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WCAP-13347

SAN ONOFRE NUCLEAR GENERATING STATION - UNIT 1 (SONGS-1)

MAIN STEAMLINE BREAK ANALYSIS

FOR

REVISED MODERATOR DENSITY COEFFICIENTS

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SONGS-1 MSLB Analysis for Revised
Moderator Density Coefficients

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1.0 INTRODUCTION

At the request of Southern California Edison (SCE), Westinghouse has performed analyses of selected Main Steamline Break (MSLB) cases for San Onofre Nuclear Generating Station - Unit 1 (SONGS-1). These analyses were performed to support revised Moderator Density Coefficients (MDCs) for End-of-Cycle (EOC) conditions.

The MSLB analyses performed include cases for Core Response evaluation and a case to determine the Mass & Energy (M&E) releases inside containment for Containment Integrity evaluations by SCE. In both the Core Response and M&E Release MSLB analyses, only the cases which represent the limiting MSLB conditions for SONGS-1 are analyzed. For Core Response, this includes three MSLB cases, each initiated from Hot Zero Power (HZP) initial conditions. For the M&E Releases inside containment, the limiting case analyzed is the MSLB initiated from a Hot Full Power (HFP) initial condition.

The analysis information contained within this report serves as the formal documentation required to support Cycle 11 operation with an EOC HFP equivalent Moderator Temperature Coefficient (MTC) less negative than or equal to -29 pcm/°F. To ensure that the -29 pcm/°F value is not exceeded during Cycle 11, a surveillance limit at 300 ppm boron should be set at -24.9 pcm/°F. This surveillance value as well as the absolute limit at EOC were determined assuming the Cycle 11 core models, and have not included any allowances for future cycles.

In addition to the revised MDCs corresponding to the above EOC MTC limit, these calculations consider an increase in the boron concentration in the Safety Injection (SI) lines to a minimum values of 3000 ppm and an increase in the HZP EOC minimum shutdown margin requirement to a value of 2.05% $\Delta k/k$. These changes are made to offset the penalty associated with the revised MDCs. These analysis assumptions along with other pertinent analysis assumptions are included in the sections that follow.

It should be noted that the increase to the boron concentration in the SI lines and the increase in the HZP EOC minimum shutdown margin requirement are both made in a direction which results in less adverse conditions for a MSLB event. Hence, these changes are conservative with respect to plant safety. Furthermore, since these changes only affect the analysis assumptions for the MSLB events, the changes in minimum shutdown margin and SI line boron concentration do not adversely affect any of the other SONGS-1 licensing basis safety analyses.

No analysis of the MSLB M&E releases outside containment is presented since the existing analyses and evaluations for this event support a HFP EOC MTC more negative than -29 pcm/°F and were conservatively analyzed assuming a lower boron concentration in the SI lines (1500 ppm) and a lower EOC HZP shutdown margin (1.9% $\Delta k/k$).

Section 2.0 of this report documents the MSLB analysis performed to address Core Response (e.g., DNBR). Section 3.0 documents the MSLB analysis performed to determine the M&E release information.

The steamline break analyses performed assume a Cycle 11 burnup of 16,015 MWD/MTU to allow for future coastdowns. The shutdown margin assumed in the analysis (2050 pcm) has already been confirmed to be met for Cycle 11 up to a burnup of 11,400 MWD/MTU in Reference 1. 11,400 MWD/MTU is the current licensing limit defined in the RSE for Cycle 11. Should SCE choose to operate Cycle 11 past 11,400 MWD/MTU, a coastdown analysis will still be required to assess the shutdown margin and other RSAC parameters, but the MSLB analysis remains bounding up to a burnup of 16,015 MWD/MTU.

2.0 MSLB Core Response

Included in this section are the results of the hypothetical main steamline break (MSLB) Core Response analysis for the San Onofre Nuclear Generating Station - Unit 1 (SONGS-1).

The analysis of the MSLB event for core response was requested by Southern California Edison (SCE) as part of the overall effort to evaluate a revision to the Moderator Density Coefficients (MDCs) for SLB conditions. This MSLB Core Response analysis assumes MDCs for SLB conditions which are equivalent to an EOC HFP MTC of -29 pcm/°F (with uncertainties).

In addition to the revised MDCs, these calculations also consider an increase in the boron concentration in the SI lines to a minimum value of 3000 ppm and an increase in the HZP EOC minimum shutdown margin requirement to a value of 2.05% $\Delta k/k$.

A total of three MSLB cases have been analyzed for core response. No analysis of the Credible SLB event was performed since this event is bounded by the MSLB cases analyzed herein. The three MSLB cases analyzed are as follows:

Break Type (1)	Break Location (2)	Loop receiving SI flow (3)
MSLB	Downstream	Intact
MSLB	Upstream	Intact
MSLB	Upstream	Faulted

(1) - MSLB is hypothetical main steamline break.
(2) - Break location relative to flow restrictor.
(3) - The faulted loop is defined as the loop in which the steamline ruptures. The other two loops are referred to as intact loops.

A hypothetical steamline break is defined as the double ended rupture of a main steamline. This event is classified as an ANS Condition IV event, a limiting fault. Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to the public health and safety in excess of guideline values of 10 CFR 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the Emergency Core Cooling System and Containment.

The purpose of this analysis is to show that the acceptance criteria stated above are met for the three cases analyzed with the revised MDCs, an increase in the SI line boron concentration, and an incrase in the HZP EOC minimum shutdown margin. The acceptance criteria for hypothetical breaks (MSLB) cases is demonstrated by showing that no DNB occurs. This ensures that there is no damage to the fuel cladding and no release of fission products from the fuel to the RCS. The acceptance criterion of no fuel rod failures for credible break case is demonstrated by showing that no DNB occurs.

The results (see Section 2.4) of these three MSLB cases showed that the minimum DNBR remained above the limit value in all cases. This ensures that DNB will not occur following the hypothetical (and credible) steamline break scenarios. Therefore, no releases of fission products from the fuel will result from a steamline break assuming the revised MDCs, 3000 ppm boron in
the SI lines, and a minimum HZP EOC shutdown margin of 2.05% $\Delta k/k$. Thus, the acceptance criteria for the steamline break core response event are met. The details of this analysis follow.

2.1 Transient Description

The steam releases arising from a rupture of a main steamline would result in an initial increase in steam flow from all three steam generators which decreases during the transient as steam pressure decreases. The increase in energy removal from the RCS causes a reduction of coolant temperature. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity which may cause a return to power. The decrease in reactor coolant temperature also causes the water in the RCS to shrink which reduces pressurizer level and pressure. The shrink in the RCS inventory may be severe enough to cause the pressurizer to empty and the fluid in the upper head of the reactor vessel to saturate.

In the event that the reactor is at power, a reactor trip would be generated manually or by the reactor protection system from one of the following signals.

1. High nuclear flux
2. Steam and feedwater flow mismatch
3. Safety injection initiation

Following the reactor trip or if the transient is initiated from zero power, there is a possibility that the core will return to power due to the positive reactivity insertion. The return to power is limited by Doppler reactivity feedback and the introduction of borated water from the safety injection system. The core is ultimately shutdown by borated water from the safety injection system and/or from the chemical and volume control system.

Safety injection may be actuated during the transient manually or by a signal generated from low pressurizer pressure or high containment pressure. Feedwater, which enhances the RCS cooldown, would be isolated manually or by safety injection initiation.

2.2 Analysis Methodology

The analysis of the steamline rupture has been performed to determine:

1. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steamline rupture. The LOFTRAN code (Reference 2) was used.

2. The thermal and hydraulic behavior of the core following a steamline rupture. A detailed thermal and hydraulic digital-computer code, THINC, was used to determine if DNB occurs for the conditions computed in item 1.

2.3 Analysis Assumptions

Studies have been performed to determine the sensitivity of steamline break analysis results to various input assumptions (Reference 3). Based on this study, the following assumptions are used for the analysis of the main steamline rupture for SONGS-1. Note that with the exception of the changes to the MDCs, boron concentration in the SI lines, and shutdown margin these analysis assumptions are consistent with corresponding cases supporting the current licensing basis for SONGS-1.

1. Initial conditions

   The plant is assumed to be operating at hot zero power (HZP) with RCS pressure equal to nominal RCS pressure, RCS flow rate equal to the Thermal Design Flow (TDF) rate, RCS vessel average temperature equal to no load Tavg, and steam generator pressure equal to the no load pressure.

   In the LOFTRAN model, the HZP initial power level is modeled as 0.01 of the nominal power level. The nominal (100%) NSSS power of 1351 MWt (core power of 1347 MWt plus 4 MWt pump heat) is assumed.

For SONGS-1, the nominal RCS pressure is 2100 psia.

A TDF of 195,000 gpm (total flow) corresponding to a steam generator tube plugging (SGTP) level of 20% was assumed. The pressure drops around the RCS loop reflect reduced flow conditions associated with the 20% SGTP level and the RCS volumes for primary side of the SGs were reduced to appropriately reflect the 20% SGTP level. These assumptions are consistent with the previous SLB core response analysis.

At no-load conditions, Tavg is 535 °F. At nominal power, a vessel Tavg of 551.5 °F corresponding to reduced Tavg operation is assumed.

The initial pressurizer water volume is assumed to be 345 ft^3. This corresponds to a pressurizer level of approximately 20%. At full power, a nominal pressurizer water volume of 602.0 ft^3 corresponding to reduced Tavg and flow conditions above is assumed.

At nominal conditions, a steam temperature of 476.36 °F corresponding to the SG pressure of 547 psia is assumed. This SG pressure corresponds to the reduced Tavg and flow conditions previously described for full power and 20% SGTP.

An initial SG mass of 68,300 lbm/SG at HZP conditions is assumed. At nominal conditions, a SG mass of 44,471 lbm/SG corresponding to reduced Tavg NSSS conditions above at 35% NRS SG level is assumed. The conservatively high initial SG mass increases the magnitude of the cooldown and, without the isolation of the steam generators, prolongs the duration of the cooldown event.

The initial core boron concentration is assumed to be 0 ppm.

2. Offsite power

Offsite power is assumed to be available throughout the transient. This results in reactor coolant pump (RCP) operation throughout the transient such that full and constant thermal design flow rate (i.e., 1.0 x TDF) is modeled throughout the event.

Actually, for SONGS-1, the RCPs will trip as a result of the SI signal even with offsite power available. However, full and constant flow during the SLB event is conservative since it enhances the heat transfer between the RCS and the secondary causing a more severe cooldown and higher subsequent return to power. This assumption is shown to be conservative in previous SONGS-1 licensing basis steamline break analyses.

3. Shutdown margin

For the HZP initial conditions assumed in the SLB core response analysis, the reactor is assumed to be tripped when the SLB event occurs. All the RCCAs are assumed to be inserted with the exception of the highest worth RCCA, which is assumed to be stuck in a fully withdrawn position. With this initial configuration, the reactor is assumed to be subcritical by the minimum required amount of shutdown margin.

The initial shutdown margin assumed for the analysis is calculated assuming no load, end of life (EOL), equilibrium xenon conditions and the most reactive RCCA stuck in its fully withdrawn position. A value of 2.05% $\Delta k/k$ is assumed.

4. Reactivity coefficients

For the SLB core response analysis, a negative moderator coefficient is assumed corresponding to the end-of-life rodded core with the most reactive RCCA in its fully withdrawn position. The k_{eff} versus temperature at 1000 psia corresponding to the negative moderator temperature coefficient used is shown in Figure 2.1. The effect of power generation in the core on overall reactivity is shown in Figure 2.2 in the form of the Doppler power defect.

The moderator density coefficients and other physics parameters used in the LOFTRAN point-kinetics model were changed from them those previously assumed in the MSLB Core Response analysis. The values used in this MSLB analysis are equivalent to an EOC HFP MTC of -31.8 pcm/°F (-29 pcm/°F with uncertainties). The resulting transient conditions calculated by LOFTRAN were confirmed to be conservative for Cycle 11 relative to predictions made in confirmatory 3D physics models.

