VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

February 6, 1995

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Serial No. 94-509A
NA&F-GLD/CGL R1'
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 & 2
CORE UPRATE LICENSING REPORT REVISION
STEAM GENERATOR TUBE RUPTURE REANALSIS

By an August 30, 1994 letter (Serial No. 94-509), the Virginia Electric and Power Company requested changes to the Operating License Nos. DPR-32 and DPR-37 and the Technical Specifications for Surry Power Station Units 1 and 2, respectively. The proposed changes support plant operation at an uprated reactor core power level of 2546 MWt. Our August 30, 1994 submittal included the Surry Core Uprate Licensing Report, which provided supporting documentation for the proposed Operating License and Technical Specification changes.

As documented in the Surry Core Uprate Licensing Report, we submitted reanalyses of the offsite and control room doses for the steam generator tube rupture (SGTR) event in support of the core power uprate license amendment request. Both the site boundary dose and control room dose calculations were based on a single RETRAN thermal/hydraulic simulation of the tube rupture event. Key assumptions for the simulation included:

- A stuck open steam generator PORV on the ruptured steam generator,
- No credit for the condenser steam dumps,
- A double ended rupture near the top of the tube bundle, and
- Continued availability of offsite power (i.e., reactor coolant pumps continue to operate throughout the event).

Based on ongoing review of our tube rupture methodology, we have concluded that the case of tube rupture with a concurrent loss of offsite power should be presented as the limiting case for site boundary dose. This is due to the sensitivity of the Westinghouse Owners Group radioiodine release methodology to break fluid flashing fraction. Loss of forced circulation flow results in slightly higher hot leg temperatures, which in turn results in higher flashed break flow.

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Therefore, we have revised our SGTR analysis to address the loss of offsite power in the site boundary dose calculations. Although the resulting doses are higher, the conclusions of the analysis remain unchanged (i.e., the doses resulting from tube rupture are a small fraction of the 10CFR100 limits). The analysis and conclusions for control room dose remain unchanged from our August 30, 1994 submittal.

Resultant revisions to the text of the Surry Core Uprate Licensing Report are listed in the attached Tabulation of Changes (Attachment 1). Attachment 2 provides revised Licensing Report pages with text changes indicated by bars in the right margins.

It has been determined that the results of the revised SGTR analysis do not alter the previous basis for finding that the proposed core power uprate does not create a significant hazards consideration as defined in 10CFR50.92. In addition, these results have been reviewed and approved by the Station Nuclear Safety and Operating Committee and presented to the Management Safety Review Committee.

Current plant operation (at unuprated conditions) continues to be supported by bounding analyses submitted by a May 5, 1988 letter (Serial No. 88-229). The conclusions reported in that letter are not impacted by the issues discussed herein and remain valid.

If you have questions regarding this SGTR-related revision to the Surry Core Uprate Licensing Report, please contact us.

Very truly yours,

James P. O'Hanlon

James P. O'Hanlon
Senior Vice President - Nuclear

Attachments:
1. Surry Core Uprate Licensing Report Tabulation of Changes - Steam Generator Tube Rupture Reanalysis
2. Surry Core Uprate Licensing Report Revised Pages - Steam Generator Tube Rupture Reanalysis

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ATTACHMENT 1

SURRY CORE UPRATE LICENSING REPORT TABULATION OF CHANGES

STEAM GENERATOR TUBE RUPTURE REANALYSIS

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151 (Figure 3.5.9-4)	151 (Figure 3.5.9-4)
152 (Figure 3.5.9-5)	152 (Figure 3.5.9-5)
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ATTACHMENT 2

SURRY CORE UPRATE LICENSING REPORT
REVISED PAGES

STEAM GENERATOR TUBE RUPTURE REANALYSIS

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LIST OF TABLES (CONTINUED)

recovery procedure can be carried out on such a time scale as to ensure that break flow to the secondary system is terminated before the water level in the affected steam generator can rise into the main steam pipe. Sufficient indications and controls are provided to enable the operator to perform these functions satisfactorily. Simulator training and classroom instruction on the tube rupture accident are a significant emphasis of the licensed operator requalification program.

3.5.9.2 Method of Analysis and Description of the Accident

A. Thermal-Hydraulic Analysis (Loss of Offsite Power Case)

The thermal hydraulic portion of the tube rupture accident is simulated with the Virginia Power RETRAN model (3.5.9-1). Key analysis assumptions were as follows:

1) A double ended tube rupture was modeled. Break flow was calculated by explicitly modelling friction losses in both segments of the ruptured steam generator tube and unchoked flow at the rupture site. This model overpredicts the actual break flows observed in the 1987 North Anna Unit 1 double-ended rupture. The resultant decrease in RCS pressure eventually reduces the overtemperature ΔT trip setpoint to the full power value resulting an a reactor and turbine trip.

2) Following reactor trip on overtemperature ΔT, the condenser dumps are assumed unavailable, and the secondary side pressurizes to steam generator atmospheric relief (PORV) setpoint following turbine trip.

3) The PORV on the main steam header nearest the ruptured generator is assumed to remain fully open until 30 minutes after event initiation. Thus atmospheric releases are assumed for the first 30 minutes. For the normal case of condensers available, a high air ejector radiation signal diverts the air ejector exhaust to containment. Following safety injection, this exhaust path is also isolated. When offsite power and the condenser are available, the volatile species undergo two stages of partitioning (i.e. in the steam generator and the condenser) prior to being released to the atmosphere. Loss of offsite power results in loss of the condenser and in coastdown of the reactor coolant pumps, which increases the break fluid flashing fraction. Flashed break flow is a major contributor to the release of radioisotopes, as discussed in Section 3.7.2.3. Thus the case of loss of offsite power is the limiting case

from the standpoint of site boundary dose, and the analysis for this case assumes loss of the condenser and coastdown of the reactor coolant pumps after reactor trip.

4) After reactor and turbine trip, the Reactor Coolant System continues to depressurize to the safety injection setpoint. Two high head safety injection pumps are assumed to operate. The Reactor Coolant System pressure stabilizes at the point where break flow and safety injection flow are essentially equal.

The thermal hydraulic results are shown in Figures 3.5.9-1 through 3.5.9-7 as follows:

Figure 3.5.9-1 - RCS Average Temperature: Following the rupture, RCS temperature is relatively stable until the unit trips on overtemperature ΔT at 73.9 seconds. The turbine stop valves are assumed to close within the next 2 seconds. Temperature continues to decrease in response to addition of cold safety injection water (safety injection occurs in response to low pressurizer pressure at 289 seconds) and the release of steam through the stuck open PORV (the PORV opens at 88.1 seconds). In actual operating practice, additional cooldown would be imposed by the operators as directed by the emergency procedures to support primary side depressurization to reduce the break flow.

