Northeast
Nuclear Energy

Rope Ferry Rd. (Route 156), Waterford, CT 06385

Millstone Nuclear Power Station
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385-0128
(860) 447-1791
Fax (860) 444-4277

The Northeast Utilities System

DEC 2 1 2000

Docket No. 50-336
B18194

RE: 10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Nuclear Power Station, Unit No. 2
License Amendment Request - Unreviewed Safety Question
Proposed Revision to Final Safety Analysis Report
Radiological Consequences of Steam Generator Tube Rupture (PLAR 2-00-4)

Northeast Nuclear Energy Company (NNECO) has determined that the increase in radiological consequences, due to changes in the assumptions and methods used in the updated thermal hydraulic analysis and in the updated dose consequence analysis of the Steam Generator Tube Rupture (SGTR) event, involves Unreviewed Safety Questions (USQs). Therefore, per 10 CFR 50.59(c), NNECO requests that the Nuclear Regulatory Commission (NRC) review and approve the changes to the Final Safety Analysis Report (FSAR) through an amendment to Operating License DPR-65, pursuant to 10 CFR 50.90. This license amendment request deals with changes in the Millstone Unit No. 2 FSAR due to revisions in the SGTR event analyses to include a loss of offsite power in addition to other changes in the assumptions and methods used in the SGTR analyses.

The proposed FSAR changes will affect section 14.6.3 of the FSAR, "Radiological Consequences of Steam Generator Tube Failure." These FSAR changes show that the dose consequences for the updated SGTR analyses are higher than the dose consequences for the previous analyses. Therefore, the results of the updated analyses represent an increase in the consequences of a previously evaluated accident and are deemed to involve an USQ in accordance with 10 CFR 50.59 (a)(2). However, the dose consequences are within the limits of 10 CFR 100, acceptance criteria of Standard Review Plan (SRP) 15.6.3, and General Design Criteria (GDC) 19. Therefore, NNECO has concluded that the proposed changes are safe. Additionally, the dose consequences were calculated with the PERC 2 computer code, which

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implements Regulatory Guide 1.4 methodology, but is different from the calculations described in the current FSAR. This change in methodology results in a departure from a method of evaluation described in the FSAR and is deemed to involve an USQ in accordance with 10 CFR 50.59 (a)(2).

Attachment 1 provides a discussion of the proposed changes and the safety summary. Attachment 2 provides the Significant Hazards Consideration. Attachment 3 provides the FSAR pages with the changes indicated.

Environmental Considerations

NNECO has reviewed the proposed license amendment request against the criteria of 10 CFR 51.22 for environmental considerations. The proposed changes show that the dose consequences for the updated SGTR analysis are higher than the dose consequences for the previous analysis. However, these changes will not significantly increase the type and amounts of effluents that may be released offsite. In addition, this amendment request will not significantly increase individual or cumulative occupational radiation exposures. Therefore, NNECO has determined the proposed changes will not have a significant effect on the quality of the human environment.

Conclusions

The proposed changes do not involve a significant impact on public health and safety (see the Safety Summary provided in Attachment 1) and do not involve a Significant Hazards Consideration pursuant to the provisions of 10 CFR 50.92 (see the Significant Hazards Consideration provided in Attachment 2).

Plant Operations Review Committee and Nuclear Safety Assessment Board

The Plant Operations Review Committee and Nuclear Safety Assessment Board have reviewed and concurred with the determinations.

Schedule

We request issuance of this amendment for Millstone Unit No. 2 by August 30, 2001, with the amendment to be implemented within 60 days of issuance.

State Notification

In accordance with 10 CFR 50.91(b), a copy of this License Amendment Request is being provided to the State of Connecticut.

There are no regulatory commitments contained within this letter.

If you should have any questions on the above, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Raymond P. Necci
Vice President - Nuclear Technical Services

Subscribed and sworn to before me

this 21st day of December, 2000

Notary Public

Date Commission Expires: SANDRA J. ANTON
NOTARY PUBLIC
COMMISSION EXPIRES
MAY 31, 2005

Attachments (3)

cc: H. J. Miller, Region I Administrator
J. I. Zimmerman, NRC Project Manager, Millstone Unit No. 2
S. R. Jones, Senior Resident Inspector, Millstone Unit No. 2

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Attachment 1

Millstone Nuclear Power Station, Unit No. 2

License Amendment Request - Unreviewed Safety Question
Proposed Revision to Final Safety Analysis Report
Radiological Consequences of Steam Generator Tube Rupture (PLAR 2-00-4)
Discussion of Changes and Safety Summary

License Amendment Request - Unreviewed Safety Question
Proposed Revision to Final Safety Analysis Report
Radiological Consequences of Steam Generator Tube Rupture (PLAR 2-00-4)
Discussion of Changes and Safety Summary

Introduction

Northeast Nuclear Energy Company (NNECO) has determined that the increase in radiological consequences, due to changes in the assumptions and methods used in the updated thermal hydraulic analysis and in the updated dose consequence analysis of the Steam Generator Tube Rupture (SGTR) event, involves Unreviewed Safety Questions (USQs). This license amendment request deals with changes in the Millstone Unit No. 2 Final Safety Analysis Report (FSAR) due to revisions in the SGTR event analyses to include a loss of offsite power in addition to other changes in the assumptions and methods used in the SGTR analyses. Therefore, per 10 CFR 50.59(c), NNECO requests that the Nuclear Regulatory Commission (NRC) review and approve the changes to the FSAR through an amendment to Operating License DPR-65, pursuant to 10 CFR 50.90.

The proposed FSAR changes will affect section 14.6.3, "Radiological Consequences of Steam Generator Tube Failure." These FSAR changes show that the dose consequences for the updated SGTR analysis are higher than the dose consequences for the previous analysis. Therefore, the results of the updated analysis represent an increase in the consequences of a previously evaluated accident and are deemed to involve an USQ in accordance with 10 CFR 50.59(a)(2). However, the dose consequences are within the limits of 10 CFR 100, the acceptance criteria of Standard Review Plan (SRP) 15.6.3, and General Design Criteria (GDC) 19. Additionally, the dose consequences calculated with the PERC 2 computer code, which implements Regulatory Guide 1.4 methodology, is different from the calculations described in the current FSAR. This change in methodology results in a departure from a method of evaluation described in the FSAR and is deemed to involve an USQ in accordance with 10 CFR 50.59 (a)(2).

Background

The Safety Evaluation Report to Amendment No. 90 documented several questions relating to the analysis of steam line and steam generator tube rupture.[(1),(2),(3),(4)] One of

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(1) K. L. Heitner to W. G. Council, "Issuance of Amendment No. 90 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit 2," dated December 30, 1983.

(2) W. G. Council to The Nuclear Regulatory Commission, "Millstone Unit No. 2, Steam Generator Tube Rupture Reassessment, Summary of Radiological Results," dated April 12, 1984.

(3) W. G. Council to The Nuclear Regulatory Commission, "Millstone Unit No. 2, Follow up Actions to Amendment No. 90 to Operating License No. DPR-65," dated September 14, 1984.

(4) W. G. Council to The Nuclear Regulatory Commission, "Millstone Unit No. 2, Follow-up Actions to Amendment No. 90 to Operating License No. DPR-65," dated January 2, 1985.

the questions raised by the staff was the exclusion of Loss Of Offsite Power (LOOP) coincident with steam generator tube rupture. In a letter to the NRC dated November 8, 1985,(5) NNECO provided results of SGTR thermal hydraulic analysis with LOOP. These results were provided in a confirmatory fashion as part of addressing the Staff's questions. However, this submittal was for information only to show that the SGTR analysis with LOOP demonstrates acceptable results. The Millstone Unit No. 2 licensing basis of the SGTR analysis was not changed to include LOOP at that time. In a letter dated January 21, 1987,(6) the NRC staff accepted the results of the confirmatory calculations and considered the issue of supplemental Cycle 6 analyses (Amendment No. 90) to be closed.

A license amendment submittal was provided in a letter dated November 13, 1998.(7) In this letter, NNECO informed the NRC that the increase in radiological consequences, due to changes in the assumptions used in the updated dose consequence analysis of the SGTR event in the Millstone Unit No. 2 FSAR, involves an USQ. Based on the preliminary review of the initial submittal, the NRC staff requested changes in the initial submittal and specific responses to four questions. This information was submitted to the NRC in a letter dated September 16, 1999.(8)

Further discussions between NNECO and the NRC following the submission of this information concluded that there is a need to update the SGTR event analysis with LOOP and to include the results of this analysis as part of the Millstone Unit No. 2 licensing basis. Therefore, by a letter dated January 25, 2000,(9) NNECO withdrew the initial and updated submittals mentioned above. In this letter, NNECO also informed the NRC that a new license amendment request pertaining to proposed revision to the FSAR that incorporate LOOP analysis results will be submitted prior to December 31, 2000. Therefore, NNECO is providing this license amendment request for the NRC review and approval.

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(5) J. F. Opeka letter to U.S. Nuclear Regulatory Commission, "Follow-up Actions to Amendment No. 90 to Operating License No. DPR-65," dated November 8, 1985.

(6) D. H. Jaffe letter to E. J. Mroczka, "Nuclear Regulatory Commission Review of Amendment No. 90 to Facility Operating License No. DPR-65, Cycle 6 Operation for Millstone Unit No. 2," dated January 21, 1987.

(7) M. L. Bowling, letter to the Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, License Amendment Request - Unreviewed Safety Question, Proposed Revision to Final Safety Analysis Report, Radiological Consequences of Steam Generator Tube Rupture," dated November 13, 1998.

(8) R. P. Necci, letter to the Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Updated Proposed Revision to Final Safety Analysis Report (Unreviewed Safety Question) and Response to Request for Additional Information, Radiological Consequences of Steam Generator Tube Rupture," dated September 16, 1999.

(9) R. P. Necci, letter to the Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Withdrawal of Pending License Amendment Request - Unreviewed Safety Question, Proposed Revision to Final Safety Analysis Report, Radiological Consequences of Steam Generator Tube Rupture," dated January 25, 2000.

Current Licensing Bases

The licensing bases for this event are contained in Millstone Unit No. 2 FSAR section 14.6.3, "Radiological Consequences of Steam Generator Tube Failure," and Technical Specifications (TS) section 3/4.4.6.2, "Reactor Coolant System, Reactor Coolant System Leakage."

Licensing Bases as Described in the FSAR

The SGTR accident is a failure of the barrier between the Reactor Coolant System (RCS) and the Main Steam System. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator tube allows the transfer of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant mixes with water in the shell side of the affected steam generator. This radioactivity is transported via the main steam dump and bypass system to the condenser or directly to the atmosphere via the atmospheric dump and safety valves. Non-condensable radioactive gases in the condenser are removed by the condenser air ejector discharge via Unit No. 1 stack until actuation of the Enclosure Building Filtration Actuation Signal (EBFAS), at which time discharge is realigned to Unit No. 2 stack. The behavior of the system will vary depending upon the size of the steam generator tube failure. For small leaks the chemical and volume control charging pumps will be able to maintain the necessary primary coolant inventory and an automatic reactor trip will not occur. The gaseous fission products will be released from the main steam system at the air ejector discharge and will be discharged via Unit No. 1 stack. Non volatile fission products will tend to concentrate in the water of the steam generators. For leaks larger than the capacity of the charging pumps, the pressurizer water level and pressure will decrease and a reactor trip will occur. Upon reactor trip, the turbine will trip and the steam system atmospheric dump valves, steam generator safety valves, and the turbine bypass valves will open. In this case it is possible that, in addition to the noble fission gases, a substantial amount of the radioiodine contained in the secondary system may also be released to the atmosphere through the steam generator safety valves, and the atmospheric dump valves. The current licensing basis of the SGTR analysis does not consider LOOP and no single failures are postulated to occur during the event as outlined in section 14.0.11 of the FSAR, "Plant Licensing Basis and Single Failure Criteria."

Licensing Bases as Described in the Technical Specifications

TS section 3.4.6.2 Limiting Condition For Operation (LCO) states that:

"Reactor Coolant System leakage shall be limited to:

a. NO PRESSURE BOUNDARY LEAKAGE,
b. 1 GPM UNIDENTIFIED LEAKAGE,
c. 0.035 GPM primary-to-secondary leakage through any one steam generator, and
d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System."

The total steam generator tube leakage limit of 0.035 GPM per steam generator ensures that the dosage contribution from the tube leakage will be less than the limits of SRP 15.6.3 and GDC 19 in the event of either a steam generator tube rupture or steam line break.

Description of Proposed Change

NNECO requests that the NRC review and approve, through an amendment to Operating License DPR-65 pursuant to 10 CFR 50.90, the proposed FSAR changes which affect section 14.6.3 of the FSAR, "Radiological Consequences of Steam Generator Tube Failure." The FSAR changes reflect changes in the updated SGTR thermal hydraulic analysis and the updated dose consequence analysis.

The following are the major changes to the SGTR thermal hydraulic analysis that are being incorporated into the FSAR:

1. The new SGTR analysis assumes a loss of offsite power occurs. The loss of offsite power leads to a loss of forced reactor coolant flow following the reactor trip which results in higher hotleg temperatures, higher fraction of the break flow flashing into the affected steam generator, slower cooldown and RCS depressurization, and a reduction in the capability to cool down the plant using the intact steam generator. These effects result in increased dose consequences.
2. The loss of offsite power will cause the circulating water pumps to stop operating which results in loss of condenser vacuum. The instrument air is assumed to be lost with the loss of offsite power. The Atmospheric Dump Valves (ADVs) are also assumed to be not available for 30 minutes from the time of reactor trip due to loss of instrument air. A local manual operator action is credited for the operation of the ADVs to cool down. A long delay time of 30 minutes for operation of the ADVs is conservative as this delays the initiation of cooldown, thus prolonging the duration of the primary to secondary break flow.
3. The new SGTR analysis uses higher High Pressure Safety Injection (HPSI) flow rates as design input. Higher HPSI flow rates will increase the primary to secondary break flow, thereby increasing the dose consequences.
4. The new SGTR analysis uses lower Auxiliary Feed Water (AFW) flow rates and a lower steam generator level for AFW initiation. Lower AFW flow rate and the added delay in AFW initiation will increase steaming and, thereby, the mass releases used in the dose consequence analysis.
5. Mass releases assuming an RCS cooldown to Shutdown Cooling (SDC) entry conditions have been used in the dose consequence analysis.
6. The analysis uses the lowest possible setpoint for the Main Steam Safety Valves (MSSVs) on the ruptured steam generator and highest possible setpoint for the MSSVs on the intact steam generator side. This maximizes the mass release from the ruptured steam generator.
7. Iodine release associated with flashing of the primary to secondary break flow has been accounted for.

8. Thyroid doses were calculated using ICRP-30 dose conversion factors. ICRP-30 dose conversion factors are not as conservative as those used in the previous analysis.
9. The offsite doses are calculated using Stone & Webster code, PERC 2, which is based on Regulatory Guide (RG) 1.4 methodology.
10. Partition factor for the noble gases is set to 1.0.
11. Since the event time is extended till shutdown cooling (SDC) entry for break flow mitigation, the atmospheric dispersion coefficient and the breathing rate are now in respective time intervals.

As the RCS is depressurized, a Safety Injection Actuation Signal (SIAS) will be initiated on low pressurizer pressure initiating both HPSI pumps. Letdown will also be isolated (analysis conservatively assumes letdown is isolated at the time of tube rupture). AFW delivery starts approximately 277 seconds after tube rupture occurs. Upon receipt of a SIAS or a high radiation signal, the control room ventilation intake is isolated and is set to emergency recirculation mode. The loss of off site power will cause a loss of instrument air as well as circulating water pumps and disable the condenser dump valves and the condenser. The ADVs will be manually opened to provide plant cooldown.

