

3.4 ENGINEERED SAFETY FEATURES

Applicability: Applies to the operating status of the Engineered Safety Features.

Objective: To define those limiting conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, and (3) to remove airborne iodine from the containment atmosphere in the event of a Maximum Hypothetical Accident.

Specification: 1. SAFETY INJECTION AND RESIDUAL HEAT REMOVAL SYSTEMS

- a. The reactor shall not be made critical, except for low power physics tests, unless the following conditions are met:
 1. The refueling water tank shall contain not less than 320,000 gal. of water with a boron concentration of at least 1950 ppm.
 2. The boron injection tank shall contain not less than 900 gal. of a 20,000 to 22,500 ppm boron solution. The solution in the tank, and in isolated portions of the inlet and outlet piping, shall be maintained at a temperature of at least 145F. TWO channels of heat tracing shall be operable for the flow path.*
 3. Each accumulator shall be pressurized to at least 600 psig and contain 875-891 ft³ of water with a boron concentration of at least 1950 ppm, and shall not be isolated.
 4. FOUR safety injection pumps shall be operable.

* See reference (11) on Page B3.4-2

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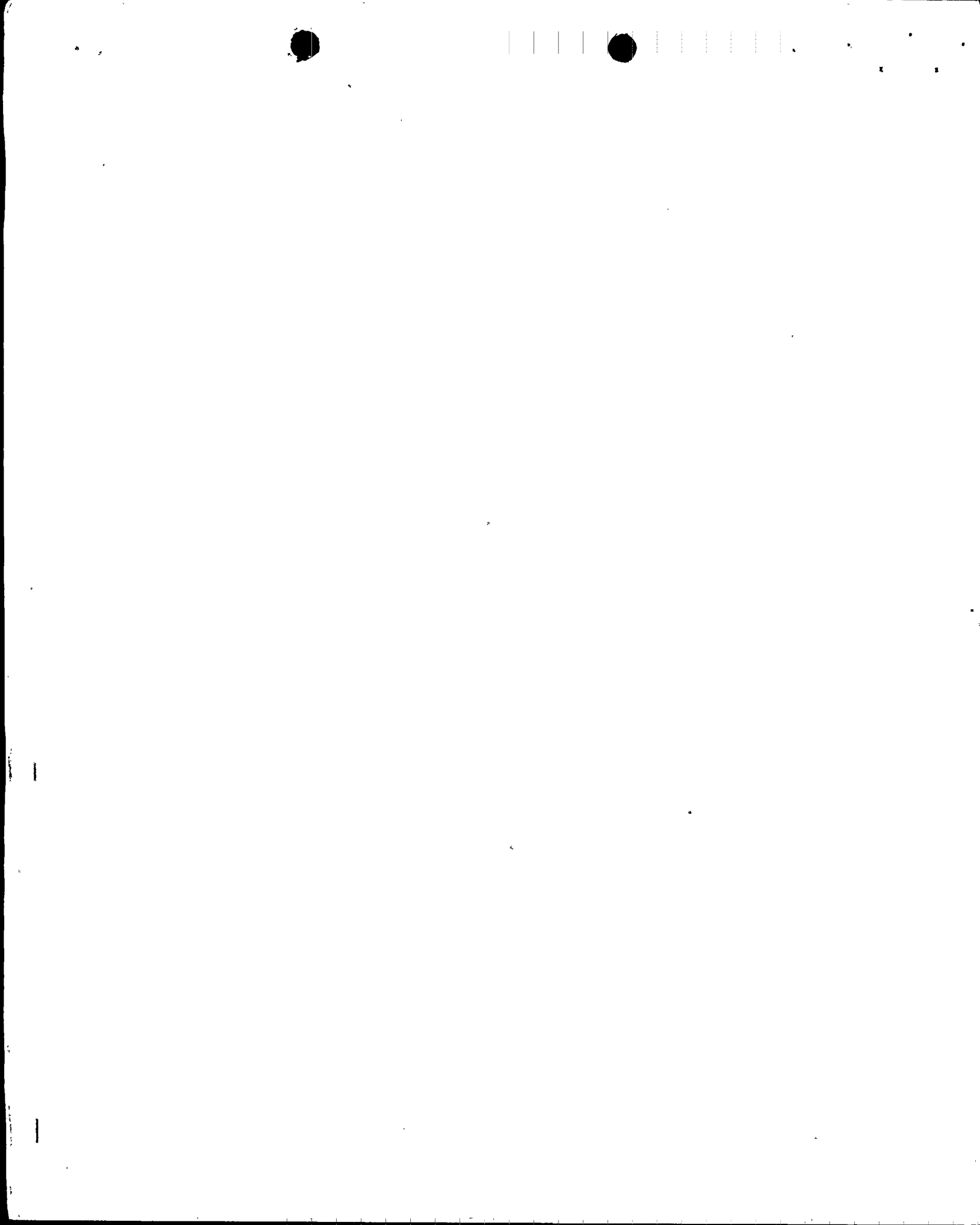
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5. TWO residual heat removal pumps shall be operable.
 6. TWO residual heat exchangers shall be operable.
 7. All valves, interlocks and piping associated with the above components and required for post accident operation, shall be operable, except valves that are positioned and locked. Valves 864-A, B; 862-A, B; 865-A, B, C; 866-A, B shall have power removed from their motor operators by locking open the circuit breakers at the Motor Control Centers. The air supply to valve 758 shall be shut off to the valve operator.
- b. During power operation, the requirements of 3.4.1a may be modified to allow one of the following components to be inoperable (including associated valves and piping) at any one time except for the cases stated in 3.4.1.b.2. If the system is not restored to meet the requirements of 3.4.1a within the time period specified, the reactor shall be placed in the hot shutdown condition. If the requirements of 3.4.1a are not satisfied within an additional 48 hours the reactor shall be placed in the cold shutdown condition. Specification 3.0.1 applies to 3.4.1.b.
1. ONE accumulator may be out of service for a period of up to 4 hours.
 2. ONE of FOUR safety injection pumps may be out of service for 30 days: A second safety injection pump may be out of service, provided the pump is restored to operable status within 24 hours. - TWO of the FOUR safety injection pumps shall be tested to demonstrate operability before initiating maintenance of the inoperable pumps.
 3. ONE channel of heat tracing on the flow path may be out of service for 24 hours.*

*See reference (11) on Page B3.4-2



2. Pumps shall start and reach required head for normal or recirculation flow, whichever is applicable to the operating condition; the instruments and visual observations shall indicate proper functioning. Test operation shall be for a least 15 minutes.

b. Valves

1. The boron injection tank isolation valves receiving a Safety Injection signal shall be cycled monthly.††*
2. The containment recirculation sump suction valves shall be cycled monthly.†
3. Accumulator check valves shall be checked for operability during each refueling shutdown.
4. The refueling water storage tank outlet valves shall be tested in performing the respective pump tests.†

† - N.A. during cold or refueling shutdowns. The specified tests, however, shall be performed within one surveillance interval prior to reactor startup.

†† - N.A. during cold or refueling shutdowns. The specified tests, however, shall be performed within one surveillance interval prior to heatup above 200 F.

* See reference (11) on Page B3.4-2



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MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

	<u>Check</u>	<u>Frequency</u>	<u>Max. Time Between Tests (Days)</u>
1. Reactor Coolant Samples	Radiochem. ($T_{1/2} > 30$ Min)	Monthly	45
	C1 & O ₂ & F	5/Week	3
	Tritium Activity	Weekly	10
	Gross β, γ Activity ($\mu\text{Ci/cc}$)	5/Week	3
	Boron Concentration	2/Week	5
	\bar{E} Determination	Semi-annually	30 Wks
2. Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly†	10
3. Boric Acid Tank	Boron Concentration	2/Week	5
4. Boron Injection Tank*	Boron Concentration	Monthly†	45
5. Control Rods	Rod drop times of all full length rods	For all rods at least once per 18 months and following each removal of the reactor vessel head. For specifically affected individual rods following maintenance on or modification of the control rod drive system which could affect the drop time of those specific rods.	NA
	Partial movement of full length rods	Biweekly while critical	20
6. Pressurizer Safety Valves	Set Point	Each refueling shutdown	NA
7. Main Steam Safety Valves	Set Point	Each refueling shutdown	NA
8. Containment Isolation Trip	Functioning	Each refueling shutdown	NA
9. Refueling System Interlocks	Functioning	Prior to each refueling	NA
10. Accumulator	Boron Concentration	At least once per 31 days and within 6 hours after each solution volume increase of $\geq 1\%$ of tank volume.†	NA

* See reference (11) on Page B3.4-2



BASES FOR LIMITING CONDITIONS FOR OPERATION, ENGINEERED SAFETY FEATURES

1. Safety Injection and Residual Heat Removal Systems

- a.1 The requirements for refueling water tank storage meet the safety analysis. (1)
- a.2 The boron injection tank contains sufficient solution to meet the steam line break accident analysis. (1)(2)(11)
- a.3 Any two accumulators meet the requirements for the MHA analysis. (1)(3)
- a.4 Any two safety injection pumps meet the requirements of the MHA analysis and the steam line break accident analysis. (2)(4)(5)
- a.5, a.6 A single residual heat removal pump and heat exchanger meets the MHA analysis requirements. (4)(5)
- b.1 See a.3 above
- b.2 See a.4 above
- b.3, b.4 See a.5 above