Figure 3.5.9-2 - Reactor Normalized Power: As discussed above, reactor trip is on overtemperature ΔT at 73.9 seconds.

Figure 3.5.9-3 - Ruptured Loop Steam Pressure: After the reactor and turbine trip, pressure in the steam generator initially increases. The expected response would be an increase followed by stabilization at the no-load pressure of about 1005 psig, but since the analysis assumes an atmospheric dump sticks open, there is a gradual depressurization.

Figure 3.5.9-4 - Pressurizer Pressure: The initial drop in pressurizer pressure results from excess of tube rupture flow over the charging flow. The pressurizer level controller, which would increase charging flow and tend to retard this initial depressurization, is not modeled. Immediately following reactor trip the depressurization rate is accelerated. Safety injection is initiated on low pressurizer pressure, the depressurization drops significantly as a result.

Figure 3.5.9-5 - Flow Through the Stuck Open PORV: This represents the primary potential source of radioactivity transport to the environment if the condenser steam dumps are not available.

Figure 3.5.9-6 - Break Flow: The initial break flow through the two ends of the ruptured steam generator tube is about 80 lbm/sec or approximately 800 gpm. The flow drops off quickly in response to the RCS depressurization until safety injection is initiated. Then the flow stabilizes, as equilibrium between the break flow and safety injection is established, at about 550 gpm. The slight increasing trend in mass flow beyond this point is a result of increased fluid density due to the RCS cooldown.

Figure 3.5.9-7 - Integrated Break Flow: At one-half hour after initiation of the event, approximately 111,500 lbm of fluid has been transferred from the RCS to the secondary side of the ruptured steam generator.

This calculation takes no credit for operator action to cool down and depressurize the Reactor Coolant System prior to steam generator isolation, i.e., for the first 30 minutes. In an actual event, within 30 minutes the operators would be expected to achieve the following:

1. Ensure that power is available to the emergency buses and that safety injection and auxiliary feedwater are actuated. Verify that main feedwater is isolated.
2. Control the reactor system cooldown to maintain no-load temperature. Stop the reactor coolant pumps if safety injection flow to the core is indicated and the minimum required RCS subcooling is not maintained.
3. If not already completed, identify the ruptured steam generator by rising water level or high steam line radiation indications and isolate flow from this steam generator. Adjust auxiliary feedwater flow to maintain the specified water levels in the ruptured and intact steam generators.
4. Initiate RCS cooldown through the intact steam generators by dumping steam to the main condenser or through the steamline PORV (depending on the availability of offsite power).

5. Depressurize the RCS to minimize break flow and refill the pressurizer using the pressurizer spray or, if spray is unavailable, the pressurizer PORVs. Maintain the RCS pressure within the pressure-temperature limit curve for the Reactor Coolant System.
6. Terminate safety injection flow upon establishing required minimum RCS subcooling, secondary heat sink requirements and level in the pressurizer.
7. Establish normal letdown and charging functions and control RCS pressure to minimize primary-to secondary leakage.
8. Initiate appropriate post - SGTR cooldown procedures.

B. Thermal-Hydraulic Analysis - Control Room Dose (Offsite Power Available) Case

The thermal/hydraulic analysis performed to support the calculation of control room dose is similar to that presented above. However, the calculated control room dose is more limiting under the assumption of offsite power continuing to remain available. Continued availability of offsite power would result in a potentially larger forced intake of unfiltered air from the normal control room air inlets prior to control room isolation than the case of concurrent loss of offsite power.

Therefore the thermal/hydraulic analysis used to develop the control room calculation assumes continued operation of the reactor coolant pumps after reactor trip. However, no credit is taken for operation of the condenser dumps. As with the previous case, releases are assumed to be via a stuck open PORV on the main steam header leading from the ruptured generator.

A revised dose analysis has been performed, using the thermal hydraulic results presented above. Details of this analysis and the resultant offsite and control room doses are presented in Section 3.7.2.3.

References

(3.5.9-1) VEP-FRD-41-A, "Reactor System Transient Analyses Using the RETRAN Computer Code," May 1985.