Once the event is diagnosed, the faulted steam generator is isolated when the hot leg temperature of both loops reaches 515°F (in approximately 3589 seconds to minimize release of radioactive contents to atmosphere). This temperature limit conservatively assumes instrument uncertainties. The manual operator action to operate the ADVs for cooldown is credited at 1800 seconds from the time of reactor trip.

Since steaming to the condenser can not be credited for RCS decay heat removal due to loss of offsite power, the ADVs are the primary source of heat removal. The MSSVs are also credited to remove decay heat for the first 30 minutes. The condensate storage tank has a minimum useable volume of 165,000 gallons (Technical Specification 3.7.1.3) which will be available for decay removal for up to 8 hours. Although not required for long term decay heat removal, a continuous supply of water to replenish the steaming from the ADVs can be achieved using the common water treatment facility which generates 400 gpm of make up water. In addition, a suction path to the AFW pumps can be established from the fire water storage tanks which get a continuous supply from the city domestic water.

For the SGTR event, the auxiliary feedwater flow to the faulted steam generator will be controlled by the operator to maintain a steam generator narrow range level of 40-70%.

Changes to the Radiological Consequences Based on the Analysis With Loss of Offsite Power

According to the SGTR analysis with offsite power available, which is the current facility licensing and design basis, the radioactive contents are released primarily via the condenser air ejector which is aligned to the Unit No. 1 stack before a reactor trip. Post trip, the contents are released via the condenser air ejector, which is aligned to the Unit No. 2 stack before loss of condenser vacuum. The radioactive gases are also released via the MSSVs/ADVs directly to the atmosphere without holdup or decontamination.

According to the updated SGTR thermal hydraulic analysis with LOOP, following reactor trip and loss of offsite power, the radioactive steam is released directly to the atmosphere via the MSSVs/ADVs.

The updated radiological consequences analysis determined that there is an increase in dose consequences in comparison with the analysis of record, which constitutes the current licensing basis. A comparison of the calculated doses as determined by the analysis of record and the updated analysis is given in the following tabulations:

CASE 1 Spike Caused by Accident Calculated Doses (Rem)					
ORGAN	Exclusion Area Boundary (Old Value)	Exclusion Area Boundary (New Value)	Low Population Zone (Old Value)	Low Population Zone (New Value)	NRC DOSE CRITERIA (SRP 15.6.3)
Thyroid	0.160	15.4	0.017	2.1	30
Whole Body	0.146	2.2	0.045	0.3	2.5

CASE 2 Pre-accident Iodine Spike Calculated Doses (Rem)					
ORGAN	Exclusion Area Boundary (Old Value)	Exclusion Area Boundary (New Value)	Low Population Zone (Old Value)	Low Population Zone (New Value)	NRC DOSE CRITERIA (10 CFR 100 limits)
Thyroid	0.813	27.8	0.085	3.7	300
Whole Body	0.146	0.8	0.045	0.1	25

Safety Summary

This license amendment request deals with changes in the Millstone Unit No. 2 FSAR due to revisions in the SGTR event analyses to include a loss of offsite power in addition to other changes in the assumptions and methods used in the SGTR analyses. The FSAR changes will affect section 14.6.3 of the FSAR. The FSAR changes involve increases in dose consequences as shown in the comparison provided in the previous section.

NNECO evaluated the radiological consequences of the SGTR and determined that the increase in dose consequences involves an USQ. However, as shown in the above tabulations, the dose consequences are within the limits of 10 CFR 100, and the acceptance criteria of SRP 15.6.3. The dose to Millstone Unit No. 2 control room operators for a SGTR was also analyzed. The results show that the consequences are bounded by the main steam line break and the Loss of Coolant Accident (LOCA) dose consequences, which are within the limits of GDC 19.

The method used to calculate the dose consequences is the PERC 2 computer code. The use of this computer code constitutes a change in the calculational methodology from the equations currently described in the FSAR. However, both the PERC 2 code and the current method described in the FSAR implement the methods described in Regulatory Guide 1.4 and are acceptable.

Therefore, NNECO has concluded that the proposed changes are safe. Attachment 3 provides FSAR section 14.6.3 pages with the proposed changes indicated.

Attachment 2

Millstone Nuclear Power Station, Unit No. 2

License Amendment Request - Unreviewed Safety Question
Proposed Revision to Final Safety Analysis Report
Radiological Consequences of Steam Generator Tube Rupture (PLAR 2-00-4)
Significant Hazards Consideration

License Amendment Request - Unreviewed Safety Question
Proposed Revision to Final Safety Analysis Report
Radiological Consequences of Steam Generator Tube Rupture (PLAR 2-00-4)
Significant Hazards Consideration

Significant Hazards Consideration

In accordance with 10 CFR 50.92, Northeast Nuclear Energy Company (NNECO) has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes to the Final Safety Analysis Report (FSAR) reflect changes due to the changes in the assumptions used in the Steam Generator Tube Rupture (SGTR) thermal hydraulic analysis and dose consequences analysis. These changes will not cause an accident to occur. In addition, the manual operator action to control the Atmospheric Dump Valves (ADV) for cooldown after an SGTR will not increase the probability of occurrence of any other accidents previously evaluated in the FSAR. Therefore, the proposed changes will not result in an increase in the probability of an accident previously evaluated.

   The proposed changes in the assumptions associated with the SGTR analyses will increase the dose consequences. However, the radiological consequences are still well within the limits of 10 CFR 100 and within the 10 CFR 50, Appendix A, General Design Criteria 19. Therefore, the proposed changes will not result in a significant increase in the consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The FSAR changes reflect changes in the updated thermal hydraulic SGTR analysis and the updated dose consequence analysis. The updated analyses do not introduce any new or unanalyzed failure modes of equipment or systems, and do not change the configuration of the plant. While the updated SGTR analysis incorporates operator actions that are in accordance with the Emergency Operating Procedures, it does not alter the way any structure, system, or component functions, and does not alter the manner in which the plant is operated. Therefore, there are no new or different types of failures of systems or equipment important to safety which could cause a new or different type of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

   10 CFR 100 establishes the accident exposure limits (300 Rem thyroid and 25 Rem whole body) for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ). The radiological consequences resulting from the updated SGTR analysis are well within these limits ("well within" is defined by Standard Review Plan 15.6.3 as 10% or less than 10 CFR 100 limits). The radiological consequences to the control room operators resulting from the updated SGTR analyses are also within the GDC 19 limit. Since these limits will not be exceeded and these limits establish the current margin of safety in the current plant licensing basis, the proposed changes will not result in a significant reduction in a margin of safety.

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Attachment 3

Millstone Nuclear Power Station, Unit No. 2

License Amendment Request - Unreviewed Safety Question
Proposed Revision to Final Safety Analysis Report
Radiological Consequences of Steam Generator Tube Rupture (PLAR 2-00-4)
Marked Up Pages

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List of Sections Affected

Section, Subsection Titles	Section Numbers	Page Numbers
Plant Licensing Basis and Single Failure Criteria	14.0.11	14.0-9
Radiological Consequences of Steam Generator Tube Failure	14.6.3	14.6-3
Event Description	14.6.3.2	14.6-3
Reactor Protection	14.6.3.3	14.6-4
Disposition and Justification	14.6.3.4	14.6-4
Definition of Events Analyzed	14.6.3.5	14.6-4, 14.6-5
Thermal-Hydraulic Calculation	14.6.3.6.1	14.6-5
Radiological Calculation	14.6.3.6.2	14.6-6, 14.6-7, 14.6-8
Conclusion	14.6.3.7	14.6-8

Table Titles	Table Numbers	Page Numbers
Available Reactor Protection for the Radiological Consequences of Steam Generator Tube Rupture Event	14.6.3-1	1 of 1
Disposition of Events for the Radiological Consequences of Steam Generator Tube Rupture Event	14.6.3-2	1 of 1
Key Parameters Assumed in the Steam Generator Tube Rupture Event	14.6.3-3	1 of 1
Sequence of Events for the Steam Generator Tube Rupture Accident	14.6.3-4	1 of 1
Assumptions for the Radiological Evaluation for the Steam Generator Tube Rupture Accident	14.6.3-5	1 of 1
Summary - Radiological Consequences of the Steam Generator Tube Rupture Accident	14.6.3-6	1 of 1

Figure Titles	Figure Numbers	Page Numbers
Steam Generator Tube Rupture Core Power Versus Time	14.6.3-1	N/A
Steam Generator Tube Rupture Pressurizer Pressure Versus Time	14.6.3-2	
Steam Generator Tube Rupture Reactor Coolant System Temperature Versus Time	14.6.3-3	
Steam Generator Tube Rupture Steam Generator Pressure Versus Time	14.6.3-4	
Steam Generator Tube Rupture Tube Leak Flow Versus Time	14.6.3-5	
Steam Generator Tube Rupture Steam Atmospheric Dump Flow per Steam Generator Versus Time	14.6.3-6	
Steam Generator Tube Rupture Safety Valve Flow per Steam Generator Versus Time	14.6.3-7	
Steam Generator Tube Rupture Steam Bypass Flow Versus Time	14.6.3-8	

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14.0.10 Effects of Mixed Assembly Types and Fuel Rod Bowing

To address the effects of mixed assembly types, current SPC methodology is to impose a penalty of 2% to MDNBR calculations even when all assemblies in the core are of similar hydraulic design. The penalty is in addition to the calculated cross flow penalty obtained by modeling the actual mixed core cross flow effects. The impact of this penalty is to effectively increase the XNB correlation limit from the calculated 95/95 limit to 1.19. 99-3

In accordance with SPC rod bow methodology (Reference 14.0-12), the magnitude of rod bow for the SPC assemblies has been estimated. The calculations indicate that 50% closure of the rod-to-rod gap occurs at an assembly exposure of about 77,000 MWd/MTU for the SPC 14x14 design. Significant impact to MDNBR due to rod bow does not occur until the gap closures exceed 50%. Since the maximum design exposure for SPC reload fuel in Millstone Unit 2 is significantly less than that at which 50% closure occurs, rod bow does not significantly impact the MDNBR for SPC fuel. Also, total peaking is not significantly impacted. 99-3

14.0.11 Plant Licensing Basis and Single Failure Criteria

Except for the steam generator tube rupture, design basis event scenarios considered in the safety analysis depend on single failure criteria. The following single failure criteria are assumed in the safety analysis for Millstone 2: 99-3

(1) The RPS is designed with redundancy and independence to assure that no single failure or removal from service of any component or channel of a system will result in the loss of the protection function.

(2) Each ESF is designed to perform its intended safety function assuming a failure of a single active component.

(3) The onsite power system and the offsite power system are designed such that each shall independently be capable of providing power for the ESF assuming a failure of a single active component in either power system.

The safety analysis is structured to demonstrate that the plant systems design satisfies these single failure criteria. The following assumptions result:

(1) The ESFs required to function in an event are assumed to suffer a worst single failure of an active component.

(2) Reactor trips occur at the specified setpoint within the specified delay time assuming a worst single active failure.

The assumptions for concurrent loss of offsite power are as follows:

~~(3)~~ (1) The following postulated accidents are considered assuming a concurrent loss of offsite power: main steam line break, control rod ejection, steam generator tube rupture, and LOCA. 99-3

~~(4)~~ (2) The loss of normal feedwater, an anticipated operational occurrence, is analyzed assuming a concurrent loss of offsite power.

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14.6.2 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside of Containment

Millstone Unit 2 does not have any instrument lines connected to the reactor coolant system (RCS) which penetrate the containment. Therefore, this event is not analyzed.

14.6.3 Radiological Consequences of Steam Generator Tube Failure

14.6.3.1 Event Initiator

The event is initiated by a loss of integrity in a single tube in a steam generator, resulting in a flow of primary side reactor coolant water into the secondary side.

14.6.3.2 Event Description

Experience with nuclear steam generators indicates that the probability of complete severance of a tube is small. The more probable modes of failure are those involving the occurrence of pinholes or small cracks in the tubes, and of cracks in the seal welds between the tubes and tube sheet.

A leaking steam generator tube would allow transport of primary coolant into the main steam system. Radioactivity contained in the primary coolant would mix with shell side water in the affected steam generator. Some of this radioactivity would be transported by steam to the turbine and then to the condenser. Noncondensible radioactive materials would then be passed to the atmosphere through the condenser air ejector discharge via the Plant stack.

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The radioactive products would be sensed by the condenser air ejector radiation monitor or the stack radiation monitor. These monitors have audible alarms that will be annunciated in the control room to alert the operator to abnormal activity levels so that corrective action could be taken.

The behavior of the systems will vary depending upon the size of the steam generator tube failure. For small leaks the chemical and volume control charging pumps will be able to maintain the necessary primary coolant inventory and an automatic reactor trip will not occur. The gaseous fission products will be released from the main steam system at the air ejector discharge and will be discharged via the Plant stack. Nonvolatile fission products will tend to concentrate in the water of the steam generators.

For leaks larger than the capacity of the charging pumps, the pressurizer water level and pressure will decrease and a reactor trip will occur. Upon reactor trip, the turbine will trip and the steam system atmospheric dump valves, steam generator safety valves and the turbine bypass valves will open. In this case it is possible that in addition to the noble fission gases a substantial amount of the radioiodines contained in the secondary system may also be released to the atmosphere through the steam generator safety valves and atmospheric dump valves. ~~dump valves.~~

The amount of radioactivity released increases with break size. For this analysis, a double-ended break of one tube was assumed. The selection of one double-ended break as an upper limit is conservatively based upon the experience obtained with other steam generators.

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The vent path is via the Unit 1 stack until actuation of an Enclosure Building Filtration Actuation Signal (EBFAS). Actuation of EBFAS automatically isolates the vent path to the Unit 1 stack after which the Operators manually align the vent path to the Unit 2 stack.

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, steam generator safety valves,

14.6.3.3 Reactor Protection

The leak rate through the double-ended rupture of one tube is greater than the maximum flow available from the charging pumps. Therefore, the Primary Coolant system pressure will decrease and a low pressurizer pressure trip or TM/LP trip will occur. The thermal margin trip has a low-pressure floor, set at 1,850 psia, below which trip will always occur. Following the reactor trip the Primary Coolant System is cooled down by exhausting steam through the atmospheric steam dump valves and turbine bypass valve. The radioactivity exhausted through the steam dump valves passes directly to the atmosphere. The radioactivity exhausted through the bypass valve flows to the condenser where the gaseous products remaining are vented to the atmosphere through the condenser air ejector and Plant stack.

s atmospheric the and steam generator safety valves turbine s

Reactor protection for the Radiological Consequences of Steam Generator Tube Failure event is summarized in Table 14.6.3-1.

14.6.3.4 Disposition and Justification

s

The radiological consequences of a steam generator tube rupture (SGTR) accident are maximized at rated power operation due to the stored energy in the primary coolant which must be removed by the intact steam generator in order to bring the primary and secondary systems into pressure equilibrium, thereby terminating the primary to secondary leak.

The challenge to the SAFDLs exists due to the depressurization prior to scram. As such, this challenge is very similar to that which exists due to the inadvertent opening of a pressurizer relief valve (Event 14.6.1). Since the depressurization rates associated with Event 14.6.1 are substantially larger than those encountered for this event, the corresponding pressure undershoot will also be greater. Event 14.6.1 will thus be characterized by lower pressures at the time of MDNBR than those obtained for this event. Therefore, the DNB aspects of this event will be bounded by those of Event 14.6.1.

