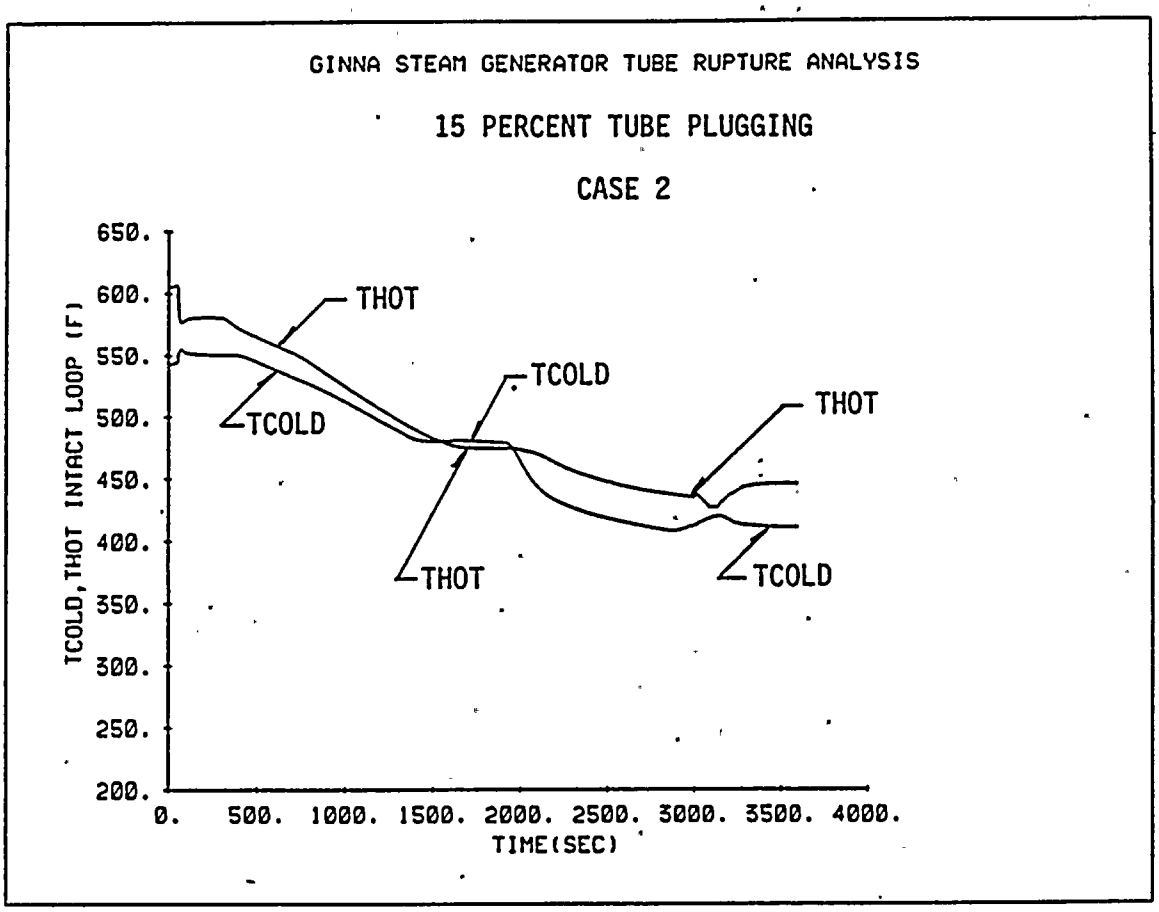


Figure II.10 Intact Loop Hot and Cold Leg RCS Temperatures - Case 2

a,c

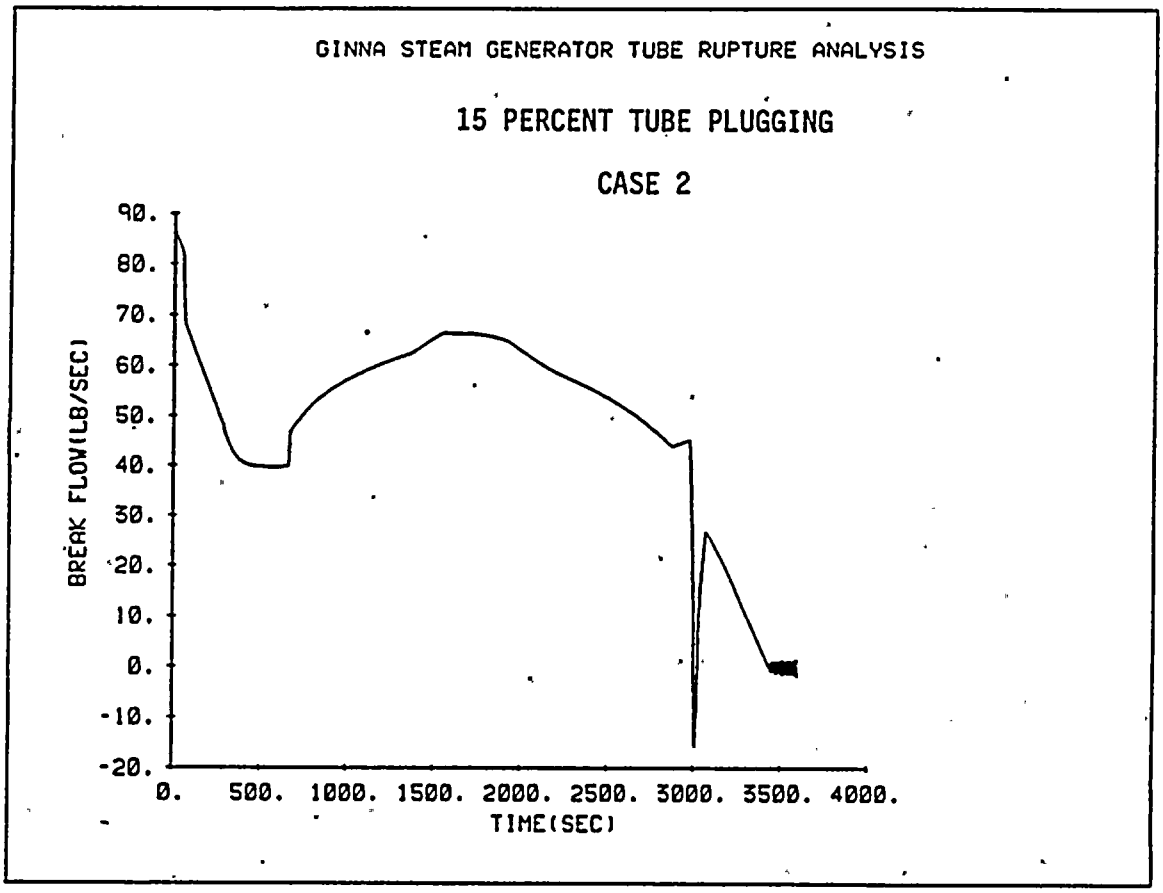


Figure II.11 Primary to Secondary Leakage - Case 2

a,c

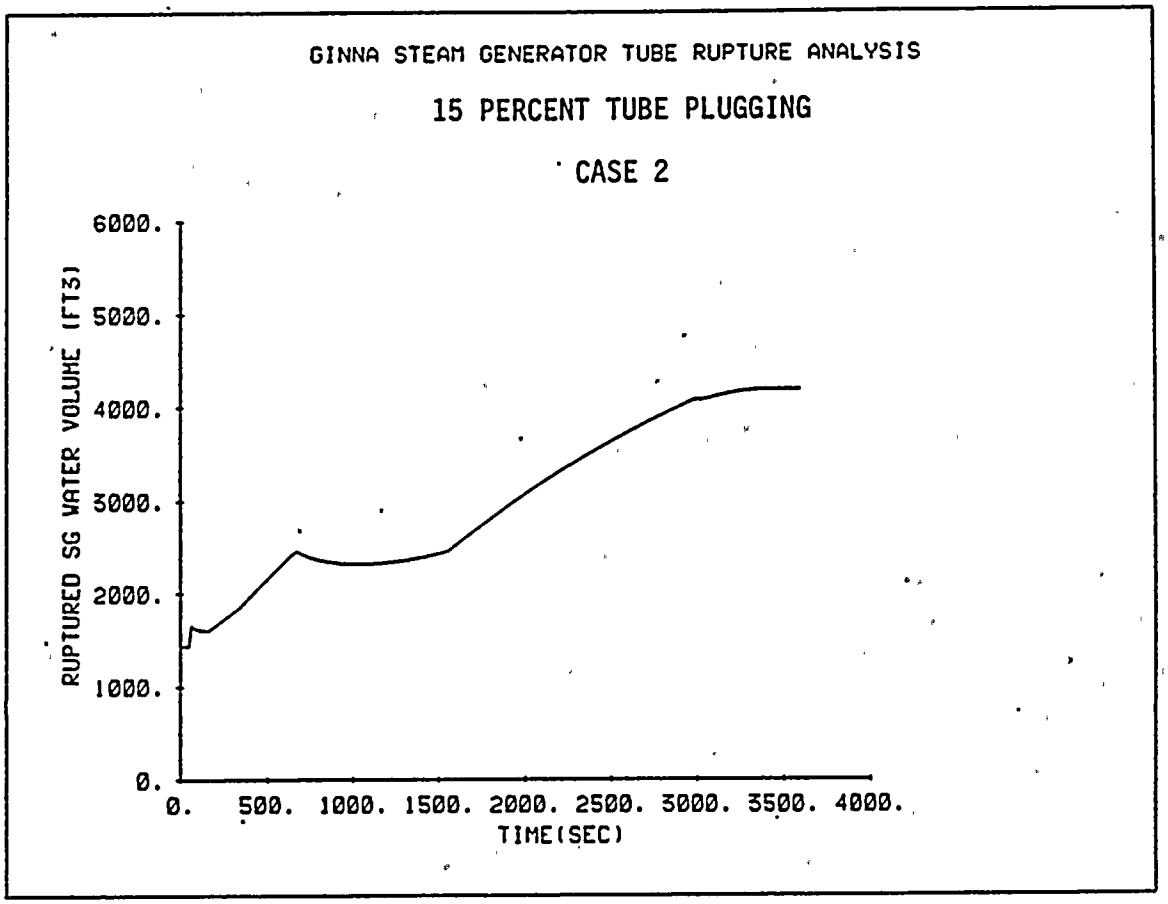


Figure II.12 Ruptured SG Water Volume - Case 2

F. Mass Releases

The mass releases were determined for each of the single failure cases for use in evaluating the exclusion area boundary and low population zone radiation exposure. The steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and primary to secondary break flow into the ruptured steam generator were determined for the period from accident initiation until 2 hours after the accident and from 2 to 8 hours after the accident. The releases for 0-2 hours are used to calculate the radiation doses at the exclusion area boundary for a 2 hour exposure, and the releases for 0-8 hours are used to calculate the radiation doses at the low population zone for the duration of the accident.