Figure 3.5.9-1

Steam Generator Tube Rupture - RCS Average Temperature

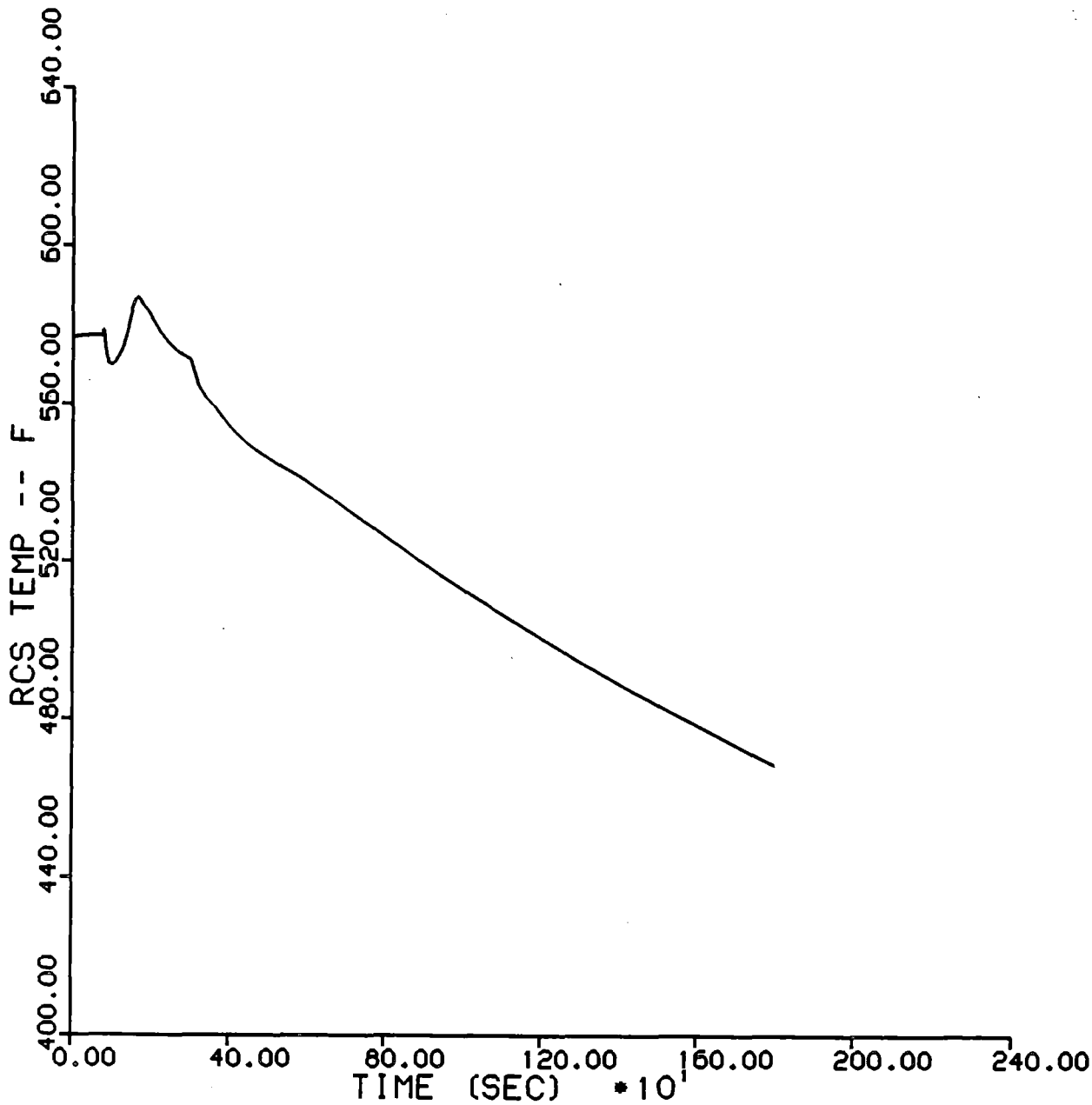

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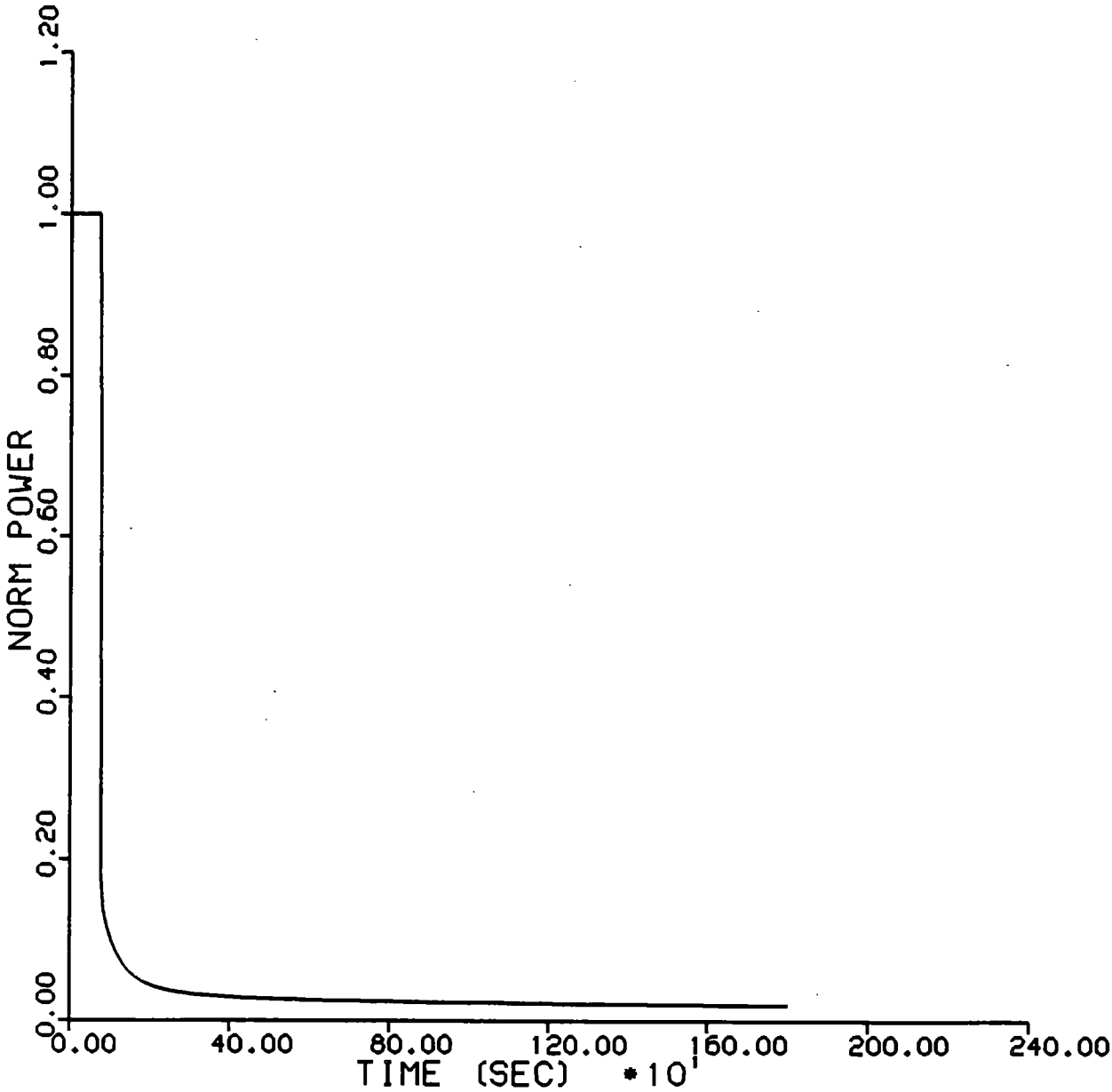
Figure 3.5.9-2