2. Emergency Containment Cooling Systems

Either two of the three emergency containment cooling units or one of the two spray pumps has the cooling capability required to meet the MHA analysis. (6)(7)(9)



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3. Emergency Containment Filtering System

Two of three filter units have capacity to meet the MHA analysis. (7)(8)

4. Component Cooling System

One pump and two heat exchangers meet the requirements of the MHA analysis. (10)

5. Intake Cooling Water System

One pump meets the requirements of the MHA analysis. (6)

References:

- (1) FSAR 6.2.2
- (2) FSAR 14.2.5
- (3) FSAR 14.3.2
- (4) FSAR 14.1.9
- (5) FSAR 6.2.3
- (6) FSAR 14.3.4
- (7) FSAR 6.3
- (8) FSAR 14.3.5
- (9) FSAR 6.4
- (10) FSAR 9.3
- (11) The requirement for use of the BIT tanks for Mitigation of the Main Steam Line Break accident has been removed following installation of the Model 44F Steam Generators. The required supporting analyses can be found in L-81-(502), dated 11/30/81. The temperature requirement above 145° F is no longer applicable.
There is no Boron Concentration requirement in the BIT.



B3.6 BASES FOR LIMITING CONDITIONS FOR OPERATION, CHEMICAL AND VOLUME CONTROL SYSTEM

The Chemical and Volume Control System provides control of the Reactor Coolant System boron inventory. There are three sources of borated water available for injection through three different paths:

- (1) The boric acid transfer pumps can deliver the boric acid tank contents to the charging pumps.
- (2) The charging pumps can take alternate suction from the refueling water storage tank.
- (3) The safety injection pumps can take their suction from the refueling water storage tank and inject the boron injection tank contents. *

The quantity of boric acid in storage from either the boric acid tanks or the refueling water storage tank is sufficient for cold shutdown at any time during core life.

One channel of heat tracing is sufficient to maintain the specified temperature limit.

* See reference (11) on Page B3.4-1

Reference

FSAR - Section 9.2



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2.1.6 STEAM BREAK ANALYSIS - BORON INJECTION TANK REMOVAL

2.1.6.1 Introduction

As part of the Florida Power and Light Uprating/Upgrading Study, Westinghouse committed to determine the minimum boron injection tank (BIT) boron concentration acceptable for the steamline break analysis (core reactivity transient). The steamline break cases were analyzed assuming the complete removal of the BIT, since this is the most limiting case.

Presently, the "hypothetical" steamline break (double ended rupture of a main steamline) and the "credible" steamline break (the failure open of a single steam generator relief, safety, or turbine bypass valve) serve as the Westinghouse steamline break licensing basis, and define the existing requirements on the minimum BIT boron concentration. Therefore, Westinghouse analyzed the following four cases assuming the removal of the BIT: "hypothetical" steamline break, with and without offsite power available, and two "credible" steamline break cases with offsite power available (uniform and non-uniform breaks).

The "credible" steamline break and the "hypothetical" steamline break cases analyzed are discussed below, respectively.

2.1.6.2 Inadvertent Opening of a Steam Generator Relief or Safety Valve

2.1.6.2.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the main steam system result from an inadvertent opening of a single steam dump, relief, or safety valve.

The steam release as a consequence of this accident 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 in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

The analysis is performed to demonstrate that the following criterion is satisfied: assuming a stuck rod cluster control assembly, with offsite power available, and assuming a single failure in the engineered safety features, there will be no consequential damage to the core or reactor coolant system after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief, or safety valve.

Accidental depressurization of the secondary system is classified as an ANS Condition II event.

2.1.6.2.2 Analysis of Effects and Consequences

A. Method of Analysis

The following analyses of a secondary system steam release are performed for this section:

1. A full plant digital computer simulation using the LOFTRAN code (Reference 1) to determine reactor coolant system temperature and pressure, during cooldown, and the effect of safety injection.
2. Analyses to determine that there is no damage to the core or reactor coolant system.

The following conditions are assumed to exist at the time of a secondary steam system release:

1. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and with the most reactive rod cluster control assembly stuck in its fully withdrawn position. Operation of



rod cluster control assembly banks during core burnup is restricted in such a way that addition of positive reactivity in a secondary system steam release accident will not lead to a more adverse condition than the case analyzed.

2. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive rod cluster control assembly in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included. The K_{eff} versus temperature at 1000 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 2.1.6-1.
3. Minimum capability for injection of concentrated boric acid solution corresponding to the most restrictive single failure in the safety injection system. This corresponds to the flow delivered by two safety injection pumps delivering their full contents to the cold leg header. Low concentration boric acid must be swept from the safety injection lines downstream of the refueling water storage tank prior to the delivery of concentrated boric acid (2000 ppm) to the reactor coolant loops. This effect has been allowed for in the analysis.
4. The case studied is a steam flow of 247 lb/sec at 1100 psia from one steam generator with offsite power available. This is the maximum capacity of any single steam dump, relief, or safety valve. Initial hot standby conditions with minimum required shutdown margin at the no-load T_{avg} is assumed since this represents the most conservative initial condition.
5. Should the reactor be just critical or operating at power at the time of a steam release, the reactor will be tripped by the normal overpower protection when 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 steam release before the no load conditions of reactor coolant system temperature and shutdown margin assumed in the analysis 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 no load condition at time zero. However, since the initial steam generator water inventory is greatest at no load, the magnitude and duration of the reactor coolant system cooldown are less for steam line release occurring at power.

6. In computing the steam flow, the Moody Curve (Reference 3) for $Fr/D = 0$ is used.
7. Perfect moisture separation in the steam generator is assumed.

B. Results

Figure 2.1.6-2 and 2.1.6-3 show the transient results for a steam flow of 247 lb/sec at 1100 psia from one steam generator.

The assumed steam release is typical of the capacity of any single steam dump, relief, or safety valve.

Safety injection is initiated automatically by low pressurizer pressure. Operation to two safety injection (SI) pumps are assumed. Boron solution at 2000 ppm enters the reactor coolant system providing sufficient negative reactivity to prevent core damage. The calculated 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 core transient occurs over a period of about 5 minutes, the neglected stored energy will have a significant affect in slowing the cooldown.

Following blowdown of the faulted steam generator, the plant can be brought to a stabilized hot standby condition through control of auxiliary feedwater flow and safety injection flow as described by plant operating procedures. The operating procedures would call for operator action to limit reactor coolant system pressure and pressurizer level by terminating safety injection flow and to control steam generator level and reactor coolant system coolant temperature using the auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following safety injection actuation.

2.1.6.2.3 Conclusions

The analysis shows that the criteria stated earlier in this section are satisfied. For an accidental depressurization of the main steam system, the minimum DNBR remains well above the limiting value and no system design limits are exceeded.

2.1.6.3 Steam System Piping Failure

2.1.6.3.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steamline would result in an initial increase in steam flow which decreases during the accident as the steam pressure decreases. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid delivered by the safety injection system.



The analysis of a main steamline rupture is performed to demonstrate that the following criteria are satisfied:

- A. Assuming a stuck RCCA, with or without offsite power, and assuming a single failure in the engineered safety features, the core remains in place and intact. Radiation doses do not exceed the guidelines of 10CFR100.
- B. Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that the DNB design basis is met for any rupture assuming the most reactive RCCA assembly stuck in its fully withdrawn position.

A major steamline rupture is classified as an ANS Condition IV event.

The rupture of a major steamline is the most limiting cooldown transient and, thus, is analyzed at zero power with no decay heat. Decay heat would retard the cooldown thereby reducing the return to power. A detailed analysis of this transient with the most limiting break size, a double ended rupture, is presented here.

2.1.6.3.2 Analysis of Effects and Consequences

A. Method of Analysis

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

1. The core heat flux and reactor coolant system temperature and pressure resulting from the cooldown following the steamline break. The LOFTRAN Code (Reference 1) has been used.
2. The thermal and hydraulic behavior of the core following a steamline break. A detailed thermal and hydraulic digital-computer code, THINC, has been used to determine if DNB occurs for the core conditions computed in Item (1) above.

The following conditions were assumed to exist at the time of a main steamline break accident:

1. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position: operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steamline break accident will not lead to a more adverse condition than the case analyzed.
2. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position: the variation of the coefficient with temperature and pressure has been included. The K_{eff} versus average coolant temperature at 1000 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 2.1.6-1. (The effect of power generation in the core on overall reactivity is shown in Figure 2.1.6-5.

The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sectors were conservatively combined to obtain average core properties for reactivity feedback calculations. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod. To verify the conservatism of this method, the reactivity, as well as the power distribution, was checked for the limiting conditions for the cases analyzed. The core analysis considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and nonuniform core inlet temperature effects. 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 conditions. These results verify conservatism: underprediction of negative reactivity feedback from power generation.