In the LOFTTR2 analyses, the SGTR recovery actions in the E-3 guideline were simulated until the termination of primary to secondary leakage. After the primary to secondary leakage is terminated, the operators will continue the SGTR recovery actions in the E-3 guideline to prepare the plant for cooldown to cold shutdown conditions. These actions include establishing normal Chemical and Volume Control System (CVCS) operation to provide reactor coolant inventory control and a boration path; restarting a reactor coolant pump (RCP), if none are running, to ensure homogeneous RCS conditions and to provide normal pressurizer spray; or stopping one RCP, if both are running, to minimize the heat input during the subsequent cooldown; and the actions necessary to minimize the spread of contamination on the secondary side. When the instructions provided in E-3 are completed, the plant should be cooled and depressurized to cold shutdown conditions. There are three alternate means of performing the post-SGTR cooldown provided in the WOG Emergency Response Guidelines. The guidelines are: ES-3.1, POST-SGTR COOLDOWN USING BACKFILL; ES-3.2, POST-SGTR COOLDOWN USING BLOWDOWN; and ES-3.3, POST-SGTR COOLDOWN USING STEAM DUMP. The preferred methods are using backfill or blowdown since these methods minimize the radioactivity released to the atmosphere. The ES-3.3 guideline using steam dump provides the fastest method for

depressurizing the RCS and ruptured steam generator. This method also results in the worst radiological releases, especially if steam dump to the condenser is unavailable. Therefore, the method using steam dump was selected for evaluation of the long-term mass releases since this produces conservative results for the offsite dose evaluation. It is noted that the use of the steam dump method would not be permitted if steam generator overfill occurs and water enters the main steamlines.

The high level actions for the ES-3.3 guideline are discussed below.

1. Prepare for Cooldown to Cold Shutdown

   The initial steps to prepare for cooldown to cold shutdown are performed in the E-3 guideline following SI termination, and these steps will be continued in ES-3.3 if they have not already been completed. A few additional steps are also performed in ES-3.3 prior to initiating cooldown. These include isolating the cold leg SI accumulators to prevent unnecessary injection, energizing pressurizer heaters as necessary to saturate the pressurizer water and to provide for better pressure control, and assuring adequate shutdown margin in the event of potential boron dilution due to in-leakage from the ruptured steam generator.

2. Cooldown RCS to Residual Heat Removal (RHR) System Temperature

   The RCS is cooled by steaming and feeding the intact steam generator similar to a normal cooldown. Since all immediate safety concerns have been resolved, the cooldown rate should be maintained less than the maximum allowable rate of 100°F/hr. The preferred means for cooling the RCS is steam dump to the condenser since this minimizes the radiological releases and conserves feedwater supply. The PORV for the intact steam generator can also be used if steam dump to the condenser is unavailable. When the RHR system operating temperature is reached, the cooldown is stopped until RCS pressure can also be decreased. This ensures that pressure/temperature limits will not be exceeded.

3. Depressurize RCS to RHR System Pressure

When the cooldown to RHR system temperature is completed, the pressure in the ruptured steam generator is decreased by releasing steam from the ruptured steam generator. Steam release to the condenser is preferred since this minimizes radiological releases. However, steam can also be released to the atmosphere using the PORV on the ruptured steam generator. An evaluation of the potential radiological consequences should be performed before releasing steam from the ruptured steam generator to the atmosphere. As the ruptured steam generator pressure is reduced, the RCS pressure is maintained equal to the pressure in the ruptured steam generator in order to prevent in-leakage of secondary side water or additional primary to secondary leakage. Normal pressurizer spray is the preferred means of RCS pressure control since this conserves coolant inventory. If pressurizer spray is not available, a pressurizer PORV or auxiliary spray can be used to control RCS pressure.

When overfill of the ruptured steam generator occurs, as with Case 1, guideline ES-3.1 POST-SGTR COOLDOWN USING BACKFILL is assumed to be used. The high level actions for ES-3.1 are similar to ES-3.3. However, the method by which ES-3.1 instructs the operator to depressurize the ruptured steam generator differs from ES-3.3. In Guideline ES-3.1 the RCS is depressurized to promote back flow through the failed tube which depressurizes the ruptured steam generator without steam releases to the atmosphere.

4. Cooldown to Cold Shutdown

When RCS temperature and pressure have been reduced to the RHR system in-service values, RHR system cooling is initiated to complete the cooldown to cold shutdown. When cold shutdown conditions are achieved, the pressurizer can be cooled to terminate the event.

F.1 Methodology for Calculation of Mass Releases

a,c

The operator actions for the SGTR recovery up to the termination of primary to secondary leakage are simulated in the LOFTTR2 analyses. Thus, the steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and the primary to secondary leakage into the ruptured steam generator were determined from the LOFTTR2 results for the period from the initiation of the accident until the leakage is terminated.

Following the termination of leakage, the operators are assumed to complete the steps in the E-3 and ES-3.3 or ES-3.1 guidelines to prepare for cooldown to cold shutdown. The time from leakage termination until the initiation of cooldown was assumed to be 20 minutes (see Reference 2). The assumed time of 20 minutes to initiate the cooldown is considered to be conservative, since the actual time is expected to be longer for an actual event because an evaluation of the potential offsite radiation doses would be required prior to using the steam dump method for the post-SGTR cooldown. It was assumed that the RCS and intact steam generator conditions are maintained stable during the 20 minute period until the cooldown is initiated. The PORV for the intact steam generator was then assumed to be used to cool down the RCS to the RHR system operating temperature of 332°F, at the maximum allowable cooldown rate of 100°F/hr. The RCS and the intact steam generator temperatures at 2 hours were then determined using the RCS and intact steam generator parameters at the time of leakage termination and the RCS cooldown rate. The steam releases and the feedwater flows for the intact steam generator for the period from leakage termination until 2 hours were determined from a mass and energy balance using the calculated RCS and intact steam generator conditions at the time of leakage termination and at 2 hours. The core decay heat and the heat added from the operation of one RCP were included in the energy balance for this time period. Since the ruptured steam generator is isolated, no change in the ruptured steam generator conditions is assumed to occur until subsequent depressurization. The assumptions of a reasonably short preparation time for cooldown and the maximum cooldown rate result in minimum RCS and steam generator temperatures at 2 hours, and therefore, a conservative estimate of the steam released to the atmosphere during the first 2 hours.