For hypothetical breaks upstream of the flow restrictor, the core properties associated with the sector nearest the faulted* steam generator and those associated with the remaining sectors were conservatively combined to obtain average core properties for reactivity feedback calculations. A non-uniform radial weighting factor of [ ][a,c] for the sector nearest the faulted SG and [ ][a,c] each for the remaining two sectors of the core were assumed for these upstream break cases to account for the non-uniform cooldown of the RCS. For the hypothetical break downstream of the flow restrictor the core power distribution was assumed to be uniform consistent with the previous analysis for this case. These two conditions cause underprediction of the Doppler reactivity feedback in the high power region near the stuck rod.

To verify the conservatism of the assumptions used in the LOFTRAN point-kinetics reactivity feedback model, the reactivity as well as the power distribution was checked for the limiting statepoints of the cases analyzed. This core analysis considered the Doppler reactivity from the high fuel temperature near the stuck RCCA,

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* The faulted loop is defined as the loop in which the steamline ruptures. The other two loops are referred to as intact loops.

moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and non-uniform core inlet temperature effects in the case of the hypothetical breaks inside the flow restrictor.

For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated including the above local effects for the statepoints. These results verify conservatism; i.e., overprediction of positive reactivity from the cooldown and underprediction of negative reactivity from power generation.

5. Feedwater

To maximize the cooldown following the SLB event, a full and constant main feedwater flow was conservatively modeled for the hypothetical breaks. Nominal feedwater flow is assumed at the transient initiation and continues until the time of feedwater isolation which occurs after receipt of a SI signal. Feedwater isolation is assumed to occur 26 seconds after the safety injection signal is generated. The 26 second delay is a conservatively long time for signal processing, valve realignment, etc. A conservatively low initial feedwater enthalpy of 40 Btu/lbm is assumed for the HZP initial conditions. This corresponds to a feedwater temperature of 72°F. A lower feedwater enthalpy is conservative for SLB since it increases the magnitude of the cooldown associated with the SLB event. For nominal conditions a feedwater enthalpy of 377.7 Btu/lbm corresponding to reduced Tavg and flow conditions is assumed.

6. Auxiliary Feedwater

Auxiliary feedwater flow (AFW) is assumed to start at the transient initiation and continue throughout the transient. A flow rate of 1419 gpm (10% of nominal feedwater flow) is assumed in all cases. The AFW flow is divided equally between all three steam generators. The temperature of the auxiliary feedwater is conservatively assumed to be 32°F and an AFW purge volume of 0 ft^2 is conservatively modeled.

7. Safety Injection

Safety injection (SI) flow is assumed to be available 26 seconds after the initiation signal is generated on low pressurizer pressure. The low pressurizer pressure setpoint assumed in the analysis is 1680 psia. This represents a nominal setpoint of 1750 psia minus uncertainties, instrument errors, etc. The 26 second delay is a conservatively long time for signal processing, valve realignment, etc. Instantaneous full SI flow is assumed to occur whenever RCS pressure falls below SI pump head at any time $\geq$ 26 seconds after the SI signal occurs.

The instantaneous full SI flow modeling assumption is consistent with the previous SLB core response analysis. With SI flow diversion considered, the SI pump head assumed is 1060 psia.

SI flow rates are calculated based on the operation of only one train of safety injection. The failure of the other train is the worst active single failure assumption. In all cases, the SI flow rates are calculated based on injection into the RCS via one line with two lines blocked and were reduced to account for mini-flow, new Byron-Jackson FW pump curve; 5% degraded, etc. For these cases which consider SI flow diversion during surveillance of the boron concentration in the SI lines, the SI flow is based on assuming the SI system is aligned for Boron

Purge Operation consistent with SCE Operating Procedure S01-4-14, Rev. 4. The SI flow rates versus RCS pressure used in the analysis are shown in Figure 2.3.

For the case of a MSLB downstream of the flow restrictor, a uniform cooldown occurs and the SI flow is modeled to be injected into one of the two RCS loop with an intact SG on the secondary side.

For the MSLB cases upstream of the flow restrictor, SI flow is modeled to be injected to either a RCS loop with an intact SG or to the RCS loop with the faulted SG loop, depending on the case being considered. Since the MSLB case upstream of the flow restrictor is modeled as a non-uniform break, the resulting cooldown of the reactor core is non-uniform. Therefore, the magnitude of the cooldown could be affected by the which RCS loop receives the SI flow. For this reason, the two upstream SLB cases are considered.

A conservatively low enthalpy of 40 Btu/lbm for the SI fluid in the RWST and the SI lines is assumed. A lower enthalpy for the SI fluid is conservative since it increases and prolongs the cooldown of the RCS.

A SI boron concentration in the RWST of 3750 ppm was assumed (corresponding to the minimum allowable RWST boron concentration requirement given in the Tech Specs) for all cases. In all cases, a boron concentration of 3000 ppm was assumed for the SI lines. This is a change from the SI line boron concentration assumption used in the previous analysis supporting the current licensing basis for SONGS-1.

8. Decay Heat

No credit is taken for decay heat since this would inhibit the cooldown of the RCS.

9. Heat Transfer Modeling

Maximum Fuel-to-Coolant overall heat transfer (UA) was assumed consistent with limiting end-of-cycle (EOC) conditions. UA is a function of Tavg. The UAs reflect maximum UA for reduced Tavg conditions.

No credit is taken for heat transfer from the thick metal throughout the RCS to the coolant.

On the secondary side, the Westinghouse Model 27 Steam Generators were modeled in the analysis consistent with the assumption used in the M&E release calculations (Section 3.0).

The SG tube metal heat capacity was calculated based on total SG tube mass reflecting the 20% SGTP level.

10. Accident Simulation

In computing the steam flow during a steamline break or the inadvertent opening of a steam dump valve, the Moody Curve (Reference 4) for f(L/D) = 0 is used.

The break area assumed for hypothetical breaks downstream of the flow restrictor is 1.12 ft^2 per loop. This is the area of the steamline flow restrictor. All three steam generators are assumed to blow down to atmospheric pressure through their respective flow restrictors. Since the break flow area is equal in each loop and no steamline isolation occurs, a fairly uniform cooldown results. Therefore, a uniform radial weighting factor was appropriately assumed for the downstream MSLB cases (see Reactivity coefficients, item 4).

The break areas assumed for hypothetical breaks upstream of the flow restrictor are 1.842 ft^2 for the faulted loop and 0.56 ft^2 for each intact loop. 1.842 ft^2 is the area of the main steamline and 0.56 ft^2 is one half the area of the steamline flow restrictor. The faulted steam generator is assumed to blow down to atmospheric pressure through the ruptured steamline and the intact steam generators are assumed to blow down through the flow restrictor in the faulted loop. Since the equivalent break flow areas are unequal, a non-uniform cooldown results. Therefore, non-uniform radial weighting factors (see item 4) were appropriately applied for the upstream MSLB cases.

11. Steam Generator Water Entrainment

Perfect moisture separation in the steam generators is assumed. This assumption leads to conservative results, especially for large breaks, since there would be considerable entrainment of the water in the steam generators following a steamline break. Entrainment of water would reduce the magnitude of the cooldown of the RCS.

2.4 MSLB Core Response Results

To facilitate presenting the results and required discussion, each of the three SLB cases analyzed herein have been numbered in the order presented in the Introduction section of this report.

Provided in Table 2.1 is a summary of sequence of events and significant parameters for all three SLB cases. In addition, the following transient information is provided in graphical form (see figure number indicated) as a function of time for each of the three cases analyzed.

Parameter	Case 1	Case 2	Case 3
Nuclear Power	2.4	2.8	2.12
Core Heat Flux	2.4	2.8	2.12
Reactivity	2.4	2.8	2.12
RCS Pressure	2.5	2.9	2.13
RV Inlet Temp	2.5	2.9	2.13
Pr'zr Water Volume	2.5	2.9	2.13
Core Average Temp	2.6	2.10	2.14
Core Flow	2.6	2.10	2.14
Core Boron	2.6	2.10	2.14
Feedwater Flow	2.7	2.11	2.15
Steam Pressure	2.7	2.11	2.15
Steam Flow	2.7	2.11	2.15

A discussion of the results for each of the three SLB cases follows.

Case 1 This case is for a hypothetical MSLB downstream of the flow restrictor in the main steamline and assuming 3000 ppm borated water in the SI lines. For this case, a SI signal is generated when the low pressurizer pressure SI setpoint of 1680 psia (including uncertainties) is reached. This setpoint condition occurs at 16.6 seconds after the initiating SLB event, just prior to the time when the pressurizer empties (17.4 seconds).

After reaching this setpoint, a 26 second delay is assumed before any SI flow is modeled. This 26 seconds allow for delays in the electronics, SI sequencing, valve alignment, etc. Since the RCS pressure falls below the SI pump head (1060 psia) at 21.2 seconds, full SI flow is first available at 42.6 seconds, 26 seconds following the SI setpoint condition. At this time, the main feedwater flow is also terminated in the analysis.

Prior to the time when SI flow is obtained, 1) the fluid in the upper head reaches saturation conditions (at 23.2 seconds), and 2) the 2.05% $\Delta k/k$ shutdown margin is lost and the reactor becomes critical (at 25.8 seconds). The saturation conditions in the upper head occur due to the decrease in the RCS pressure which results from the RCS cooldown.

The minimum RCS pressure reached for this case is 732.6 psia which occurs at 48.2 seconds into the event. With the SI system injecting into the RCS, the RCS pressure begins to increase and the pressurizer begins to refill at 48.6 seconds. For this case, the SI flow is assumed to be injected into one of the two RCS loops with an intact steam generator. Borated water reaches the core at 44.2 seconds and begins to turn the event around.

The peak heat flux reached is 0.1637 of nominal and occurs at 45.2 seconds into the event.

Case 2 This case is for a hypothetical MSLB upstream of the flow restrictor in the main steamline and assuming 3000 ppm borated water in the SI lines. For this case, a SI signal is generated when the low pressurizer pressure SI setpoint of 1680 psia (including uncertainties) is reached. This setpoint condition occurs at 18.4 seconds after the initiating SLB event, just prior to the time when the pressurizer empties (19.4 seconds).

After reaching this setpoint, a 26 second delay is assumed before any SI flow is modeled. This 26 seconds allow for delays in the electronics, SI sequencing, valve alignment, etc. Since the RCS pressure falls below the SI pump head (1060 psia) at 23.6 seconds, full SI flow is first available at 44.4 seconds, 26 seconds following the SI setpoint condition. At this time, the main feedwater flow is also terminated in the analysis.

Prior to the time when SI flow is obtained, 1) the fluid in the upper head reaches saturation conditions (at 26.0 seconds), and 2) the 2.05% $\Delta k/k$ shutdown margin is lost and the reactor becomes critical (at 27.4 seconds). Like Case 1, the saturation conditions in the upper head occur due to the decrease in the RCS pressure which results from the RCS cooldown.

The minimum RCS pressure reached for this case is 755.2 psia which occurs at 50.6 seconds into the event. With the SI system injecting into the RCS, the RCS pressure begins to increase and the pressurizer begins to refill at 45.0 seconds. For this case, the SI flow is assumed to be injected into one of the two RCS loops with an intact steam generator. Borated water reaches the core at 46.2 seconds and begins to turn the event around.

The peak heat flux reached is 0.1610 of nominal and occurs at 57.8 seconds into the event.

Case 3 This case is identical to Case 2 with the exception that the SI flow is delivered to the RCS loop with the faulted steam generator. Therefore, up until 44.4 seconds into the event at which time the SI flow is delivered, the sequence of events are the same as those for Case 2.

The minimum RCS pressure reached for this case is 750.9 psia which occurs at 57.2 seconds into the event. This minimum RCS pressure is slightly lower than the Case 2 minimum pressure (by 4.3 psi) and occurs 6.6 seconds later. This is due to the differences in the RCS loop receiving the SI flow. This difference also results in the pressurizer beginning to refill 7.4 seconds later (at 52.4 seconds). Borated water reaches the core at 46.2 seconds, the same time as Case 2.

