VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

W. L. STEWART
VICE PRESIDENT
NUCLEAR OPERATIONS

September 13, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 521
PSE/NAS/cdk:0008N
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Gentlemen:

AMENDMENT TO OPERATING LICENSES DPR-32 AND DPR-37
SURRY POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGE

Pursuant to 10CFR 50.90, the Virginia Electric and Power Company requests an Amendment, in the form of changes to the Technical Specifications, to Operating Licenses DPR-32 and DPR-37 for the Surry Power Station, Units No. 1 and 2, respectively. The proposed changes and the supporting safety evaluation are enclosed.

There are many problems associated with the high boric acid concentration which must be maintained in the boron injection tanks and concentrated boric acid system. During normal operation reactor coolant letdown is concentrated and recycled to the boric acid tanks via the Boron Recovery System. As a result the systems attain high radiation levels. These high radiation levels compound the maintenance problems caused by the high boric acid concentrations. For example, boric acid is a highly corrosive fluid and leakage has led to the degradation of carbon steel components. Leakage has also led to the failure of heat tracing which is required to maintain solution solubility. Failure of heat tracing results in additional maintenance problems such as boron plateout and potential line plugging as the solution temperature drops. The increased maintenance causes increased personnel radiation exposures. A reduction in boric acid concentration would reduce maintenance requirements and the associated exposures to plant personnel.

Attachment 1 provides the detailed justification for a proposed reduction in the minimum boron injection tank (BIT) concentration from 11.5 wt % to 0 wt % and a change in the minimum boric acid system concentration from 11.5 wt % to 7.0 wt %. A general description of the current design of the Boron Injection Tank and Boric Acid System is given. The proposed physical changes to each system are described and operational and maintenance benefits are discussed.

A001
1/44

Rec'd w/check
$4,400

Analyses have been performed to determine the impact of the proposed changes on the appropriate Surry licensing bases, including a reanalysis of the steamline break accidents discussed in Chapter 14 of the Surry Updated Final Safety Analysis Report (UFSAR). The analysis has been performed by Vepco, using the RETRAN Computer Code and the reactor system transient analysis methodology described in our topical report which was transmitted by letter, dated April 14, 1981 (Serial No. 215). The methodology, assumptions and results of the analysis are discussed in detail in Attachment 1. This documentation will be incorporated into the Surry Updated FSAR during the next annual update.

The changes to the Technical Specifications associated with the proposed boron concentration reduction are provided in Attachment 2. The proposed changes and the safety evaluation have been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Safety Evaluation and Control Staff. It has been determined that this request does not pose a significant hazards consideration as defined in 10CFR 50.92 or an unreviewed safety question as defined in 10CFR 50.59.

We have evaluated this request in accordance with the criteria in 10CFR 170.22. Since this request involves a safety issue which the staff should be able to determine does not involve a significant hazards consideration for Unit 1 and a duplicate safety issue for Unit 2, a Class III license amendment fee and a Class I license amendment fee are required for Unit 1 and Unit 2, respectively. Accordingly, a voucher check in the amount of $4,400.00 is enclosed in payment of the required fee.

Inasmuch as the proposed changes will result in significant operational benefits and reduction in personnel exposure, we solicit your expeditious review and approval by December 15, 1983. We are interested in meeting with you during the month of September, to discuss the details of this proposed change and the supporting safety analysis.

Very truly yours,

W. L. Stewart

Enclosures:
(1) Safety Evaluation for Proposed Changes
(2) Proposed Technical Specifications Changes
(3) Voucher Check $4,400.

cc: Mr. James P. O'Reilly
Regional Administrator
Region II

Mr. D. J. Burke
NRC Resident Inspector
Surry Power Station

Mr. J. Don Neighbors
NRC Project Manager - Surry
Operating Reactors Branch No. 1
Division of Licensing

Mr. Charles Price
Department of Health
109 Governor Street
Richmond, Virginia

COMMONWEALTH OF VIRGINIA)
)
CITY OF RICHMOND)

The foregoing document was acknowledged before me, in and for the City and Commonwealth aforesaid, today by W. R. Cartwright who is Manager-Nuclear Operations Support, of the Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 13th day of September, 19 83.

My Commission expires: 2-26, 19 85.

Ann C. McSw...

Notary Public

(SEAL)

S/001

ATTACHMENT 1

SAFETY EVALUATION FOR REDUCTION IN BORON CONCENTRATION IN THE BORON INJECTION TANK AND CONCENTRATED BORIC ACID SYSTEM

SURRY POWER STATION

UNITS 1 AND 2

A. INTRODUCTION

1. Objective

Surry Power Station has experienced problems associated with the high boric acid concentrations which must be maintained in the Boron Injection Tanks and concentrated Boric Acid System. The highly concentrated acid is required to be heated to a minimum Technical Specifications limit of 145 F in order to maintain solution solubility. Numerous failures of the heat tracing circuitry have occurred throughout the life of the plant. These heat tracing failures lead to line plugging and restrictions as the solution temperature decreases and boric acid plateout occurs. These problems necessitate extensive maintenance and operational attention which causes excessive radiation exposure to plant personnel.

Vepco has been evaluating different methods for alleviating these problems. The following sections decribe the design functions of the Boron Injection Tank and the concentrated Boric Acid System and provide the justification for reducing the minimum boron concentration requirements in order to greatly reduce these maintenance problems and thus the associated personnel radiation exposure. The proposed reduction consists of a change in the minimum Boron Injection Tank (BIT) concentration from 11.5% to 0% and a change in the minimum Boric Acid System concentration from 11.5% to 7%. The reduction in BIT concentration can be achieved by taking credit for the

Integral Flow Restrictors in the safety analysis of the main steam line break accident; this accident is discussed in Chapter 14 of the Surry Updated Final Safety Analysis Report (UFSAR). The Integral Flow Restrictors were installed during the steam generator repair outage. The reduction in boric acid system concentration can be accomplished by increasing the minimum allowable Boric Acid Tank inventory associated with each unit from 4200 gallons to 6000 gallons, thereby preserving the capability for cold safe shutdown at any time in life with the most reactive control rod assembly withdrawn from the core.

Section A.2 provides a general description of the current design of the Boron Injection Tank and Boric Acid System and describes the proposed physical changes to each system; operational and maintenance benefits of the proposed changes are discussed in Section A.3.

Analyses have been performed to determine the impact of the proposed changes on the appropriate Surry licensing bases. A boron concentration reduction in the BIT affects only the steamline break transient results. A detailed discussion of the supporting analyses performed for this transient is provided in section B of this attachment. The proposed boron concentration reduction in the BAT does not impact any of the accident analyses presented in Chapter 14 of the UFSAR.

Section C presents an evaluation of the impact of the proposed plant modifications on plant operations and the results of a

review of the FSAR.

A compilation of the required Technical Specification changes to implement the proposed concentration reductions is presented as a separate attachment.

A.2 Changes to Current System Operations

BORON INJECTION TANK

The Boron Injection Tank (BIT) is a 900 gallon carbon steel tank which is internally clad with stainless steel and is part of the Safety Injection System; it contains boric acid solution at a minimum of 11.5% by weight boric acid. Redundant tank heaters and line heat tracing are provided to maintain a minimum solution temperature at a Technical Specifications limit of 145 degrees F, thus preventing boron plateout. Recirculation from the BIT to the Boric Acid Tanks is maintained continuously via a Boric Acid Transfer Pump to ensure the BIT is full of concentrated boric acid at all times and to prevent cold spots and stratification within the tank. The BIT is isolated from the Reactor Coolant System and the Charging Pumps during normal plant operation by two sets of parallel isolation valves. Figure 1 illustrates the system design as described above.

The purpose of the BIT is to provide injection of highly concentrated boric acid to the Reactor Coolant System to mitigate the reactivity addition resulting from a main steam line break accident. Operation of the BIT, which takes place upon actuation of a Safety Injection Signal, does not impact any of the accident analysis results presented in Chapter 14 of the Updated Final Safety Analysis Report other than the steamline break.

During Safety Injection, the suction of the high head safety injection/charging pumps is diverted from the normal suction at the Volume Control Tank (VCT) to the Refueling Water Storage Tank (RWST). The Safety Injection flow path through the BIT is established by the opening of redundant parallel isolation valves upon a Safety Injection signal. Concurrent with the opening of the BIT isolation valves is the closing of redundant isolation valves in the recirculation line to the BAT. Flow from the safety injection/charging pumps is introduced into the BIT via a sparger internal to the vessel. This sparger is designed to disperse the fluid and create a front, prevent channeling and therby cause slug flow to pass through the tank and into the Reactor Coolant System.

Current plans are to reduce the required boric acid concentration in the BIT to 0% with only a minor physical modification to the plant. This modification would entail (1) cutting the recirculation lines (to the BAT and Volume Control Tank) and welding the ends closed to ensure concentrated boric acid will not leak from the Boric Acid Tank to the BIT and (2) removal of electrical power to the recirculation line isolation valves. Electrical power will also be terminated to the BIT heaters and heat tracing circuits of the recirculation lines. The BIT would remain in place. All other system components will remain unaffected, including the actuation of the BIT inlet and outlet isolation valves.