3. Minimum capability for injection of high concentration boric acid (2000 ppm) solution corresponding to the most restrictive single failure in the safety injection portion of the emergency core cooling system (ECCS). The ECCS consists of three systems: 1) the passive accumulators, 2) the residual heat removal (low head safety injection system), and 3) the safety injection system. Only the safety injection system and the passive accumulators are modeled for the steamline break accident analysis.

The modeling of the SI system in LOFTRAN is described in Reference 1. The flow corresponds to that delivered by two SI pumps delivering full flow to the cold leg header. No credit has been taken for the low concentration borated water, which must be swept from the lines downstream of the refueling water storage tank prior to the delivery of concentrated boric acid to the reactor coolant loops.

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 SI 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 SI system due to changes in the reactor coolant system pressure. The SI system flow calculation includes the line losses in the system as well as the SI pump head curve. ;

The boric acid solution from the safety injection system is assumed to be uniformly delivered to the three reactor coolant



loops. The boron in the loops is then delivered to the inlet plenum where the coolant (and boron) from each loop is mixed and delivered to the core. The stuck RCCA is conservatively assumed to be located in the core sector near the broken steam generator. Because the cold leg pressure is lowest in the broken loop due to larger loop flow and a larger loop pressure drop, more boron would actually be delivered to the core sector where the power is being generated, enhancing the effect of the boric acid on the transient. No credit was taken for this in the analysis. Furthermore, sensitivity studies have demonstrated that the transient is insensitive to boron worth or distribution.

For the cases where offsite power is assumed, the sequence of events in the SI system is the following. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the SI pumps starts. In 12 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing the low concentration borated water is swept into the core before the 2000 ppm borated water reaches the core. This delay, described above, is inherently included in the modeling.

In cases where offsite power is not available, a 10 second delay to start the standby diesel generators in addition to the time necessary to start the safety injection equipment (mentioned above) is included.

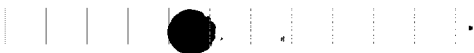
4. Design value of the steam generator heat transfer coefficient including allowance for fouling factor.
5. Since the steam generators are provided with integral flow restrictors with a 1.4 ft² throat area, any rupture with a break area greater than 1.4 ft², regardless of location, would

have the same effect on the nuclear steam supply system (NSSS) as the 1.4 ft² break. The following cases have been considered in determining the core power and reactor coolant system transients:

- a. Complete severance of the pipe, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available.
 - b. Case a with loss of offsite power simultaneous with the initiation of the safety injection signal. Loss of offsite power results in reactor coolant pump coastdown.
6. Power peaking factors corresponding to one stuck RCCA and non-uniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow, and thus, are different for each case studied.

The core parameters used for each of the two cases correspond to values determined from the respective transient analysis.

Both the cases above assume initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when 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



flow exists. The transient shown assumes an uncontrolled 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 steam flow coincident with low steamline pressure or low T_{avg} will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast acting isolation valves in the steamlines by HI and HI-HI containment pressure signals or by high steam flow coincident with low steamline pressure or low T_{avg} signals. Even with the failure of one valve, release is limited to no more than 10 seconds for the other steam generators while the one generator blows down. The steamline stop valves are designed to be fully closed in less than 5 seconds from receipt of a closure signal.

As shown in Figure 2.1.6-7, the core attains criticality with the RCCA's inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 2000 ppm enters the reactor coolant system. A peak core power less than the nominal full power value is attained.

Figures 2.1.6-8 through 2.1.6-10 show the response of the salient parameters for case (b), which corresponds to the case discussed above with additional loss of offsite power at the time the safety injection signal is generated. The safety injection system delay time includes 10 seconds to start the standby diesel generator and 12 seconds to start the safety injection pump and open the valves. Criticality is achieved later and the core power increase is slower than in the similar case with offsite power available. The ability of the emptying steam generator to extract heat from the reactor coolant system is reduced by the decreased flow in the reactor coolant system. The power transient shown in Figure 2.1.6-8 is conservative due to the underprediction of the feedback in the low flow condition. For the DNBR evaluation, a power feedback and power shape analysis consistent with the fluid conditions was used.

It should be noted that following a steamline break only one steam generator blows down completely. Thus, the remaining steam generators are still available for dissipation of decay heat after the initial transient is over. In the case of loss of offsite power this heat is removed to the atmosphere via the steamline safety valves.

Following blowdown of the faulted steam generator, the plant can be brought to a stabilized hot standby condition through control of the auxiliary feedwater flow and safety injection flow as described by plant operating procedures. The operating procedures would call for operator action to limit reactor coolant system pressure and pressurizer level by terminating safety injection flow and to control steam generator level and reactor coolant system coolant temperature using the auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following safety injection actuation.

Margin to Critical Heat Flux

A DNB analysis was performed for all of these cases. It was found that the DNB design basis was met for all cases.

2.1.6.3.3 Conclusions

The analysis has shown that the criteria stated earlier in the accidental depressurization of the secondary system section are satisfied.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive RCCA stuck in its fully withdrawn position.



no-load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the 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 insertion proceed in the same manner as in the analysis which assumes no load condition at time zero.

7. In computing the steam flow during a steamline break, the Moody Curve (Reference 3) for $FL/D = 0$ is used.
8. Perfect moisture separation in the steam generator is assumed.
9. Feedwater addition aggravates cooldown accidents like the steamline rupture. Therefore, the maximum feedwater flow is assumed. All the main and auxiliary feedwater pumps are assumed to be operating at full capacity when the rupture occurs, even though the plant is assumed to be in a hot standby condition.
10. The effect of heat transferred from thick metal in the pressurizer and reactor vessel upper head is not included in the cases analyzed. Studies previously performed have shown that the heat transferred to the coolant from these latent sources is a net benefit in DNB and reactor coolant system energy when the effect of the extra heat on reactivity and peak power is considered.

8. Results

Core Power and Reactor Coolant System Transient

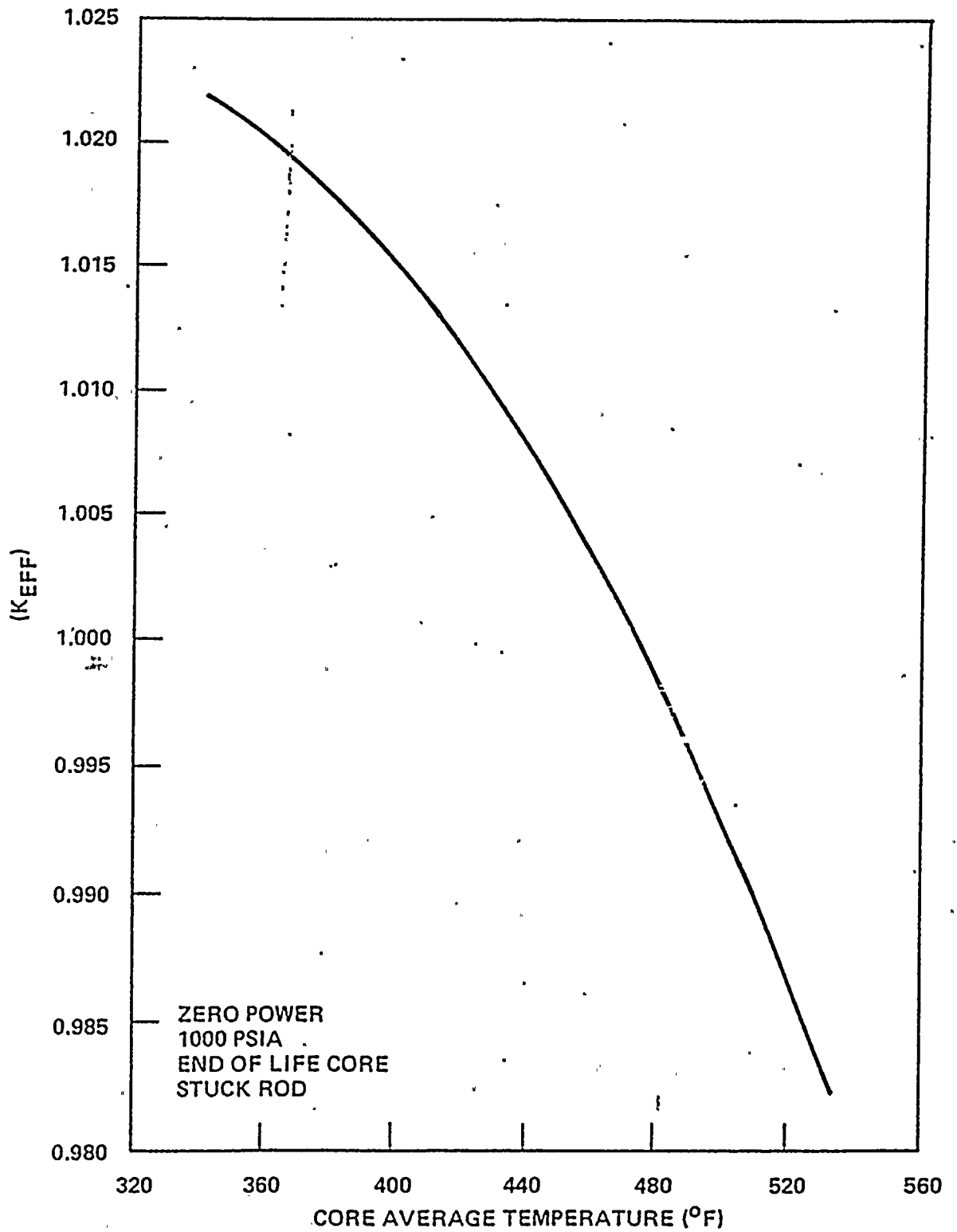
Figures 2.1.6-5 through 2.1.6-7 show the reactor coolant system transient and core heat flux following a main steamline rupture (complete severance of a pipe) at initial no-load conditions (case a). Offsite power is assumed available so that full reactor coolant



REFERENCES

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907, June 1972. Also supplementary information in letter from T. M. Anderson, NS-TMA-1802, May 26, 1978 and NS-TMA-1824, June 16, 1978.
2. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.
3. Moody, F. S., "Transactions of the ASME, Journal of Heat Transfer," Figure 3, Page 134, February 1965.