However, due to the non-uniform cooldown associated with the upstream break case and the radial weighting factors applied to the faulted loop as described in the analysis assumptions, the peak heat flux reached (0.1384 of nominal) for the case where SI flow is delivered to the faulted loop is slightly lower than the Case 2 value and occurs 11.2 seconds sooner (i.e., at 46.6 seconds into the event).

Based on the analysis performed and the results summarized above, the following is observed.

1. On the basis of peak heat flux reached, the downstream MSLB case is limiting compared to the upstream MSLB cases.

2. As evident by the SLB pressure transient conditions presented in the figures provided, the SLB events result in a decrease in both RCS pressure and steam pressure. Hence, the RCS pressure and steam pressure never exceed their respective initial condition values.

3. Finally, in all the SLB cases analyzed herein, the analysis of the thermal and hydraulic behavior of the core following the steamline break demonstrated that the safety analysis minimum DNBR limit is met and, therefore, the occurrence of DNB is precluded for all of the cases analyzed.

FIGURE 2.1

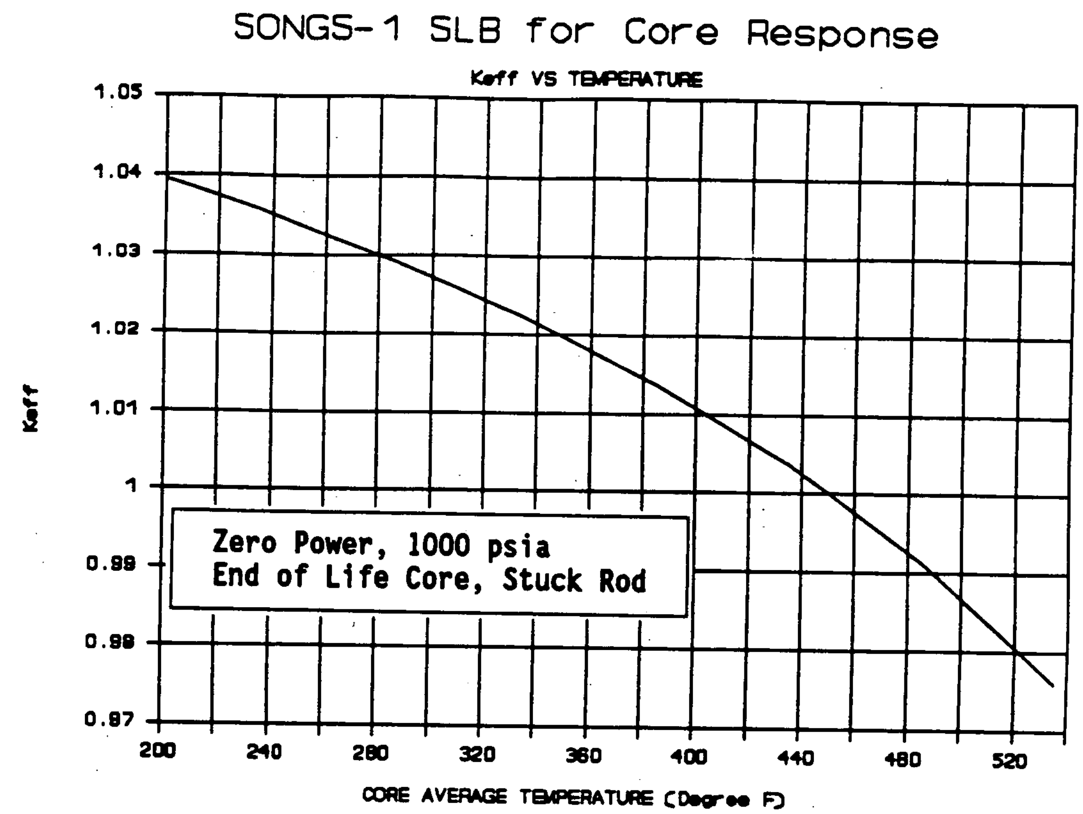

SONGS-1 SLB for Core Response
Keff VS TEMPERATURE
1.05
1.04
1.03
1.02
1.01
1
0.99
0.98
0.97
Keff
200
240
280
320
360
400
440
480
520
CORE AVERAGE TEMPERATURE (Degree F)
Zero Power, 1000 psia
End of Life Core, Stuck Rod

FIGURE 2.2

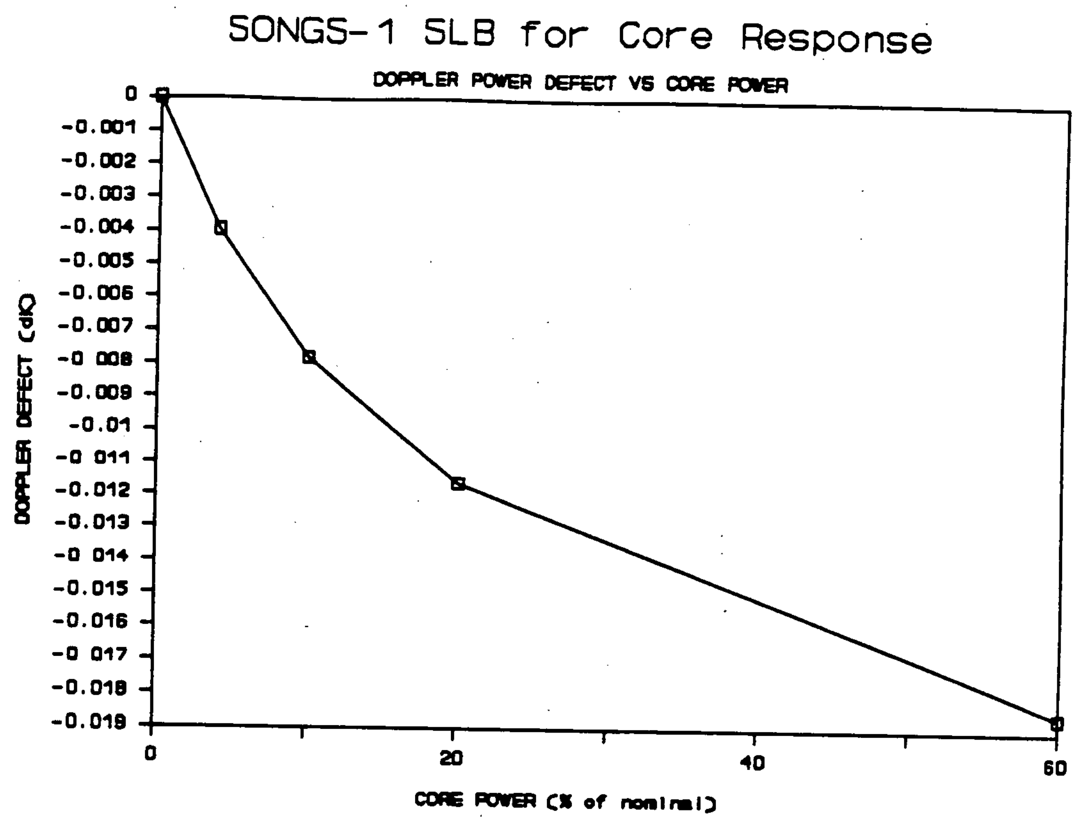

SONGS-1 SLB for Core Response
DOPPLER POWER DEFECT VS CORE POWER
DOPPLER DEFECT (dK)
0
-0.001
-0.002
-0.003
-0.004
-0.005
-0.006
-0.007
-0.008
-0.009
-0.01
-0.011
-0.012
-0.013
-0.014
-0.015
-0.016
-0.017
-0.018
-0.019
0
20
40
60
CORE POWER (% of nominal)

FIGURE 2.3

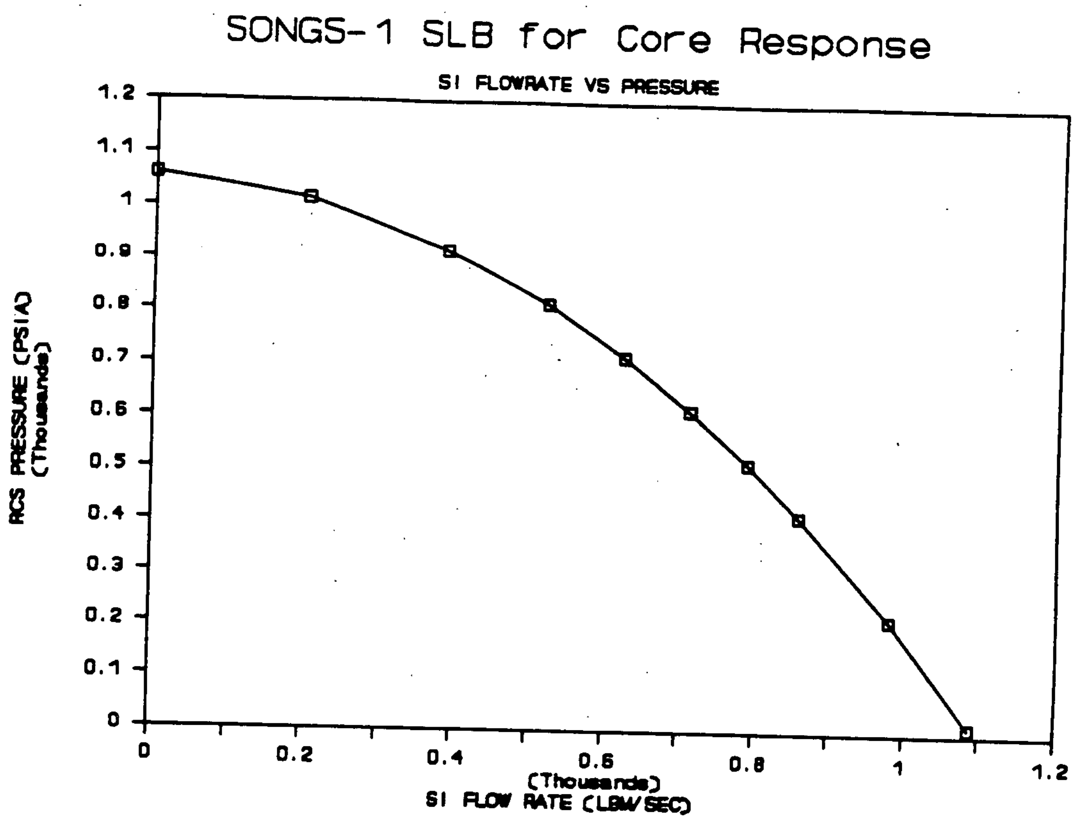

SONGS-1 SLB for Core Response
SI FLOWRATE VS PRESSURE
RCS PRESSURE (PSIA)
(Thousands)
1.2
1.1
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0
0.2
0.4
0.6
0.8
1
1.2
(Thousands)
SI FLOW RATE (LBM/SEC)