Steam Generator Tube Rupture - Reactor Normalized Power



12/19/94 16.14.03

Figure 3.5.9-3

Steam Generator Tube Rupture - Ruptured Steam Generator Pressure

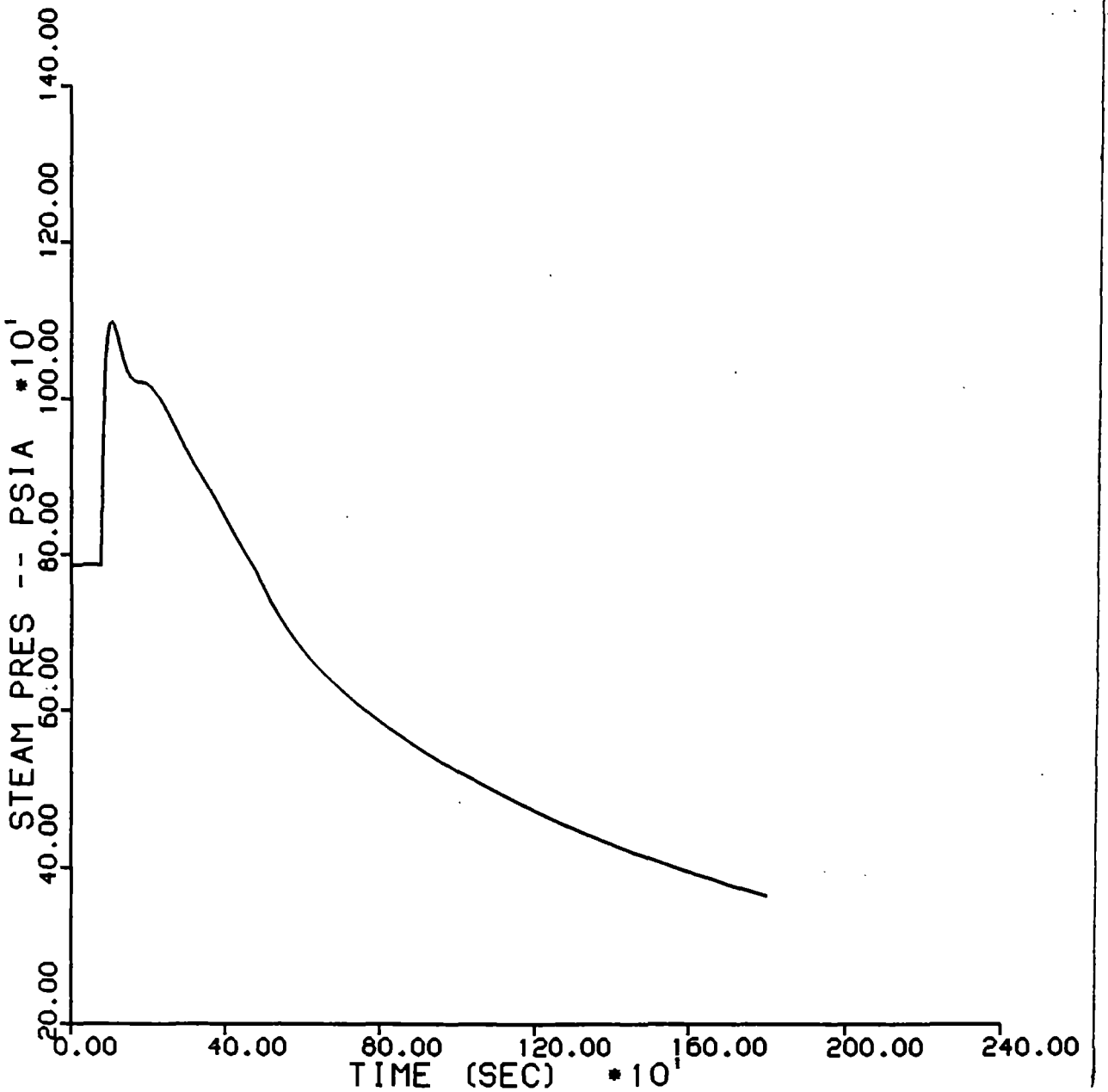


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Figure 3.5.9-4

Steam Generator Tube Rupture - Pressurizer Pressure

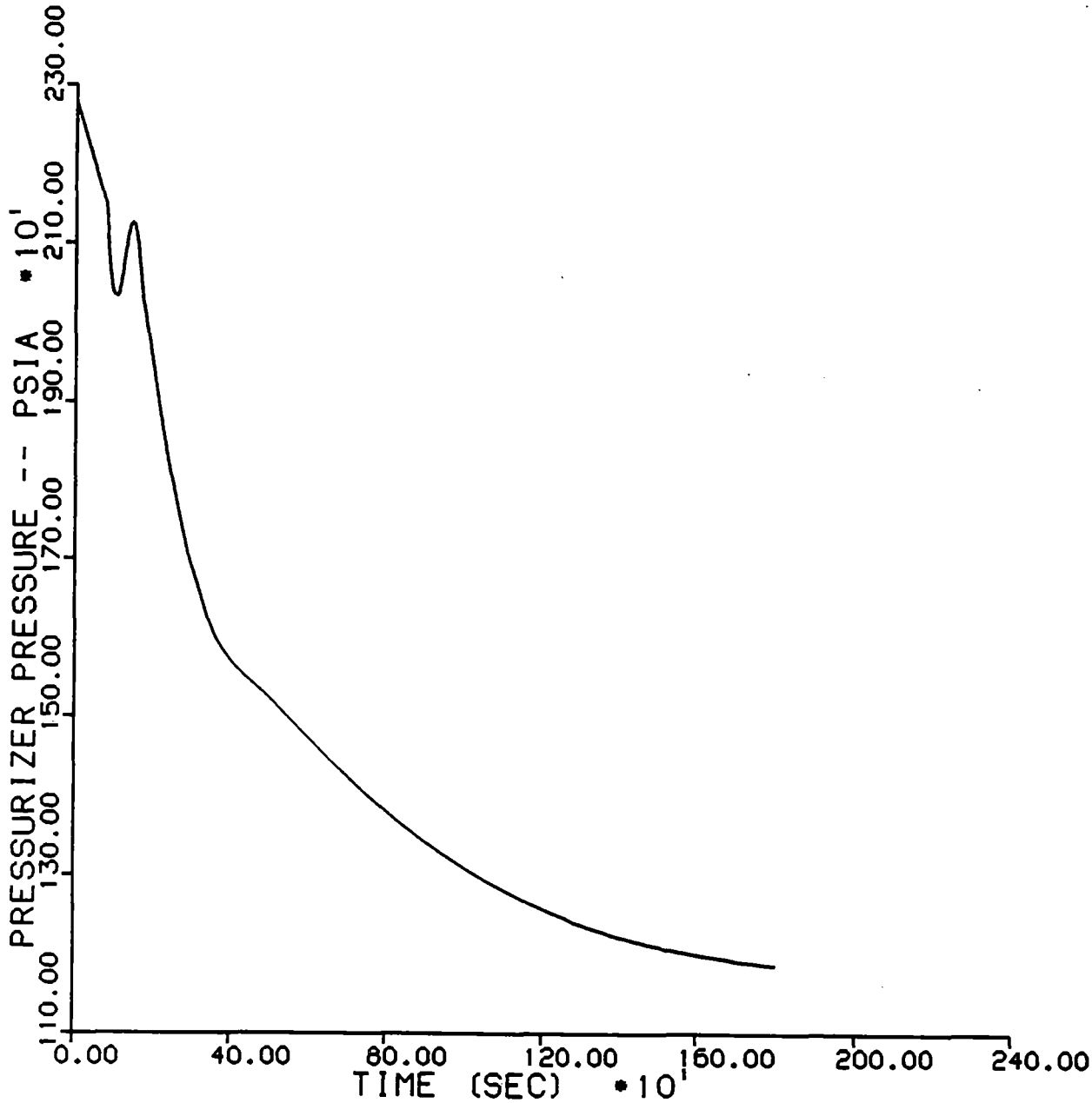

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Figure 3.5.9-5

Steam Generator Tube Rupture - Open SG PORV Mass Flow Rate

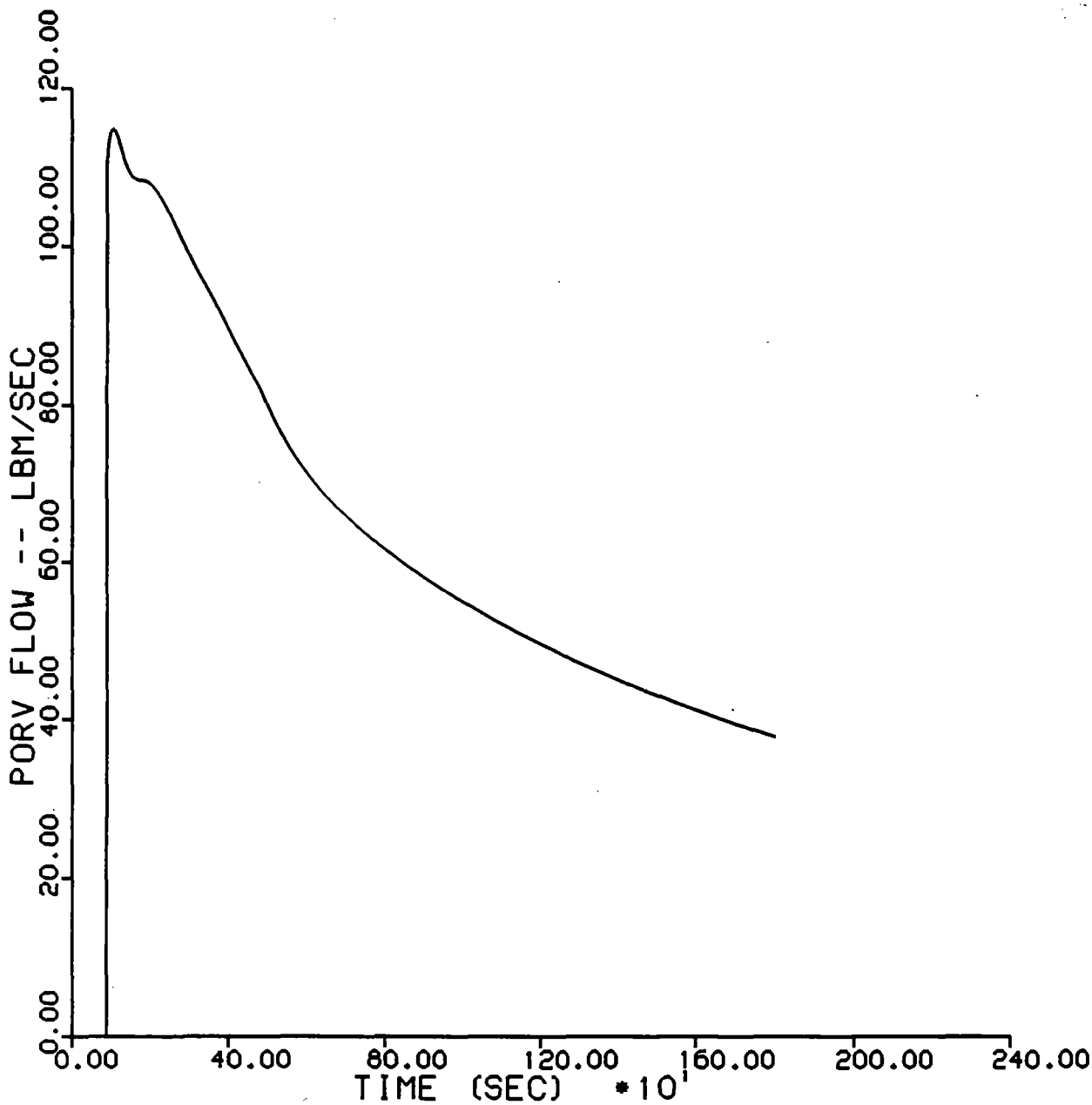

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Figure 3.5.9-6

Steam Generator Tube Rupture - Break Mass Flow Rate

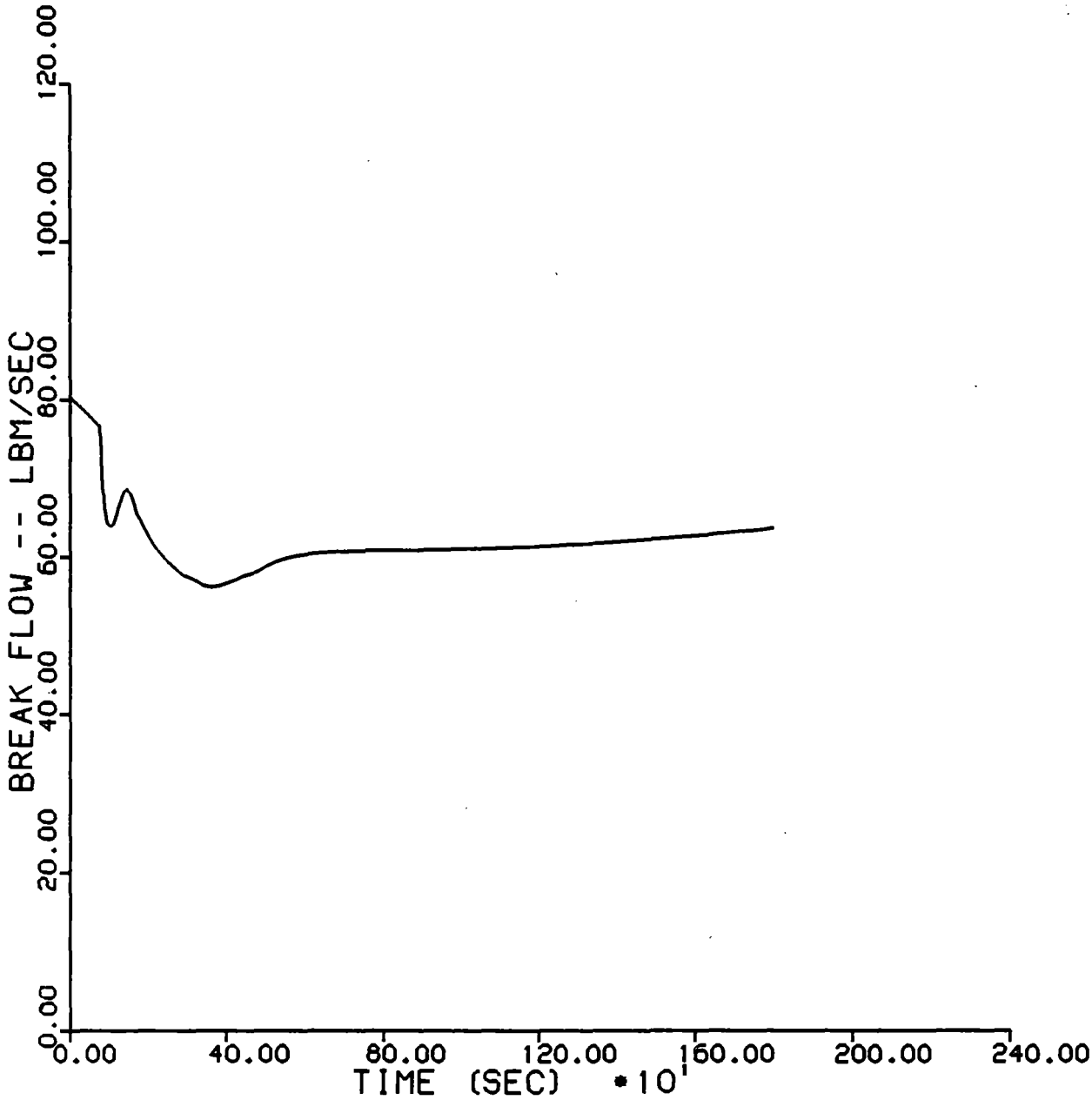