Figure 2.1.6-1 K_{EFF} Vs. Coolant Average Temperature



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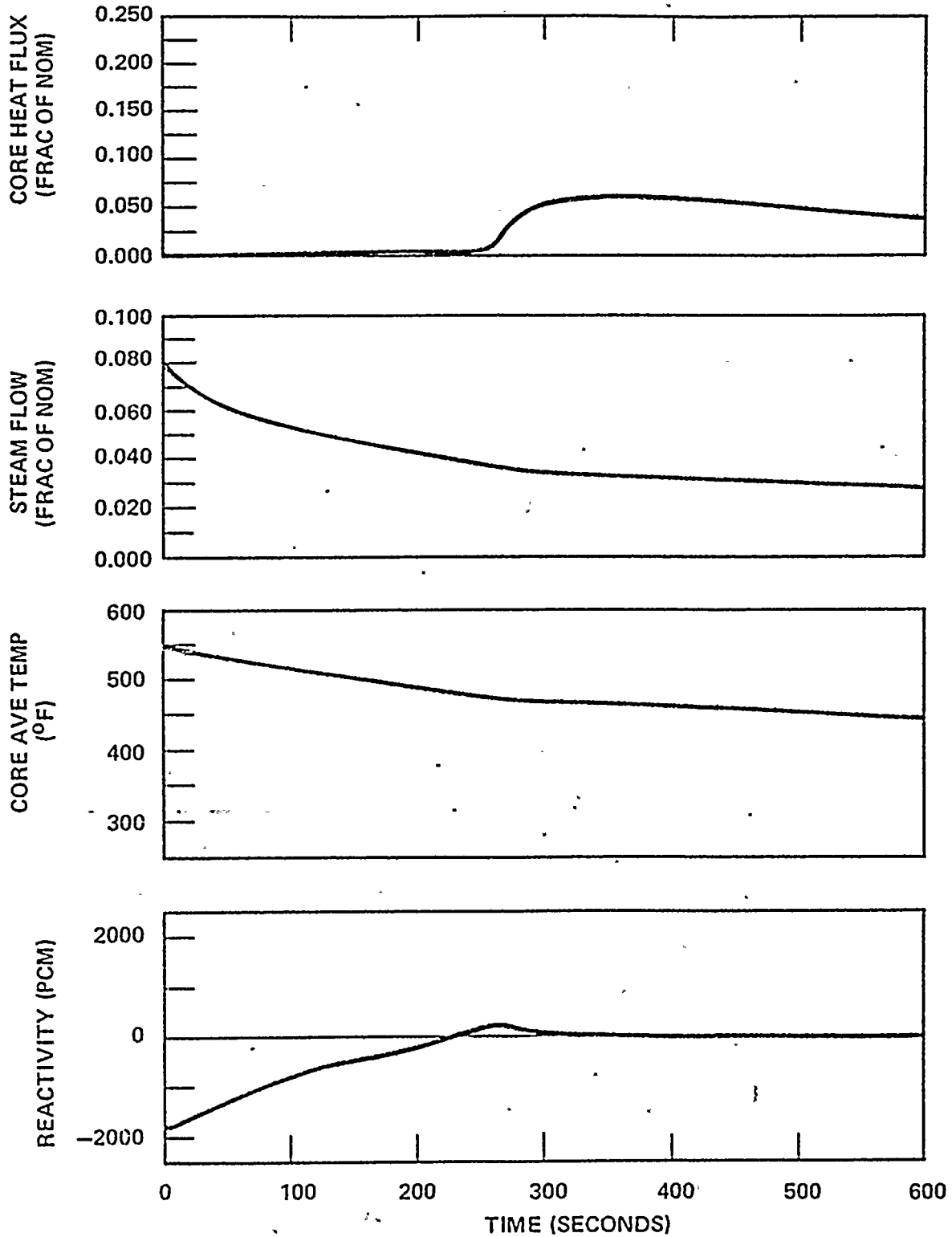


Figure 2.1.6-2 Failure of a Steam Generator Safety or Dump Valve-Heat Flux Vs. Time, Steam Flow Vs. Time, Average Temperature Vs. Time, Reactivity Vs. Time

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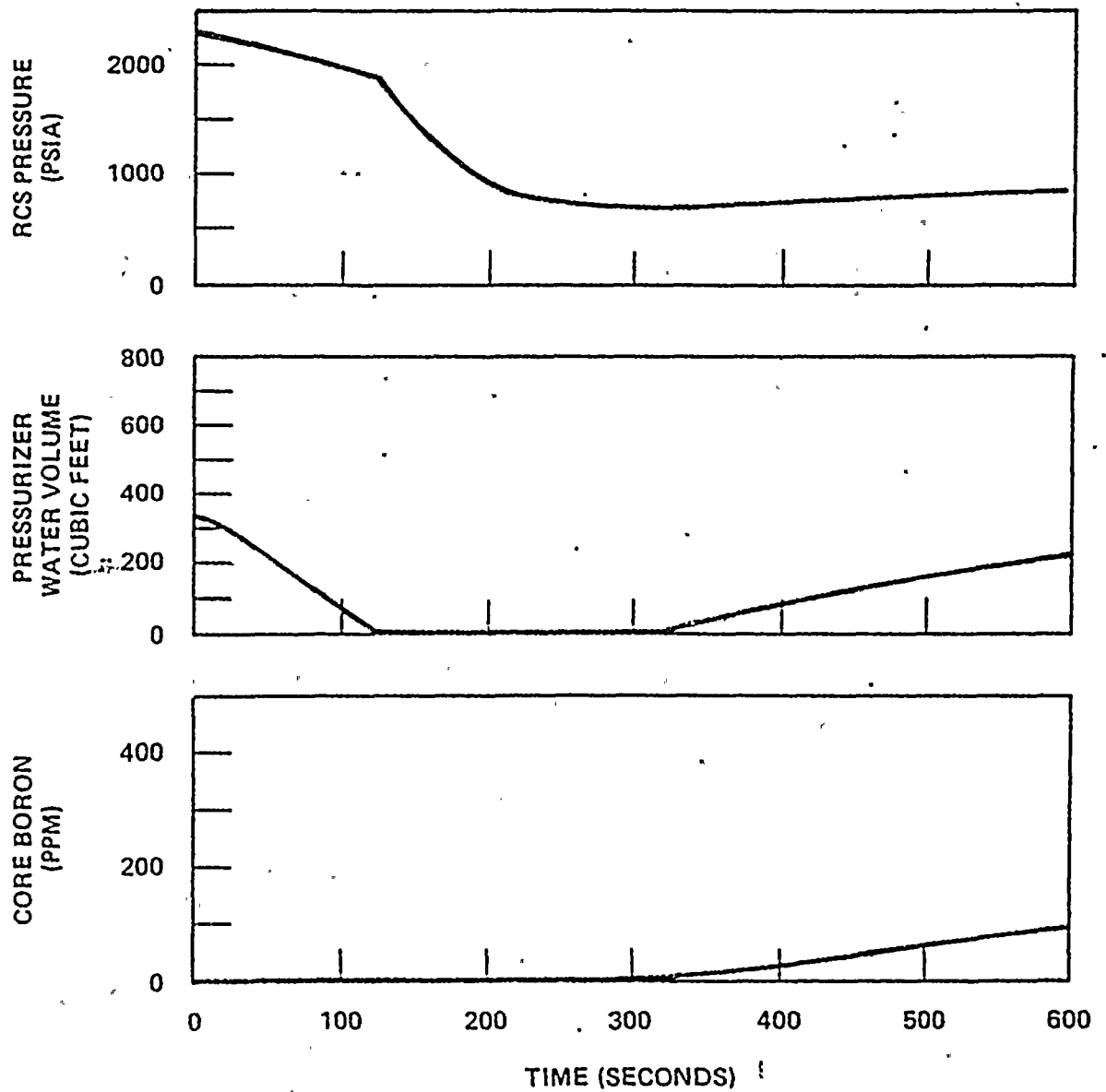