The disposition of events for the Radiological Consequences of Steam Generator Tube Failure event is summarized in Table 14.6.3-2.

see Insert A

14.6.3.5 Definition of Events Analyzed

The analysis of an SGTR accident was performed using RETRAN 02 MOD 3 (Reference 14.6-4) computer code. The plant simulation includes modeling of the RCS, the steam generators, the main steam and feedwater systems, the charging and letdown systems, the High Pressure Safety Injection (HPSI) System, and reactor core kinetics including fuel and moderator temperature feedback. The pressurizer was modeled as a nonequilibrium volume. The analysis also considers automatic initiation of auxiliary feedwater system (AFWS). (93-3 (93-

The following assumptions are made to ensure a conservative estimate of the radioactive release to the atmosphere:

(1) The initial core power is 2754 Mwt;

(2) The initial reactor pressure is the nominal value, 2300 psia;

(3) The initial main steam pressure is 933 psia. An uncertainty of +45 psi was placed on the nominal steam generator pressure in order to maximize steam generator safety valve flow following reactor trip.

(4) A double-ended rupture of one steam generator tube occurs instantaneously.

(5) Under full load operating conditions, the steam mixture containing reactor coolant passes through the turbine and condenser;

(6) Following the reactor and turbine trip, the main steam dump and bypass system is automatically actuated for removal of decay heat from the RCS. The steam generator safety valves and atmospheric dump valves also are operable.

(7) The main steam safety valves were modeled to reset at a pressure 12 percent below the opening pressure. This assumption is conservative and is based on information discussed in Reference 14.6-5. The analysis also assumes conservative safety valve lift setpoints by including a minus 3 percent drift. (93-42)

(8) Both backup charging pumps are assumed to be operable to increase the time to reactor trip. Letdown flow was isolated at the time of the SGTR. Following the safety injection actuation signal (SIAS) operation of the HPSI and charging pumps is assumed;

(9) Safety injection flow is initiated upon a low low pressurizer pressure safety injection signal.

(10) Automatic reactor trip for the accident is initiated when RCS pressure decreases to 1728 psia by the minimum setpoint of the TM/LP trip. The 1728 psig setpoint conservatively bounds the actual 1850 psia setpoint minus 22 psi uncertainty.

(11) Reactor coolant pumps (RCP) are conservatively assumed to be on throughout the transient.

(12) At the end of 60 minutes, it is assumed that releases from the affected SG have been terminated. While the time to accomplish this action is dependent on operator action, it is a reasonable estimate which, when combined with other conservative assumptions, ensures that the calculated doses will be bounding. (96-15) (93-42)

14.6.3.6 Analysis Results

14.6.3.6.1 Thermal-Hydraulic Calculation

The design basis SGTR is a double-ended break of one steam generator U-tube. Table 14.6.3-3 lists the key transient related parameters used in this analysis. In the analysis, it is assumed that the initial RCS pressure is as high as 2300 psia. This initial RCS pressure maximizes the amount of primary coolant transported to the steam

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system since the amount of leak is directly proportional to the difference between the primary and secondary pressures. Also, the higher pressure delays the low pressurizer pressure trip. Steam generator pressure is maximized in this analysis as well. Increasing this pressure serves to decrease break flow, however, it results in the operation of the steam generator safety valves and thus increases steam releases to the atmosphere. Additionally, the AFWS is modeled with a conservatively long delay time of 240 seconds.

For this event, the DNBR SAFDL is not exceeded due to the action of the TM/LP trip which provides a reactor trip to maintain the Departure from Nucleate Boiling Ratio (DNBR) above 1.30. Therefore, no fuel failure occurs during the transient. Inherent in the TM/LP trip is the explicit calculation of the limiting radial and axial peaks, maximum inlet temperature, RCS pressure, core power, and conservative Control Element Assemblies (CEA) scram characteristics.

The sequence of events for this transient is given in Table 14.6.3-4. Figures 14.6.3-1 through 14.6.3-8 present the transient behavior of core power, the RCS pressure, the RCS coolant temperatures, the steam generator pressure, ruptured tube leak rate, atmospheric dump flow, safety valve flow and steam bypass to condenser flow rates.

14.6.3.6.2 Radiological Calculation

The radiological consequences for this accident were based on the analysis presented here. The intent of this analysis is to verify that the site boundary doses do not exceed the guidelines of 10CFR100. (5/9c)

The SGTR accident is a penetration of the barrier between the RCS and the main steam system. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator tube allows the transfer of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant mixes with water in the shell side of the affected steam generator. This radioactivity is transported by steam to the turbine and then to the condenser, directly to the condenser via the main steam dump and bypass system or directly to the atmosphere via the steam dumps and safety valves. Noncondensible radioactive gases in the condenser are removed by the condenser air ejector discharge via the Unit 1 stack.

The concentration of I-131 in the steam generators was calculated by solving the following differential equation over discrete intervals of time:

$$\frac{dA_{SG_L}(t)}{dt} = f_B(t)C_B(t) + f_{fw}(t)C_{fw}(t) - \left[f_{BD}(t) + \frac{f_{STM}(t)}{\rho f_{SG}} K_{CO} - f_{STM}(t)\right] \frac{A_{SG_L}(t)}{M_{SG_L}(t)}$$

Where: $A_{SG_L}(t)$ = activity in liquid on secondary side of steam generator;

$f_B(t)$ = break flow rate;

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$C_B(t)$ = break flow iodine concentration;

$f_{fw}(t)$ = feedwater flow rate;

$C_{fw}(t)$ = feedwater flow iodine concentration;

$f_{BD}(t)$ = blow down flow rate;

$f_{STM}(t)$ = steam flow rate;

ρf_{SG} = iodine partition factor between the liquid and vapor phases in the steam generator;

K_{CO} = carry over fraction defined as the ratio of the amount of liquid mass carry over to the amount of steam flow;

$M_{SG_L}(t)$ = mass of liquid on secondary side of steam generator.

The effects of iodine spiking were also accounted for in the analysis. Two different spiking models were evaluated. The first assumes that the primary coolant concentration of I-131 (DEQ) is at the technical specification limit of 1.0 uCi/gm and the resulting tube rupture causes the iodine release rate to increase by a factor of 500 over the equilibrium release rate. The second spiking model assumes that a preaccident iodine spike causes the primary coolant to reach an I-131 (DEQ) concentration of 60 uCi/gm at the time of tube rupture. This concentration is assumed to last for the entire accident. (519c)

The thyroid dose is then calculated using the following equation:

$$D_{THY} = \sum_{i=1}^{n} Ci \times M_{Si} \times \frac{1}{\rho f} \times BR_i \times X/_{Qi} \times DCF$$

Where:

D_{THY} = thyroid dose (rems);

C_i = average concentration of steam generator secondary side water during time interval i;

M_{S_i} = average mass of steam released during time interval i;

ρf = partition factor between the liquid and vapor phases in the steam generator;

BR_i = breathing rate during interval i;

$X/_{Q_i}$ = relative atmosphere dispersion coefficient during interval i;

DCF = thyroid dose conversion factor.

The whole body dose is calculated using the following equation:

$$D_{WB} \sum_{i=1}^{n} 0.25 \times E\overline{\gamma} \times C_{\rho_i} \times M_{\rho_i} \times X/_{Q_i}$$

Where:

D_{WB} = whole body dose (rems);

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$\bar{E}_\gamma$	=	average energy of gammas per disintegration (Mev);
C_{ρ_i}	=	primary coolant activity concentration during time interval i;
M_{ρ_i}	=	atmospheric dispersion coefficient during time interval i.
X/Q_i	=	primary coolant mass leaked to secondary side of steam generator during time interval;

In determining the whole body dose, the major assumption made is that all noble gases leaked to the steam generators will be released to the atmosphere. Table 14.6.3-5 summarizes the assumptions used in the calculation for the radiological release. Based on output from the RETRAN model and the values presented in Table 14.6.3-5, the site boundary and Low Population Zone (LPZ) doses calculated are presented in Table 14.6.3-6.

14.6.3.7 Conclusion

The radiological release criterion for this analysis is also presented in Table 14.6.3-6. As can be seen the calculated doses are a small fraction of the NRC criteria for both cases evaluated.

(5190)

14.6.4 Radiological Consequences of a Main Steam Line Failure Outside Containment

This event is only applicable to Boiling Water Reactors (BWR). As such, this event is not applicable to Millstone Unit 2.

14.6.5 Loss of Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

This event is initiated by a breach in the primary coolant system pressure boundary. Basically, a range of break sizes from small leaks up to a complete double-ended severance of a primary coolant system pipe must be considered. Typically, these breaks are classified as large breaks or small breaks. Large break loss of coolant accidents (LBLOCA) are discussed in Section 14.6.5.1. Small break loss of coolant accidents (SBLOCA) are discussed in Section 14.6.5.2.

14.6.5.1 Large Break Loss of Coolant Accidents

14.6.5.1.1 Event Initiator

This event is initiated by a large break in the primary coolant system pressure boundary. The size of breaks typically considered to be large breaks are from 0.5 ft^2 up to a double ended severance of a primary coolant system pipe.

INSERT A

14.6.3.5 Definition of Events Analyzed

The analysis of the SGTR event was performed with assumptions regarding system operation that were chosen to maximize the radiological doses. The analysis assumed a loss of forced circulation on reactor trip. This results in a higher hotleg temperature, larger portion of the break flow flashing, slower cooldown and RCS depressurization, and reduced capability to cool down the plant via the unaffected SG. All of these effects result in higher doses.

The plant simulation includes modeling of the RCS, the steam generators, the main steam and feedwater systems, the charging and letdown systems, and the HPSI System. The pressurizer was modeled as a non-equilibrium volume. Single failure is not postulated in conjunction with the SGTR event. The following assumptions are made to ensure a conservative estimate of the radiological consequences:

1) The initial core power is 2754 Mwt;
2) The initial reactor pressure is 2300 psia including instrument uncertainty;
3) The initial main steam pressure is 933 psia including instrument uncertainty;
4) The initial inlet temperature is 551°F including instrument uncertainty;
5) A double-ended rupture of one steam generator tube occurs instantaneously;
6) On reactor trip and turbine trip, loss of offsite power is assumed along with loss of instrument air and the condenser. The ADVs may be operated by local manual action due to loss of instrument air;
7) Following the reactor trip, the MSSVs lift for removal of decay heat from the RCS;
8) The analysis assumed the lowest allowed opening setpoint (-3% drift) for the ruptured steam generator MSSVs and the highest allowed opening setpoint (+3% drift) for the intact steam generator MSSVs;
9) The reseat pressure of the MSSVs on the ruptured steam generator is 12% below the opening pressure, Reference 14.6-5, while the reseat pressure of the MSSVs on the intact steam generator is nominal 6% below the opening pressure. This maximizes the releases to the atmosphere from the ruptured steam generator;
10) All three charging pumps are assumed to be operable, which will lead to a larger primary to secondary break flow. Letdown is conservatively isolated at the time of tube rupture;
11) SIAS is initiated on low pressurizer pressure which starts two HPSI pumps to deliver maximum flow;
12) AFW auto-initiates, accounting for system delay, and delivers a minimum flow;

The operator actions assumed in this analysis are consistent with the EOPs. The major post-trip analysis assumptions regarding operator actions are:

1. Commence Cooldown to Hotleg Temperature Less Than 515°F.

   Once the event is diagnosed, the operators will cool the RCS at a maximum controllable rate until the hotleg temperature of both loops reaches 515°F, for ruptured steam generator isolation. This temperature assumed in the analysis conservatively includes instrument uncertainties to delay the time till the ruptured steam generator can be isolated. The analysis

INSERT A (continued)

assumes a loss of offsite power leading to a loss of the condenser. Therefore, the cooldown is performed using the ADVs. Since the analysis assumes a loss of offsite power, a loss of instrument air is postulated, requiring a local manual control of the ADVs for this cooldown. The analysis assumes that, to account for local manual operator action, the cooldown starts 30 minutes from the time of reactor trip.

2. Reduce and Control RCS Pressure

   The analysis conservatively does not depressurize the RCS till after the hotleg temperature is less than 515°F and the ruptured steam generator is isolated. In the EOPs, the RCS depressurization begins just after the cooldown to 515°F commences. It is more conservative for dose consequences to delay the RCS depressurization since this will provide a larger primary to secondary break flow rate.

3. Determine and Isolate the Most Affected Steam Generator

   The operator isolates the most affected steam generator once the hotleg temperature of the loops have reached the isolation temperature of hotleg less than 515°F.

4. Cooldown and Depressurize RCS to SDC Entry Condition

   Cooldown and depressurization to SDC entry would minimize the primary to secondary break flow. The analysis assumes that cooldown to SDC entry is achieved by steaming just the intact steam generator per the EOPs. This is performed for 16 hours from the time of the tube rupture. The analysis conservatively assumes that the hotleg temperatures of the two loops fail to stay coupled, impeding the depressurization to SDC condition. There are five options available in the EOPs to cool and depressurize the isolated steam generator: 1) if RCPs are operating, use at least one RCP and perform a back flow into the RCS; 2) if time permits, allow ambient cooling; 3) if the condenser is available, steam to the condenser; 4) feed and bleed via steam generator blowdown; 5) steam to the atmosphere using ADV and feeding. Since the last option would lead to a larger offsite dose, it is the method modeled. Given a loss of offsite power / loss of instrument air condition, the feed and bleed via the steam generator blowdown or ambient cooling may be chosen to limit the offsite dose as well as dose to the operator for performing a local manual operation of the ruptured steam generator ADV.

5. Maintain Isolated Steam Generator Level Less Than 90%

   The EOPs prevent the ruptured steam generator from overfill by maintaining the pressurizer pressure within 50 psi of the isolated steam generator pressure or backflow into the RCS, in order to minimize the primary to secondary break flow. Alternatively, the steam generator blowdown may be used to restore level less than 90% narrow range level. For offsite dose purposes, this is not explicitly modeled. However, the model assumes primary side depressurization, facilitated by steaming of the isolated steam generator. Therefore, the pressurizer pressure can be maintained within 50 psi of the isolated steam generator, avoiding ruptured steam generator overfill.

INSERT A (continued)

14.6.3.6 Analysis Results

14.6.3.6.1 Thermal Hydraulic Calculation

The portion of the SGTR analysis, till the time the hotleg temperatures reach less than 515°F, was performed using RETRAN-02 MOD 3 (Reference 14.6-4) computer code. The sequence of results for this transient is presented in Table 14.6.3-3. Figures 14.6.3-1 through 14.6.3-9 present the dynamic behavior of important NSSS parameters during this event.

Following a doubled-ended break of a steam generator tube rupture, reactor coolant flows from the primary side into the secondary side of the ruptured steam generator (see Figure 14.6.3-5). A portion of this break flow is released as flashed steam (see Figure 14.6.3-6). The model has the reactor tripping at the time of tube break. This is conservative, since any pre-trip mass releases would be via the condenser air ejector where a partition factor would greatly reduce the iodine releases (this is further discussed in Section 14.6.3.6.2). Therefore, to conservatively maximize the direct atmospheric releases, the earliest possible trip is limiting. A loss of offsite power at the time of trip leads to a loss of forced flow and a momentary spike in the coldleg temperature as shown in Figure 14.6.3-1. The pressurizer level decreases as the reactor coolant shrinks post-trip. Also, the break flow is greater than the capacity of the charging pumps. As a result, the pressurizer level decreases as shown in Figure 14.6.3-2. The pressurizer pressure also drops as shown in Figure 14.6.3-3. While all three charging pumps and pressurizer heaters attempt to maintain level and pressure, letdown is conservatively isolated at the time of tube break. The pressurizer heaters are turned off as the pressurizer level decreases towards heater uncovery.