a,c

The RCS cooldown was assumed to be continued after 2 hours until the RHR system in-service temperature of 332°F is reached. Depressurization of the ruptured steam generator was then assumed to be performed immediately following the completion of the RCS cooldown. The ruptured steam generator was assumed to be depressurized to the RHR in-service pressure of 343 psia via steam release from the ruptured steam generator PORV, since this maximizes the steam release from ruptured steam generator to the atmosphere which is conservative for the evaluation of the offsite radiation doses. The RCS pressure is also assumed to be reduced concurrently as the ruptured steam generator is depressurized to minimize flow between the RCS and the ruptured steam generator. It is assumed that the continuation of the RCS cooldown and depressurization to RHR operating conditions are completed within 8 hours after the accident since there is ample time to complete the operations during this time period. The steam releases and feedwater flows from 2 to 8 hours were determined for the intact steam generator from a mass and energy balance using the RCS and steam generator conditions at 2 hours and at the RHR system in-service conditions. The core decay heat and the heat addition due to the operation of one RCP were also included in the energy balance for this time interval. The steam released from the ruptured steam generator from 2 to 8 hours was determined based on a mass and energy balance for the ruptured steam generator using the conditions at the time of leakage termination and saturated conditions at the RHR in-service pressure.

After 8 hours, it is assumed that further plant cooldown to cold shutdown as well as long-term cooling is provided by the RHR system. Therefore, the steam releases to the atmosphere are terminated after RHR in-service conditions are assumed to be reached at 8 hours.

F.2 Mass Release Results

The mass release calculations were performed for both single failure cases using the methodology discussed above. For the time period from initiation of the accident until leakage termination, the releases were determined from the LOFTTR2 results for two separate periods for use in the dose calculations. The first time period considered is from accident initiation until reactor trip. Since the condenser is in service until reactor trip, any radioactivity

released to the atmosphere prior to reactor trip will be through the condenser air ejector. After reactor trip, the releases to the atmosphere are assumed to be via the steam generator PORVs. The mass releases calculated from the time of leakage termination until 2 hours and from 2-8 hours are also assumed to be released to the atmosphere via the steam generator PORVs. The mass releases for the SGTR event [assuming delayed use of the intact steam generator PORV][a,c] (Case 1) are presented in Table II.5. The results indicate that approximately [21,990][a,c] lbm of steam and [23,710][a,c] lbm of water is released from the ruptured steam generator to the atmosphere in the first 2 hours. A total of [129,300][a,c] lbm of primary water is transferred to the secondary side of the ruptured steam generator before the break flow is terminated.

The mass releases for the SGTR event assuming [failure and isolation of the ruptured steam generator PORV][a,c] (Case 2) are presented in Table II.6. The results indicate that approximately [62,480][a,c] lbm of steam is released to the atmosphere from the ruptured steam generator within the first 2 hours. After 2 hours [33,300][a,c] lbm is released to the atmosphere from the ruptured steam generator. A total of [172,800][a,c] lbm of primary water is transferred to the secondary side of the ruptured steam generator before break flow is terminated.

TABLE II.5

CASE 1 MASS RELEASES

TOTAL MASS FLOW (POUNDS)

TIME PERIOD

a,c

	0-TRIP	TRIP TMSEP	TMSEP - OVFILL	OVFILL - TTBRK	TTBRK - T2HRS	T2HRS TRHR
Faulted SG						
- Condenser	47,800	0	0	0	0	0
- Atmosphere	0	15,450	6540	23,710*	0	0
- Feedwater	43,600	30,500	0	0	0	0
Intact SG						
- Condenser	47,150	0	0	0	0	0
- Atmosphere	0	14,730	32,950	13,450	151,870	505,500
- Feedwater	47,150	46,040	29,610	17,900	158,700	513,500
Break Flow	4216	47,574	53,710	23,800	0	0

TRIP = Time of reactor trip = [50]^{a,c} sec.
TMSEP = Time when water reaches the moisture separators = [1086]^{a,c} sec.
OVFILL = Time when steam generator overfills = [2372]^{a,c} sec.
TTBRK = Time when break flow is terminated = [3428]^{a,c} sec.
T2HRS = Time at 2 hours = 7200 sec.
TRHR = Time to reach RHR in-service conditions, 8 hours = 28,800 sec.

*Water

TABLE II.6

CASE 2 MASS RELEASES

TOTAL MASS FLOW (POUNDS)

TIME PERIOD

a,c

	0-TRIP	TRIP TMSEP	TMSEP - TTBRK	TTBRK - T2HRS	T2HRS TRHR
Faulted SG					
- Condenser	46,880	0	0	0	0
- Atmosphere	0	62,090	390	0	33,300
- Feedwater	42,740	33,260	0	0	0
Intact SG					
- Condenser	46,220	0	0	0	0
- Atmosphere	0	35,430	18,450	153,920	457,000
- Feedwater	46,220	97,780	19,700	160,300	457,700
Break Flow	4134	132,666	36,000	0	0

TRIP = Time of reactor trip = [49.4]^{a,c} sec.
TMSEP = Time when water reaches the moisture separators = [2372]^{a,c} sec.
TTBRK = Time when break flow is terminated = [3438]^{a,c} sec.
T2HRS = Time at 2 hours = 7200 sec.
TRHR = Time to reach RHR in-service conditions, 8 hours = 28,800 sec.