Vepco is evaluating the feasibility of physically removing the BIT and associated BIT/BAT recirculation piping at some future date.

BORIC ACID SYSTEM

The concentrated boric acid system is a part of the Chemical and Volume Control System described in Section 9.1 of the Updated Final Safety Analysis Report. The purpose of the system is to provide an inventory of concentrated boric acid for (1) chemical shim reactivity control, (2) providing makeup to the Reactor Coolant System, Refueling Water Storage Tank, spent fuel pit and refueling cavity as necessary and (3) recirculation of boric acid through the BIT via the boric acid transfer pumps. The system consists of three Boric Acid Tanks, four boric acid transfer pumps, one batch tank, boric acid filters and associated piping, valves, heat tracing, controls and instrumentation. The Boric Acid Tanks are sized to provide sufficient boric acid to bring the reactors to cold shutdown conditions assuming a stuck control rod. A simplified schematic of the system is shown in Figure 2.

The three Boric Acid Tanks (BAT) are 7500 gallon stainless steel tanks which are designed for atmospheric pressure. They serve as the reservoirs for boric acid inventory; the three tanks serve both units. A boric acid solution of 11.5% to 13% by weight is maintained at all times. The upper concentration

limit of 13% is established to ensure concentrations low enough to remain soluble at a 145 degrees F minimum temperature, which is maintained by redundant tank immersion heaters and line heat tracing.

During normal operation, boric acid is supplied to each BAT from the Boric Acid Batching Tank via the Boric Acid Transfer Pumps or from the Boron Recovery System to maintain a minimum level of 4200 gallons dedicated to each unit.

Reduction in the boron concentration requirement for the Boric Acid system to 7% will require increasing the minimum volume of boric acid stored for each operating unit to 6000 gallons. This can be accomplished by resetting the existing level instrumentation and alarms for the new minimum low level. With this increase in volume requirement for each unit, the capability to bring the units to cold shutdown conditions with the most reactive control rod assembly withdrawn from the core is preserved. BAT heater controls and the system heat tracing controls will be reset to maintain a minimum temperature of 112 degrees F to maintain solution solubility.

In summary, the proposed Boric Acid System changes will require the following plant modifications:

1. Reset Boric Acid Tank level instrumentation and alarms for a minimim volume of 6000 gallons.
2. Reset BAT heater and heat tracing controls to maintain a minimum solution temperature of 112

degrees F.

A.3 Benefits

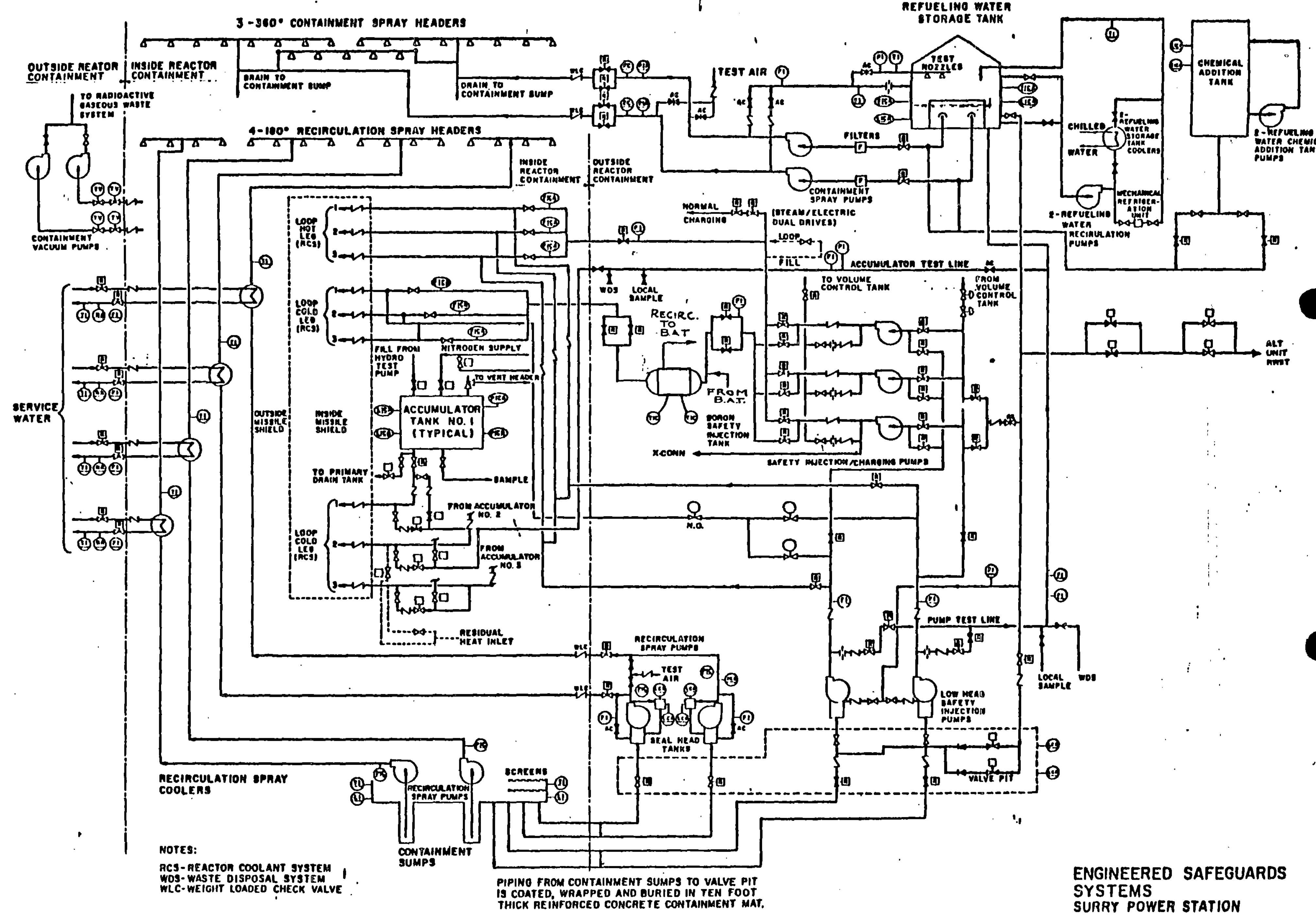
The high boric acid concentrations in the boron injection tank and the concentrated boric acid system is causing numerous maintenance problems, resulting in increased radiation exposures to plant personnel. The concentrated boric acid is a corrosive fluid. Leakage from the systems has been linked to (1) corrosion of carbon steel components and supports and (2) failure of heat tracing equipment.

The lowering of the minimum required boric acid concentration in the BIT to zero (0) ppm (1) reduces the potential for degradation of carbon steel components and supports due to leakage, (2) eliminates the need to maintain BIT heaters and heat tracing on the associated safety injection piping and recirculation lines and (3) eliminates the need for periodic checks of the BIT concentration, thereby reducing radiation exposures for plant personnel. The reduction in boron concentration in the BIT will also reduce the RCS dilution required for a return to power in the event of an inadvertant Safety Injection. This will reduce the amount of letdown which must be processed by the Boron Recovery and Waste Handling systems.

Reducing the minimum required boric acid concentration for the concentrated boric acid system will improve heat tracing system performance, which in turn decreases the potential for system line blockage due to boron plateout. This will increase system

reliability, reduce maintenance requirements and therefore personnel radiation exposure.

In summary, the proposed reductions in BIT and Boric Acid System concentrations offer significant benefits to Vepco in terms of increased operational reliability, reduced maintenance costs and decreased personnel radiation exposures.