As the steam bypass to the condenser is assumed to be unavailable, the post-trip steaming is accomplished via the ADVs and the MSSVs. However, the ADVs require instrument air, which is postulated to be lost with the loss of offsite power. Therefore, no releases from the ADVs are modeled till 1,800 seconds from trip, when local manual operator action can be credited (see Figure 14.6.3-7). Hence, the post-trip steaming to remove decay heat is accomplished, during the initial 30 minutes, by the MSSVs (see Figure 14.6.3-8). The turbine valve closure, due to reactor trip, causes the steam generator pressure to rise, as shown in Figure 14.6.3-4, till the MSSV lift pressure is reached. The main feedwater flow is terminated at the time of trip and the AFW initiates on low steam generator level at 277 seconds accounting for system response time. As the colder AFW is delivered, a hotter volume of feedwater is swept in first. Two AFW pumps deliver a minimum flow rate.

The pressurizer level and pressure continue to decrease as the energy transfer to the secondary side shrinks the reactor coolant and the tube break flow continues to deplete the primary inventory. The decrease in pressure results in actuation of SIAS at 312 seconds. Once RCS pressure decreases below the HPSI shutoff head pressure, two HPSI pumps deliver maximum flow to slow the decrease in pressurizer pressure. The pressurizer pressure approaches an equilibrium pressure as the combined HPSI and charging flow rate matches the break flow rate. The hotleg temperature of less than 515°F is reached in 3,589 seconds post-trip.

Specific analyses of the potential for fuel failure is not performed for the steam generator tube accident. The potential for fuel failure is bounded by the analysis for the inadvertent opening of the pressurizer relief valve (Event 14.6.1). The analyses for Event 14.6.1 show that fuel failure does not occur for that event, therefore, fuel failure does not occur following a steam generator tube rupture.

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14.6.3.6.2 Radiological Calculation

The intent of this radiological consequences analysis is to verify that the site boundary doses do not exceed the guidelines of 10 CFR 100.

The mass releases following a SGTR were determined for use in evaluating the exclusion area boundary (EAB) and the low population zone (LPZ) radiation exposure. Figures 14.6.3-5 through 14.6.3-8 show the break mass flow rate and the steam mass flow rate predicted by the thermal hydraulic analysis. This includes the flashing of the break flow as it enters the secondary side of the steam generator. Table 14.6.3-4 summarizes the mass releases for the SGTR event. This includes 92,000 lbm additional mass releases from the ruptured steam generator associated with facilitating cooldown and depressurization for SDC entry, as well as 2,014,040 lbm released from the intact steam generator for cooldown to SDC entry. Also, the liquid break flow and the flashed break flow is assumed for 20 additional minutes from the time that the hotleg temperature reached less than 515°F, till RCS depressurization may equalize the primary and secondary pressures.

The SGTR accident is a penetration of the barrier between the RCS and the main steam system. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator tube allows the transfer of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant mixes with water in the shell side of the affected steam generator. This radioactivity is transported by steam to the turbine and then to the condenser, directly to the condenser via the main steam dump and bypass system, or directly to the atmosphere via the ADVs and MSSVs. Noncondensable radioactive gases in the condenser are normally removed by the condenser air ejector via the Unit 1 stack until an actuation of EBFAS. On EBFAS, the discharge is manually realigned to the Unit 2 stack.

The effects of iodine spiking were accounted for in the analysis. Two different spiking models were evaluated. The first assumes that the primary coolant concentration of I-131 (DEQ) is at the Technical Specification limit of 1.0 μCi/gm and the resulting tube rupture causes the iodine appearance rate to increase by a factor of 500 over the equilibrium appearance rate corresponding to the 1.0 μCi/gm (DEQ) I-131 coolant concentration. The duration of the spike is assumed to be 4 hours. The second spiking model assumes that a pre-accident iodine spike causes the primary coolant to reach an I-131 (DEQ) concentration of 60 μCi/gm at the time of the tube rupture. This concentration is assumed to last for the duration of the accident.

The PERC2 computer program is used to calculate the thyroid, gamma and beta dose due to halogens and noble gases. PERC2 is a multiple compartment activity transport code with the dose model consistent with the Regulatory Guideline 1.4 model. The PERC2 activity transport model first calculates the integrated activity, using a closed form integration solution, then calculates the cumulative doses.

The thyroid dose is calculated using the following equation:

$$D_{THY} = A \times h \times C2 \times C3 \times CB \times CO$$

where:

D_{THY}	=	thyroid dose (rem)
A	=	integrated activity (Ci-sec/m^3)
h	=	thyroid dose conversion factor (ICRP30)
C2	=	unit conversion 10^{12} pCi/Ci

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C3	=	unit conversion 10^{-3} rem/mrem
CB	=	breathing rate (m^3/sec)
CO	=	occupancy factor

The whole body dose is calculated using the semi-infinite cloud model outlined in Regulatory Guideline 1.4 and the integrated concentration. In determining the whole body dose, the major assumption is that all noble gases leaked to the steam generators will be released to the atmosphere resulting in a partition factor of 1.0. Table 14.6.3-5 summarizes the assumptions used in the calculation for the radiological releases. As shown on Table 14.6.3-5, the condenser air ejector partition factor is 0.15. If the SGTR occurred with the condenser available, then the iodine releases would be reduced by this 0.15 factor. For a SGTR with a loss of offsite power, the condenser is lost post-trip. However, steaming to the condenser would be assumed for pre-trip, and the iodine releases will be factored down. Therefore, the analysis assumes that the reactor trip is at the time of the tube break, allowing for all of the releases to be released directly to the atmosphere via the ADVs and the MSSVs.

The EAB and the LPZ doses calculated are presented in Table 14.6.3-6. The results are bounding for the assumptions on Table 14.6.3-5 and the thermal-hydraulic results presented in Table 14.6.3-4.

14.6.3.7 Conclusion

The radiological release criterion for this analysis, as well as the calculated results, are presented in Table 14.6.3-6. The calculated results are less than the NRC criteria for both the cases evaluated. The dose to the Unit 2 control room operators for a SGTR was also analyzed. The results show that the consequences are bounded by the main steam line break and the LOCA dose consequences.

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TABLE 14.6.3-1

AVAILABLE REACTOR PROTECTION FOR THE RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE RUPTURE EVENT

Operational Modes

Reactor Operating Conditions	Reactor Protection
1	Thermal Margin/Low Pressure Trip
	Safety Injection Actuation Signal
2, 3	Safety Injection Actuation Signal
4-6	No Significant Consequences for These Reactor Operating Conditions

Operational Modes

5/10

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TABLE 14.6.3-2

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DISPOSITION OF EVENTS FOR THE RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE RUPTURE EVENT

Operational Modes

Reactor Operating Conditions	Disposition
1	Bounded by the analysis of record and by that of Event 14.6.1
2-6	Bounded by the above

Analyze radiological consequences. Fuel performance bounded by Event 14.6.1

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e TABLE 14.6.3-3

KEY PARAMETERS ASSUMED IN THE STEAM GENERATOR TUBE RUPTURE EVENT

Parameter	Units	Value
Initial Core Power Level	MWt	2754
Core Inlet Temperature	°F	551
RCS Pressure	psia	2300
Initial Steam Generator Pressure	psia	933
Plugged U-Tubes SG1/SG2	#	1300/1200
Moderator Temperature Coefficient (ARO)	$x10^{-4}$ Δp	-2.5
Doppler Multiplier		1.15

93-42 (6/94)

93-42

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SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE EVENT

EVENT	TIME [1] (sec)
Tube Rupture Occurs	0
Reactor Trip	0
AFW Delivery Starts	277
SI Actuated	312
RCS Cooldown to T_{HOT} < 515°F Initiated	1800 [2]
T_{HOT} < 515°F Achieved; Ruptured Steam Generator Isolated	3589

Notes:
1. Time values are rounded to the nearest second.
2. This is an assumed analytical time and is not a required operator action time.

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TABLE 14.6.3-4

SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE ACCIDENT

TIME, sec	EVENT	SETPOINT OR VALUE	
0	Tube rupture occurs.	--	
815.0	Low pressure trip condition.	1728 psia	(93-39 5/94) + (93-42 6/94)
815.8	CEAs begin dropping into core.	--	
818.0	Bypass valves begin opening.	--	
818.0	Atmospheric dump valves begin opening.	--	
818.0	Steam generator safety valve lifts.	970 psia	96-15 4/96
820.0	Maximum steam generator pressure.	986 psia	
824.0	Pressurizer empties.	--	
NA	Operator trips Reactor Coolant Pumps.	--	
834.0	Atmospheric dump valves close.	--	(93-39) & (93-42)
862.0	Bypass valves close.	--	
902.0	Steam generator safety valves reseat.	853.6 psia	96-15
912.0	Pressurizer begins refilling.	--	
950.0	Bypass valves reopen.	--	
974.0	Atmospheric dump valves reopen.	--	
1242.9	Auxiliary feedwater flow begins.	--	
1244.0	Atmospheric dump valves close.	--	
3600.0	Simulation Ended	--	

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MASS RELEASES FOR THE STEAM GENERATOR TUBE RUPTURE EVENT

	Total Mass Flow (lbm)	
	0-2 hours	2-16 hours
Ruptured Steam Generator - ADVs and MSSVs	219,882	0
Intact Steam Generator - ADVs and MSSVs	670,856	2,014,036
Break Flow	183,451	0
Flashed Break Flow	5,240	0

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TABLE 14.6.3-5

(5/96)

ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION FOR THE STEAM GENERATOR TUBE RUPTURE ACCIDENT

Conservative Assumption	Basis
1) Reactor Coolant System Maximum Allowable Concentration I-131 (DEQ) = 1.0uCi/gm	Tech Specs
2) Steam Generator Maximum Allowable Concentration I-131 (DEQ) = .1uCi/gm	Tech Specs
3) Reactor Coolant System Maximum Allowable Concentration of Noble Gases Xe-133 (DEQ) = 100/E uCi/gm	Tech Specs
4) Steam Generator Partition Factor = .01	
5) Air Ejector Partition Factor - .0005	
6) Atmospheric Dispersion Coefficient:	95% maximum X/Q's for the years 1974-1977

Nair (4/96)

Receptor Location	Elevated (sec/m^3)	Ground Level (sec/m^3)
Site Boundary	1.03×10^{-4}	5.41×10^{-4}
LPZ	3.41×10^{-5}	5.55×10^{-5}

Conservative Assumption	Basis
7) Breather Rate = 3.47×10^{-4} m^3/sec	SRP 15.6.3
8) I-131 dose conversion factor = 1.49×10^{6} rem Ci	Reg. Guide 1.109- Adult-Thyroid Inhalation
9) Iodine Spiking Factors	NRC Criterion

a) Case 1: Concurrent iodine spike equivalent to 500 times equilibrium iodine appearance rate at Technical Specification limit.

b) Case 2: Preaccident iodine spike concentration based upon 60 times Technical Specification limit.

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ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION FOR THE STEAM GENERATOR TUBE RUPTURE EVENT

Conservative Assumption	Basis
1) Reactor Coolant System Maximum Allowable Concentration I-131 (DEQ) = 1.0 μCi/gm	Technical Specifications
2) Steam Generator Maximum Allowable Concentration I-131 (DEQ) = .1 μCi/gm	Technical Specifications
3) Reactor Coolant System Maximum Allowable Concentration of Noble Gases (DEQ) = 100/E_{bar} μCi/gm	Technical Specifications
4) Steam Generator Partition Factor Iodine 0.01 Noble Gases 1.0	SRP 15.6.3
5) Air Ejector Partition Factor Iodine 0.15 Noble Gases 1.0	NUREG 0017
6) Atmospheric Dispersion Coefficient	95% maximum X/Q's for the years 1974-1981

Receptor Location		Ground Level (sec/m^3)
EAB		3.66×10^{-4}
LPZ	0-4 hr	4.80×10^{-5}
	4-8 hr	2.31×10^{-5}
	8-24 hr	1.60×10^{-5}
	24-96 hr	7.25×10^{-6}
	96-720 hr	2.32×10^{-6}

Conservative Assumption		Basis
7) Breathing Rate		Reg. Guide 1.4
0-8 hr	3.47×10^{-4} m^3/sec	
8-24 hr	1.75×10^{-4} m^3/sec	
24-720 hr	2.32×10^{-4} m^3/sec	
8) Dose Conversion factor		ICRP 30
I-131	1.073×10^{6} rem Ci	
I-132	6.290×10^{3} rem Ci	
I-133	1.813×10^{5} rem Ci	
I-134	1.073×10^{3} rem Ci	
I-135	3.145×10^{4} rem Ci	

9) Iodine Spiking Factors — Basis: SRP 15.6.3

a) Case 1: Concurrent iodine spike equivalent to 500 times equilibrium iodine appearance rate at Technical Specification limit.

b) Case 2: Preaccident iodine spike concentration based upon 60 times Technical Specification limit.