TABLE II.7

SUMMARIZED MASS RELEASES

TOTAL MASS FLOW (POUNDS)

	CASE 1			CASE 2		
	0 - TTBRK	TTBRK - 2HRS	2HRS - 8HRS	0 - TTBRK	TTBRK - 2HRS	2HRS 8HRS
Faulted SG						
- Condenser	47,820	0	0	46,880	0	0
- Atmosphere	45,700*	0	0	62,480	0	33,300
- Feedwater	74,100	0	0	76,000	0	0
Intact SG						
- Condenser	47,150	0	0	46,220	0	0
- Atmosphere	61,130	151,870	505,500	53,880	153,920	457,000
- Feedwater	122,800	158,700	513,500	163,700	160,300	457,700
Break Flow	129,300	0	0	172,800	0	0

* 23,710 lbm of this is water.

III. RADIOLOGICAL CONSEQUENCES ANALYSIS

The evaluation of the radiological consequences of a steam generator tube rupture, assumes that the reactor has been operating at the proposed Technical Specification limit for primary coolant activity and at the existing Technical Specification limit for primary to secondary leakage for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant. 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 and via the condenser air ejector exhaust.

The quantity of radioactivity released to the environment, due to a SGTR, depends upon primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow, break flow flashing fractions, attenuation of iodine carried by the flashed portion of the break flow, partitioning of iodine between the liquid and steam phases, the mass of fluid released from the generator and liquid-vapor partitioning in the turbine condenser hot well. All of these parameters were conservatively evaluated for a design basis double ended rupture of a single tube.

A. Design Basis Analytical Assumptions

The major assumptions and parameters used in the analysis are itemized in Table III.1. The following is a discussion of the source term.

Source Term Calculations

The radionuclide concentrations in the primary and secondary system, prior to and following the SGTR are determined as follows:

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

i. Accident Initiated Spike - The initial primary coolant iodine concentration is 1 μCi/gm of Dose Equivalent (D.E.) I-131. Following the primary system depressurization associated with the SGTR, an iodine spike is initiated in the primary system which increases the iodine release rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to the initial primary system iodine concentration. The duration of the spike, [3.3][a,c] hours, is sufficient to increase the initial RCS I-131 inventory by a factor of [100][a,c].

ii. Preaccident Spike - A reactor transient has occured prior to the SGTR and has raised the primary coolant iodine concentration from 1 to 60 μCi/gram of D.E. I-131.

b. The initial secondary coolant iodine concentration is 0.1 μCi/gram of D.E. I-131.

c. The chemical form of iodine in the primary and secondary coolant is assumed to be elemental.

Dose Calculations

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

1. The mass of reactor coolant discharged into the secondary system through the rupture and the mass of steam released from the intact and ruptured steam generators to the atmosphere are presented in Table II.5 and II.6.

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

3. The time dependent iodine removal efficiency for scrubbing of steam bubbles as they rise from the leak site [(assumed to be at the top of the tube bundle)][a,c] to the water surface was also determined for each

case. The iodine removal efficiency is determined by the method suggested by Postma and Tam (Ref. 6). The iodine removal efficiencies are shown in Figure III.2.

4. The 0.2 gpm primary to secondary leak is assumed to be split evenly between the steam generators.

5. The iodine partition factor between the liquid and steam of the ruptured and intact steam generators is assumed to be 100.

6. No credit was taken for radioactive decay during release and transport, or for cloud depletion by ground deposition during transport to the site boundary or outer boundary of the low population zone.

7. Short-term atmospheric dispersion factors (x/Qs) for accident analysis and breathing rates are provided in Table III.4. The breathing rates were obtained from NRC Regulatory Guide 1.4, (Ref. 4).

Offsite Thyroid Dose Calculation Model

Offsite thyroid doses are calculated using the equation:

$$D_{Th} = \sum_i \left[DCF_i \left(\sum_j (IAR)_{ij} \quad (BR)_j \quad (x/Q)_j \right) \right]$$

where

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

---

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

$(BR)_j$ = breathing rate during time interval j in meter^3/second (Table III.4)

$(x/Q)_j$ = atmospheric dispersion factor during time interval j in second/meter^3 (Table III.4)

$(DCF)_i$ = thyroid dose conversion factor via inhalation for isotope i in rem/Ci (Table III.5)

D_{Th} = thyroid dose via inhalation in rem

Results

Thyroid doses at the Exclusion Area Boundary and Low Population Zone are presented in Table III.6. All doses are well within the allowable guidelines as specified by Standard Review Plan 15.6.3 and 10CFR100.

TABLE III.1

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

I.	Source Data	
	A. Core power level, MWt	1520
	B. Total steam generator tube leakage, prior to accident, gpm	0.2
	C. Reactor coolant iodine activity:	
	1. Accident Initiated Spike	The initial RC iodine activities based on 1 μCi/gram of D.E. I-131 are presented in Table III.3. The iodine appearance rates assumed for the accident initiated spike are presented in Table III.2.
	2. Pre-Accident Spike	Primary coolant iodine activities based on 60 μCi/gram of D.E. I-131 are presented in Table III.3.
	D. Secondary system initial activity	Dose equivalent of 0.1 μCi/gm of I-131, presented in Table III.3.