ENGINEERED SAFEGUARDS SYSTEMS
SURRY POWER STATION

SECTION A -- FIGURE 1

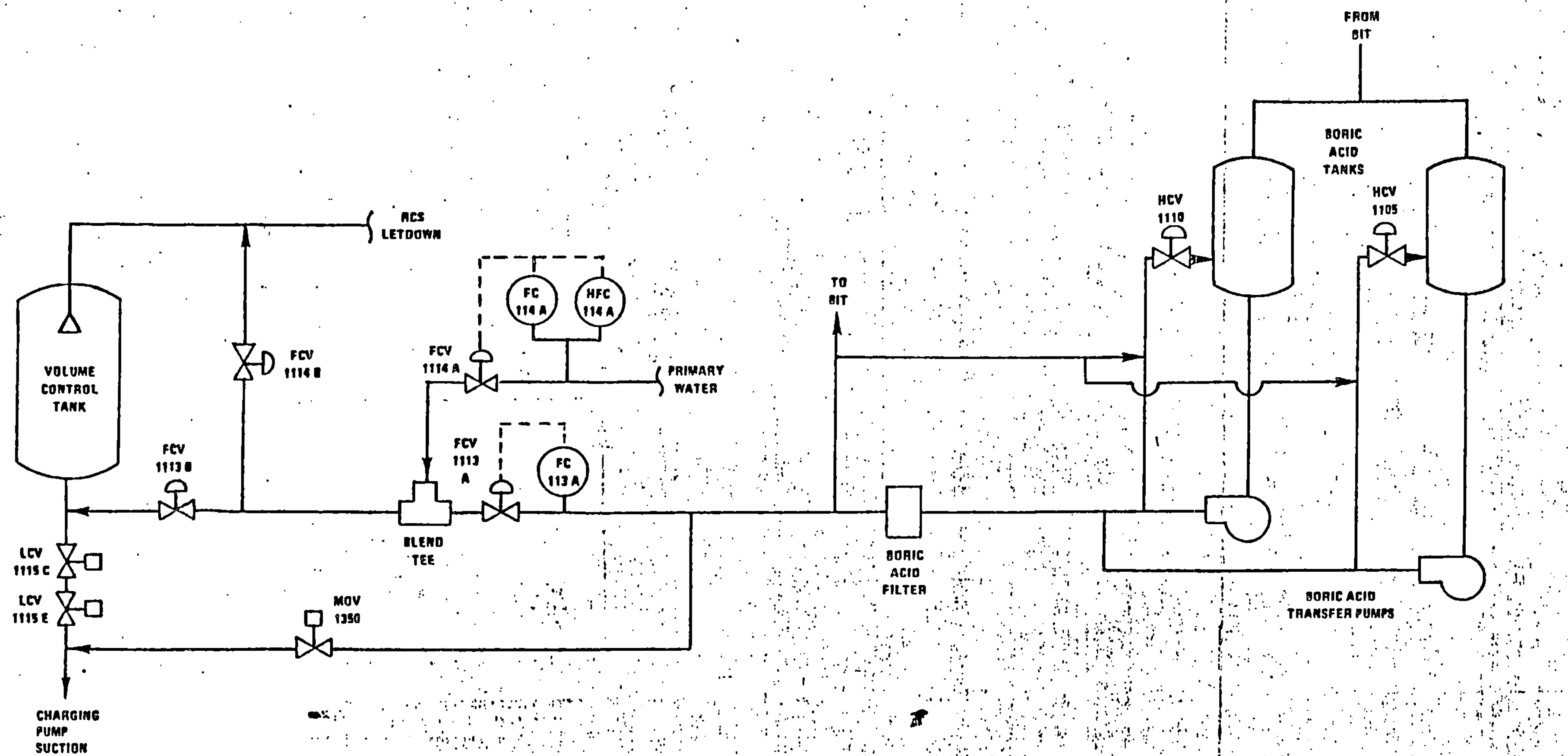


FIGURE 2
BORIC ACID SYSTEM

B. ACCIDENT ANALYSIS -RUPTURE OF A MAIN STEAM PIPE ASSUMING 0 PPM BORON IN THE BORON INJECTION TANK

1. INTRODUCTION AND BACKGROUND

A rupture of a main steam pipe is assumed to include any accident which results in an uncontrolled steam release from a steam generator. The release can occur due to a break in a pipe line or due to a valve malfunction. The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of reactor coolant temperature and pressure. With a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive control rod assembly is assumed stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power even with the remaining control rod assemblies inserted. A return to power following a main steam pipe rupture is a potential problem mainly because of the high hot channel factors which exist when the most reactive rod is assumed stuck in its fully withdrawn position. Assuming the worst combination of circumstances which could lead to resumption of power generation following a main steam line break, the core is ultimately shut down by the boric acid in the Safety Injection System.

The analysis of a main steam pipe rupture is performed to

demonstrate that even with a boron concentration of zero in the Boron Injection Tank (BIT):

a. Assuming a stuck control rod assembly with or without offsite power, and assuming a single failure in the engineered safety features there is no consequential damage to the primary system and the core remains in place and intact.

b. With no stuck control rod assembly, and all equipment operating at design capacity, insignificiant (or no) cladding rupture occurs.

c. There will be no DNB or clad perforation resulting from any single active failure of the main steam system. The single active failure is the opening, with failure to close, of the largest of any single steam bypass, relief or safety valve.

Although DNB and possible clad perforation following a main steam pipe rupture are not necessarily unacceptable, the following analysis shows that no DNB occurs for any rupture assuming the most reactive control rod assembly stuck in its fully withdrawn position.

The following systems provide the necessary protection against a main steam pipe rupture:

a. Safety Injection System actuation from any

of the following*:

(1). Two out of three pressurizer low pressure signals.

(2). Two out of three differential pressure signals between any main steam line and the main steam header.

(3). High steam flow in two out of three main steam lines (one out of two per line) in coincidence with either low Reactor Coolant System average temperature (two out of three) or low main steam line pressure (two out of three).

(4). Three out of four high containment pressure signals.

b. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring upon actuation of the Safety Injection System.

---

*The details of the logic used to actuate Safety Injection are discussed in Section 7 of the FSAR.

c. Redundant isolation of the steam generator feedwater lines. Sustained high feedwater flow would cause additional cooldown, thus, in addition to the normal control action which closes the main feedwater valves, any safety injection signal rapidly closes all feedwater control valves, trips the steam generator feedwater pumps, and closes the feedwater pump discharge valves.

d. Trip of the fast acting main steam line trip valves (designed to close in less than 5 seconds) on:

(1). High steam flow in two out of three main steam lines (one out of two per line) in coincidence with either low Reactor Coolant System average temperature (two out of three) or low steam line pressure (two out of three).

(2). Three out of four high containment pressure signals.

Each main steam line has a fast closing trip valve and a non-return valve. These valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For example, for a break upstream of the trip valve in one line, closure of either the non-return valve in that line or the trip valves in the other lines prevent blowdown of the other steam generators.

All Surry steam generators are equipped with integral flow restrictors at the generator outlet. The restrictors have a smaller flow area than the main pipe and serve to reduce the largest effective break area which must be considered to 1.4 square feet.

2. METHOD OF ANALYSIS

The analysis of the main steam pipe rupture has been performed to determine:

a. The core heat flux and Reactor Coolant System temperature and pressure resulting from the cooldown following the steam line break. The analysis was performed with the RETRAN computer code. The calculation describes the plant neutron kinetics, the reactor coolant system including natural circulation, the pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the break flow rate, core power and point kinetics reactivity and primary coolant temperatures.

b. The thermal and hydraulic behavior of the core following the steam line break. A detailed COBRA core thermal and hydraulic digital computer calculation has been used to determine if DNB occurs for the core conditions computed in (1) above. This calculation solves the continuity, momentum and energy equations of fluid flow in the core and with the W-3 correlation (See reference in Paragraph g below) determines the margin to DNB.

The following assumptions were made:

a. A 1.77% shutdown reactivity from all but one control

rod assembly at no load conditions. This is the end of life design value including design margins with the most reactive control rod assembly stuck in its fully withdrawn position. The actual shutdown capability is expected to be significiantly greater.

b. A negative moderator coefficient representative of end of life core conditions with all but the most reactive control rod assembly inserted. The variation of the coefficient with temperature and pressure has been included. The reactivity versus temperature corresponding to the negative moderator coefficient used is shown in Figure 1. In computing the power generation following a steam line break, the local reactivity feedback from the high neutron flux in the region of the core near the stuck control rod assembly has been included in the overall reactivity balance. The local reactivity feedback is composed of Doppler reactivity from the high fuel temperatures near the stuck control rod assembly and moderator feedback from the high water temperature near the stuck control rod assembly. For the cases analyzed where steam generation occurs in the high flux regions of the core, the effect of void formation on the reactivity has also been included. The effect of power generation in the core on overall reactivity is shown in Figure 2 The curve

assumes end of life core conditions with all control rod assemblies in except the most reactive control rod assembly which is assumed stuck in its fullly withdrawn position (completely removed from core).

c. Minimum safety injection capability corresponding to the operation of only one high head safety injection pump. A boron concentation of 2000 ppm was assumed in the Refueling Water Storage Tank, from which the safety injection pumps take suction. The initial boron concentration in the Boron Injection Tank (BIT) and the associated safety injection piping is assumed to be zero. The time delays required to sweep this unborated water from the piping prior to the delivery of the 2000 ppm boron have been included in the analysis.

d. A conservatively high steam generator heat transfer coefficient (UA) which is representative of nucleate boiling in the secondary side of the generator through out the transient. This is conservative since no allowance for reduction of the heat transfer UA as the water level falls into the tube region has been made. The effects of main feedwater flow (prior to feedline isolation) and auxiliary feedwater flow have been included in the analysis.

e. Hot channel factors corresponding to one stuck control

rod assembly -- the control rod assembly giving the highest factor at end of life. The hot channel factors account for the void existing local to the stuck control rod assembly at the pressure that occurs during the return to power phase following the steam break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck control rod assembly. The hot channel factors depend upon the core temperature, pressure and flow and are therefore different for each case studied. The calculations used to obtain the hot channel factors again assume end of life core conditions with all control rod assemblies in except the most reactive control rod assembly.

f. Three combinations of break sizes and initial unit conditions have been considered in determining the core power and Reactor Coolant System transient.