FIGURE 2.4

MSLB Downstream / 3000 ppm in SI Lines / SI to Intact SG Loop

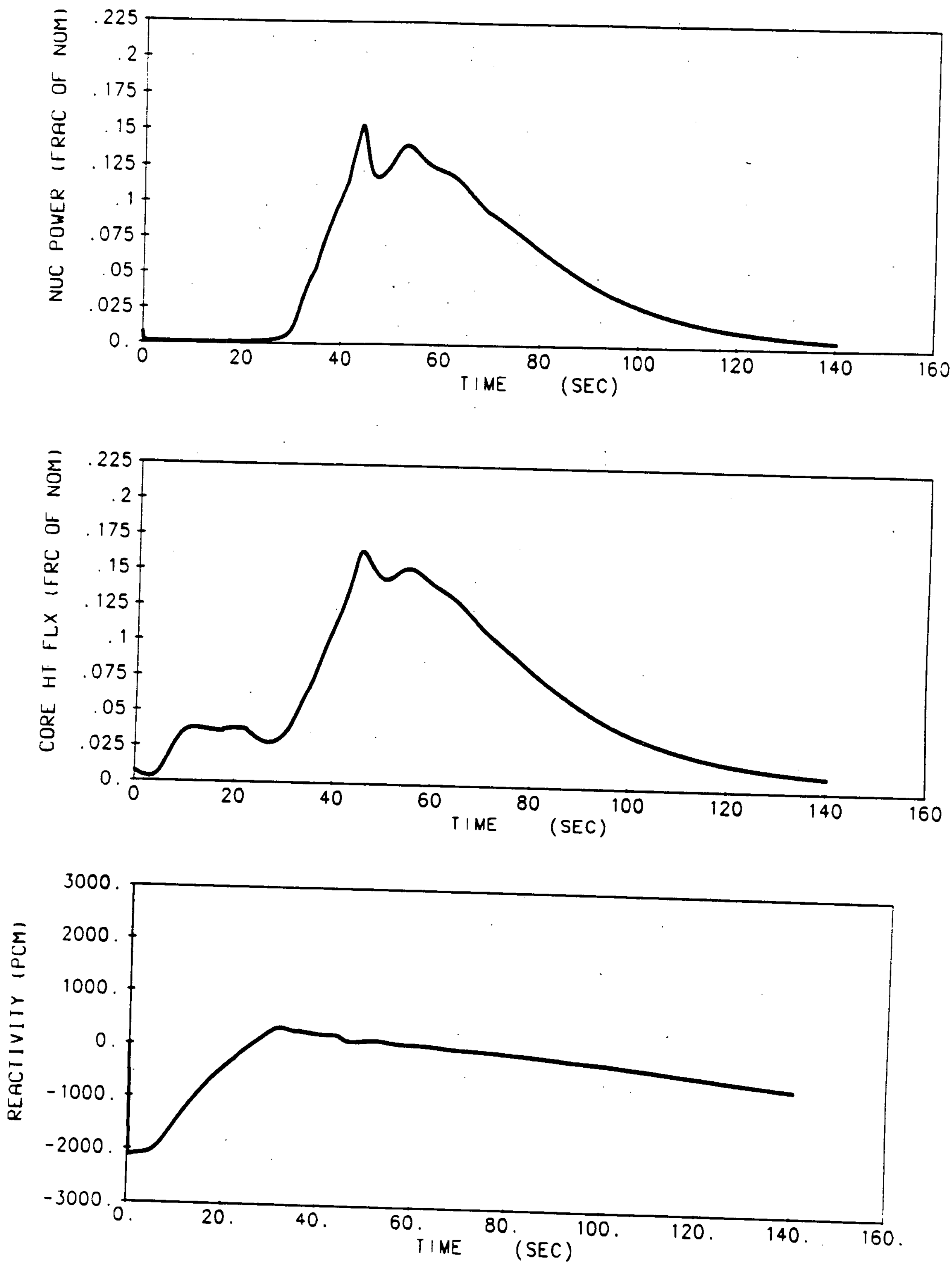


FIGURE 2.5

MSLB Downstream / 3000 ppm in SI Lines / SI to Intact SG Loop

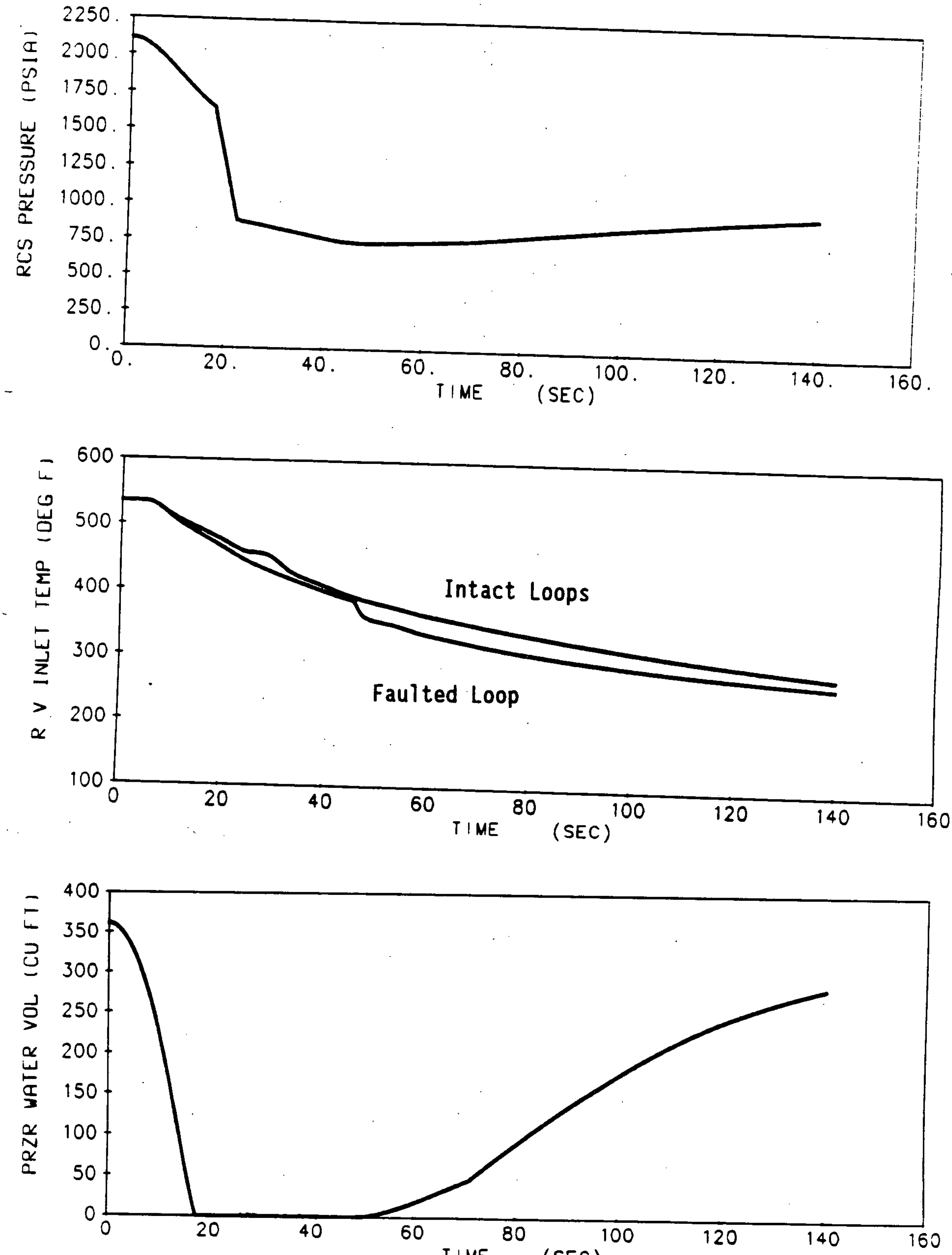


FIGURE 2.6

MSLB Downstream / 3000 ppm in SI Lines / SI to Intact SG Loop

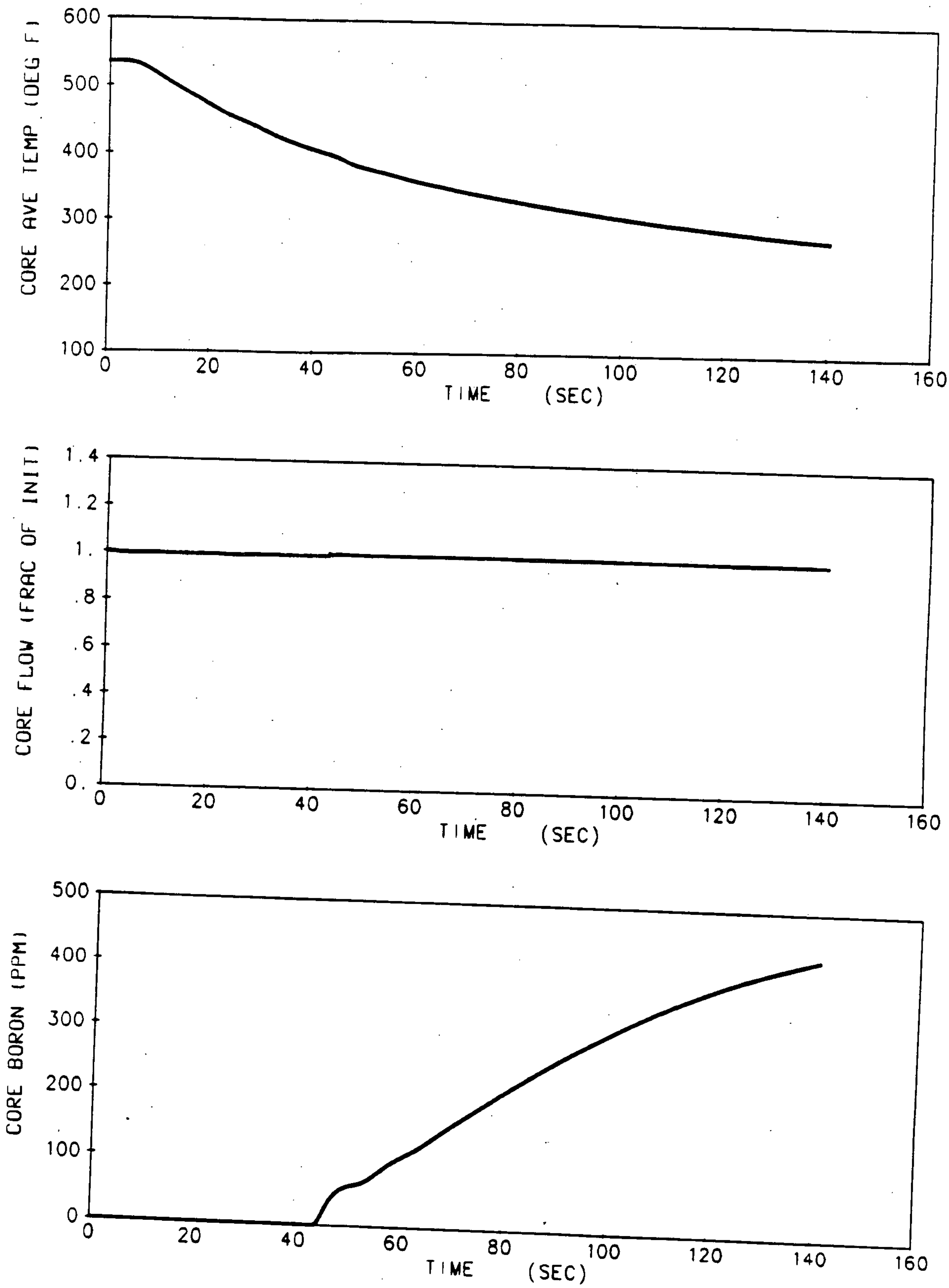


FIGURE 2.7

MSLB Downstream / 3000 ppm in SI Lines / SI to Intact SG Loop

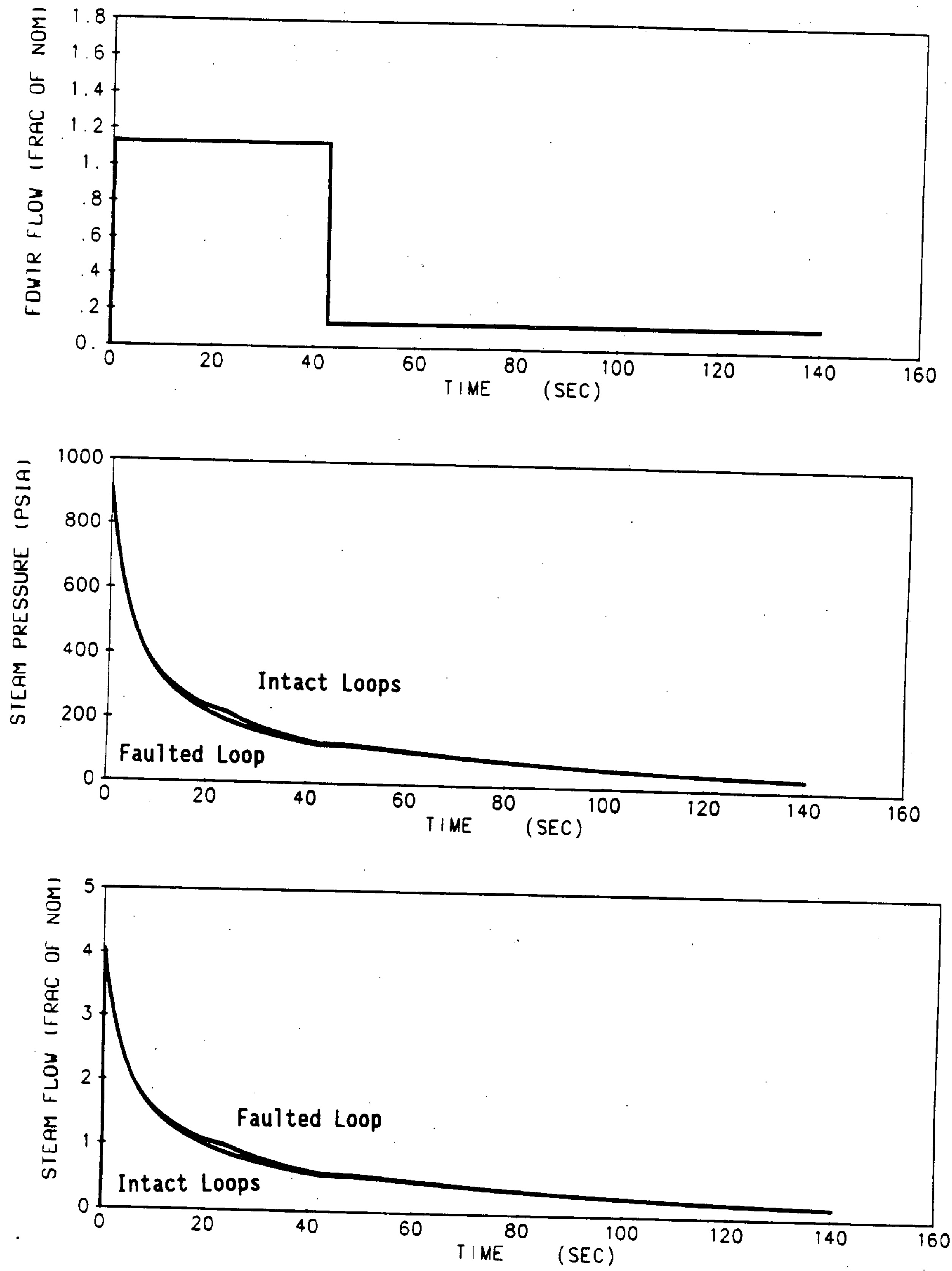


FIGURE 2.8

MSLB Upstream / 3000 ppm in SI Lines / SI to Intact SG Loop

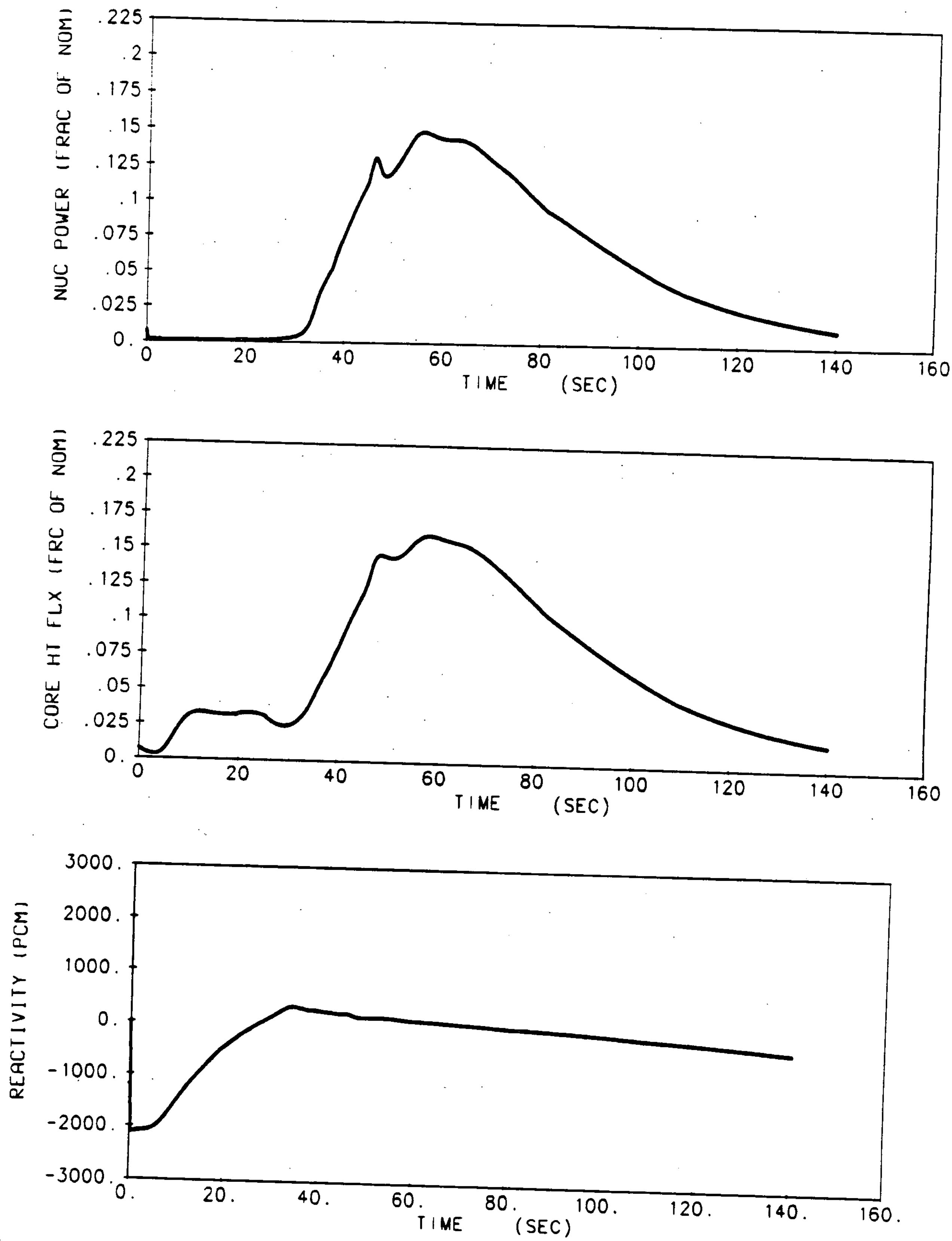


FIGURE 2.9

MSLB Upstream / 3000 ppm in SI Lines / SI to Intact SG Loop

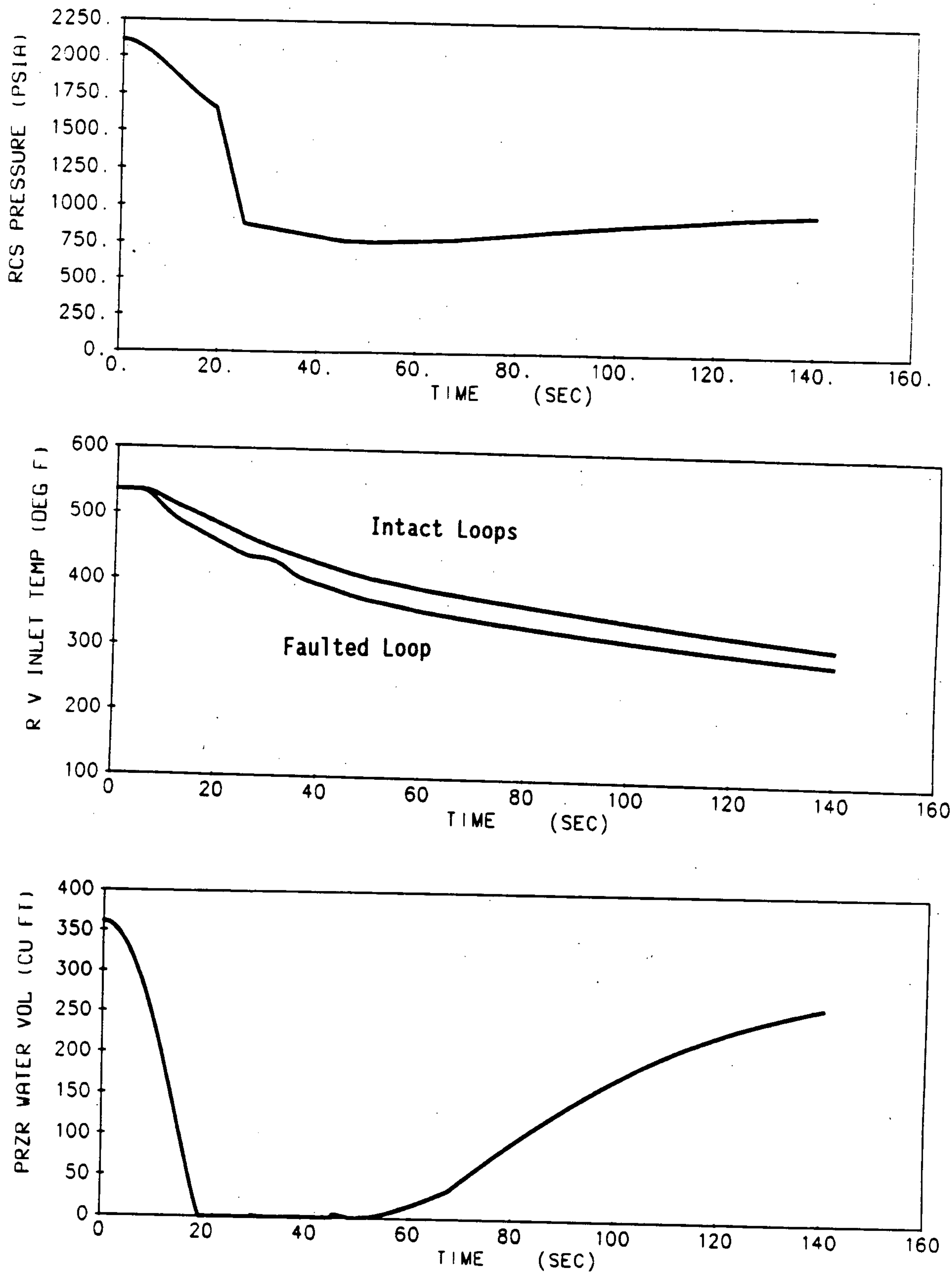


FIGURE 2.10

MSLB Upstream / 3000 ppm in SI Lines / SI to Intact SG Loop