Figure 2.1.6-3 Failure of a Steam Generator Safety or Dump Valve — RCS Pressure vs. Time, Pressurizer Water Volume vs. Time, Boron Concentration vs. Time



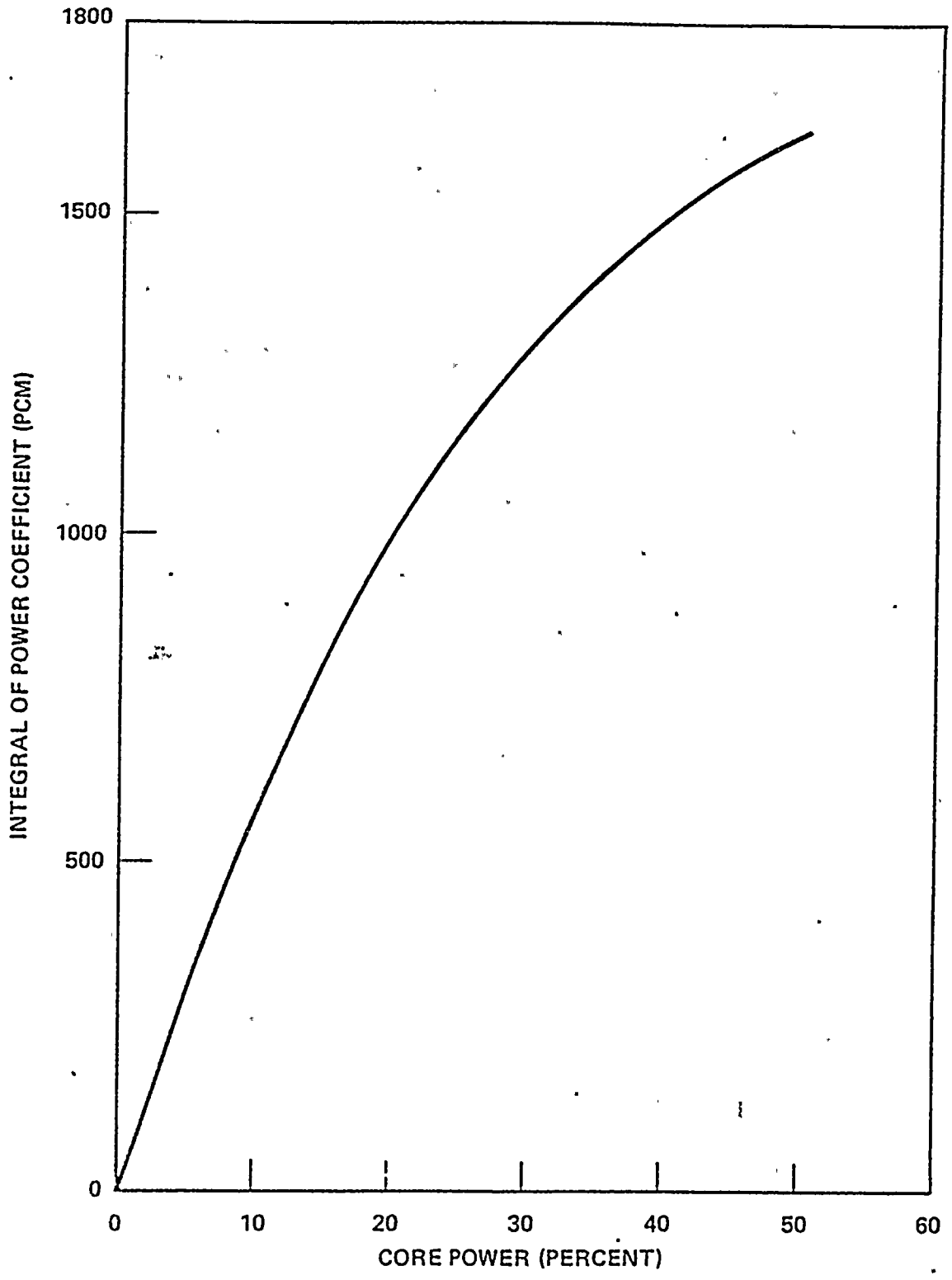


Figure 2.1.6-4 Doppler Power Feedback



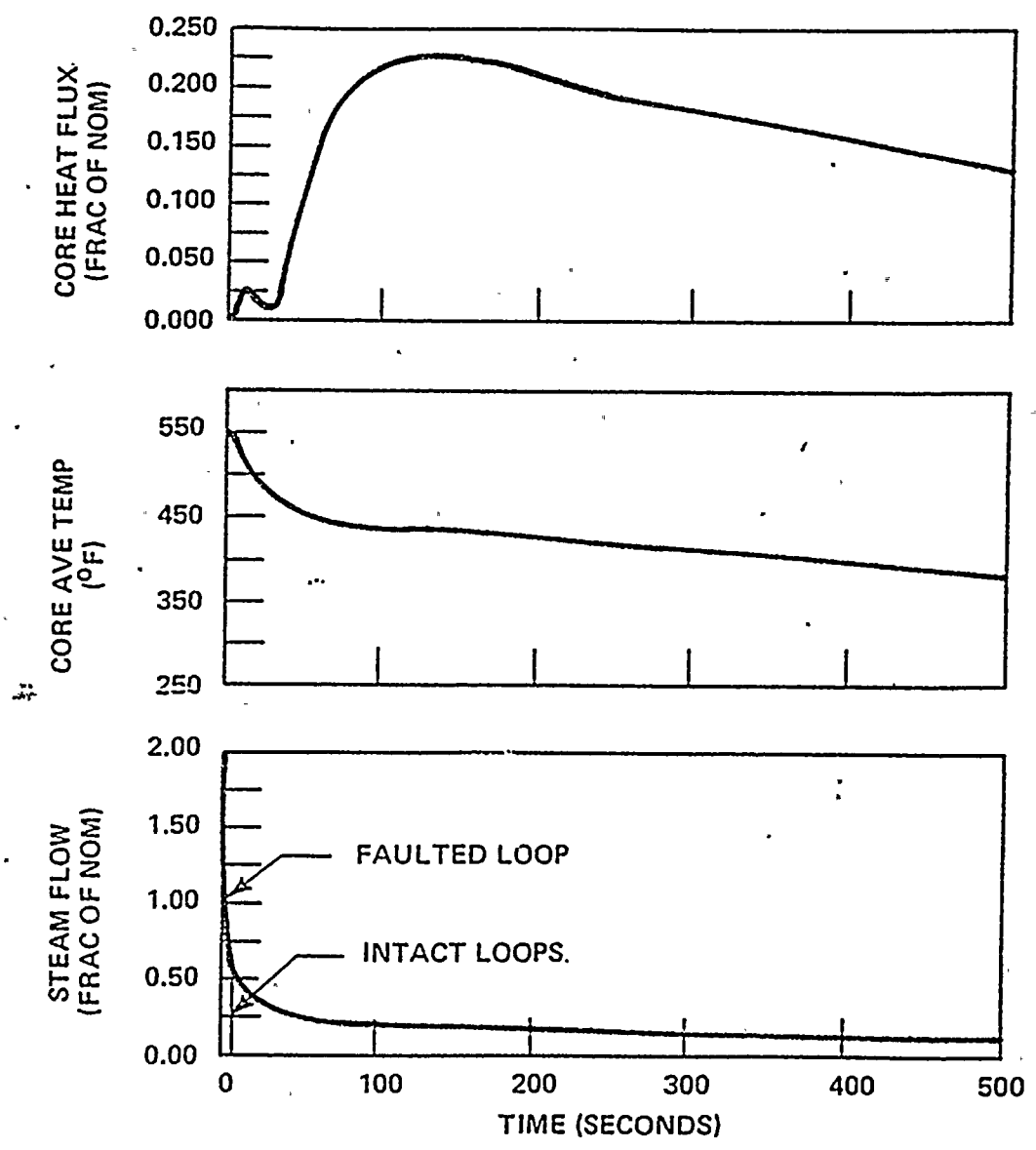


Figure 2.1.6-5 1.4 Ft² Steamline Rupture, Offsite Power Available - Heat Flux Vs. Time, Average Temperature Vs. Time, Steam Flow Per Loop Vs. Time