00-MP2-47

TABLE 14.6.3-6

SUMMARY - RADIOLOGICAL CONSEQUENCES OF THE STEAM GENERATOR TUBE RUPTURE ACCIDENT

EVENT

ORGAN	CASE 1			CASE 2		
	Spike Caused by Accident			Preaccident Iodine Spike		
	Calculated Doses (Rem)		EAB	Calculated Doses (Rem)		
EAB	SITE BOUNDARY	LPZ	NRC DOSE CRITERIA	SITE BOUNDARY	LPZ	NRC DOSE CRITERIA
Thyroid	0.160	0.017	30	0.813	0.085	300
Whole Body	0.146	0.045	2.5	0.146	0.045	25

96-15 4/96

15.4 2.1 27.8 3.7

(93-39 5/94) & (93-42 6/94)

2.2 0.3 0.8 0.1

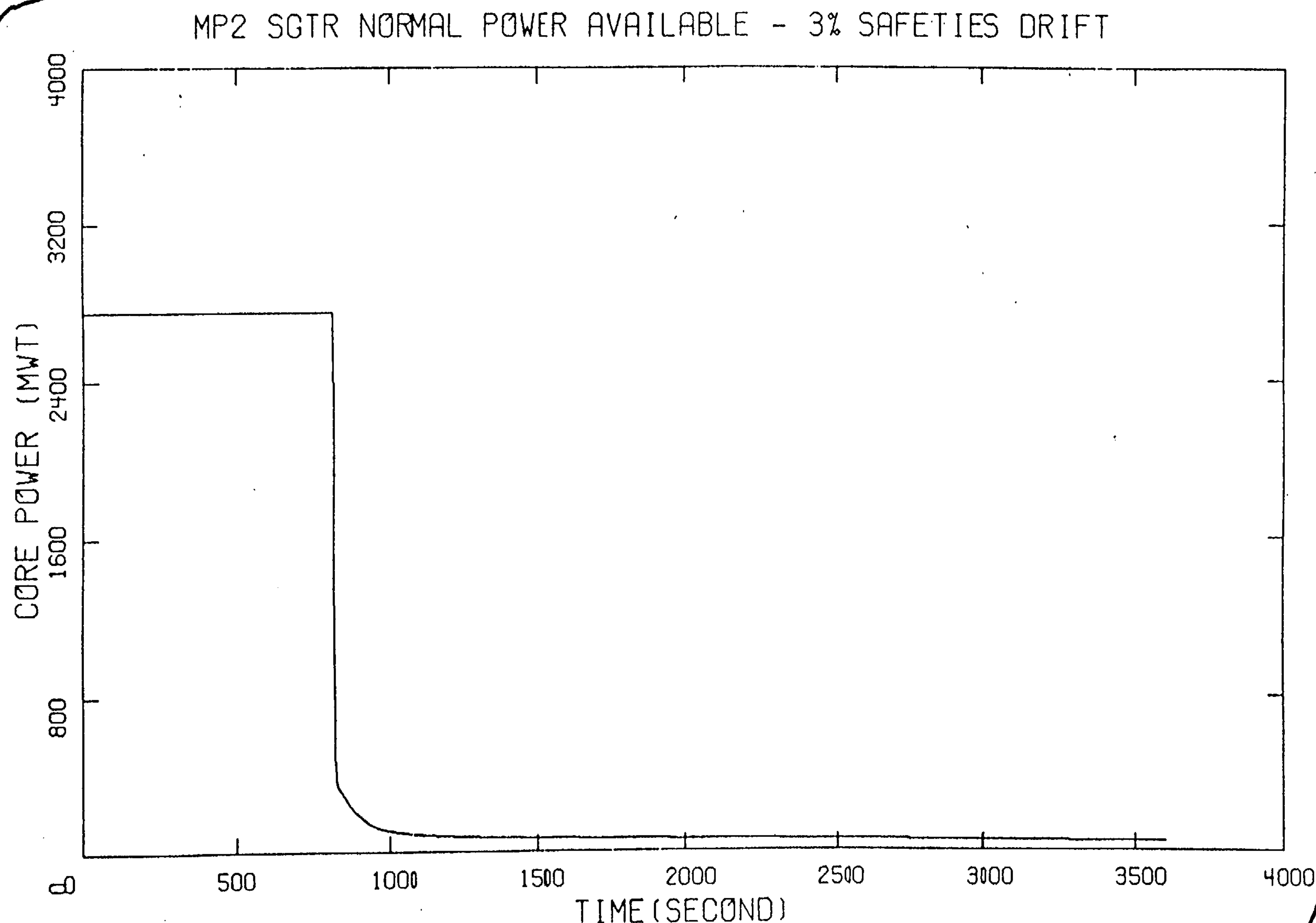

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FIGURE 14.6.3-1
STEAM GENERATOR TUBE RUPTURE CORE POWER VS. TIME

00-MW-47

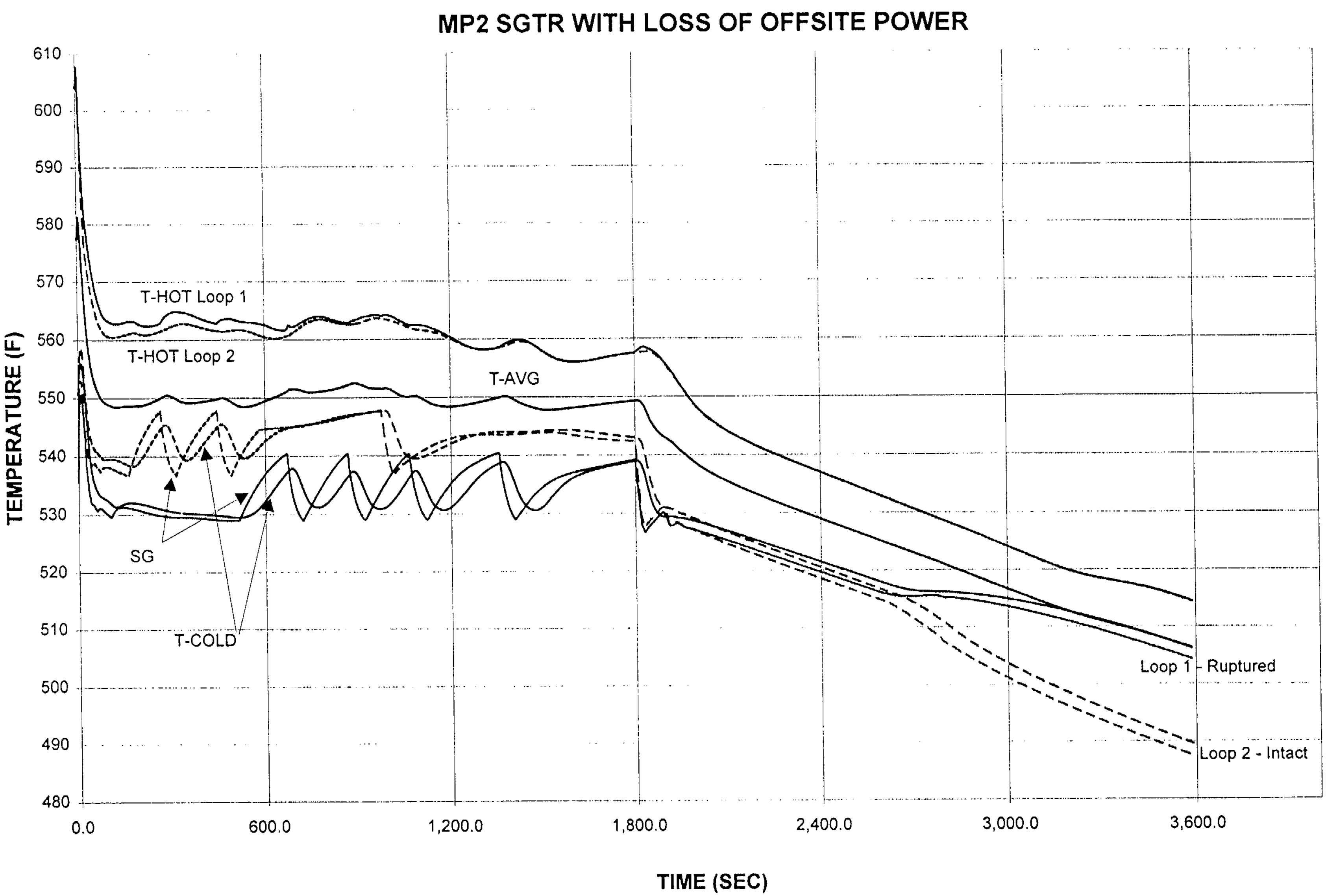

FIGURE 14.6.3-1
STEAM GENERATOR TUBE RUPTURE
RCS TEMPERATURE VS. TIME

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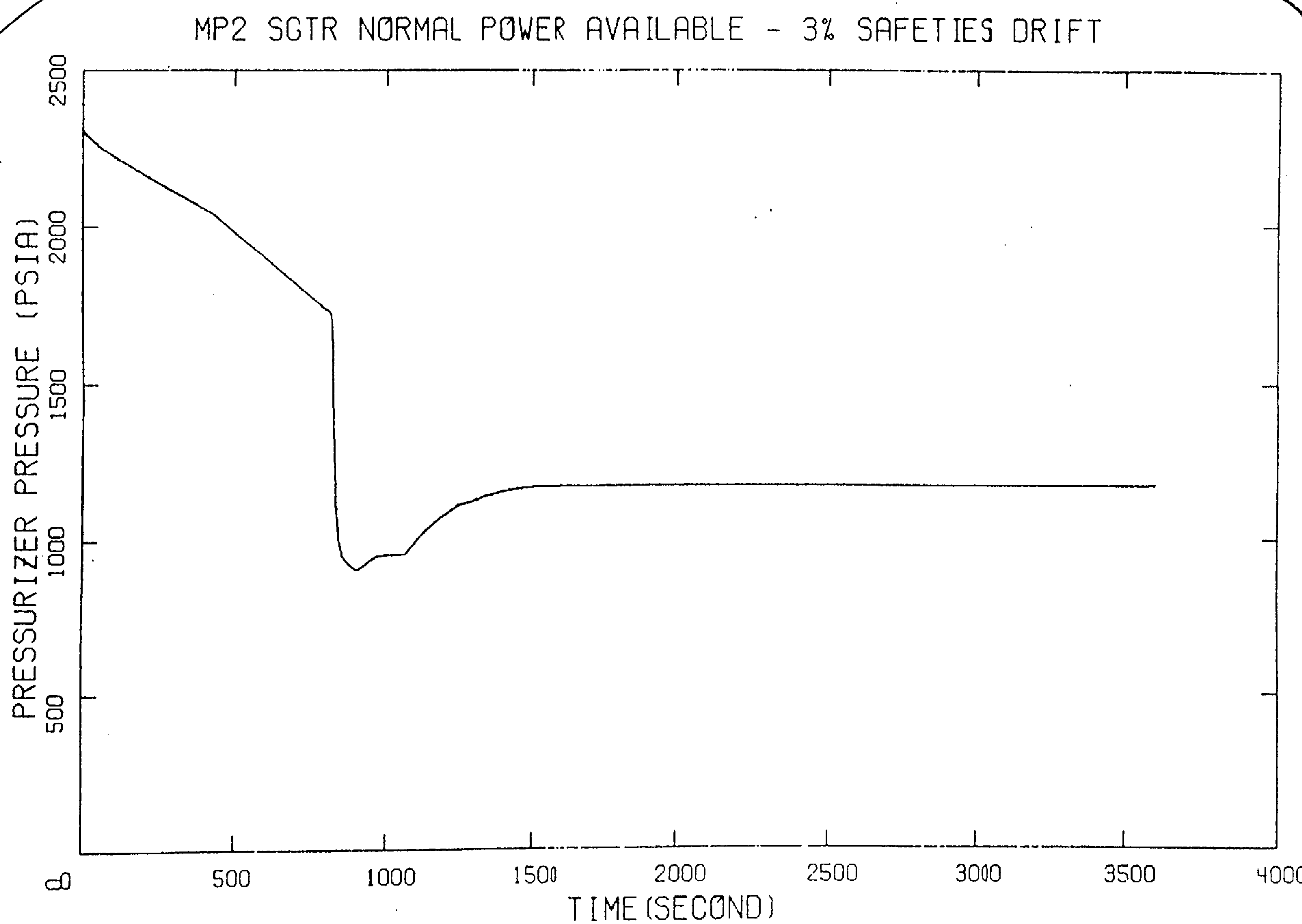