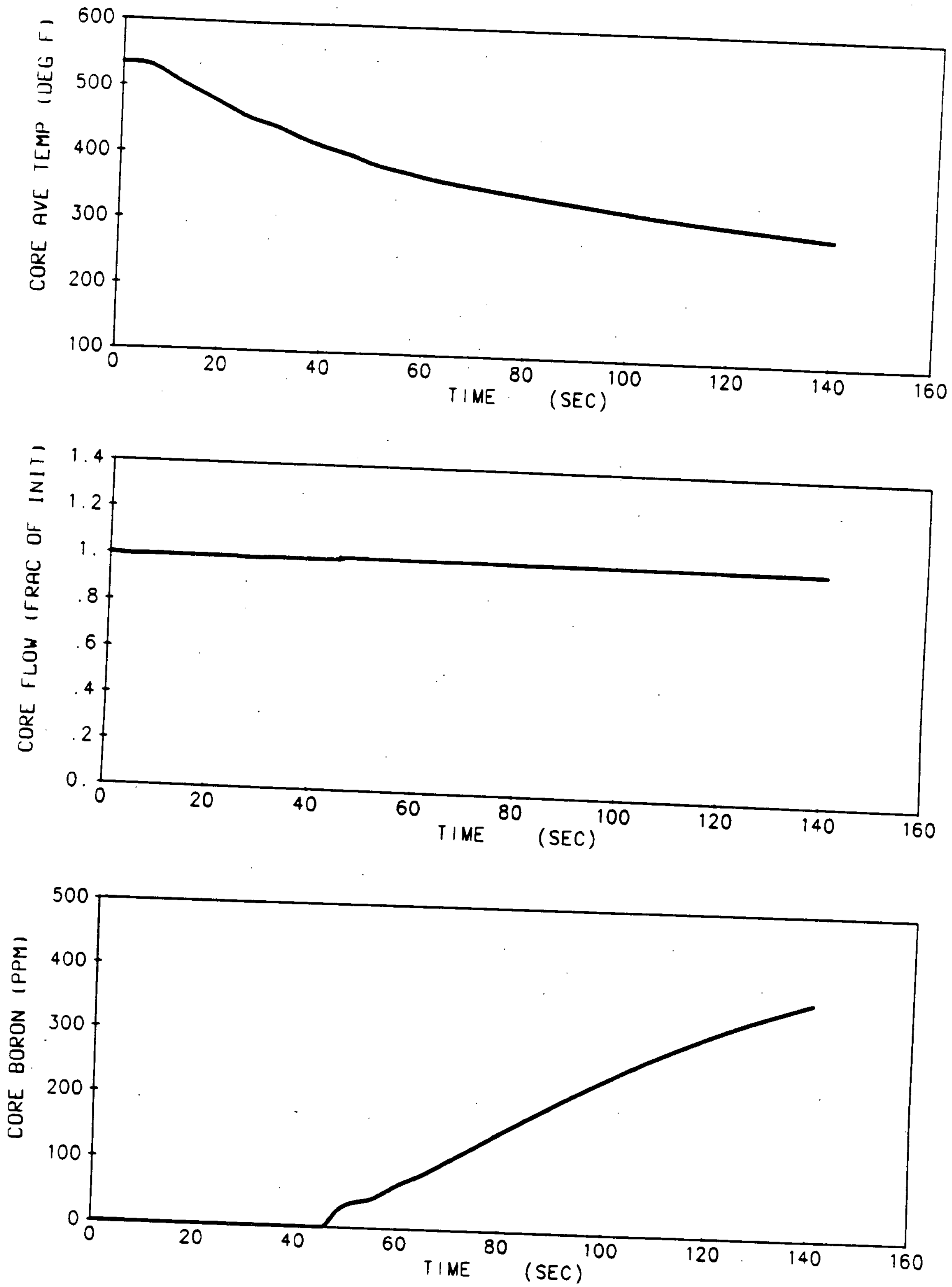


FIGURE 2.11

MSLB Upstream / 3000 ppm in SI Lines / SI to Intact SG Loop

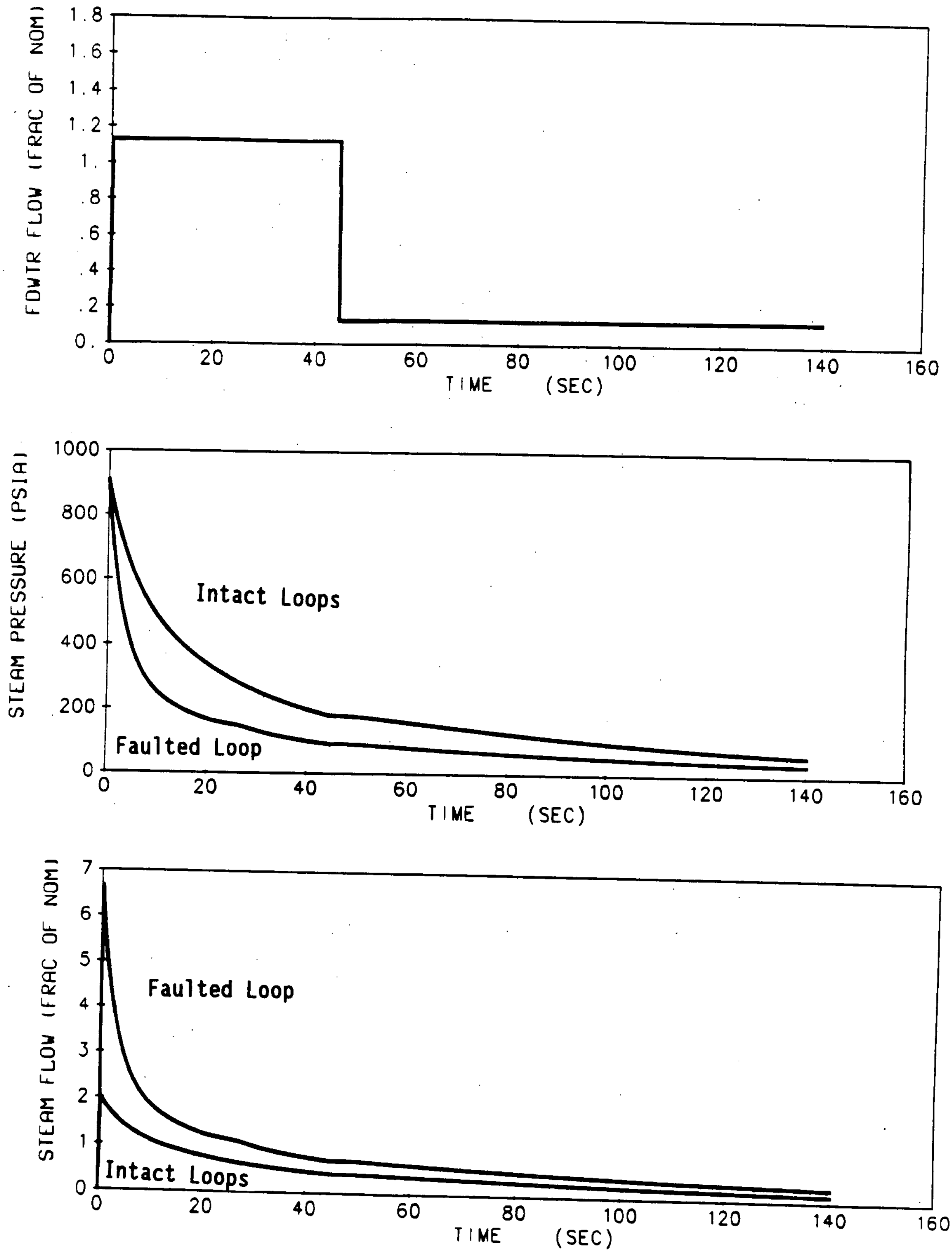


FIGURE 2.12

MSLB Upstream / 3000 ppm in SI Lines / SI to Faulted SG Loop

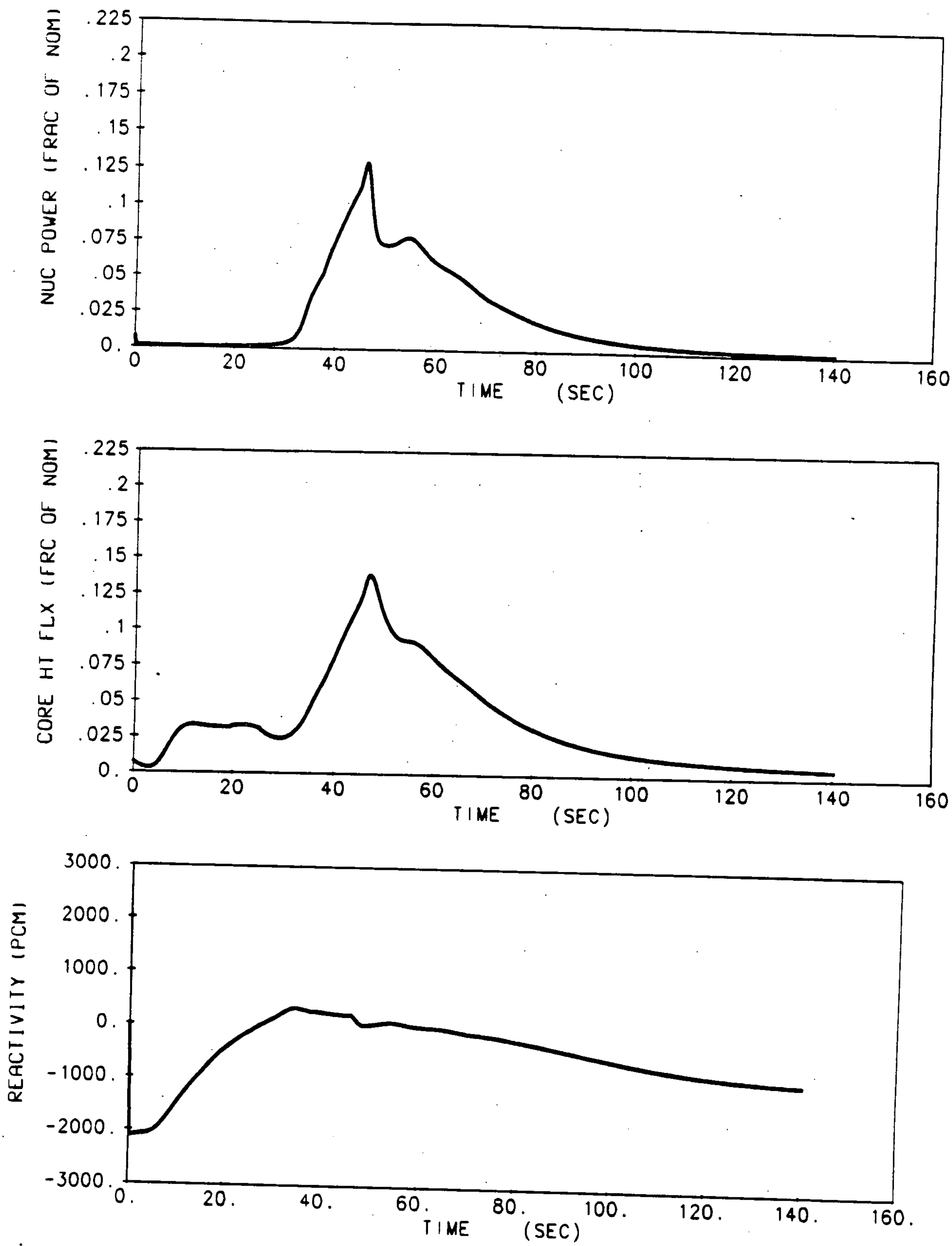


FIGURE 2.13

MSLB Upstream / 3000 ppm in SI Lines / SI to Faulted SG Loop

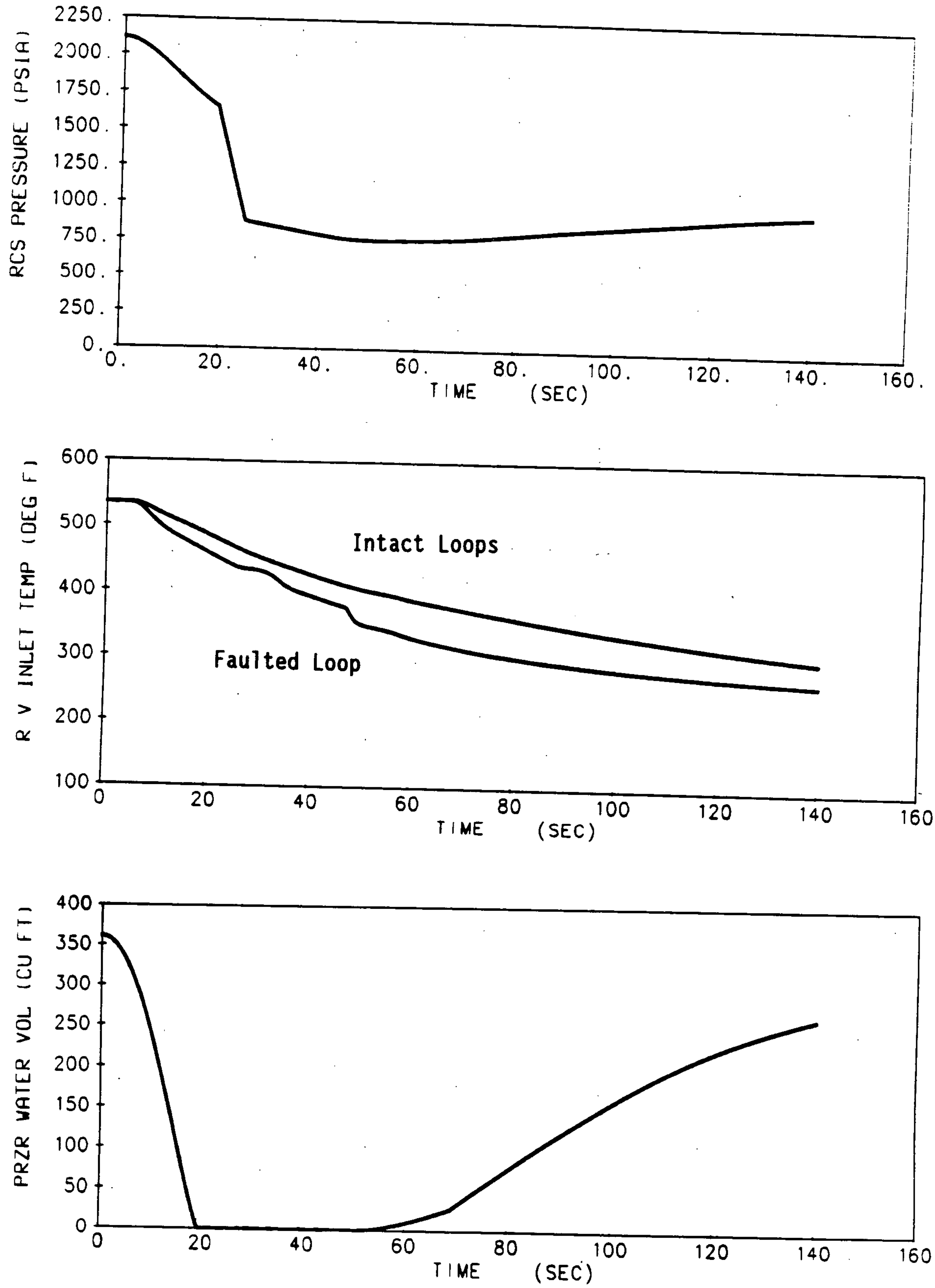


FIGURE 2.14

MSLB Upstream / 3000 ppm in SI Lines / SI to Faulted SG Loop

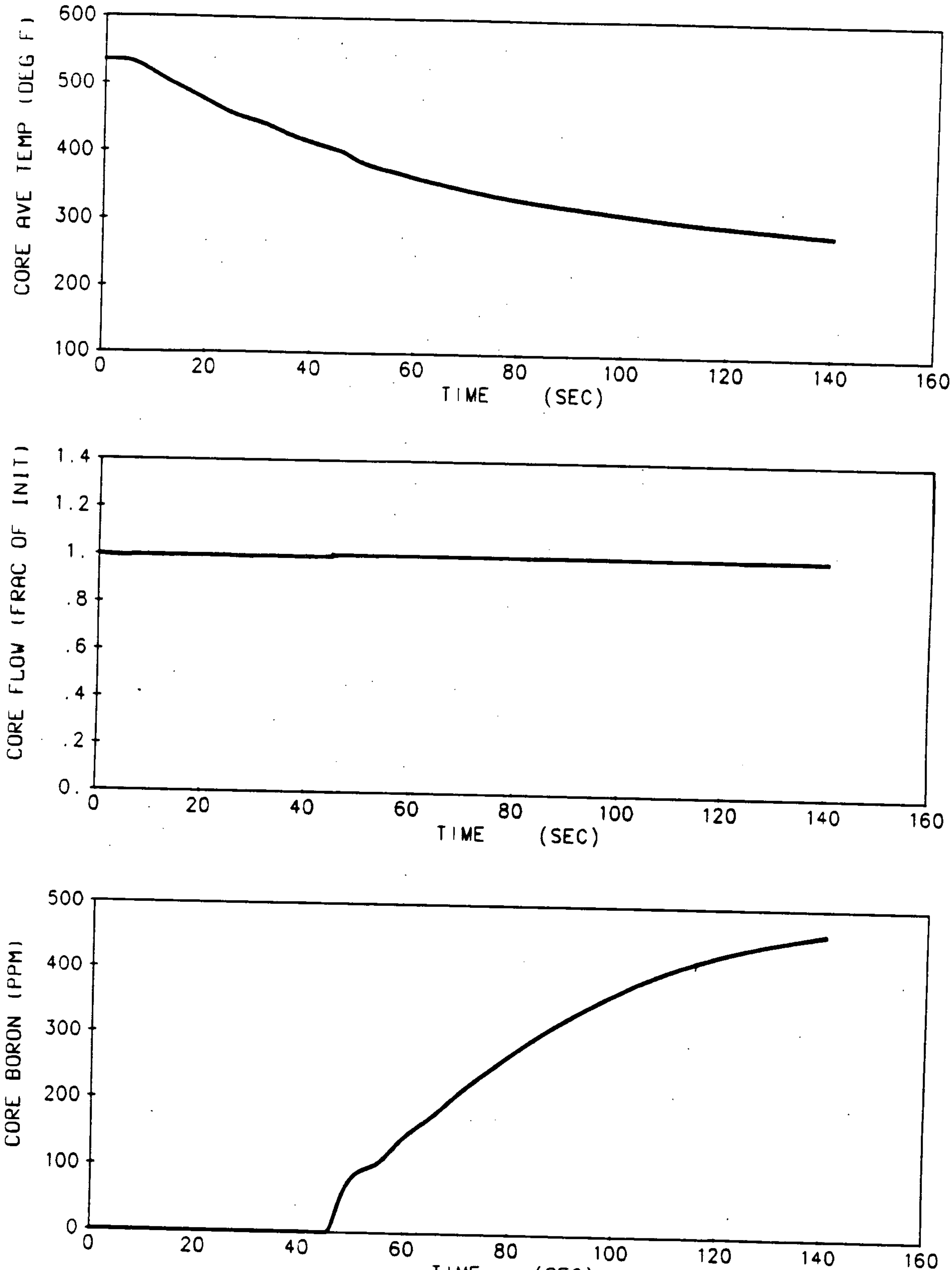
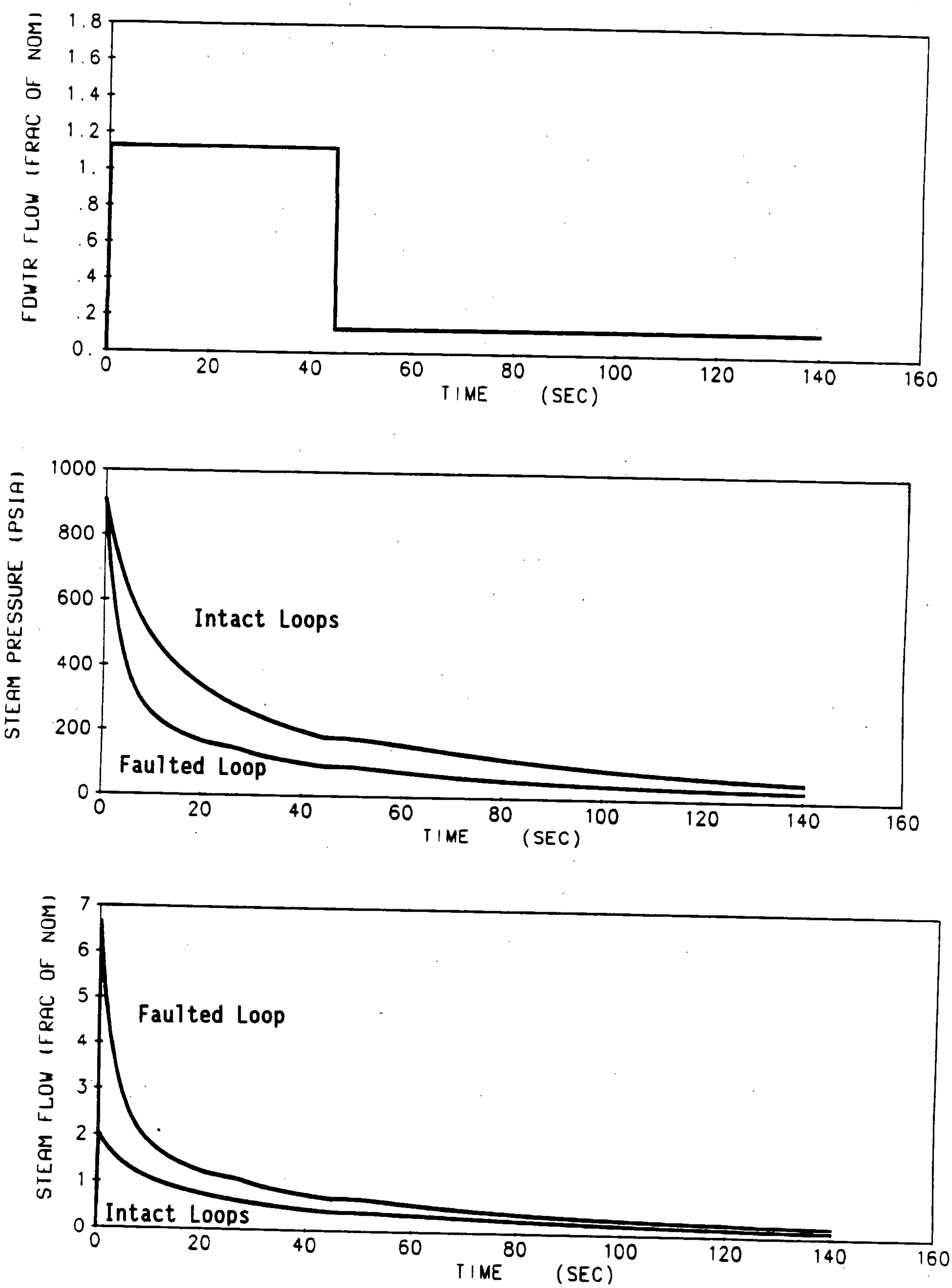


FIGURE 2.15

MSLB Upstream / 3000 ppm in SI Lines / SI to Faulted SG Loop



3.0 HFP MSLB M&E Releases

Included in this section are the final results of the Mass & Energy (M&E) Release analysis for hypothetical main steamline break (MSLB) inside containment of the San Onofre Nuclear Generating Station - Unit 1 (SONGS-1).