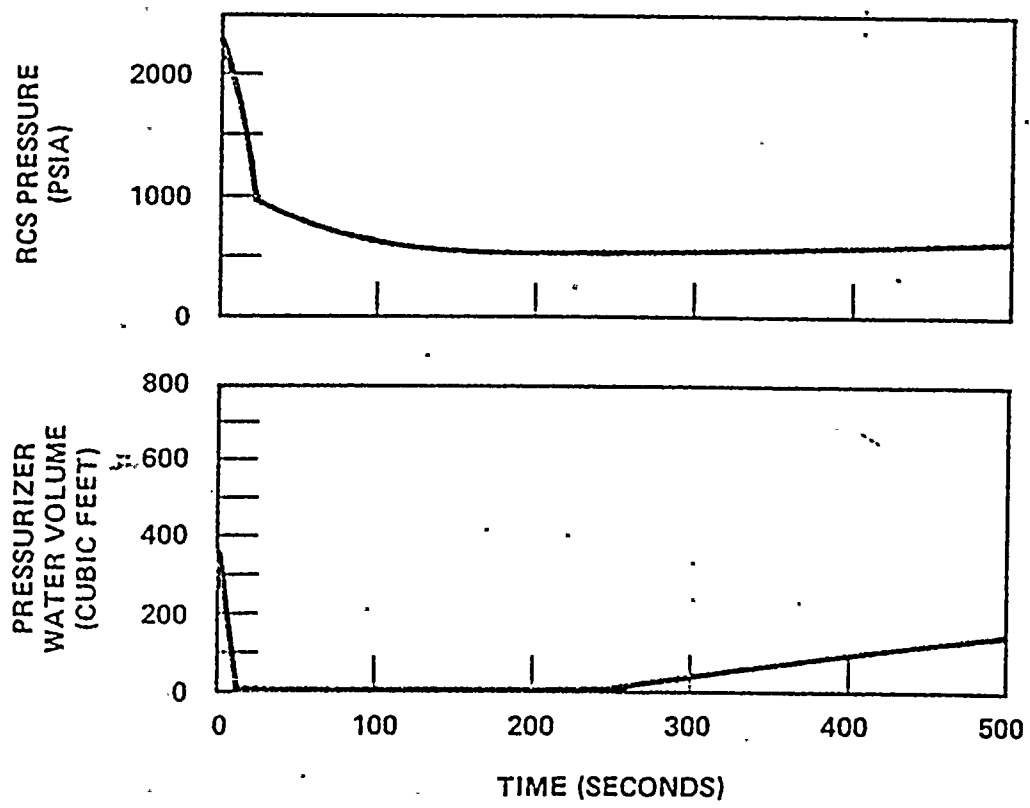


Figure 2.1.6-6 1.4 FT² Steamline Rupture, Offsite Power Available, RCS Pressure vs. Time, Pressurizer Water Volume vs. Time



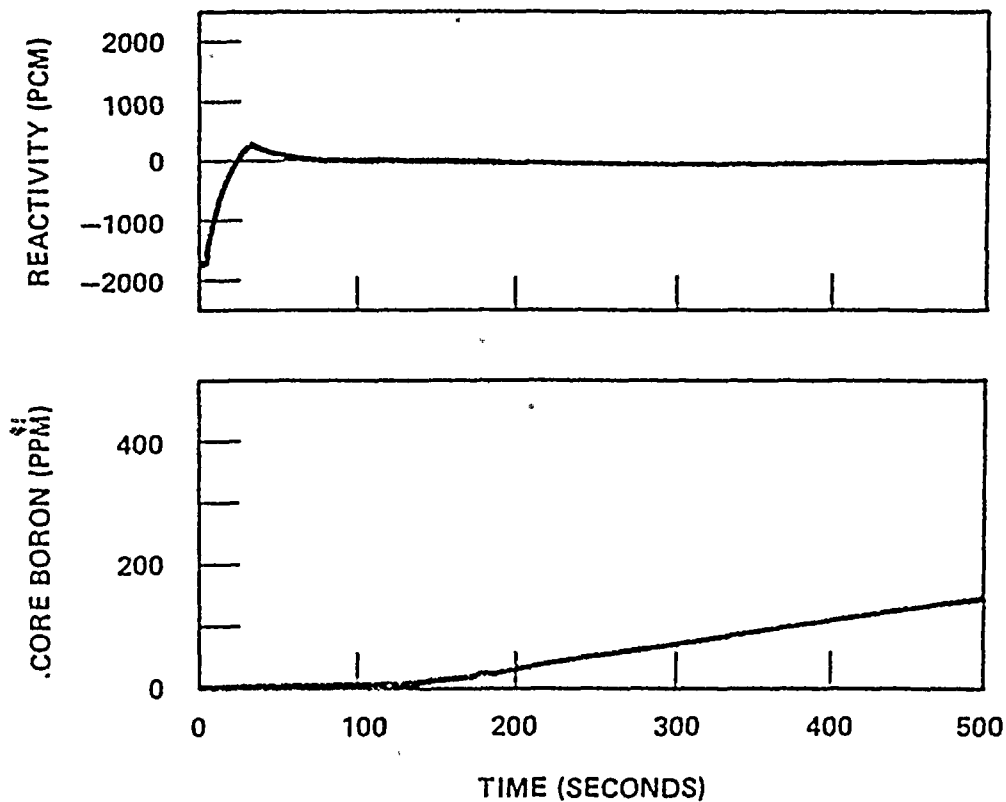


Figure 2.1.6-7 1.4 FT² Steamline Rupture, Offsite Power Available, Boron Concentration vs. Time, Reactivity vs. Time



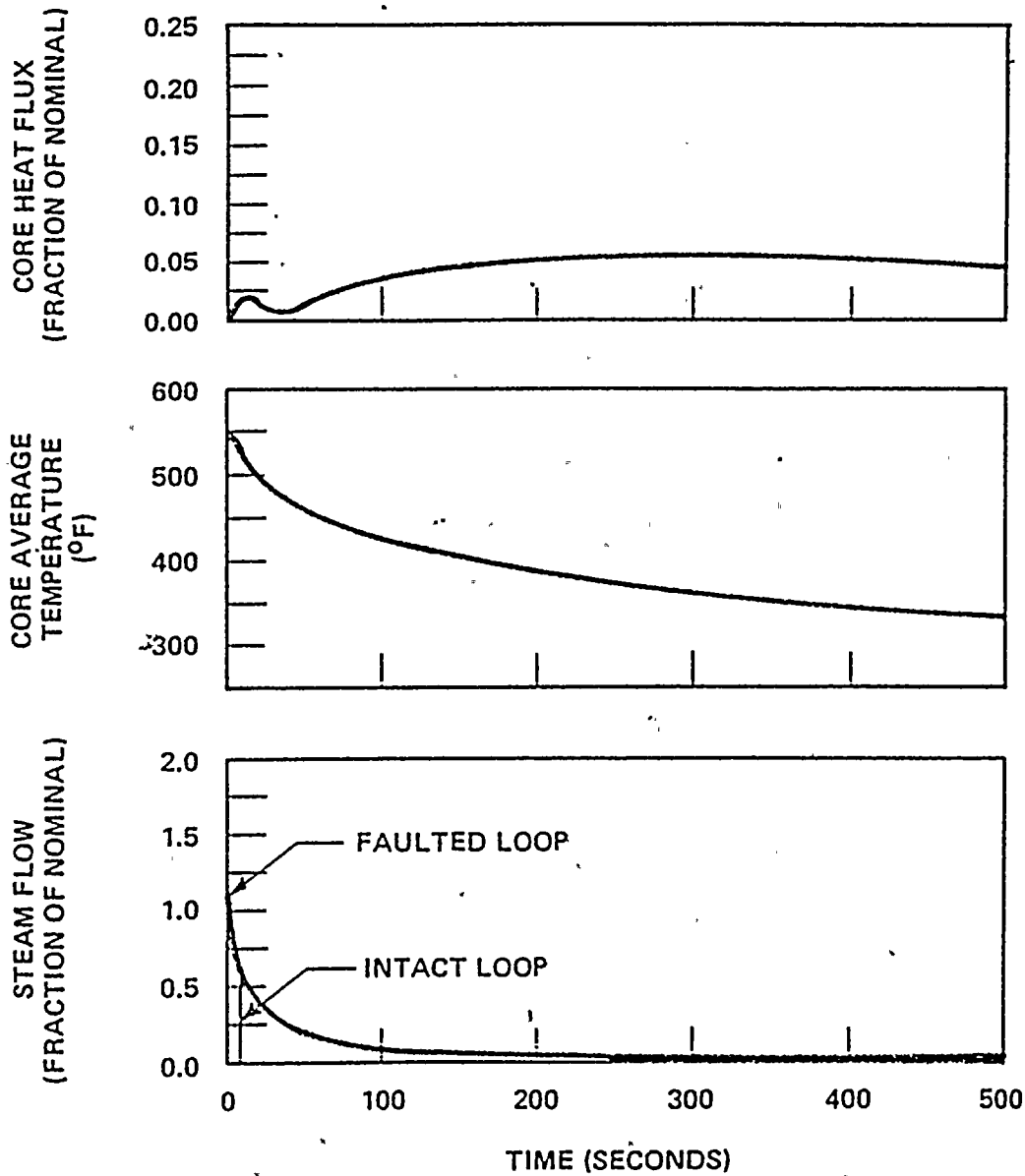


Figure 2.1.6-8 1.4 FT² Steamline Rupture, Offsite Power Not Available — Heat Flux vs. Time, Average Temperature vs. Time, Steam Flow Per Loop vs. Time