FIGURE 14.6.3-2
STEAM GENERATOR TUBE RUPTURE PRESSURIZER PRESSURE VS. TIME

L7-GOW-00

MP2 SGTR WITH LOSS OF OFFSITE POWER

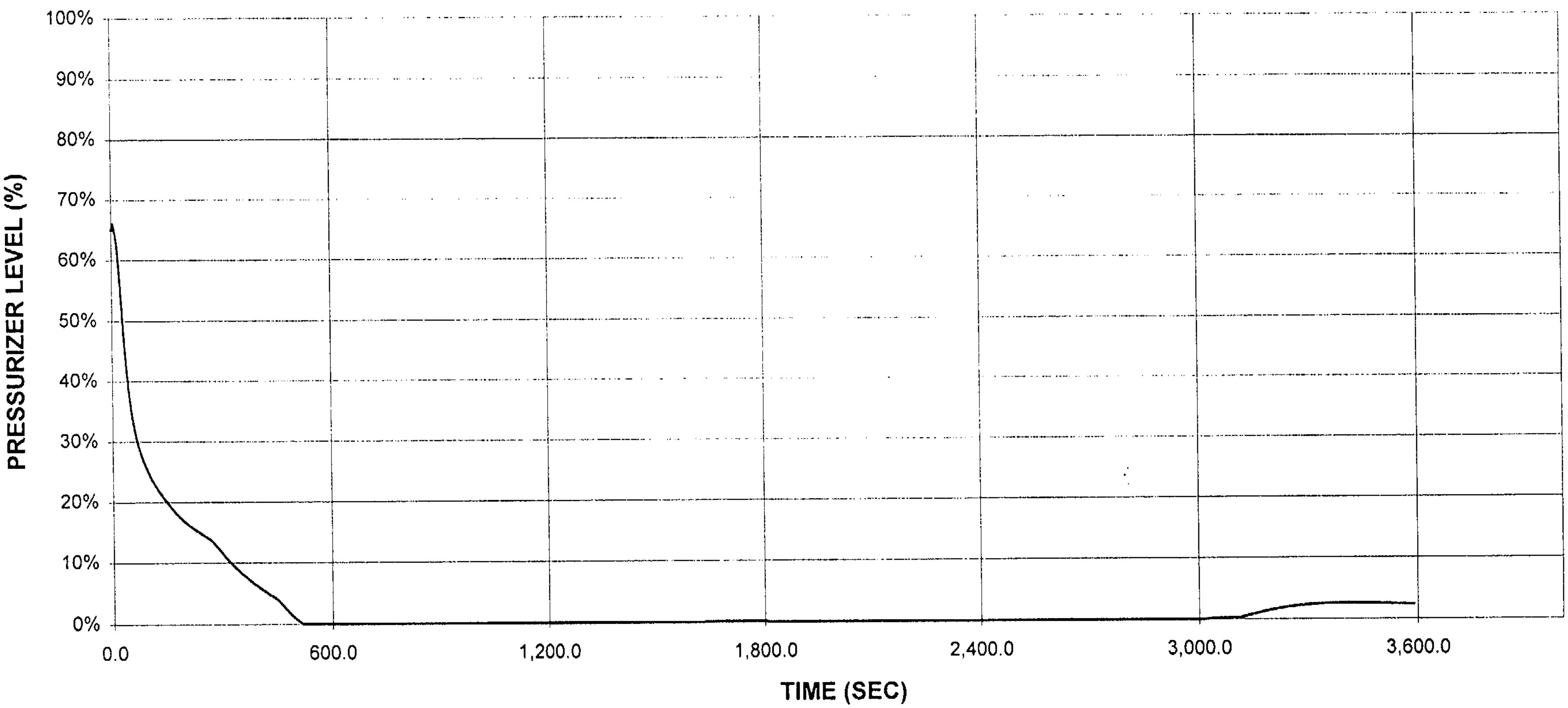

FIGURE 14.6.3-2
STEAM GENERATOR TUBE RUPTURE
PRESSURIZER LEVEL VS. TIME

00-MP2-47

2

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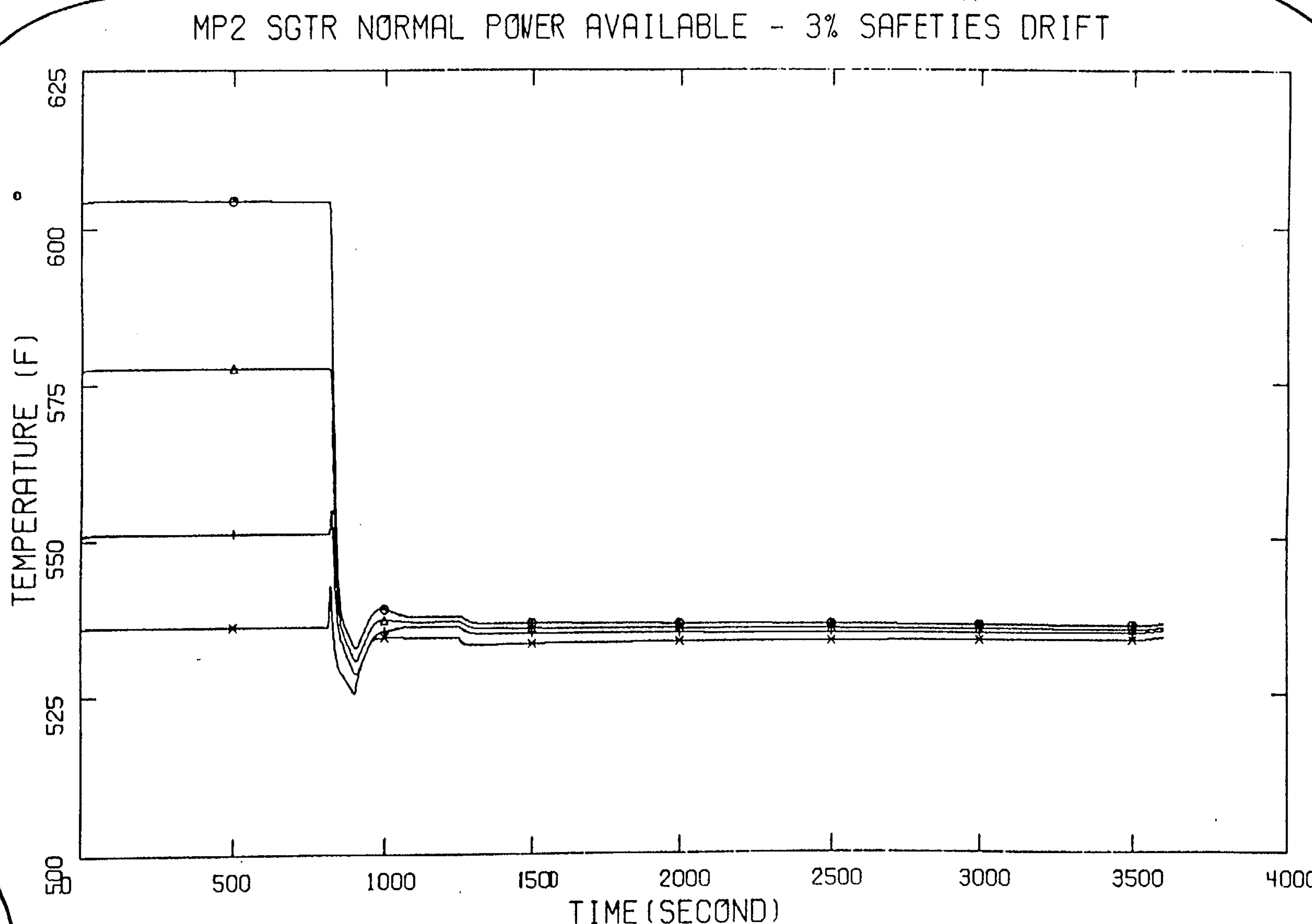

FIGURE 14.6.3-3
STEAM GENERATOR TUBE RUPTURE RCS TEMPERATURE VS. TIME

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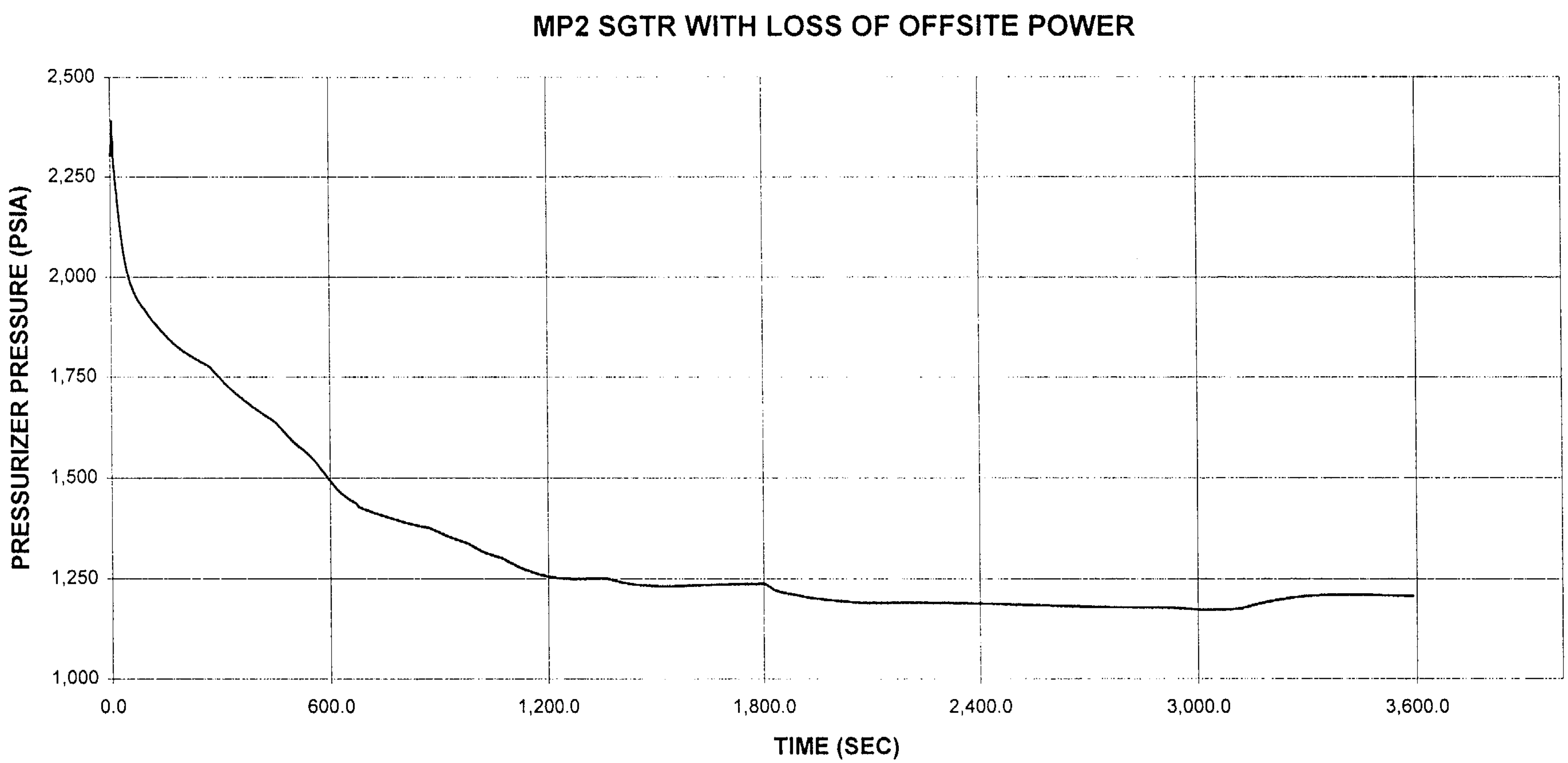

FIGURE 14.6.3-3
STEAM GENERATOR TUBE RUPTURE
PRESSURIZER PRESSURE VS. TIME

00-MP2-47

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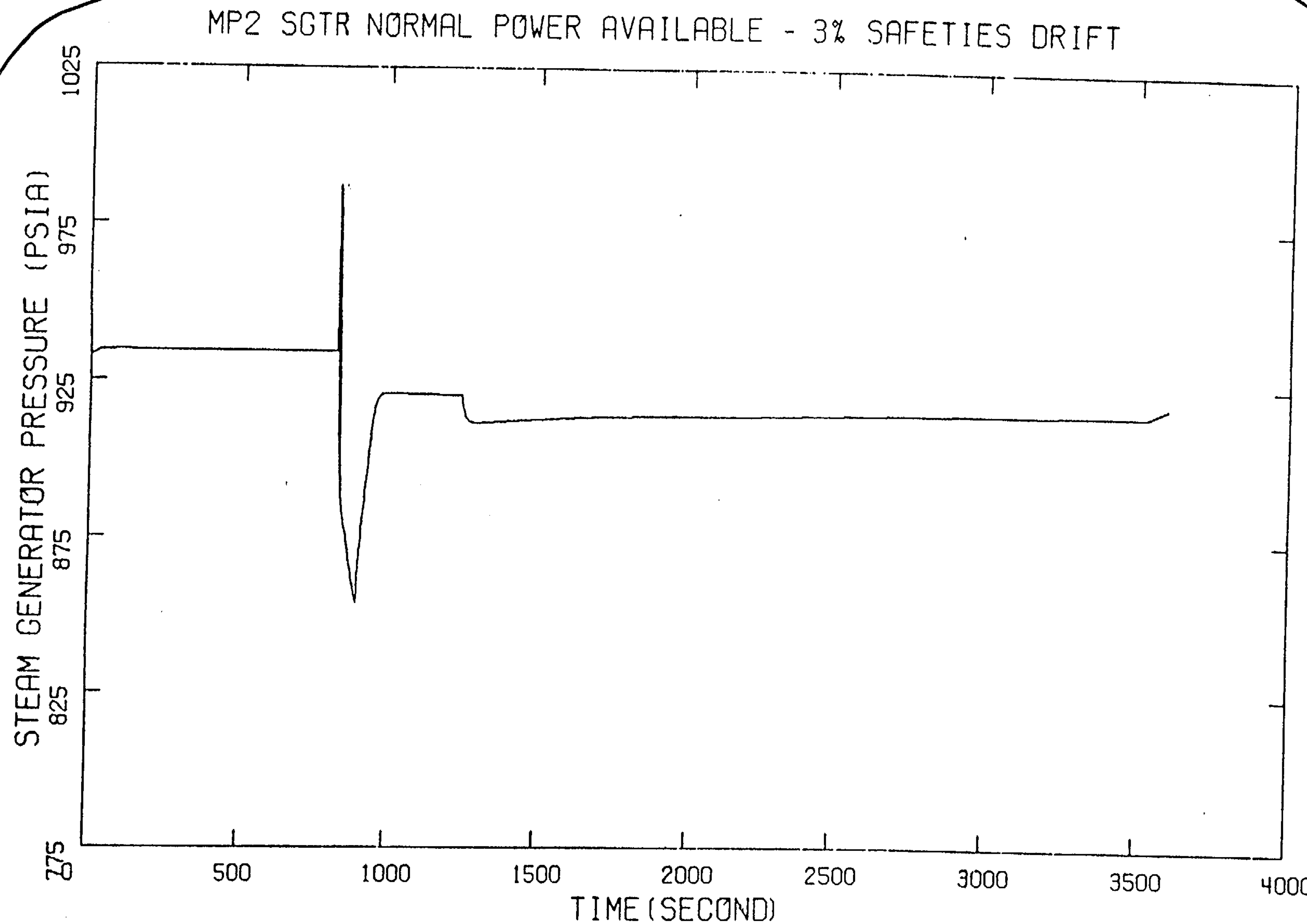