(a). Complete severence of a main steam pipe, initially at no load conditions with outside power available. The presence of the integral flow restrictors in the steam generators will control the steam release rates for all break locations, both inside and outside the containment

(b). Case (a) above with loss of outside power simultaneous with the steam break.

(c). A break equivalent to a steam flow of 247 lbs per second at 1100 psia from one steam generator with outside power available. This is larger than or equal to the capacity of any single dump or safety valve.

All the cases above assume initial hot shutdown conditions with the control rod assemblies inserted (except for one stuck control rod assembly) at time zero. Should the reactor be just critical or operating at power at the time of a main steam line break, the reactor is tripped by the normal overpower protection system when the power level reaches a trip point. Following a trip at power the Reactor Coolant System contains more stored energy than at no load, the average coolant temperature is higher than at no load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the main steam line break before the no load conditions of Reactor Coolant System temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes a no-load condition at time zero. However, since the initial steam generator mass is greatest at no load, the magnitude and duration of the Reactor Coolant System cooldown are less for main steam line breaks occurring at power.

g. In determination of the critical flux at which burnout could occur the W-3 correlation was used. It was considered to be the correlation which most accurately represents the range of parameters produced in the transients analyzed.

h. In computing the steam flow during a steam line break, the Moody critical flow model (Moody, F. J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, 93, pp 179-187, 1965) was used.

3. Results

The results presented are a conservative indication of the events which would occur following a main steam line rupture. The worst case assumes that all of the following occur simultaneously.

(1). Minimum shutdown reactivity margin of 1.77%.

(2). An end-of-life, rodded core moderator temperature coefficient; use of end-of-life conditions maximizes the positive reactivity insertion resulting from cooldown.

(3). The highest worth control rod assembly stuck in its fully withdrawn position.

(4). The single most restrictive failure of the Engineered Safety Features.

A. Core Power and Reactor Coolant System Transient

Figures 3-4 show the Reactor Coolant System transient and core heat flux following a main steam pipe rupture (complete severence of a pipe) at initial no load conditions (Case A). The break assumed is the largest break which can occur anywhere in the system. Outside power is assumed available such that full reactor coolant flow exists. The transient shown assumes the control rod assemblies inserted at time 0 (with one control rod assembly stuck in its fully withdrawn position) and steam

release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by high differential pressure between any steam generator and the main steam header or by high steam flow signals in coincidence with either low Reactor Coolant System temperature or low steam line pressure trips the reactor. Steam release from at least two steam generators is prevented by either the non-return valves or by automatic trip of the fast acting trip valves in the steam lines by the high steam flow signals in coincidence with either low Reactor Coolant System temperature or low main steam line pressure. Even with the failure of one valve, release is limited to no more than 5 seconds for two steam generators while the third generator blows down. The main steam line trip valves are designed to be fully closed in less than 5 seconds with no flow through them. With the high flow existing during a main steam line rupture, the valves will close considerably faster since closure is flow assisted.

As shown in Figure 3, the core attains criticality with the control rod assemblies inserted (with the design shutdown margin, assuming one stuck control rod assembly) at 22 seconds. Boron solution at 2000 ppm enters the Reactor Coolant System from the Safety Injection System, is diluted and mixed with RCS water and reaches the core at 231 seconds. This reflects a delay of 224.5 seconds to purge the boron injection tank and associated piping of 0 ppm water, and 3.5 seconds to transport

dilute boron from the injection point in the cold legs to the core. The calculation also accounts for a 3 second delay to receive and actuate the safety injection signal and 10 seconds to completely open valve trains in the safety injection lines. Since the safety injection pump accelerates to full speed in less than the time required to open the valve train, and since it is expected that the safety injection signal will be generated in less than 3 seconds, the overall delay time is considered conservative.

The calculation assumes the boric acid is mixed with and diluted by the water flowing in the Reactor Coolant System prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the Reactor Coolant System and in the Safety Injection System. The variation of mass flow rate in the Reactor Coolant System due to water density changes is included in the calculation as is the variation of flow rate in the Safety Injection System due to changes in the Reactor Coolant System pressure. The Safety Injection System flow calculation includes the line losses in the system as well as the pump head curve.

No credit has been taken for any boron in the BIT or safety injection lines which enters the Reactor Coolant System prior to the 2000 ppm boric acid from the Refueling Water Storage Tank. The heat flux achieved was 23.7% of the value at 2441 MWt (a summary of conditions is listed in Table 1).

Figures 5-7 show the responses for the previous break except with a loss of outside power (Case B). Reactor Coolant system flow coastdown is assumed to occur simultaneously with the break. The Safety Injection System delay time includes the time required to start safety injection pumps with emergency power from the diesel generators. Only one safety injection pump is assumed. Criticality is attained at 31 seconds and the peak heat flux is 8.1% of the values at 2441 MWt. A summary of the time sequence for the above case is given in Table 1.

Figures 8-9 show the transient following a break equivalent to a steam flow of 247 lbs per sec. at 1100 psia with steam release from one steam generator (Case C). The assumed steam release is larger than or equal to the capacity of any single dump or safety valve. In this case, safety injection is initiated automatically by low pressurizer pressure at 140 seconds. Operation of one safety injection pump is considered, since this will provide the most conservative results. Criticality is attained at 280 seconds; dilute boron solution reaches the core at 370 seconds and limits the peak heat flux to 4.06% of the value at 2441 mwt. The cooldown for the case shown in Figures 8-9 is more rapid than the case of steam release from all steam generators through one relief, bypass or safety valve. The transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel

elements or the energy stored in the other steam generators. Since the transient occurs over a period of about five minutes, the neglected stored energy is likely to have a significant effect in slowing the cooldown.

It should be noted that following a main steam line break only one steam generator blows down completely. Thus, two steam generators are still available for dissipation of decay heat after the initial transient is over. In the case of loss of outside power this heat is removed to the atmosphere and the atmospheric safety valves have been sized to cover this condition.

B. Margin to Critical Heat Flux

Using the transients shown in Figures 3 through 9 the Westinghouse W-3 correlations was used in conjunction with the Vepco version of the COBRA core thermal hydraulics code to determine the margin to DNB. Carefully chosen points from each transient were examined and the results are presented in Table 2. The power and flow conditions are shown together with pressure and core inlet temperatures. It was found that all cases had a minimum DNBR greater than 1.30.

TIME SEQUENCE OF EVENTS FOR MAJOR SECONDARY SYSTEM PIPE RUPTURE

TABLE 1

ACCIDENT	EVENT	TIME(SEC.)
Major Secondary System Pipe Rupture		
1. Case a	Steam line ruptures	0
	Pressurizer empty	11
	Criticality attained	22
	Dilute boron reaches core	231
2. Case b	Steam line ruptures	0
	Pressurizer empty	12
	Criticality attained	31
	Dilute boron reaches core	245
4. Case c	Steam line ruptures	0
	Pressurizer empty	100
	Criticality attained	280
	Dilute boron reaches core	370