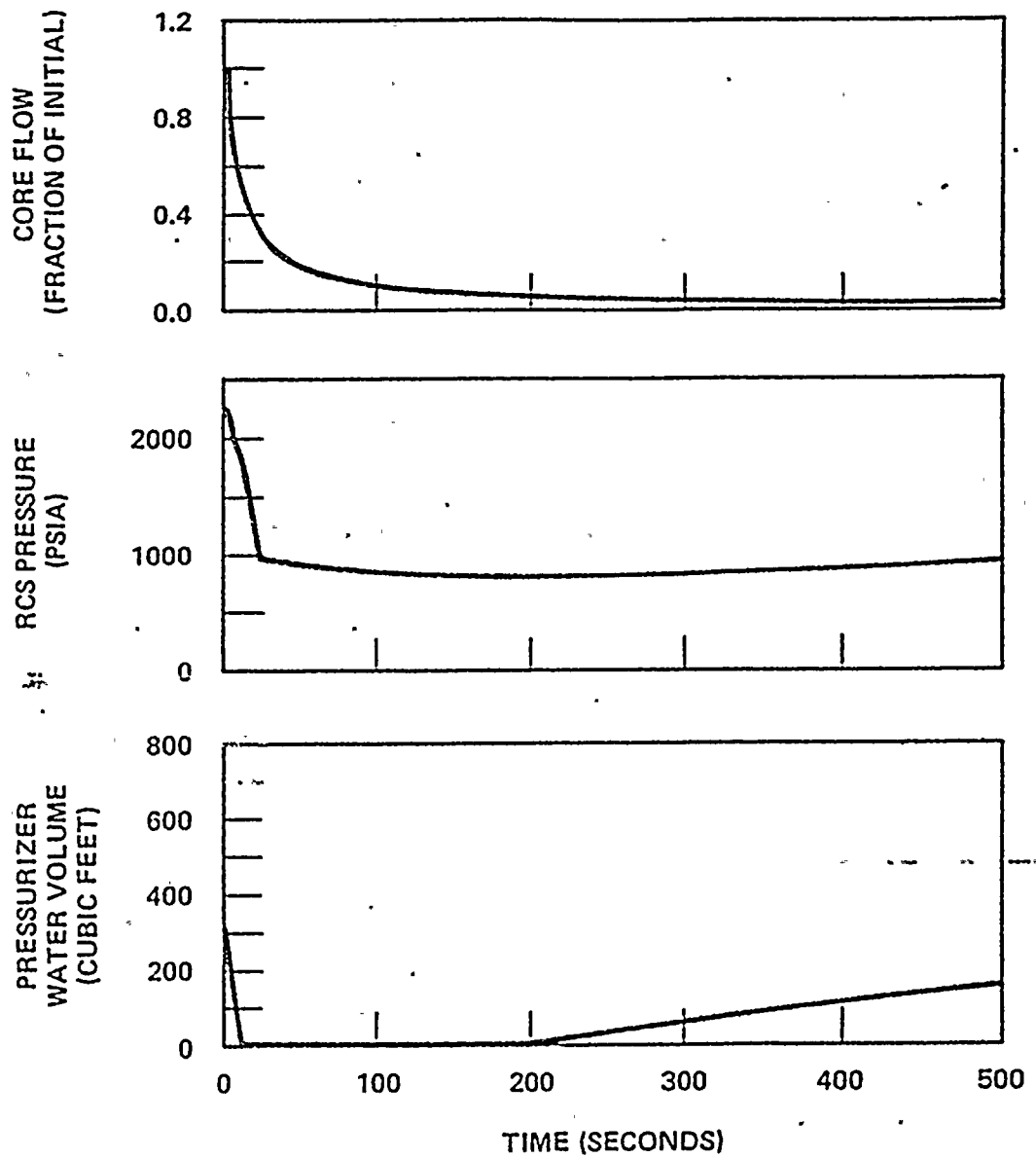


Figure 2.1.6-9 1.4 FT² Steamline Rupture, Offsite Power Not Available — Core Flow vs. Time, RCS Pressure vs. Time, Pressurizer Water Volume vs. Time



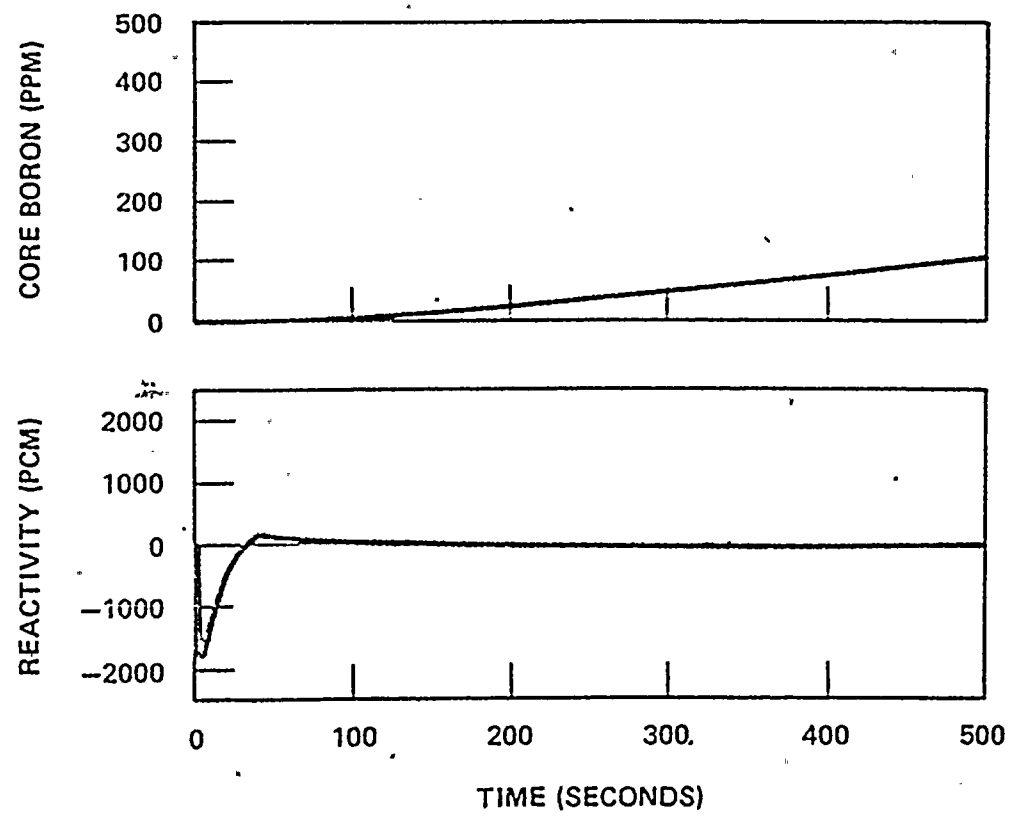
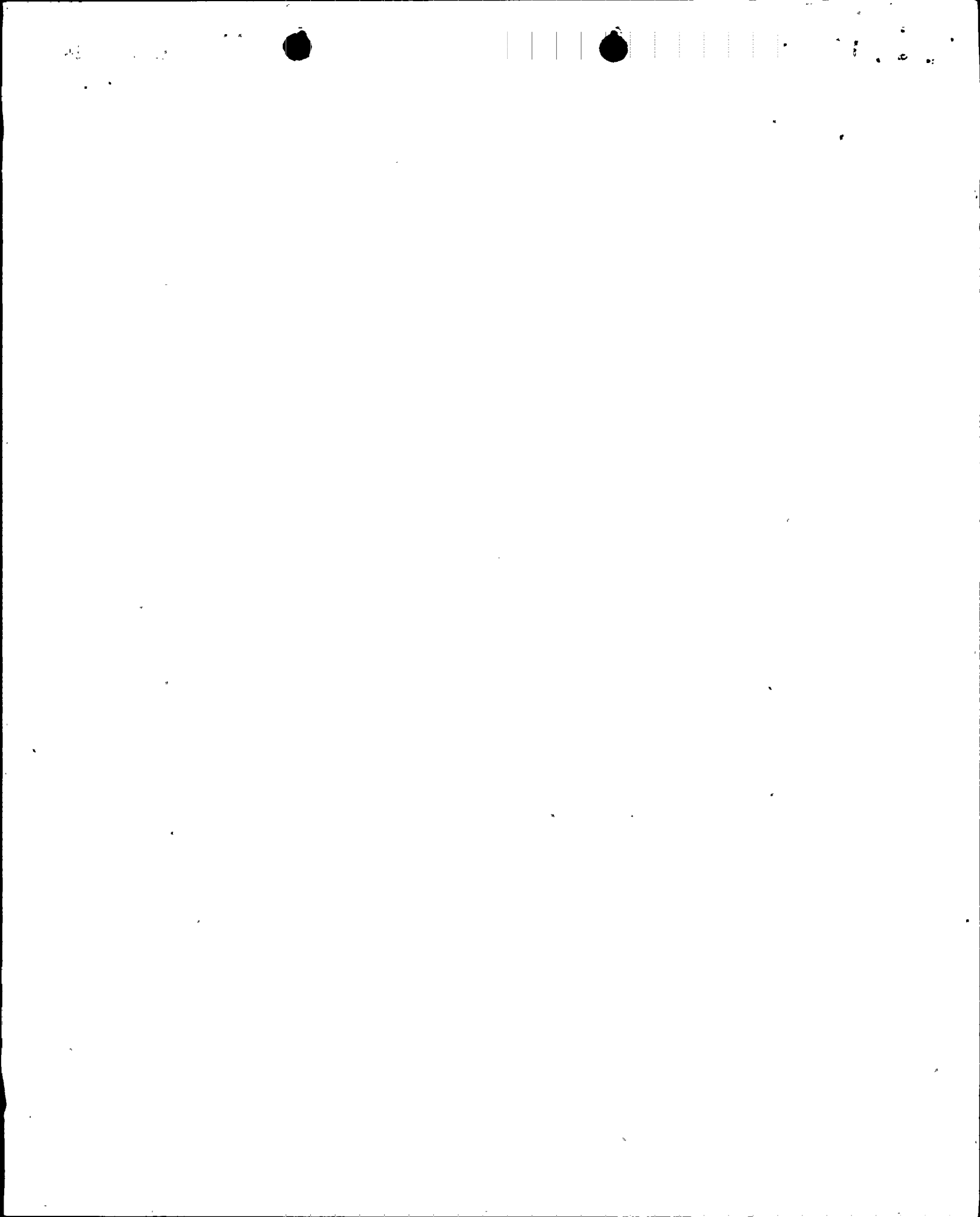