FIGURE 14.6.3-4
STEAM GENERATOR TUBE RUPTURE STEAM GENERATOR PRESSURE VS. TIME

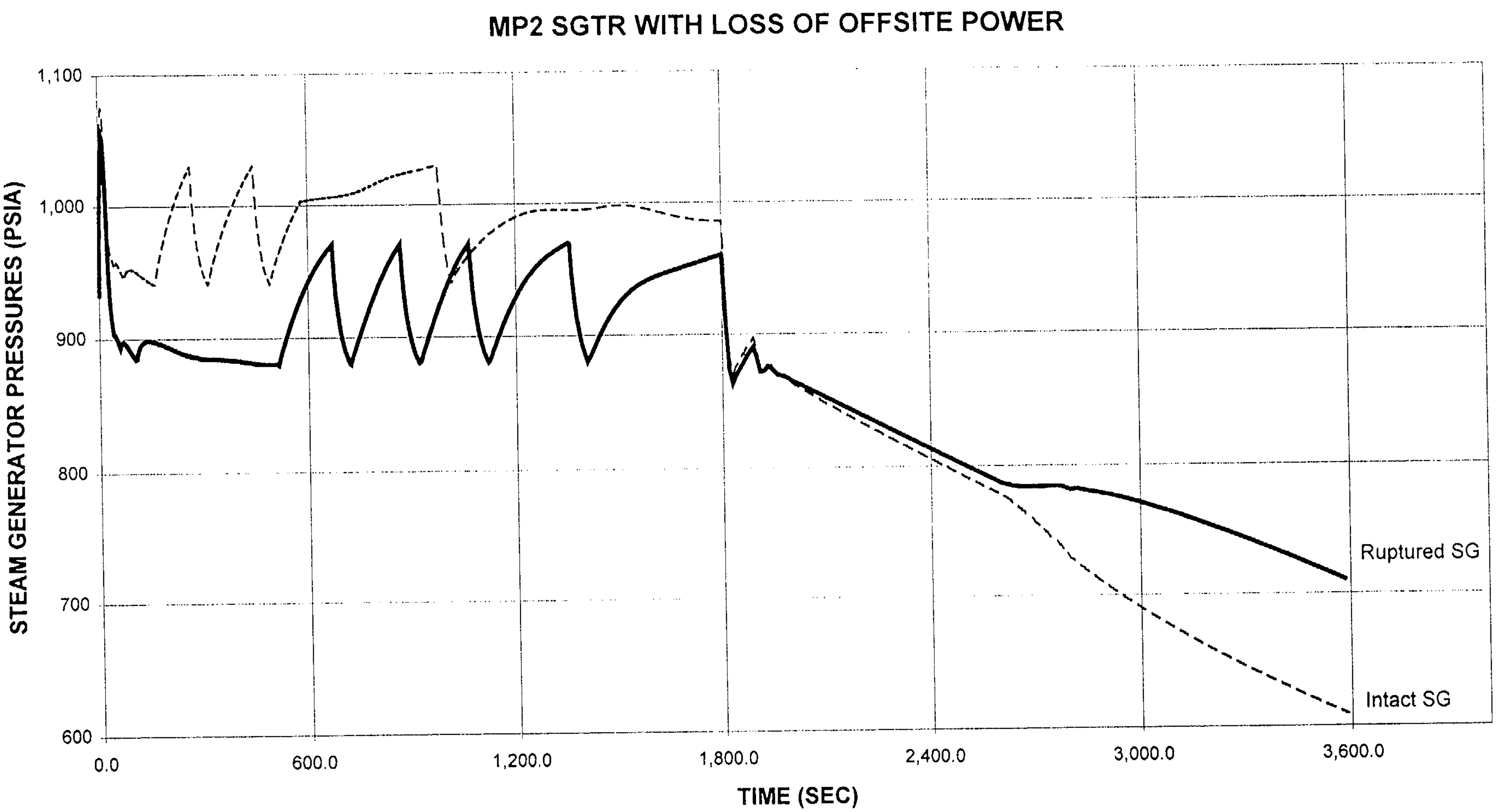

FIGURE 14.6.3-4
STEAM GENERATOR TUBE RUPTURE
STEAM GENERATOR PRESSURE VS. TIME

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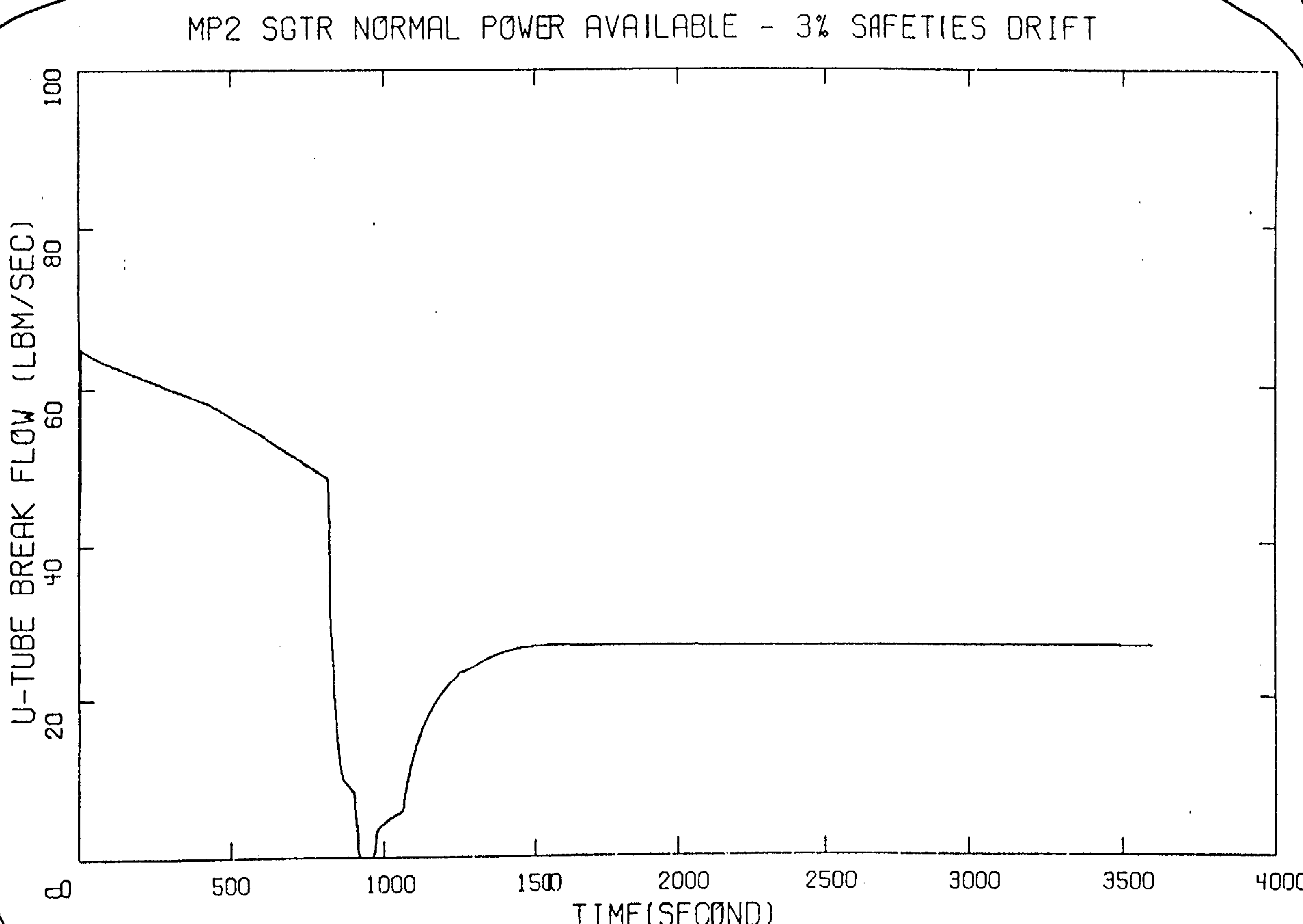

FIGURE 14.6.3-5
STEAM GENERATOR TUBE RUPTURE TUBE LEAK FLOW VS. TIME

00-MP2-47

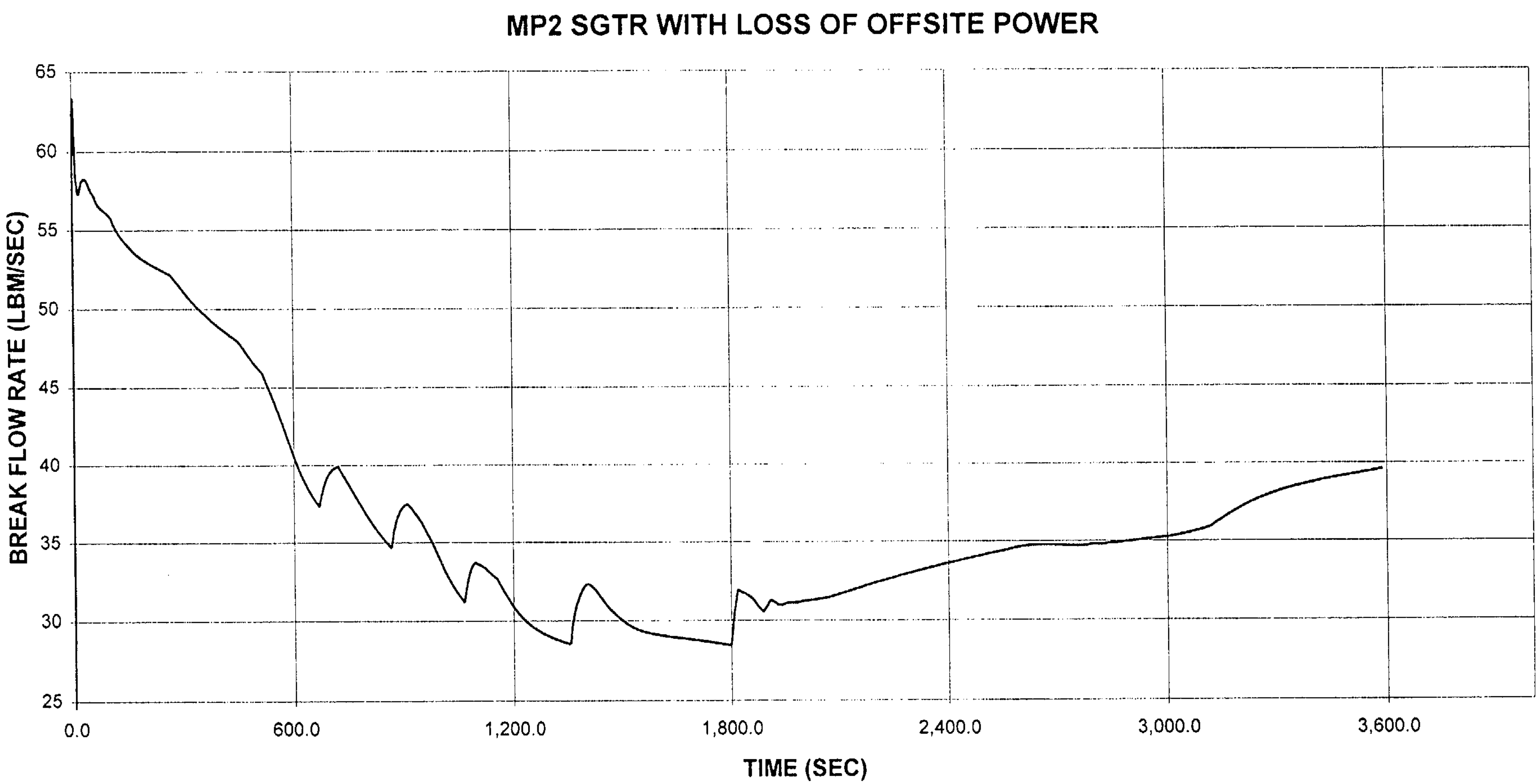

FIGURE 14.6.3-5
STEAM GENERATOR TUBE RUPTURE
BREAK FLOW RATE VS. TIME

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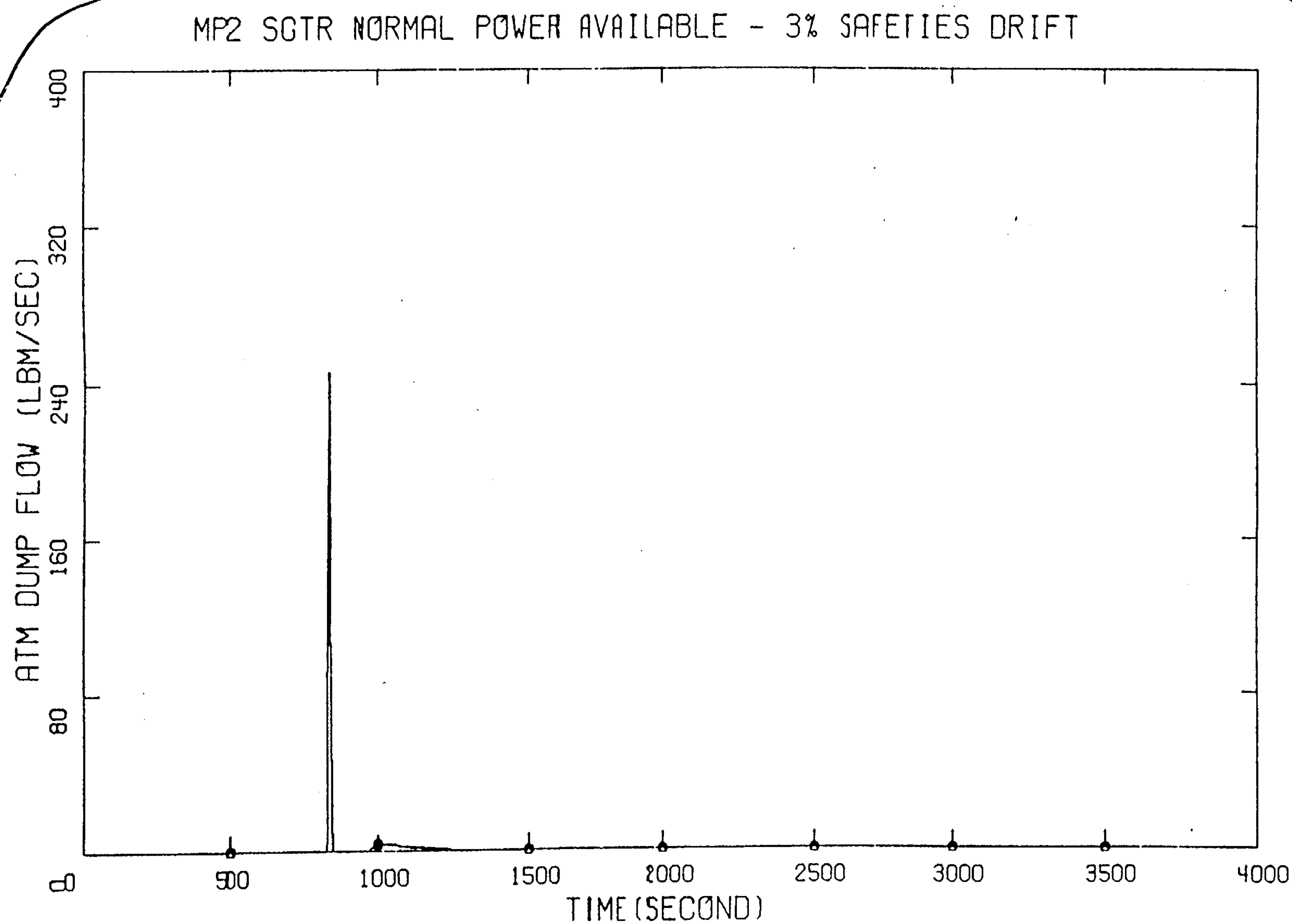