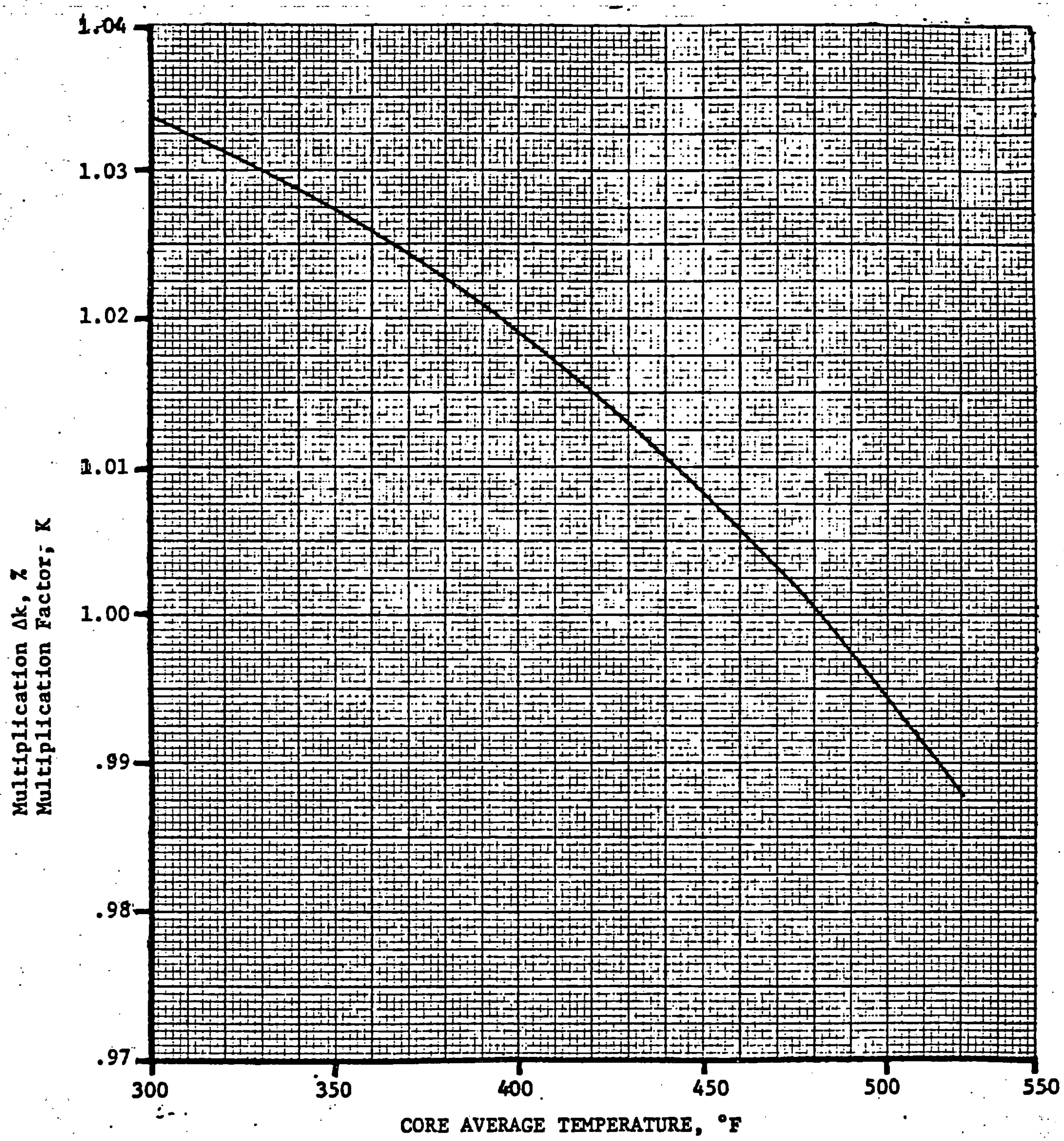
STEAMBREAK ACCIDENT STATEPOINTS

TABLE 2

	Hypothetical Break		Credible Break
	With Power	Without Power	With Power
	Case A	Case B	Case C
Core Heat Flux, % of 2441 MWT	23.7	8.1	4.1
RCS Pressure, psia	959	853	733
Loop A Inlet Temp, °F	398	276	460
Loop B Inlet Temp. °F	469	497	482
Core Boron Concentration, PPM	0.0	0.4	8.2
RCS Flow, %	100	6.4	100
Reactivity, % deltaK/K	.007	.003	.026
Time, sec.	201	250	395
DNBR	>1.30	>1.30	>1.30

FIGURE 1

(SPS UFSAR Figure 14.3-5)



VARIATION OF REACTIVITY WITH CORE TEMPERATURE
AT 1000 PSIA FOR THE END OF LIFE RODDED CORE
WITH ONE CONTROL ROD ASSEMBLY STUCK
(ASSUMES 0 POWER)

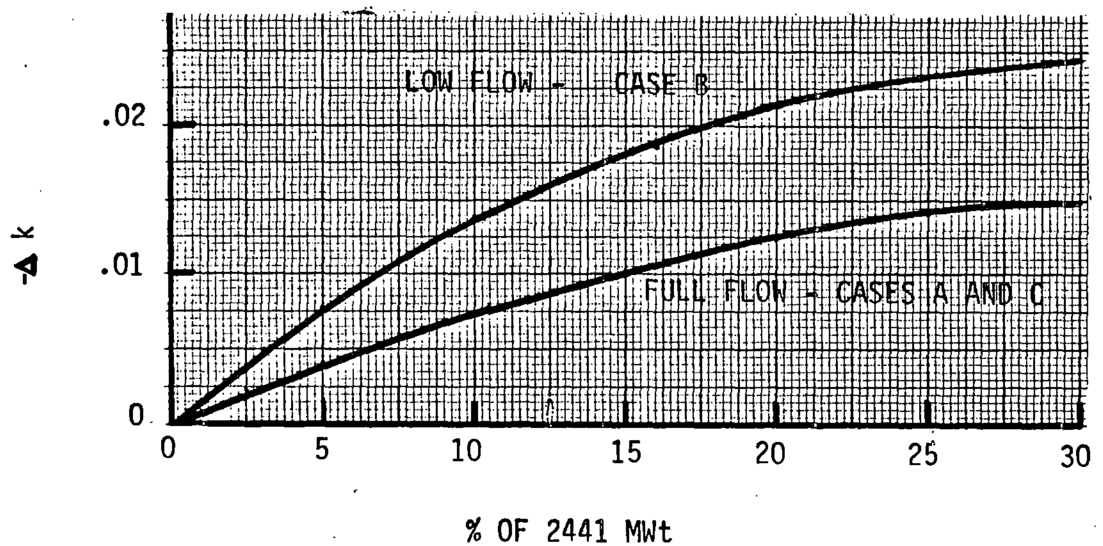


FIGURE 2

VARIATION OF REACTIVITY WITH POWER AT CONSTANT CORE AVERAGE TEMPERATURE. VALUES INDICATED WERE USED IN STEAM PIPE RUPTURE ANALYSIS FOR THE END OF LIFE RODDED CORE WITH ONE CONTROL ROD ASSEMBLY STUCK

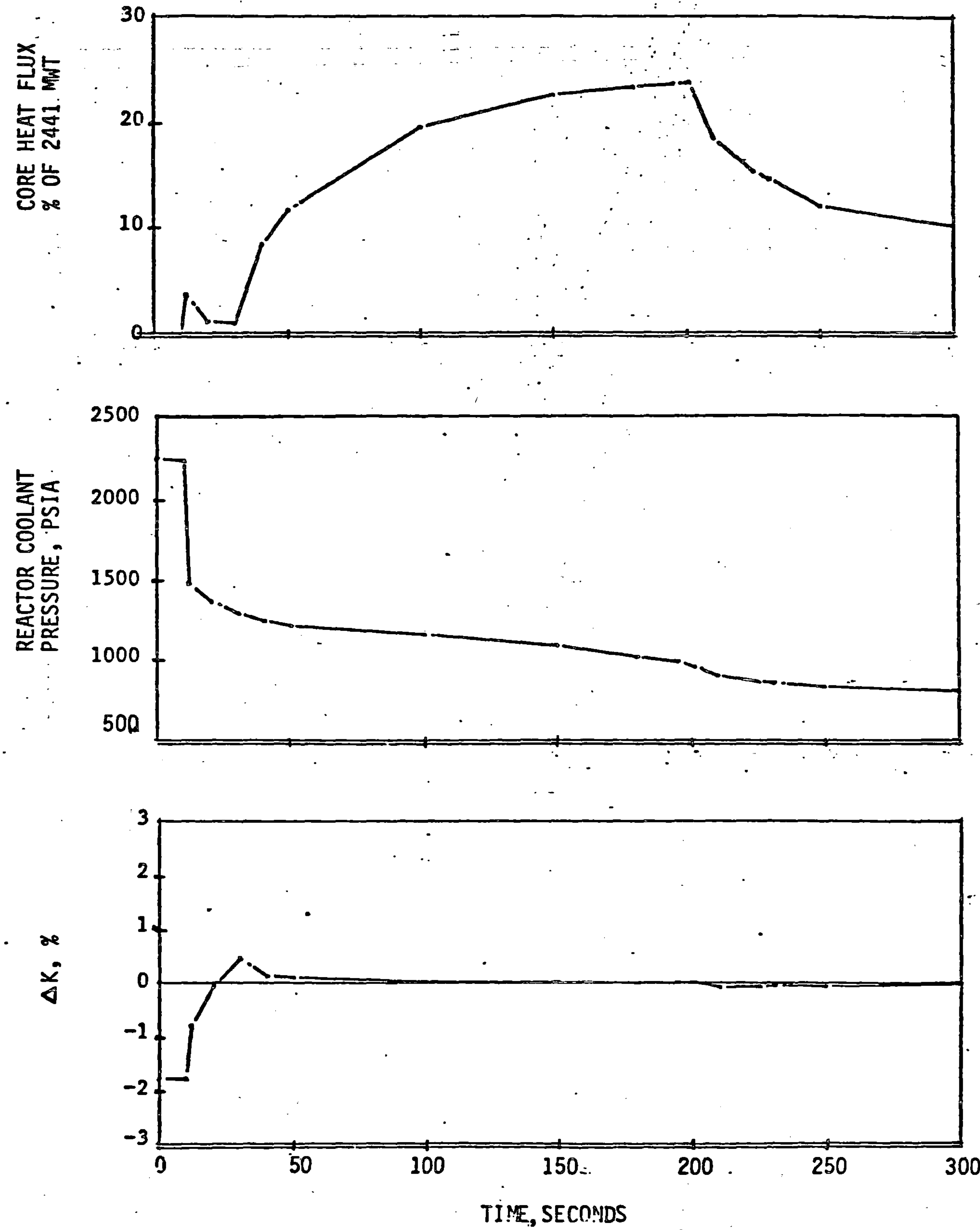


FIGURE 3. STEAM LINE BREAK AT EXIT OF STEAM GENERATOR WITH ZERO PPM IN BIT AND OUTSIDE POWER (CASE A)