TABLE III.1 (Sheet 2)

E. Reactor coolant mass, grams	1.27×10^8
F. Steam generator mass (each), grams	3.39×10^7
G. Offsite power	Lost at time of reactor trip
H. Primary-to-secondary leakage duration for intact SG, hrs.	8
I. Species of iodine	100 percent elemental
II. Activity Release Data	
A. Faulted steam generator	
1. Rupture flow	See Table II.5 or II.6
2. Rupture flow flashing fraction	See Figure III.1
3. Iodine scrubbing plus moisture separator removal efficiency	See Figure III.2
4. Total steam release, lbs	See Table II.5 or II.6
5. Iodine partition factor	
a. Prior to overfill	100
b. After overfill	1.0 - See Figure III.3
6. Location of tube rupture	[Top of Bundle][a,c]

TABLE III.1 (Sheet 3)

B. Intact steam generator	
1. Primary-to-secondary leakage, gpm	0.1
2. Total steam release, lbs	See Table II.5 or II.6
3. Iodine partition factor	100
C. Condenser	
1. Iodine partition factor	100
D. Atmospheric Dispersion Factors	See Table III.4

TABLE III.2

IODINE SPIKE APPEARANCE RATES
(CURIES/SECOND)

I-131	I-132	I-133	I-134	I-135
0.94	2.22	1.74	3.07	2.34

TABLE III.3

IODINE SPECIFIC ACTIVITIES IN (μCi/gm) THE PRIMARY AND SECONDARY COOLANT BASED ON 1, 60 AND 0.1 μCi/gram OF D.E. I-131

	Primary Coolant		Secondary Coolant
Nuclide	1 μCi/gm	60 μCi/gm	0.1 μCi/gm
I-131	0.79	47.1	0.079
I-132	0.35	20.7	0.035
I-133	1.01	60.7	0.101
I-134	0.20	12.2	0.020
I-135	0.79	47.1	0.079

TABLE III.4

ATMOSPHERIC DISPERSION FACTORS AND BREATHING RATES

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

TABLE III.5

THYROID DOSE CONVERSION FACTORS
(Rem/Curie) [5]

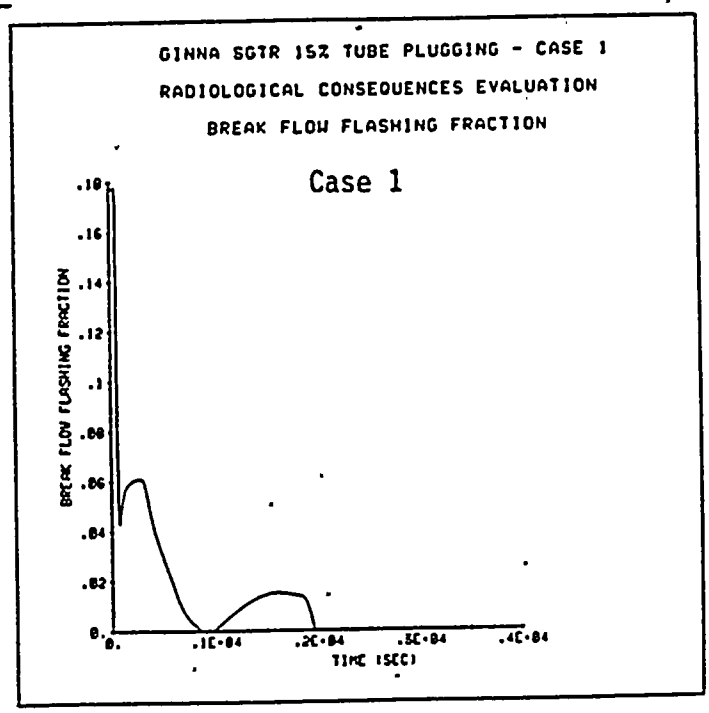
Nuclide	
I-131	1.49×10^6
I-132	1.43×10^4
I-133	2.69×10^5
I-134	3.73×10^3
I-135	5.60×10^4

TABLE III.6

RESULTS

	Doses (Rem)		
	Case 1	Case 2	Allowable Guideline Value
1. Accident Initiated Iodine Spike			
Exclusion Area Boundary (0-2 hr.)			
Thyroid	26.4	3.8	30
Low Population Zone (0-8 hr.)			
Thyroid	1.7	0.3	30
2. Pre-Accident Iodine Spike			
Exclusion Area Boundary (0-2 hr.)			
Thyroid	102.1	22.1	300
Low Population Zone (0-8 hr.)			
Thyroid	6.4	1.4	300