The results of this MSLB M&E Release analysis is presented in Section 3.2. These results reflect the calculation of M&E release rates inside containment for the MSLB event initiated from Hot Full Power (HFP) conditions as requested by Southern California Edison (SCE). The M&E release rates for this case result in the most limiting containment pressure conditions following a MSLB for SONGS-1 and are presented for use by SCE for evaluating containment integrity for Cycle 11. This MSLB M&E Release analysis assumes revised Moderator Density Coefficients (MDCs) for SLB conditions which are equivalent to an EOC HFP MTC of -31.8 pcm/°F (-29 pcm/°F with uncertainties).

In addition to the revised MDCs, these calculations also consider an increase in the boron concentration in the SI lines to a minimum value of 3000 ppm and an increase in the HZP EOC minimum shutdown margin requirement to a value of 2.05%$\Delta k/k$.

The pertinent analysis assumptions for the HFP SLB M&E case analyzed herein follow.

3.1 HFP MSLB M&E Analysis Assumptions

The following presents the pertinent analysis assumptions used in the HFP MSLB M&E analysis inside containment for SONGS-1. With the exception of changes in the reactivity feedback modeling, the minimum shutdown margin, and the minimum boron concentration in the SI lines as noted, these analysis assumptions are consistent with those used in the current licensing basis analysis for SONGS-1.