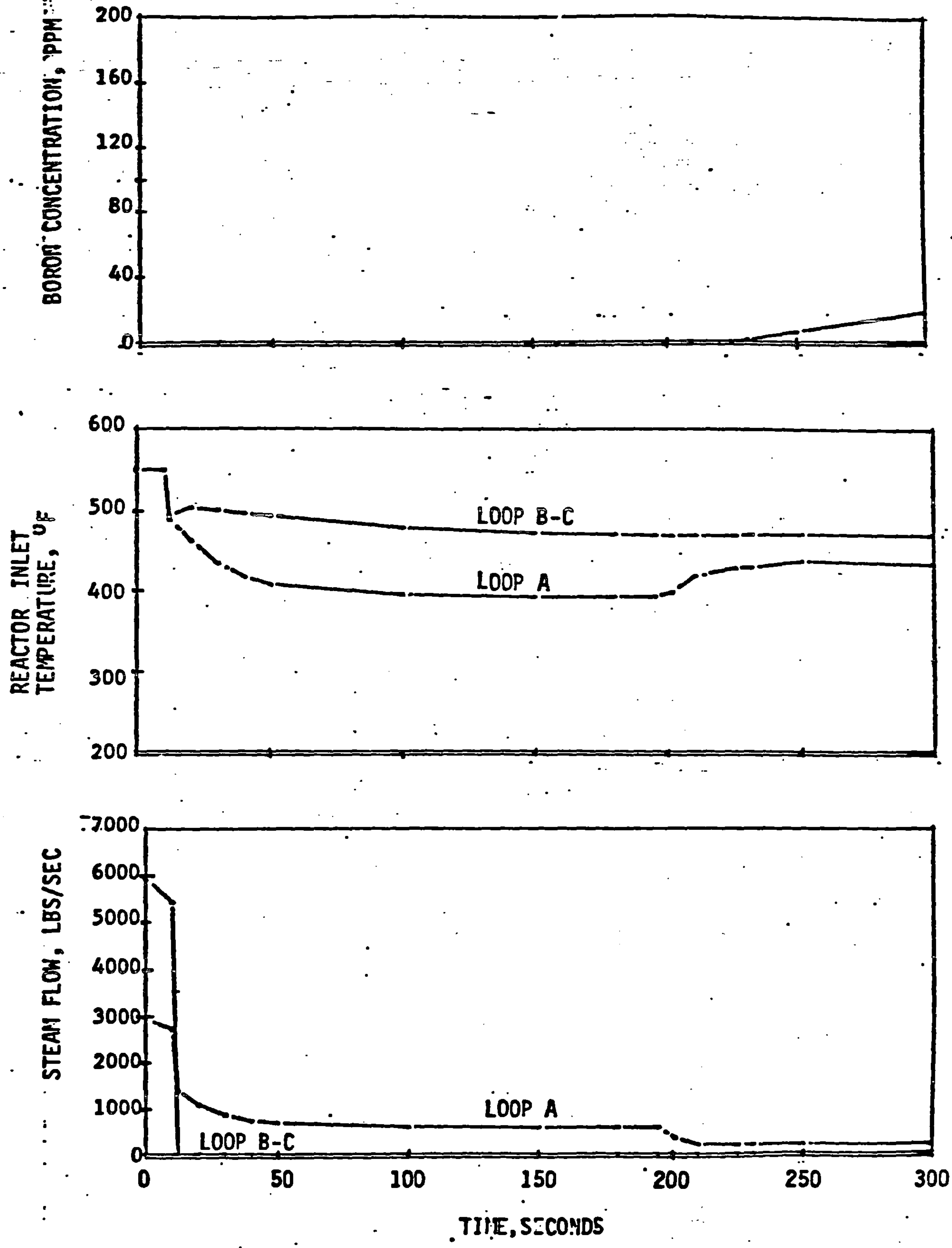


FIGURE 4. STEAM LINE BREAK AT EXIT OF STEAM GENERATOR WITH ZERO PPM IN BIT AND OUTSIDE POWER (CASE A)

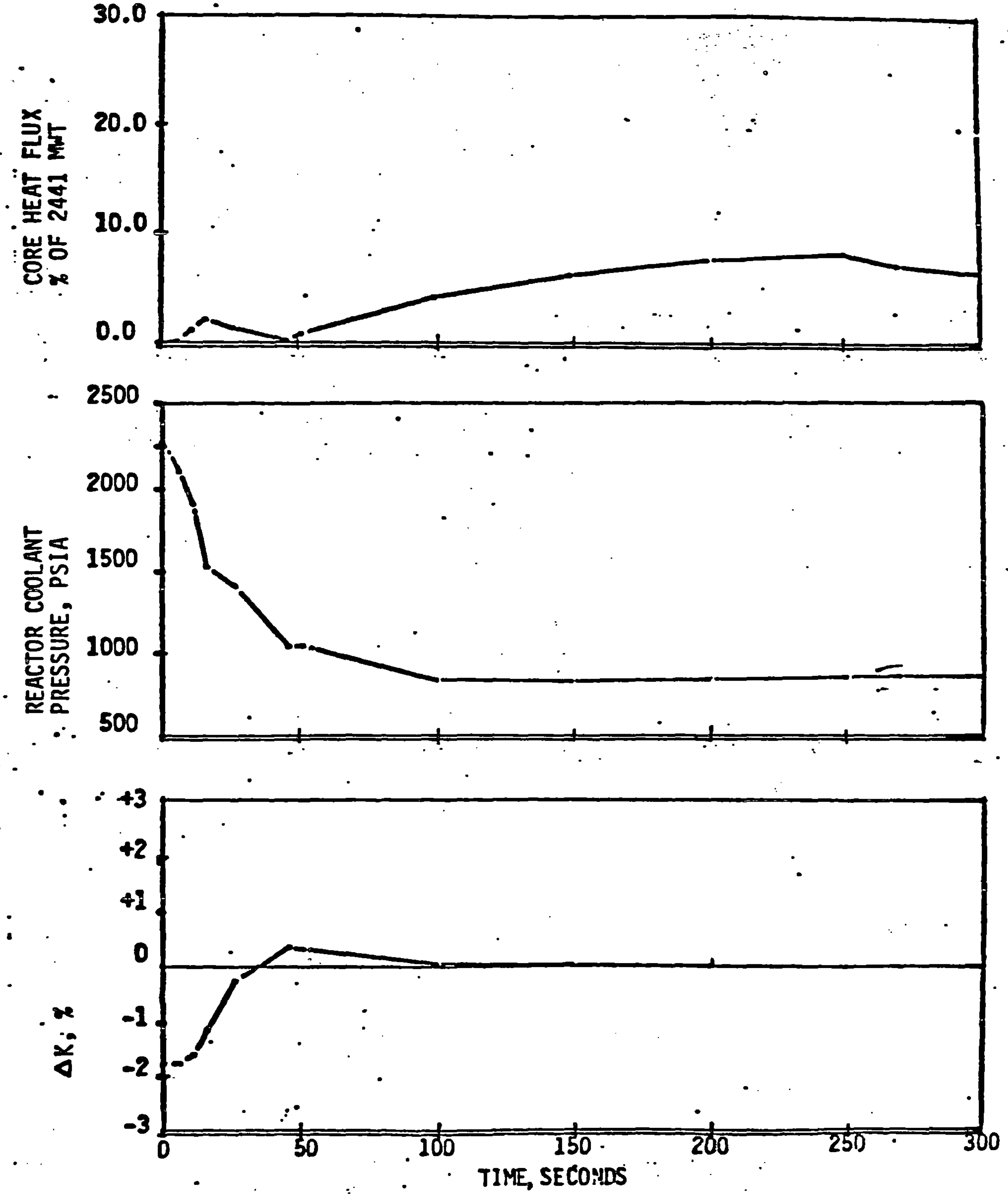


FIGURE 5. STEAM LINE BREAK AT EXIT OF STEAM GENERATOR WITH ZERO PPM IN BIT WITHOUT OUTSIDE POWER (CASE B)