a,c



a,c

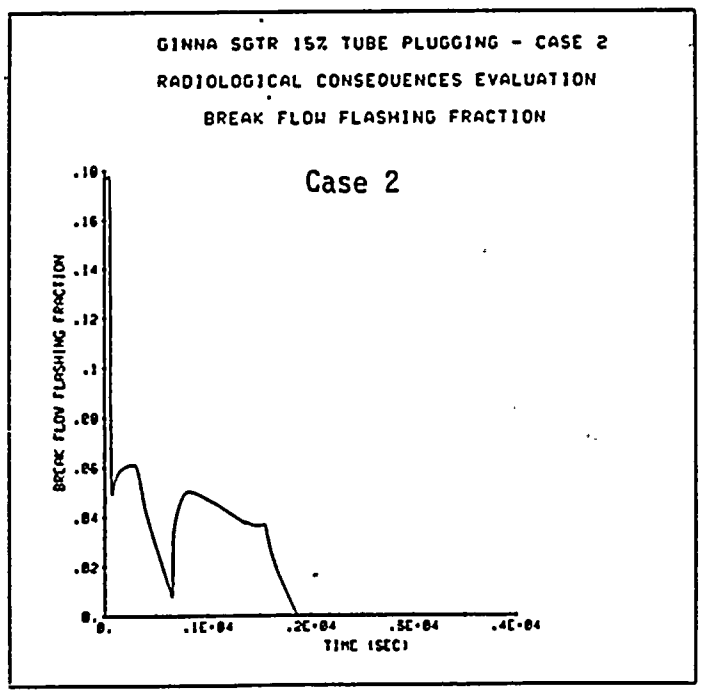
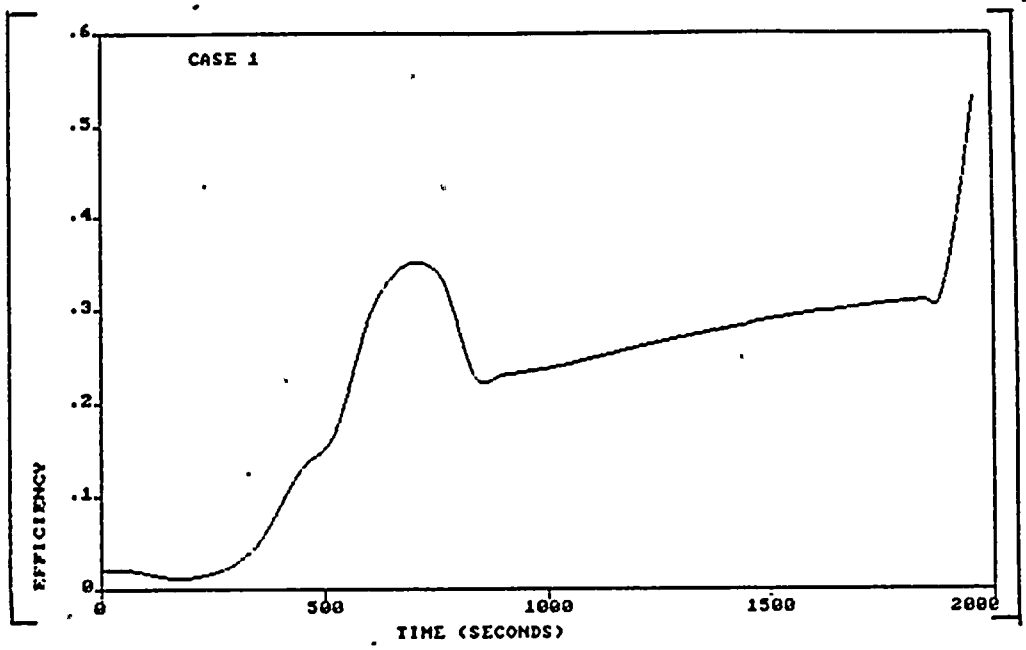


Figure III.1 Break Flow Flashing Fraction

a,c



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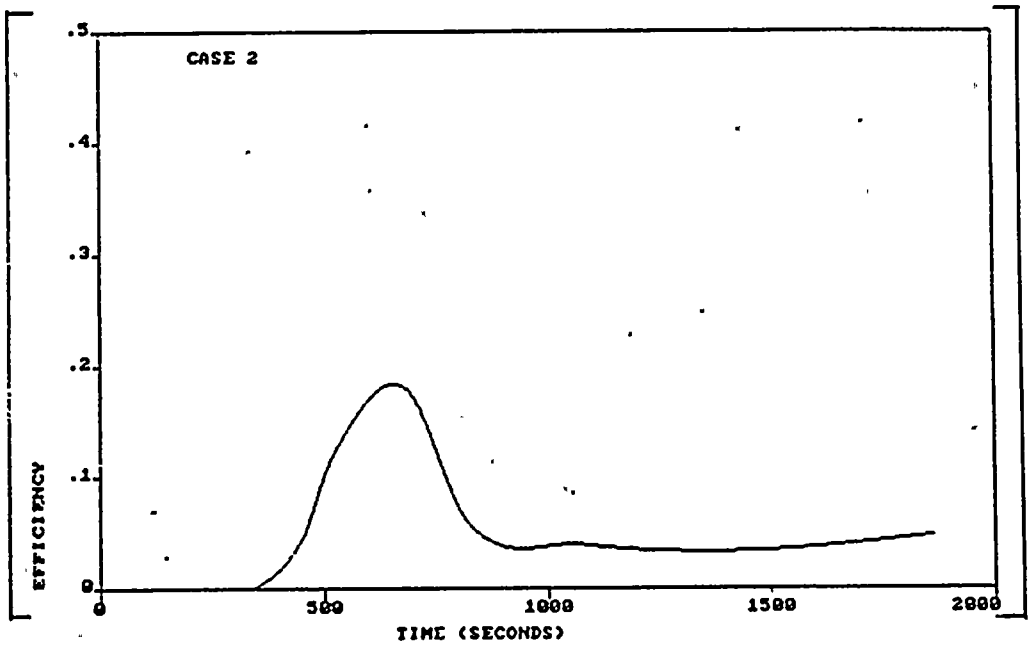
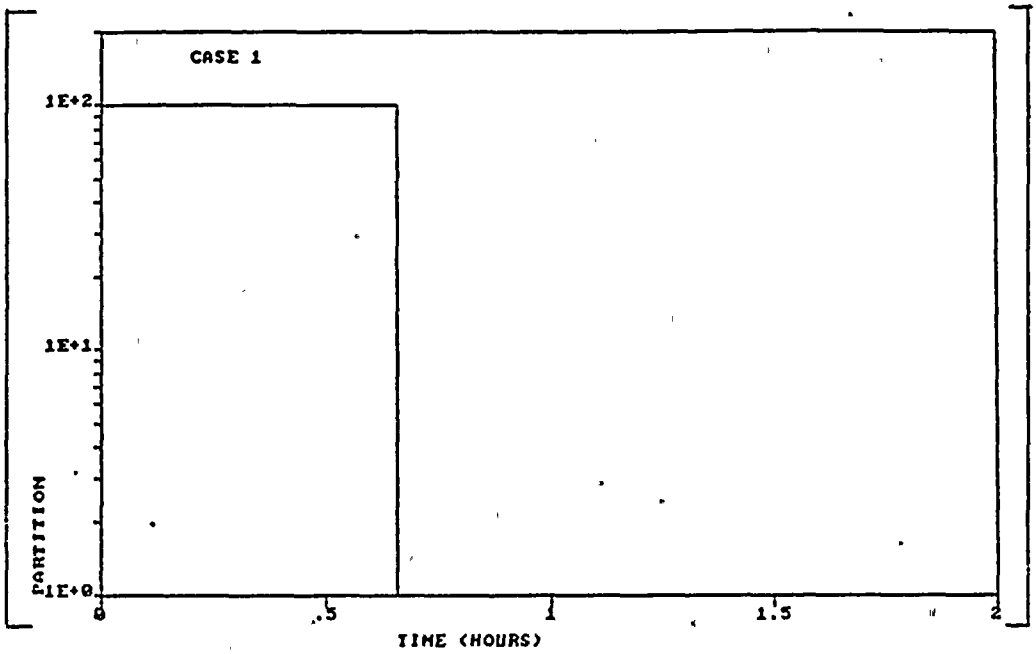