FIGURE 14.6.3-6
STEAM GENERATOR TUBE RUPTURE
STEAM ATMOSPHERIC DUMP FLOW PER STEAM GENERATOR VS. TIME

MP2 SGTR WITH LOSS OF OFFSITE POWER

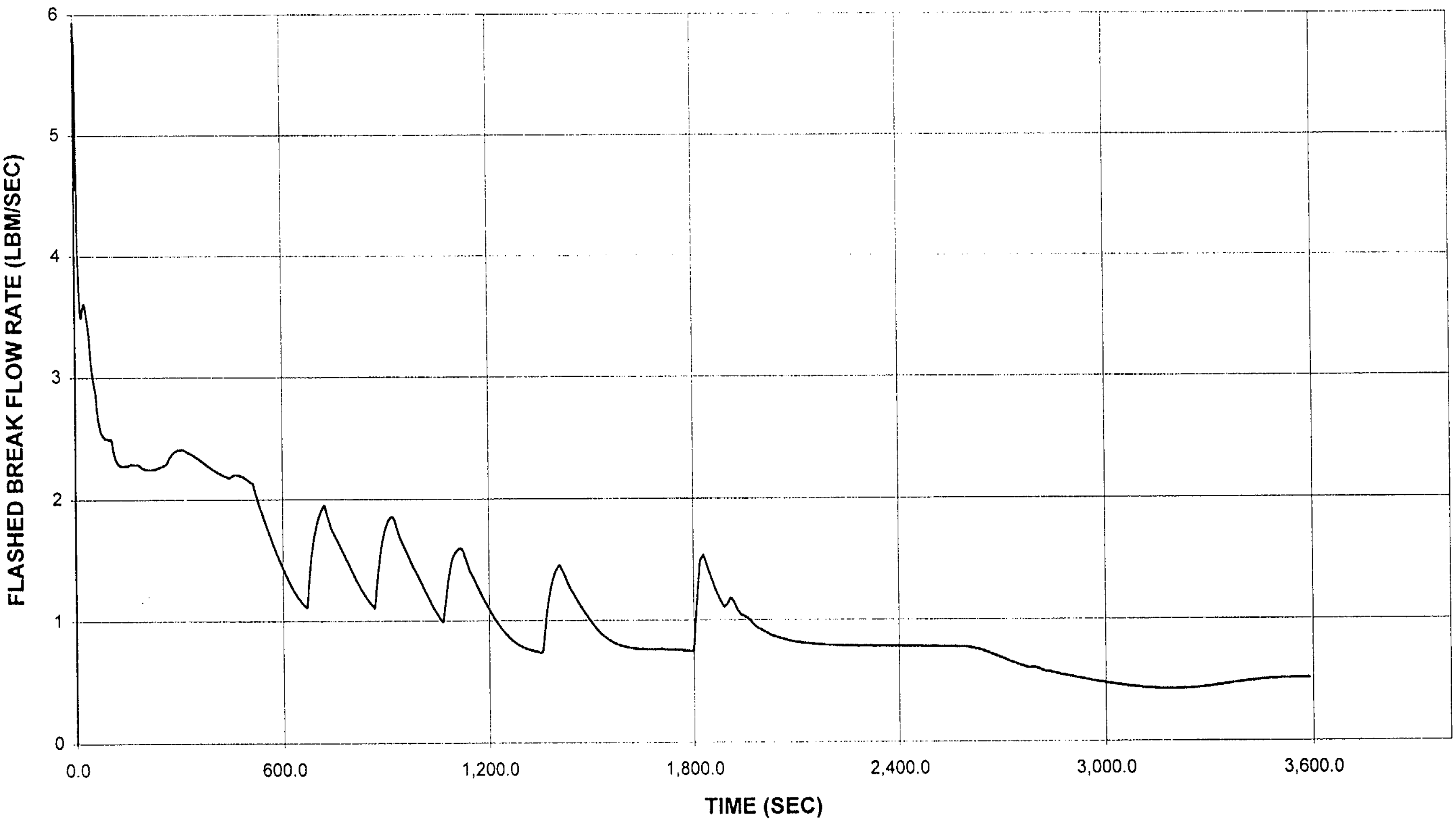

FIGURE 14.6.3-6
STEAM GENERATOR TUBE RUPTURE
FLASHED BREAK FLOW RATE VS. TIME

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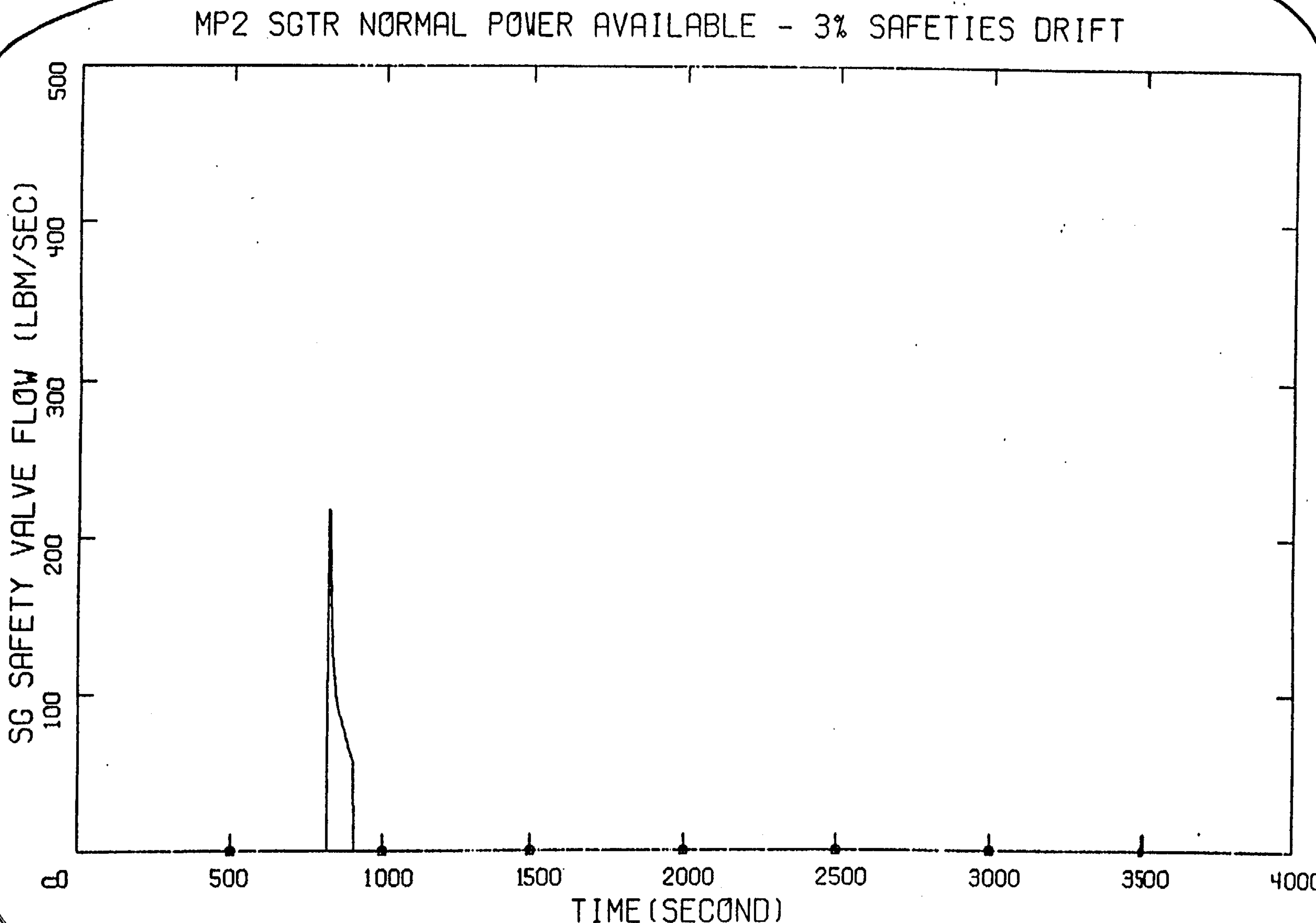

FIGURE 14.6.3-7
STEAM GENERATOR TUBE RUPTURE
SAFETY VALVE FLOW PER STEAM GENERATOR VS. TIME

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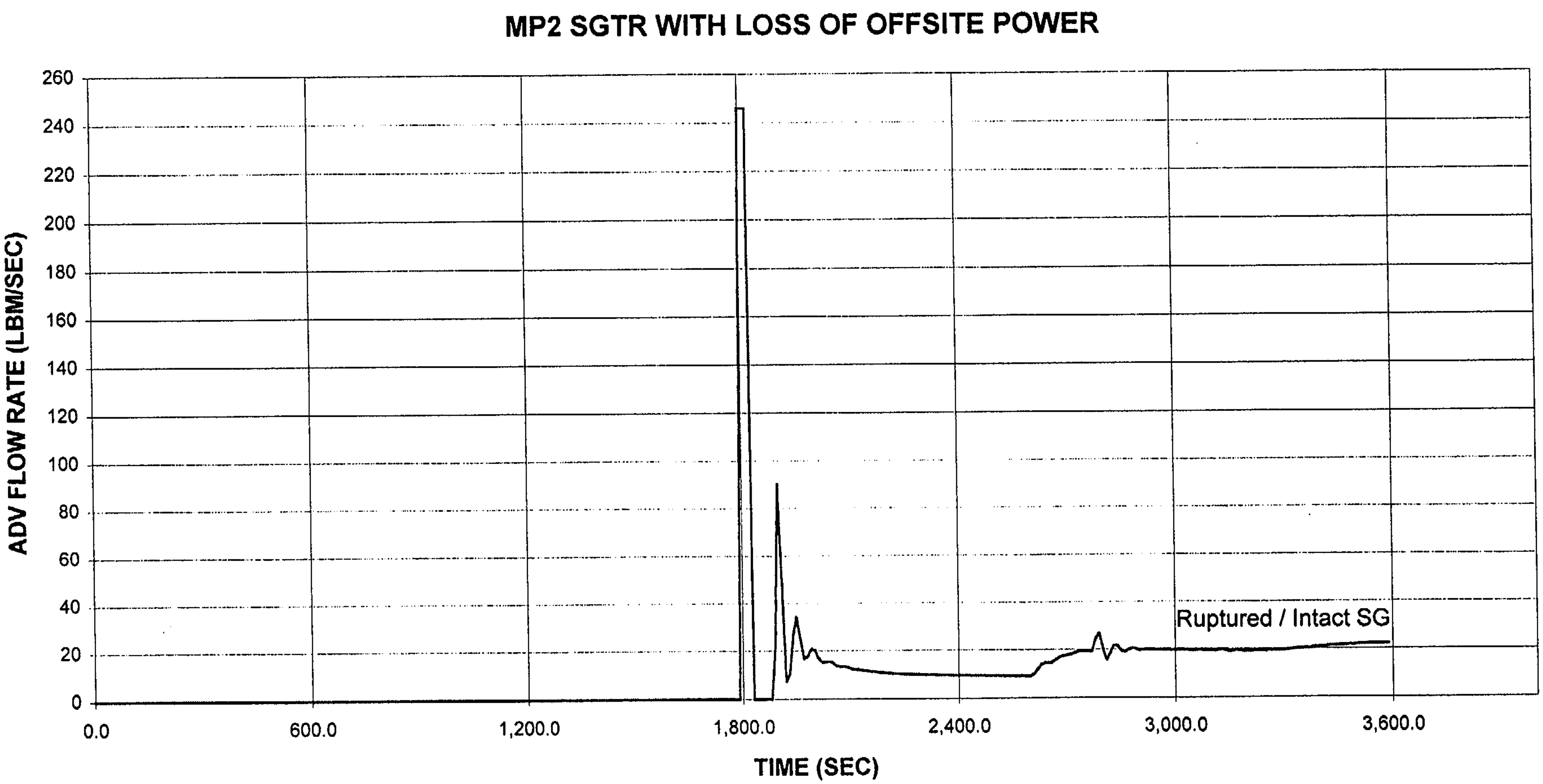

FIGURE 14.6.3-7
STEAM GENERATOR TUBE RUPTURE
ATMOSPHERIC DUMP VALVE FLOW RATE PER STEAM GENERATOR VS. TIME

00-MP2-47

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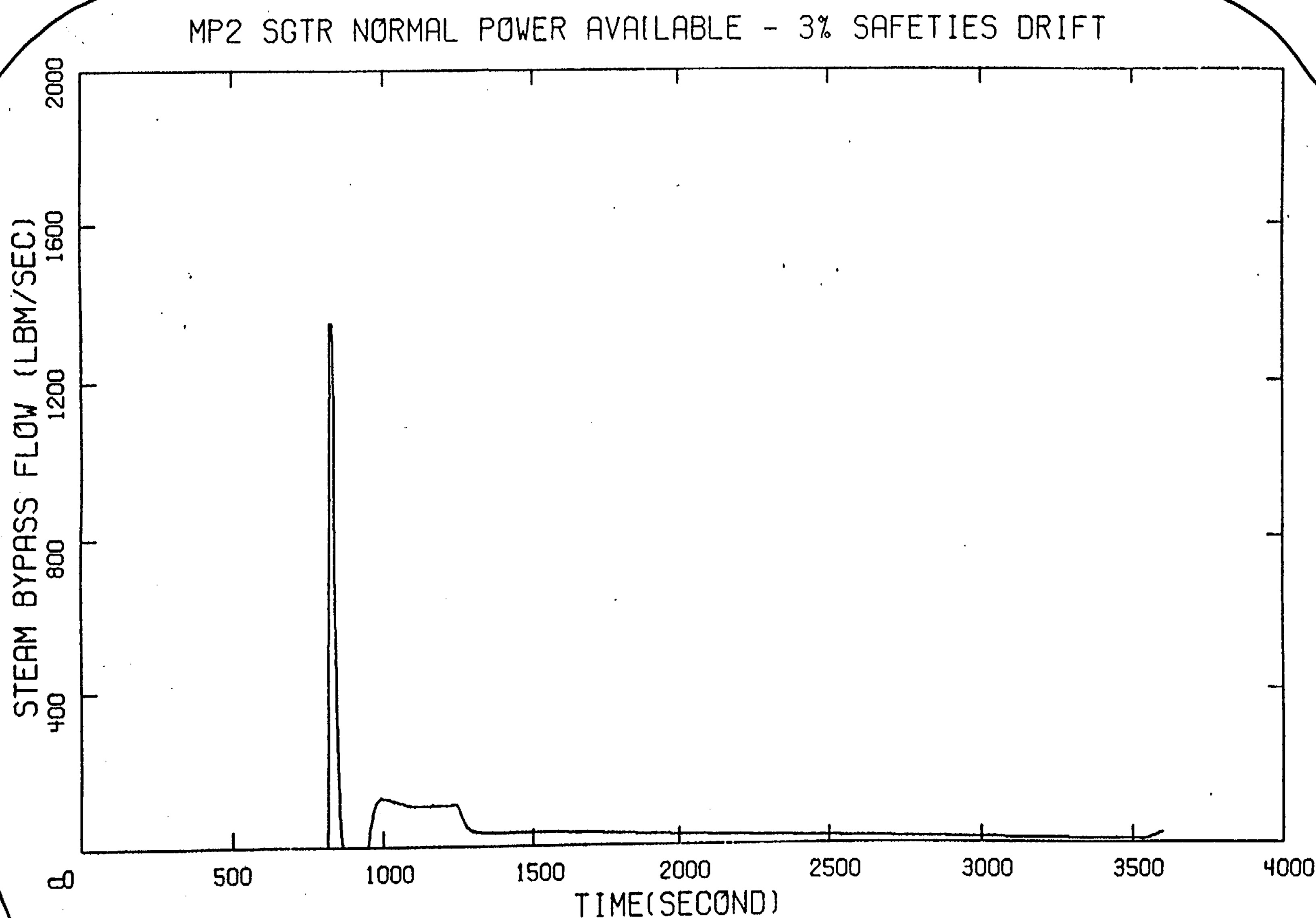

FIGURE 14.6.3-8
STEAM GENERATOR TUBE RUPTURE STEAM BYPASS FLOW VS. TIME

00-MP2-47

MP2 SGTR WITH LOSS OF OFFSITE POWER

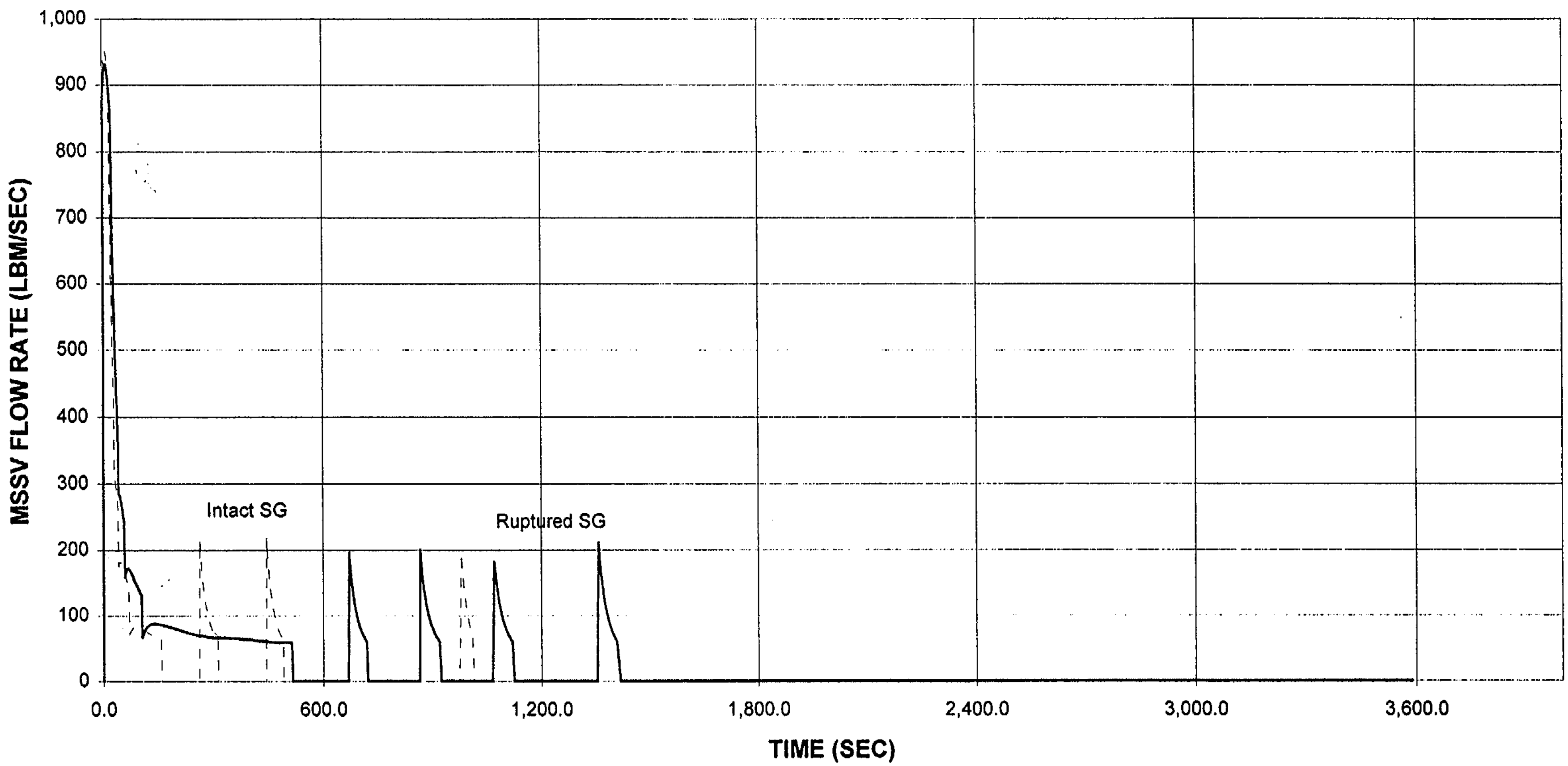

FIGURE 14.6.3-8
STEAM GENERATOR TUBE RUPTURE
MAIN STEAM SAFETY VALVE FLOW RATE PER STEAM GENERATOR VS. TIME

00-MP2-47