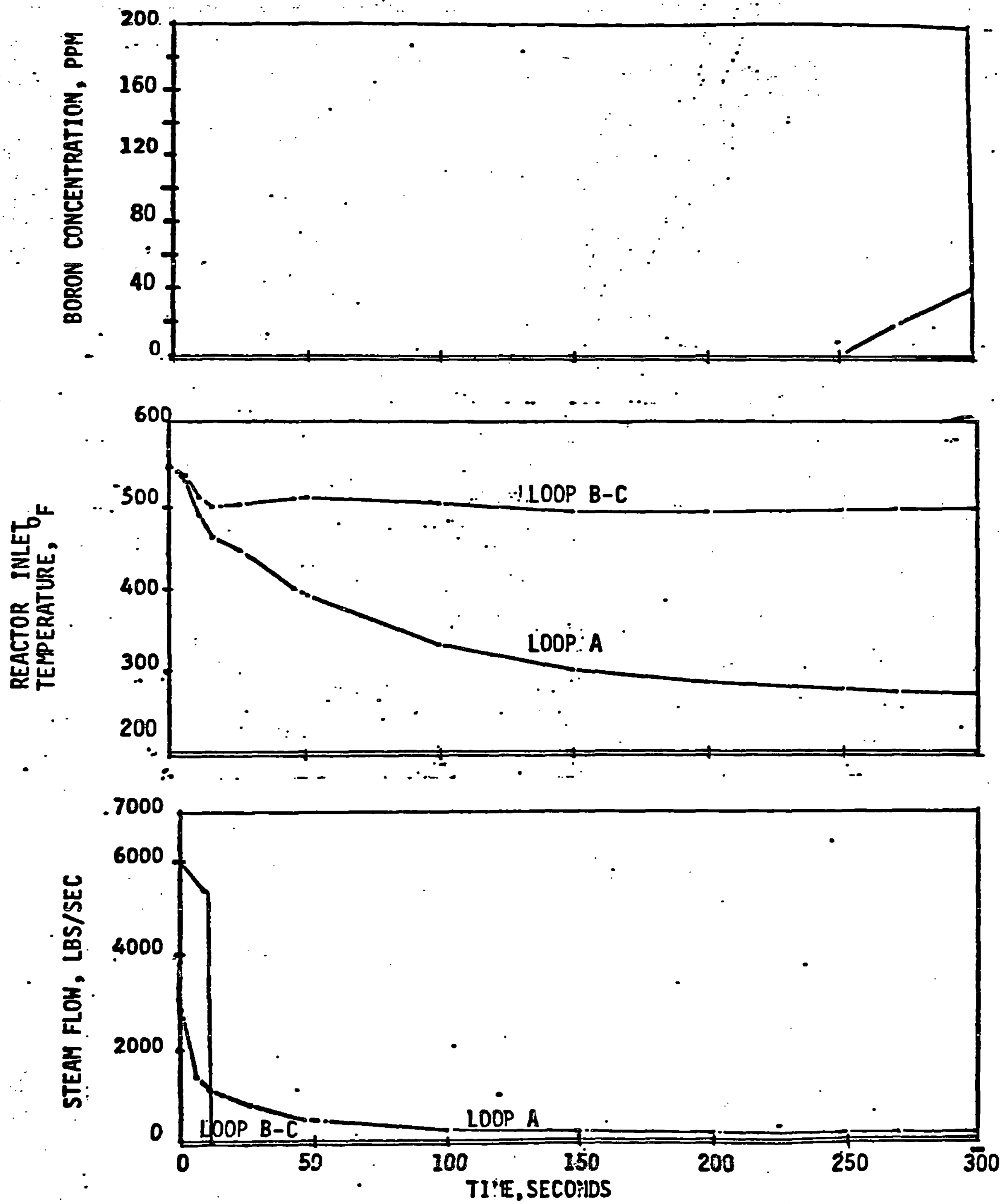


FIGURE 6. STEAM LINE BREAK AT EXIT OF STEAM GENERATOR WITH ZERO PPM IN BIT WITHOUT OUTSIDE POWER (CASE B)

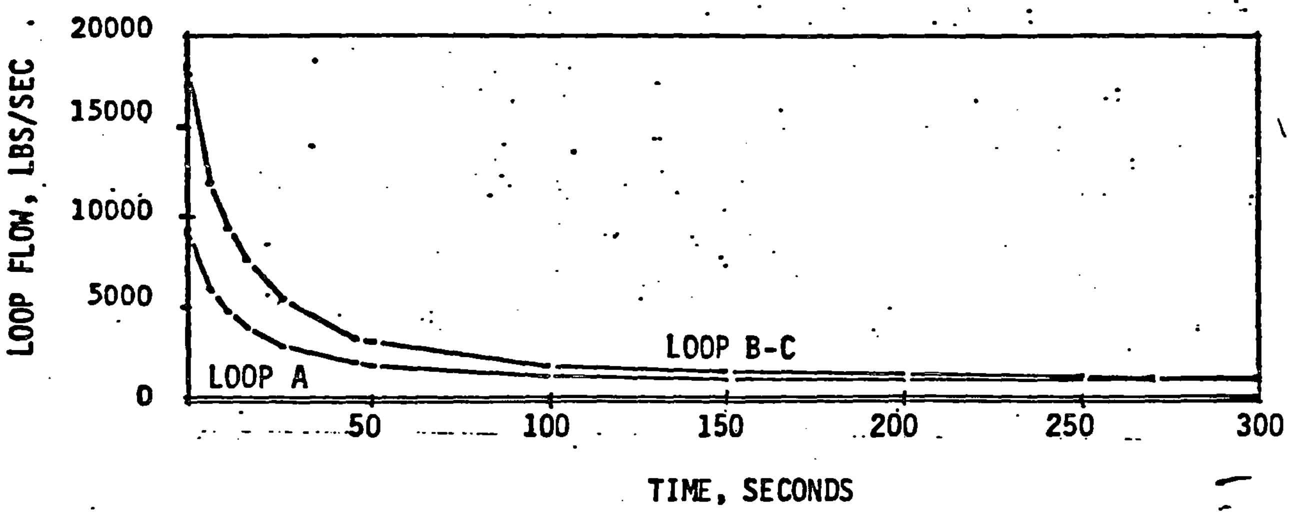


FIGURE 7. STEAM LINE BREAK AT EXIT OF STEAM GENERATOR WITH ZERO PPM IN BIT WITHOUT OUTSIDE POWER (CASE B)