Figure III.2 Srubbing Efficiency

a,c



a,c

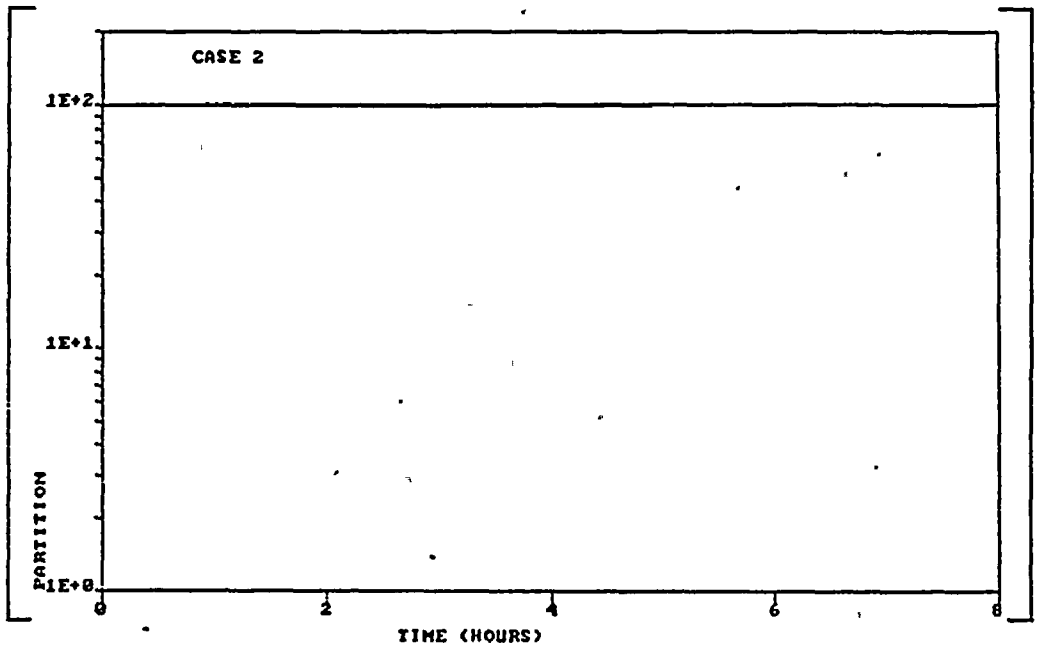


Figure III.3 Faulted SG Iodine Partition Factors

IV. CONCLUSION

The potential radiological consequences of a steam generator tube failure were evaluated for the R.E. Ginna nuclear power plant to demonstrate that the use of the Standard Technical Specification (STS) primary coolant activity limit of 1 μCi/gram of dose equivalent I-131 will result in offsite doses that are within the appropriate guidelines. The mass releases for a design basis double ended rupture of a single tube with a loss of offsite power were conservatively calculated using the computer code LOFTTR2. Two cases were considered: [1) intact steam generator PORV fails closed and must be locally opened, and 2) ruptured steam generator PORV fails open and must be locally isolated.][a,c] The analysis explicitly modeled the time needed for the operators to perform the recovery steps outlined in guideline E-3 of Revision 1 of the Westinghouse Owners Group Emergency Response Guidelines. The resulting doses at the exclusion area boundary and low population zone are within the allowable guidelines as specified by Standard Review Plan 15.6.3 and 10CFR100. Consequently, the STS primary coolant activity limit is sufficiently low to ensure that the radiological consequences of a steam generator tube rupture at the R.E. Ginna plant will be within the guidelines.

V. REFERENCES

1. Lewis, Huang, Behnke, Fittante, Gelman, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," WCAP-10698-P-A, August 1987. [PROPRIETARY]

2. Lewis, Huang, Rubin, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Supplement 1 to WCAP-10698-P-A, March 1986. [PROPRIETARY]

3. R. Elaisz, Letter from RGE to Westinghouse concerning Ginna specific operator action times for SGTR analysis, February 7, 1985.

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

5. 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", October 1977.

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