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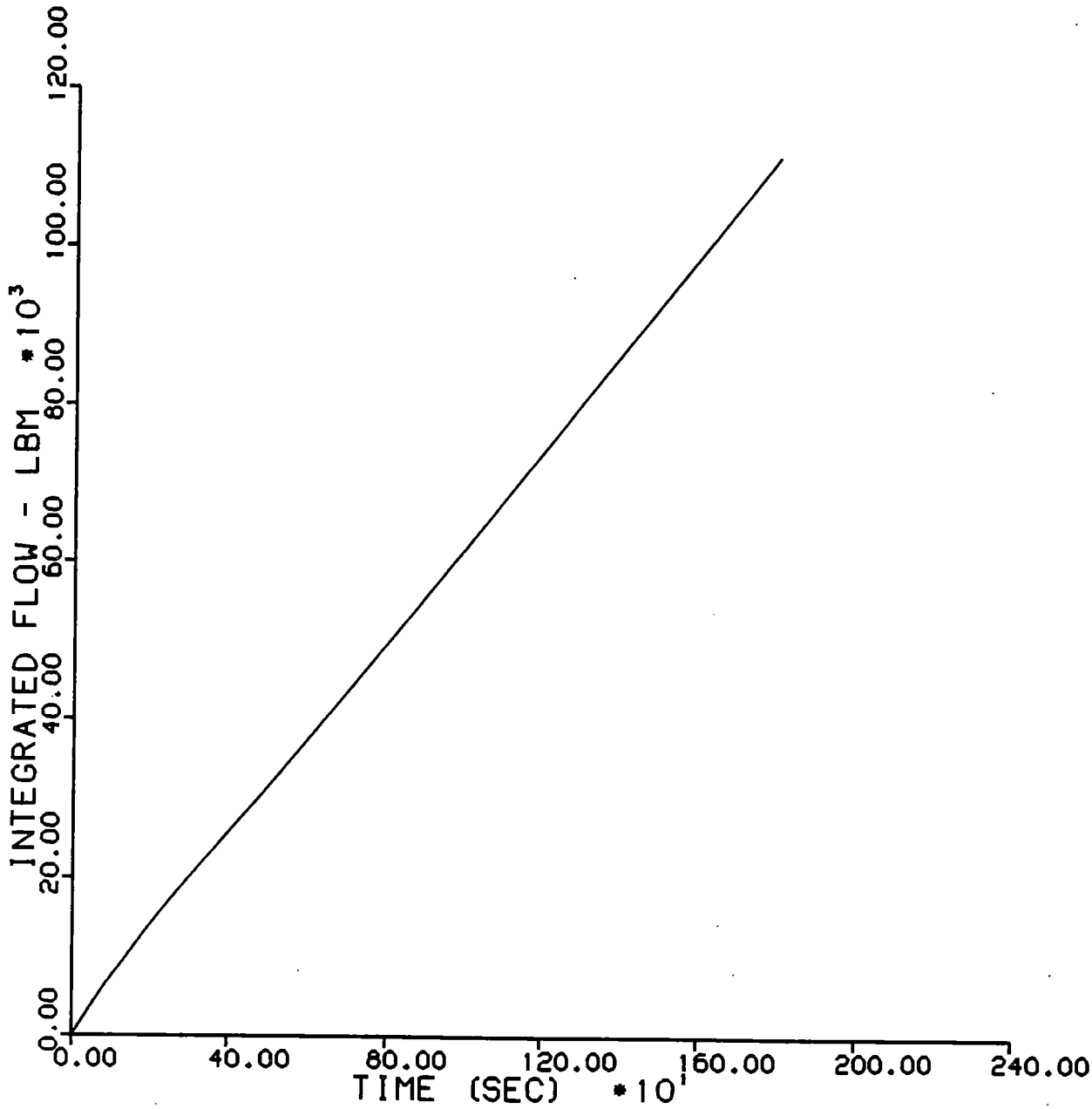
Figure 3.5.9-7

Steam Generator Tube Rupture - Break Integrated Mass Flow



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3.7.2.3 Steam Generator Tube Rupture (SGTR)

A steam generator tube rupture (SGTR) is a break in a tube carrying primary coolant through the steam generator. This postulated break allows primary liquid to leak to the secondary side of the steam generator with an assumed release to the environment through the steam generator Power Operated Relief Valves (PORVs) or the steam generator safety valves. Steam is assumed to be discharged from the affected generator to the environment for 30 minutes until the generator is isolated. As required by the NRC Standard Review Plan, the SGTR analysis was performed assuming both a pre-accident iodine spike and a concurrent accident iodine spike.

In a SGTR, the release point for all three steam generators is the same as for the unaffected steam generators in the MSLB. However, since there is a short time delay between the tube rupture and isolation of the control room inlet during a SGTR, a control room χ/Q for the normal inlet also had to be calculated. The control room χ/Qs used after isolation for the SGTR are the same as those used for releases from the unaffected steam generators during a MSLB (Section 3.7.2.2).