1) Nominal (100%) NSSS power of 1351 MWt = Core Power of 1347 MWt + 4 MWt pump heat.

2) Thermal Design Flow (total) = 195,000 gpm corresponding to a steam generator tube plugging (SGTP) level of 20%.

3) Nominal Vessel Tavg at 100% power = 551.5 °F corresponding to reduced Tavg operation. At no-load conditions, Tavg = 535 °F.

4) Nominal RCS pressure = 2100 psia.

5) Nominal Pressurizer Water Volume = 602.0 ft^3 corresponding to reduced Tavg and flow conditions above.

6) Nominal Steam Temperature = 476.36 °F corresponding to reduced Tavg and flow conditions above.

7) Nominal Feedwater Enthalpy = 377.6 Btu/lbm corresponding to reduced Tavg and flow conditions above.

8) Nominal SG Mass = 43,193 lbm/SG corresponding to reduced Tavg NSSS conditions above at 30% NRS SG level.

9) RCS volumes for primary side of SG reduced to reflect 20% SGTP.

10) Maximum Fuel-to-Coolant overall heat transfer (UA) has been assumed consistent with limiting end-of-cycle (EOC) conditions. UA is a function of Tavg. The UAs reflect maximum UA for reduced Tavg conditions.

11) SG tube metal heat capacity calculated based on total SG tube mass reflecting 20% SGTP.

12) Pressure drops around the RCS loop reflect reduced flow conditions associated with 20% SGTP.

13) Westinghouse Model 27 Steam Generators were modeled in the analysis consistent with the assumption of the previous M&E release calculations.

14) Thick metal heat capacities and effective UAs for major RCS components were modeled and the values assumed are unchanged from those used in the previous M&E release calculations.

15) The feedwater flashing volume per loop (82 ft^3) and feedline flashing volume pressure drop at nominal feed flow (30 psi) were modeled and the values assumed are unchanged from those used in the previous M&E release calculations. The initial enthalpy of flashing volume based is on the initial feedwater enthalpy and initial steam generator steam pressure.

16) Initial power level of 103% of 1351 MWt NSSS power was assumed for the full power condition. This is consistent with full power level assumption used in the previous M&E release calculations.

17) A full and constant thermal design flow rate (i.e., 1.0 x TDF) was assumed consistent with the RCS flow assumption in the previous M&E release calculations.

18) An initial RCS pressure of 2130 psi was assumed. This value includes a 30 psi increase from nominal RCS pressure to account for any pressure uncertainties at HFP conditions. An increase in RCS pressure is conservative for M&E release calculation since it delays the time to reach the RCS pressure corresponding to the SI pump head pressure. The uncertainty value of 30 psi is consistent with the pressure uncertainty assumed in other SONG-1 analyses and assumed in the previous M&E release analysis.

19) Initial Tavg of 555.5 °F = Nominal Tavg of 551.5 °F (i.e., Reduced Tavg) plus 4 °F for temperature uncertainties. For M&E release calculations, a higher Tavg will maximize the potential secondary side energy release.

20) The initial Pressurizer Water Volume = 602.0 ft^3 corresponding to nominal volume (item 5 above).

21) The initial Feedwater Enthalpy = 377.6 Btu/lbm corresponding to nominal FW enthalpy (item 7 above). This is consistent with the previous M&E release calculations.

22) The initial steam generator (SG) Mass = 43,193 lbm/SG at HFP corresponding to the nominal SG mass (item 8 above).

23) Reactor trip on a SI signal from High Containment Pressure at 2 seconds is assumed with rod motion occurring at 4.0 seconds for the full power case. This is based on assuming a high containment pressure signal is reached in 2 seconds plus a total time for signal delays (i.e. electronics) of 2 seconds prior to rod motion. This is consistent with time of rod motion assumed in previous M&E release calculations.

24) Earliest availability of full SI flow actuated on a high containment pressure SI signal at 2 seconds is at 28 seconds (i.e., 26 second delay assumed). Instantaneous full SI flow assumed to occur whenever RCS pressure falls below SI pump head at any time $\geq$ 28 seconds after initiation of event (i.e., t=0). The instantaneous full SI flow modeling assumption is consistent with the previous M&E release calculations.

25) SI flow versus pressure reflect 1 SI train injecting into 2 lines and 1 line blocked with reduced SI flow rates to account for mini-flow, new Byron-Jackson FW pump curve; 5% degraded, etc. Since

borated water is assume in the SI lines, SI flow diversion during surveillance of the boron concentration in the SI lines is considered. SI flow is based on assuming the SI system is aligned for Boron Purge Operation consistent with SCE Operating Procedure S01-4-14, Rev. 4. For these conditions, a SI pump head of 1050 psia is assumed.

26) SI flow is modeled to inject into 2 (intact) RCS loops. This is consistent with the previous M&E release analysis.

27) An enthalpy of 40 Btu/lbm for the SI fluid in the RWST and the SI lines is assumed. This is consistent with the assumption used in the previous M&E release calculations.

28) A SI boron concentration in the RWST of 3750 ppm is assumed (corresponding to the minimum allowable RWST boron concentration requirement given in the Tech Specs). A boron concentration of 3000 ppm is assumed for the SI lines. This is a change from the previous M&E release calculations which assumed 1500 ppm in the SI lines.

29) A large SLB with an equivalent break flow area of 1.12 ft^2 for each loop is assumed. This represents a break inside containment and downstream of the flow restrictor in the faulted loop and is consistent with break area modeling for the large SLB in the previous M&E release calculations. Since the equivalent break flow is equal in each loop and no steamline isolation occurs, a fairly uniform cooldown results. However, a non-uniform radial weighting factor was conservatively assumed.

30) The physics parameters used in the LOFTRAN point-kinetics model were revised from them those previously assumed in the MSLB M&E release analysis.

The moderator density coefficients and other physics parameters used in this MSLB analysis are those equivalent to an EOC HFP MTC of -31.8 pcm/°F (-29 pcm/°F with uncertainties). An EOC shutdown margin of 2.05% $\Delta k/k$ was assumed.

The resulting transient conditions calculated by LOFTRAN were confirmed to be conservative for Cycle 11 relative to predictions made in confirmatory 3D physics models used in the reload design process.

31) Feedwater flow as a function of time was modeled to account for an initial increase in feedwater flow resulting from a decrease in steam pressure following the SLB (at t=0) and for a subsequent decrease resulting from the reactor trip signal and initiation of the SI system on the high containment pressure SI signal. The specific FW flow modeling assumptions used in the HFP cases are shown in Table A1.2. These FW flow assumptions are consistent with those used in the previous M&E release analysis (Reference A1-1) and are based on FW flow transient information provided by SCE and conservatively included a constant 500 gpm AFW flow rate for the duration of the event (i.e., $\leq$ 600 seconds).

32) An initial enthalpy of 560.0 Btu/lbm corresponding to a maximum fluid temperature of 560.22°F was assumed for the fluid conditions in the reactor vessel upper head region. This analysis assumption is consistent with that assumed in the HFP MSLB M&E Release Analysis supporting the current licensing basis for SONGS-1.

3.2 HFP MSLB M&E Release Results

Table 3.2 provides the M&E release rates following a MSLB with an equivalent break flow area of 1.12 ft^2 per loop (i.e., at a location downstream of the flow restrictor) for the HFP MSLB case. M&E release rates are provided for the first 300 seconds following the break. Figure 3.2 shows the integrated mass and energy releases over the range of interest in comparison to the previous integrated M&E releases for the equivalent case in the SONGS-1 current licensing basis. Since the HFP SLB M&E releases calculated herein for the revised MDCs and other changes are bounded by the equivalent HFP SLB M&E case in the current licensing basis for SONGS-1, and, since the current licensing basis for SONGS-1 shows that the HFP SLB M&E case bounds the HZP SLB M&E releases, the HFP SLB M&E case presented herein would still bound the SONGS-1 HZP SLB M&E case with the equivalent changes.

TABLE 3.1

SONGS-1

Mass and Energy Release Rates

1.12 ft^2 MSLB - HFP Initial Conditions

Time (seconds)	Mass Release Rate (lbm/sec)	Energy Release Rate (10^6 Btu/sec)	Integrated Total Mass (10^3 lbm)	Integrated Total Energy (10^6 Btu)

a,c

TABLE 3.2

HFP MSLB M&E Release

Feedwater Flow Modeling

Time (sec)	Relative Flow		AFW Flow		Total Relative FW Flow
0.0	1.2240	+	0.044	=	1.2680
0.5	1.2997	+	0.044	=	1.3437
1.0	1.3502	+	0.044	=	1.3942
1.5	1.4006	+	0.044	=	1.4446
2.0	1.4259	+	0.044	=	1.4699
2.5	1.4763	+	0.044	=	1.5263
3.0	1.4890	+	0.044	=	1.5330
8.0	0.6814	+	0.044	=	0.7254
14.0	0.2208	+	0.044	=	0.2648
18.0	0.0	+	0.044	=	0.0440
600.0	0.0	+	0.044	=	0.0440
600.1	0.0	+	0.022	=	0.0220

Notes:

The MFW flow initially increases due to the drop in steam pressure caused by the steamline break. The MFW pumps are modeled to trip on a reactor trip signal on the High Containment Pressure SI signal assumed at 2 seconds.

The AFW flow added corresponds to 500 gpm until 600 seconds followed by 250 gpm (reduced via operator action). However, the HFP MSLB M&E analysis was terminated well before 600 seconds. This modeling of AFW addition is consistent with the M&E release calculations supporting the current licensing basis for SONGS-1.

FIGURE 3.1

SONGS-1 HFP SLB M&E ANALYSIS

a,c

INTEGRATED MASS (lbm - E03)

TIME (seconds)

a,c

INTEGRATED ENERGY (Btu - E06)

TIME (seconds)

4.0 CONCLUSIONS

The results of the MSLB Core Response analysis presented in Section 2.0 show that the DNBR remained above the limit value for all of the cases analyzed. Since the DNB criterion is met, no fuel rod failures or releases of fission products from the fuel are expected following a credible or hypothetical steamline break, respectively.

The results also show that there is no overpressurization of the Reactor Coolant System or secondary system as a result of a steamline break.

Therefore, it is concluded that the applicable acceptance criteria is met for the hypothetical breaks with the revised analysis assumptions as specified within. Since the current SONGS-1 licensing basis analysis results for the hypothetical breaks clearly bound those for the credible steamline break case, these conclusions are also applicable for the credible steamline break event.

However, these conclusions are based on the assumption the that the boron concentration in the SI lines is maintained at or above a minimum value of 3000 ppm and that the HZP EOC shutdown margin is maintained at or above a minimum value 2.05% $\Delta k/k$ as described earlier in the Introduction section of this report.

The M&E Release analysis for the most limiting SONGS-1 MSLB case for containment intergrity is reported in Section 3.0. Table 3.1 provides the M&E release rates following a MSLB from HFP initial conditions for an equivalent break flow area of 1.12 ft^2 per loop (i.e., at a location downstream of the flow restrictor). Figure 3.1 illustrates the total integrated M&E releases (over the time period of interest) in comparison to the equivalent case supporting the current licensing basis for SONGS-1. Based on Figure 3.1, it can be seen that the M&E releases determined herein are bounded by those supporting the current licensing basis for SONGS-1.

5.0 REFERENCES

1. 90SC*-G-0025, "SONGS-1 Cycle 11 Reload Safety Evaluation Report," M. F. Muenks (Westinghouse) to Mr. Paul. D. Myers (SCE), September 26, 1990.

2. Burnett, T. W. T., et al., "LOFTRAN Code Description", WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

3. Hollingsworth, S. D. and Wood, D. C., "Reactor Core Response To Excessive Secondary Steam Releases", WCAP-9226 (Proprietary) and WCAP-9227 (Non-Proprietary), January 1978.

4. Moody, F. S., "Transactions of the ASME, Journal of Heat Transfer", Figure 3, page 134, February 1965.