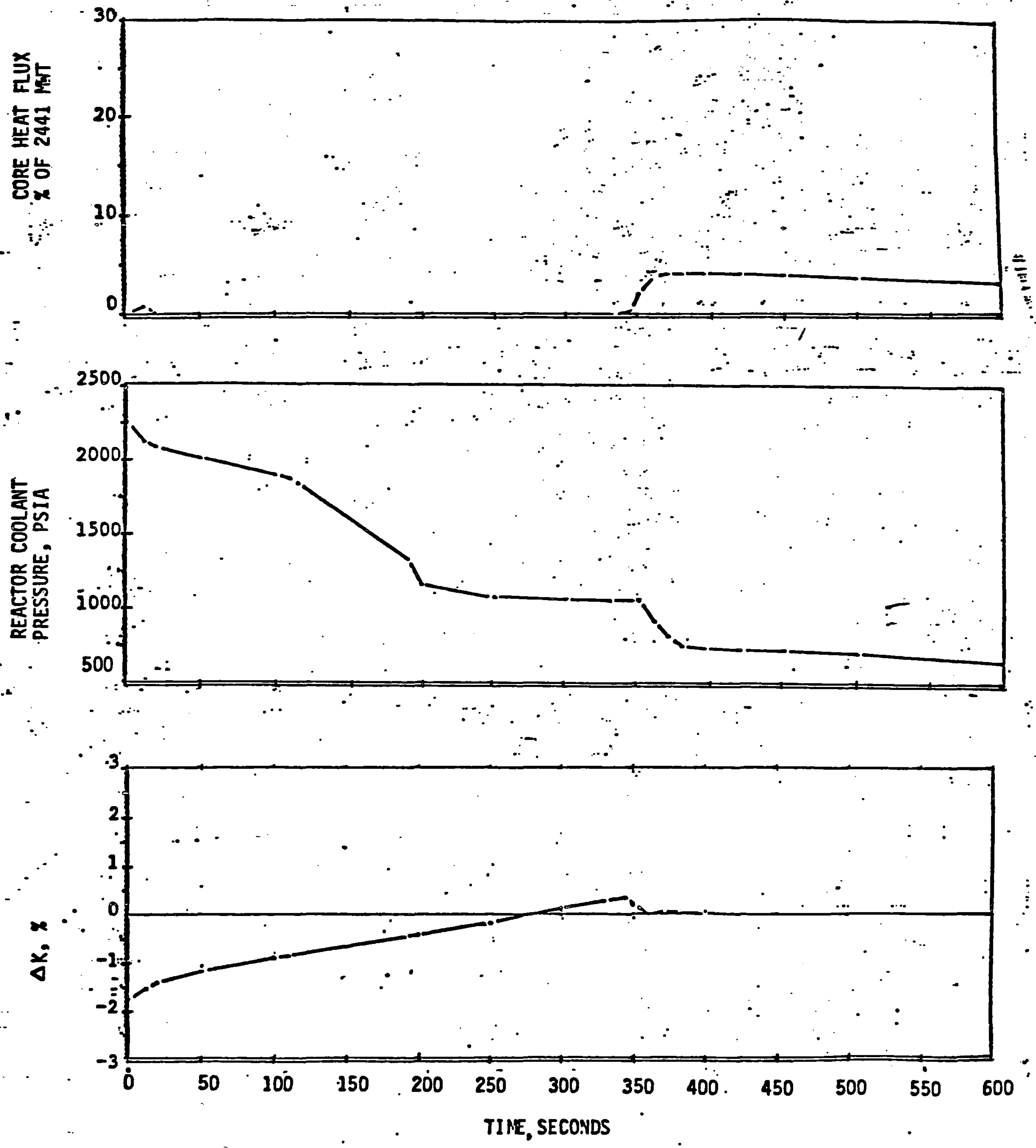


FIGURE 8. STEAM LINE BREAK EQUIVALENT TO 247 LBS/SEC AT 1100 PSIA WITH ZERO

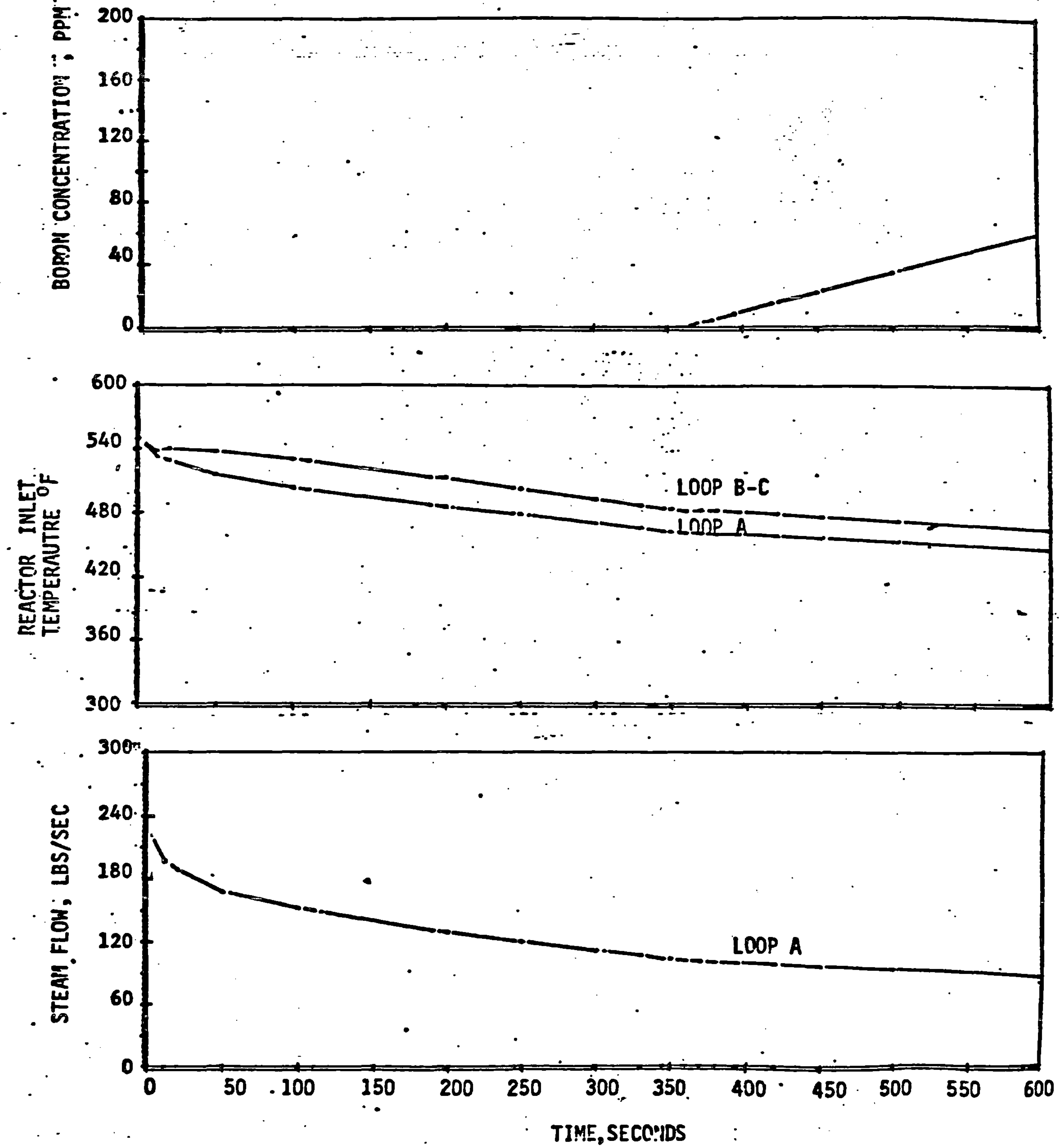


FIGURE 9 . STEAM LINE BREAK EQUIVALENT TO 247 LBS/SEC AT 1100 PSIA WITH ZERO PPM IN BIT AND OUTSIDE POWER (CASE C)

C. OPERATIONS/FSAR REVIEW

1. Operations Impact

An evaluation of effects on plant operations was made to determine all positive or negative implications of reducing boron concentration in the Boric Acid Tank and the Boron Injection Tank. A summary of those changes is listed below:

a. The reduction of the Boric Acid Tank concentration and the increase in tank volume was evaluated and the following operations were found to be impacted:

(1). Increase in time to borate under normal and emergency operating conditions- Station Curve Book nomographs for boron addition will be revised for the decreased minimum concentration of 7%. All increases in times required to borate or makeup were found to be satisfactory from a plant operational standpoint.

(2). Increased Boric Acid Tank volume- The increase in BAT volume to 6000 gallons (previously 4200) was evaluated relevant to overflow considerations. It was determined that sufficient tank capacity is available to replenish the tank in standard batch volumes without

overflowing the tank.

(3). Setpoint, chemistry, and operating procedure changes have been identified and will be revised in accordance with approved procedures.

b. The reduction of the minimum boric acid concentration in the Boron Injection Tank (BIT) to 0% boron concentration was evaluated and found to effect the following:

(1). Recirculation of the BIT will no longer be required to maintain BIT concentration.

(2). Periodic sampling of BIT concentration will no longer be required.

2. UFSAR REVIEW

The Updated Final Safety Analysis Report for Surry has been reviewed for required changes as a result of the boron concentration reduction. Several areas will require revision due to these changes. Upon approval of this submittal, these changes will be submitted with the normal yearly UFSAR update.

For example, a change required to Section 9.1.1.2-Chemical and Volume Control- is the time required to shut the reactor down (i.e., to hot shutdown) with no rods inserted. This condition

has been analyzed to show that the increase in time required to shut down (from approximately 15 minutes to approximately 25 minutes) is acceptable.

An evaluation of the time required to borate to cold shutdown conditions with the reduced boric acid concentration has also been performed. The results of the evaluation show that, for conservative worst-case conditions, the boric acid volume required to borate to cold shutdown conditions will be less than 6000 gallons, and the associated boration times will be less than 100 minutes. This is well within the requirements of the most restrictive action statement in the Technical Specifications (i.e., be in cold shutdown within 30 hours).

The evaluation was based on a review of design data for several recent Surry core reloads. Significant conservatisms in the analysis include the following:

a. The initial condition assumes the peak xenon concentration which would occur following a reactor trip from full power; the final condition is assumed to correspond to no xenon.

b. The total shutdown margin available following reactor trip is assumed to correspond to the Technical Specifications limit; for most reload cores the available margin is significantly higher.

c. The moderator temperature defect is based on end of life core conditions, where it will be largest.

The calculated reacitvity requirements were increased by an uncertainty factor of 20% to cover calculational uncertainties and cycle-to-cycle variations in the requirements.

e. A volumetric boron mixing model is used which conservatively neglects the fact that the boric acid is being introduced at colder temperatures and therefore higher densities than exist in the Reactor Coolant System.

An evaluation of the maximum reactivity insertion due to boron dilution resulting from inadvertent discharge of the BIT to the Reactor Coolant System has been performed. The results show that inadvertent criticality cannot result from such a discharge at either refueling or cold shutdown conditions. At hot shutdown or at-power conditions, the maximum reactivity insertion rates realized from such a discharge are well within the range considered in the rod withdrawal analyses of Chapter 14 of the UFSAR.

D. CONCLUSIONS

The reduction of minimum boric acid concentration requirements from 11.5% to 0% in the Boron Injection Tank and from 11.5% to 7.0% in the Concentrated Boric Acid System at Surry Power Station offers significant benefits to Virginia Electric and Power Company. These benefits include increased operational reliability, reduced maintenance costs and decreased personnel radiation exposure.

Additionally, a detailed operational review was conducted; it has been concluded that the plant can continue to be operated in a safe and efficient manner following the change.

Analyses of the Main Steam Pipe Rupture have been performed to demonstrate that the conclusions reached in Chapter 14 of the Updated Final Safety Analysis report will not be impacted by the change. As such, the change will not introduce any unreviewed safety questions as defined in 10 CFR 50.59. Also, the change does not involve a significant hazards consideration. There is a relaxation in the limiting condition for operation; however, a commensurate level of safety is maintained by the presence of the steam generator integral flow restrictors, thus, allowing zero ppm concentration of boron in the boron injection tank.