3.7.2.3.1 SGTR Analysis Assumptions

Virginia Power recognized in 1988 that tube bundle uncovery could impact doses from a Steam Generator Tube Rupture (SGTR). In a letter from D. S. Cruden (Virginia Power) to NRC (3.7-1), Virginia Power committed to analyze SGTR doses with tube bundle uncovery and submit this analysis for NRC review after the Westinghouse Owners Group (WOG) developed a methodology for this analysis. This methodology was submitted to NRC in March 1992 (3.7-2). Reference 3.7-1 also discussed the results of an interim, conservative analysis of the steam generator tube rupture which was based on bounding estimates of the impact of tube uncovery on radioidine releases. As discussed in that submittal, the results showed doses at the Exclusion Area and Low Population Zone boundaries which were within the 10 CFR 100 limits, but which exceeded the Standard Review Plan Section 15.6.3 criteria (i.e., well within the 10 CFR 100 limits). The analysis presented below shows that with the use of the approved WOG methodology, calculated LPZ and EAB doses are indeed within the Standard Review Plan guidelines. As indicated in the WOG methodology, SGTR releases consist of four components:

1) Releases from secondary liquid boiling including allowance for a partition factor of 0.01 for iodine between secondary liquid and steam.
2) Releases from the fraction of primary liquid break flow that flashes to steam. A partition factor of 1 is assumed for this flashing fraction.
3) Releases from primary liquid bypassing the secondary side.
4) Releases caused by secondary moisture carryover.

As shown in Reference (3.7-2), releases from a SGTR are dominated by the first two terms above for a case with a stuck open PORV. A stuck open PORV also produces a larger radionuclide release than a cycling PORV or a PORV that fails closed and causes the steam generator safety valves to open to relieve secondary side pressure.

Uncovery of the tube bundle in a SGTR does not significantly increase radionuclide releases for the stuck open PORV case. If the tube bundle is uncovered in a SGTR and the PORV is stuck open, the third release term described above increases, but it is still only a small part of the total release.

The LOCADOSE computer models for the SGTR analysis shown below are based on the methodology developed by the Westinghouse Owner's group (3.7-2). These models include only the first two terms discussed above. This does not significantly affect the model results because these two terms dominate the releases for the stuck open PORV case, which is the limiting case for radionuclide releases.

3.7.2.3.2 Initial Radioisotope Concentrations

Initial radionuclide concentrations of the primary and secondary systems for the SGTR accident are the same as those for the MSLB. The analyses of both the SGTR and the MSLB accidents indicate that no additional fuel rod failures occur as a result of these transients. Thus, radioactive material releases are determined by the radionuclide concentrations initially present in primary liquid, secondary liquid and secondary steam, plus any releases from fuel rods that have failed before the transient. These radionuclide inventories and concentrations are shown in Tables 3.7.2.2-1 and 3.7.2.2-2.

3.7.2.3.3 Determination of χ/Q values

During a SGTR with loss of offsite power, or the condenser otherwise unavailable, the steam generators release steam through the secondary system PORVs. The control room χ/Q values for releases from the steam generator PORVs to the control room emergency air inlet are discussed in Section 3.7.2.2.3 and shown in Table 3.7.2.2-4. The limiting case for control room dose assumes loss of the main condenser; however, offsite power is assumed to remain available to power the normal control room ventilation fans.

The distance from the closest PORV to the control room normal air inlet is shorter than the distance to the emergency inlet, so a different χ/Q value is applicable for releases when the normal control room ventilation system is in use. Based on the Reference (3.7-8) methodology, the control room χ/Q for the normal inlet was determined to be 7.71×10^{-3} sec/m^3. This χ/Q is only used for the short time (247 seconds) before the control room is isolated from the normal inlet air by a SI signal.

3.7.2.3.4 Steam Generator Tube Rupture LOCADOSE Models

The LOCADOSE computer code system (3.7-3) (3.7-4) (3.7-5) was used to model the SGTR. Models were developed for both a pre-accident iodine spike case and a concurrent accident iodine spike case. The two models are identical except for the initial radioisotope inventories and the inclusion of modeling of iodine release from the fuel rods for four hours for the concurrent accident case.

The primary system, steam generator and control room volumes for the SGTR are the same as for the MSLB (Table 3.7.2.2-3). The liquid properties are also the same. As for the MSLB analysis, the release of the radionuclides contained in the steam from all three steam generators was modeled as essentially a puff release occurring when the PORVs open.

The primary coolant leakage to the unaffected steam generators was based on the maximum leakage allowed by Technical Specifications. The maximum leakage allowed from all three generators in Surry Technical Specification 3.1.C-6 is 1 GPM.

For conservatism, all of this leakage was assumed to occur into the two unaffected steam generators. This assumption is conservative because the unaffected generators release steam to the environment for 8 hours compared to 30 minutes for the affected generator.

The break flow rates through the ruptured tube to the affected steam generator were based on the thermal hydraulic analysis of a complete double ended tube rupture presented in Section 3.5.9. To be consistent with the regulatory guidance in SRP Section 15.6.3 (3.7-6), the liquid and steam break flows are modeled separately. The break flow rates and release rates to the environment are summarized in Tables 3.7.2.3-1 and 3.7.2.3-2 for the cases with and without continued availability of offsite power (used for the control room and site boundary dose calculations, respectively).

The liquid break flow from the primary system is modeled as mixing with the secondary liquid in the affected steam generator. The flow from the secondary liquid to the secondary steam is then modeled assuming a partition factor of 0.01 for iodine. This technique for modeling a SGTR with uncovery of the tube bundle was developed in a generic study by the Westinghouse Owners Group (3.7-2).

The fraction of the break flow that flashes to steam is modeled as being transferred to the affected steam generator steam space. Once in the steam generator steam space, the radionuclides in this part of the break are almost immediately released to the environment.

The primary and secondary system releases are replaced with safety injection and auxiliary feedwater flows. Therefore, the volume of the primary and secondary liquids remains relatively constant during this transient.

The flow from the affected generator through the condenser was represented for the time interval between the tube rupture and the opening of the PORV. During this period, there is some build-up of radionuclide inventory in the affected generator liquid and steam volumes. A very small volume and a large return flow to the steam generator liquid space was used for the condenser. This conservatively ignores the dilution and retention of radionuclides in the condenser. The flow through the condenser for the unaffected generators is not modeled because there is no rapid build-up of radionuclides in these generators. The radionuclide inventory in these generators is modeled based on the initial inventory and the primary to secondary leakage.

The model for the control room ventilation system for the SGTR is similar to that used for the LOCA and MSLB analyses, with some differences incorporated to more accurately model the timing of the sequence of events of the SGTR. The start of the accident is the tube rupture itself. The PORV on the faulted steam generator was determined to open 88 seconds after the break, and the SI signal is generated at 247 seconds for the case assuming continued availability of offsite power. The timing of these events was extracted from the thermal-hydraulic analysis presented in Section 3.5.9. During this time, the control room is being supplied via the normal ventilation system, with a 3000 cfm intake air flow rate. The control room isolates automatically on initiation of the SI signal and is then assumed to be on bottled air until 1.0 hour after the tube rupture. From 1.0 hour until the end of the accident, the control room is provided

with a filtered air supply of 1000 cfm. An unfiltered inleakage of 10 cfm was assumed for the entire time the control room is isolated. The control room intake filter efficiency assumed was 90% and 30% for elemental and organic iodine respectively.

3.7.2.3.5 Results of Dose Calculations for SGTR

Both pre-accident and concurrent accident iodine spike cases were analyzed for the steam generator tube rupture. The skin and whole body doses for both cases were below 0.1 Rem. These low doses are well below the regulatory criteria.

The limiting case for the control room thyroid doses was determined to be for a pre-accident iodine spike with continued availability of offsite power. The calculated thyroid dose for this case is less than the value reported to the NRC in Reference (3.7-10). A comparison of the doses calculated for the limiting SGTR accident scenario with the GDC-19 criteria is shown in Table 3.7.2.3-4. All calculated control room doses for the Surry steam generator tube rupture remain below the GDC-19 criteria.

As discussed in Reference 3.7-1, the EAB doses reported in the Surry UFSAR did not consider tube uncovery. As noted in this reference, tube uncovery has the potential to increase EAB doses. The revised doses shown in Table 3.7.2.3-2 are greater than the values shown in Surry UFSAR Section 14.3.1, but these doses are less than the 10 CFR 100 limits and meet the SRP 15.6.3 review criteria of less than a small fraction (10%) of 10 CFR 100.

Table 3.7.2.3-1

Steam Generator Tube Rupture Break Flow Rates and Releases (Case with Offsite Power Not Available)

Affected Steam Generator

Time (sec)	RCS to SG Liquid Flow (lb_m)	RCS to SG Liquid Flow (cfm)	RCS to SG Steam Flow (lb_m)	RCS to SG Steam Flow (cfm)	SG Liquid to Steam (cfm)	Steam Release (cfm)
0 - 88	6075	91.6	747	11.26	1330	0
88 - 289	12020	79.4	687	4.54	134	3735
289 - 1800	87733	77.0	4221	3.71	73	2038

Time (sec)	SG Liquid to Steam Flow (cfm)	SG Steam to Condenser Flow (cfm)
0 - 88	----	37080

Unaffected Steam Generators

Time Period	Steam Mass (lb_m)	Flow Rate (cfm)
0 sec - 88 sec	0	0
88 sec - 500 sec	41906	127
500 sec - 30 min	0	0
30 min - 2 hr	179398	41
2 hr - 8 hr	632579	37

Table 3.7.2.3-2

Steam Generator Tube Rupture Break Flow Rates and Releases (Offsite Power Available)

Affected Steam Generator

Time (sec)	RCS to SG Liquid Flow (lb_m)	RCS to SG Liquid Flow (cfm)	RCS to SG Steam Flow (lb_m)	RCS to SG Steam Flow (cfm)	SG Liquid to Steam (cfm)	Steam Release (cfm)
0 - 88	6238	94.1	501	7.55	1330	0
88 - 247	9997	83.4	82	0.68	145	4036
247 - 1800	90270	77.1	677	0.58	113	3156

Time (sec)	SG Liquid to Steam Flow (cfm)	SG Steam to Condenser Flow (cfm)
0 - 88	----	37080

Unaffected Steam Generators

Time Period	Steam Mass (lb_m)	Flow Rate (cfm)
0 sec - 88 sec	0	0
88 sec - 393 sec	43315	177
393 sec - 30 min	0	0
30 min - 2 hr	179398	41
2 hr - 8 hr	632579	37

Table 3.7.2.3-3

Steam Generator Tube Rupture Offsite Doses (Case with Loss of Offsite Power, Concurrent Accident Iodine Spike

Dose Type	EAB 2-hour Dose (Rem)	LPZ 30-Day Dose (Rem)	10 CFR 100 Limit (Rem)
Thyroid	15.4	0.7	300
Skin	<0.1	<0.1	---
Whole Body	<0.1	<0.1	25

Table 3.7.2.3-4

Steam Generator Tube Rupture Control Room Doses (Offsite Power Available, Pre-Accident Iodine Spike)

Dose Type	30-day Dose (Rem)	GDC-19 Criteria[5] (Rem)
Thyroid	8.1	30
Skin	<0.1	30
Whole Body	<0.1	5

[5]Control room skin and thyroid dose criteria are not specified in GDC-19; values shown are taken from SRP Section 6.4.

Table 3.7.3-1

Control Room and Off Site Doses Previously Reported to NRC

CONTROL ROOM

Accident	Thyroid Dose (Rem)	Whole Body Dose (Rem)	Skin Dose (Rem)
LOCA	26.6	0.5	1.3
MSLB	1.7	<0.1	<0.1
SGTR	16.2	<0.1	0.3
FHA	0.9	<0.1	0.1
WGDT		0.5	19.7

EXCLUSION AREA BOUNDARY

Accident	Thyroid Dose (Rem)	Whole Body Dose (Rem)
LOCA	248.0	5.0
MSLB	12.3	
SGTR	---[8]	0.3
FHA	171.2	7.5
WGDT	0.1	3.7[9]

LOW POPULATION ZONE

Accident	Thyroid Dose (Rem)	Whole Body Dose (Rem)
LOCA	23.3	0.4

[8]Less than 10CFR100 but in excess of SRP 15.6.3 review criteria, per Reference 3.7-1.

[9]Based on a postulated tank content of 95,400 Ci of Xe-133.

Table 3.7.3-2

Summary of Control Room and Offsite Doses

CONTROL ROOM

Accident	Thyroid Dose (Rem)	Whole Body Dose (Rem)	Skin Dose (Rem)
LOCA	29.0	0.2	0.1
MSLB	3.6	<0.1	<0.1
SGTR	8.1	<0.1	<0.1
LRA	10.6	0.2	1.0
FHA	2.4	0.1	0.1

EXCLUSION AREA BOUNDARY

Accident	Thyroid Dose (Rem)	Whole Body Dose (Rem)	Skin Dose (Rem)
LOCA	224.0	6.0	3.0
MSLB	3.6	<0.1	<0.1
SGTR	15.4	<0.1	<0.1
LRA	2.1	0.3	0.2
FHA	55.0	1.6	1.8
WGDT	--	<0.5	

LOW POPULATION ZONE

Accident	Thyroid Dose (Rem)	Whole Body Dose (Rem)	Skin Dose (Rem)
LOCA	12.0	0.3	0.2
MSLB	0.4	<0.1	<0.1
SGTR	0.7	<0.1	<0.1
LRA	0.7	<0.1	<0.1
FHA	2.4	0.1	<0.1