Figure 2.1.6-10 1.4 FT² Steamline Rupture, Offsite Power Not Available – Boron Concentration vs. Time, Reactivity vs. Time



CONTAINMENT PRESSURE RESPONSE TO STEAM LINE BREAK

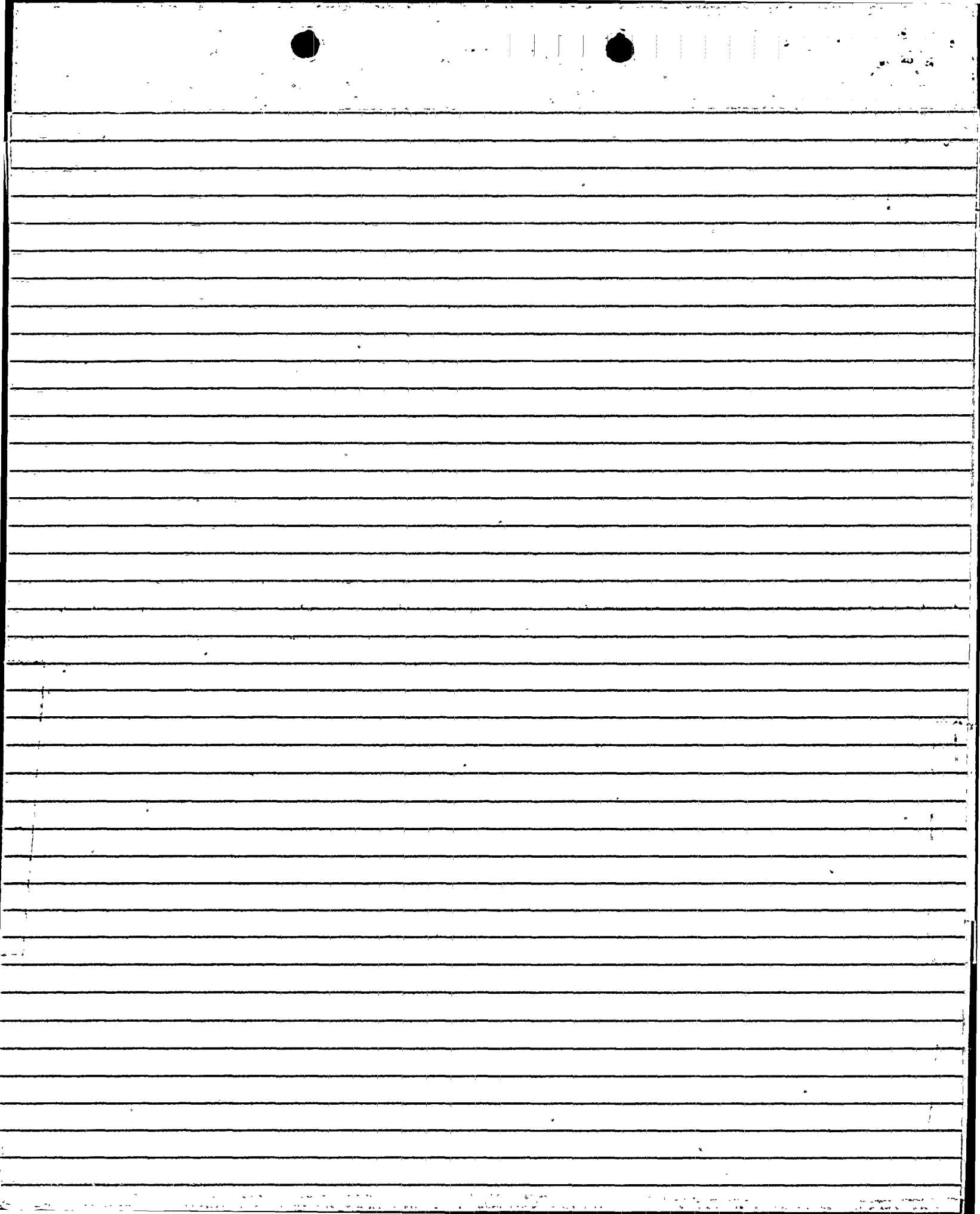
An analysis was performed to provide an estimate of the containment pressure response during a steamline break. Although this analysis did not include a full spectrum of break sizes, initial power levels, and single failures which would be performed for a full scope analysis, the results do provide a high degree of confidence that a steam line break would not cause the containment design pressure of 59 psig or the vessel test pressure of 65 psig to be exceeded. The analyses did specifically account for main feedwater flow and auxiliary feedwater flow.

The mass/energy release portion of the transient was calculated using the LOFRAN code. LOFRAN has been used for accident analyses in numerous safety analysis reports. The containment pressure and temperature transients are calculated using the COCO code. COCO has been used and found acceptable to calculate containment pressure transients for the H. B. Robinson and Zion plants.

Cases were analyzed at zero power and full power (2500 Mwt) to evaluate the sensitivity to initial power level. Conservatively high steam generator masses were assumed. A full double-ended break was analyzed assuming dry steam blowdown, i.e. no credit was taken for liquid entrainment in the mass/energy releases. Credit was taken for integral flow restrictors in the steam generator outlet nozzles. The assumption of dry steam in conjunction with a double ended break typically provides a pressure transient which bounds smaller breaks. No credit was taken for steamline check valves to prevent reverse flow from the intact steam generators. It was assumed there was no BIT in the Safety Injection System, resulting in a conservatively high return to power. Conservatively high main feedwater flow was assumed prior to feedline isolation. Analyses were run assuming various auxiliary feed flows. The 1200 gpm is well in excess of either the 800 gpm or the 1000 gpm which could be supplied by the existing auxiliary feed system. Credit was taken for operator action at 10 minutes to isolate auxiliary feed flow to the faulted steam generator.

For the containment pressure transient calculation, a conservatively low value for containment heat sinks was assumed. The containment atmosphere was conservatively assumed to reach a maximum of only 280°F for the purpose of calculating the heat removal capability of the fan cooler. The spray pumps were assumed to supply only 400 gpm. For most cases, failure of a spray pump and/or a fan cooler were assumed as the most limiting single failure.

In addition, a zero power steam break (which the sensitivity studies indicate is more limiting) with the assumptions listed above was performed. This analysis specifically considered 800 gpm auxiliary feedwater to the faulted steam generator, and operation of one containment spray pump and two fan coolers. The results of this analysis indicate a peak containment pressure of 56.1 psig.



W-PTP-32-URGENT
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Pittsburgh Pennsylvania 15230

Mr. C. O. Woody, Manager
Power Resources, Nuclear
Florida Power and Light Company
P.O. Box 529100
Miami, FL 33152

September 23, 1981

Dear Mr. Woody:

FLORIDA POWER AND LIGHT COMPANY
TURKEY POINT UNITS 3 AND 4
Upgrading Study
Steam Line Break Analysis

The following information is being furnished in response to requests made by Mr. Shepherd relative to the subject analysis.

The Westinghouse Steam Line Break Analysis prepared for Turkey Point Units 3 and 4 as part of the Upgrading Engineering Evaluation assumed 2000 ppm boron concentration. However, the Turkey Point Technical Specification allows 1950 ppm minimum boron concentration in the refueling water storage tank. The conclusions reached in the Westinghouse analysis therefore required re-evaluation.

Westinghouse has reviewed the Steam Line Break Analysis to assess the impact of reducing the minimum boron concentration by 50 ppm to 1950 ppm in the refueling water tank. The conclusion is that the results of the analysis remain valid. This is due to conservative values used for critical parameters and the resulting margins available.

With regard to the boron concentration in the BIT, accident analyses performed by Westinghouse in conjunction with BIT boron reduction assumes zero boron concentration for the in-line 900-gallon boron injection tank and lines leading to the primary piping injection point. Acceptable results were obtained for all accident conditions analyzed. Therefore, with steam generator steam nozzle flow restrictors, the Westinghouse analysis demonstrates adequacy for ranges of boron concentration in the BIT from 0 to 2000 ppm boron.

Should additional information or clarification be required, please contact the Westinghouse Project Office.

Very truly yours,

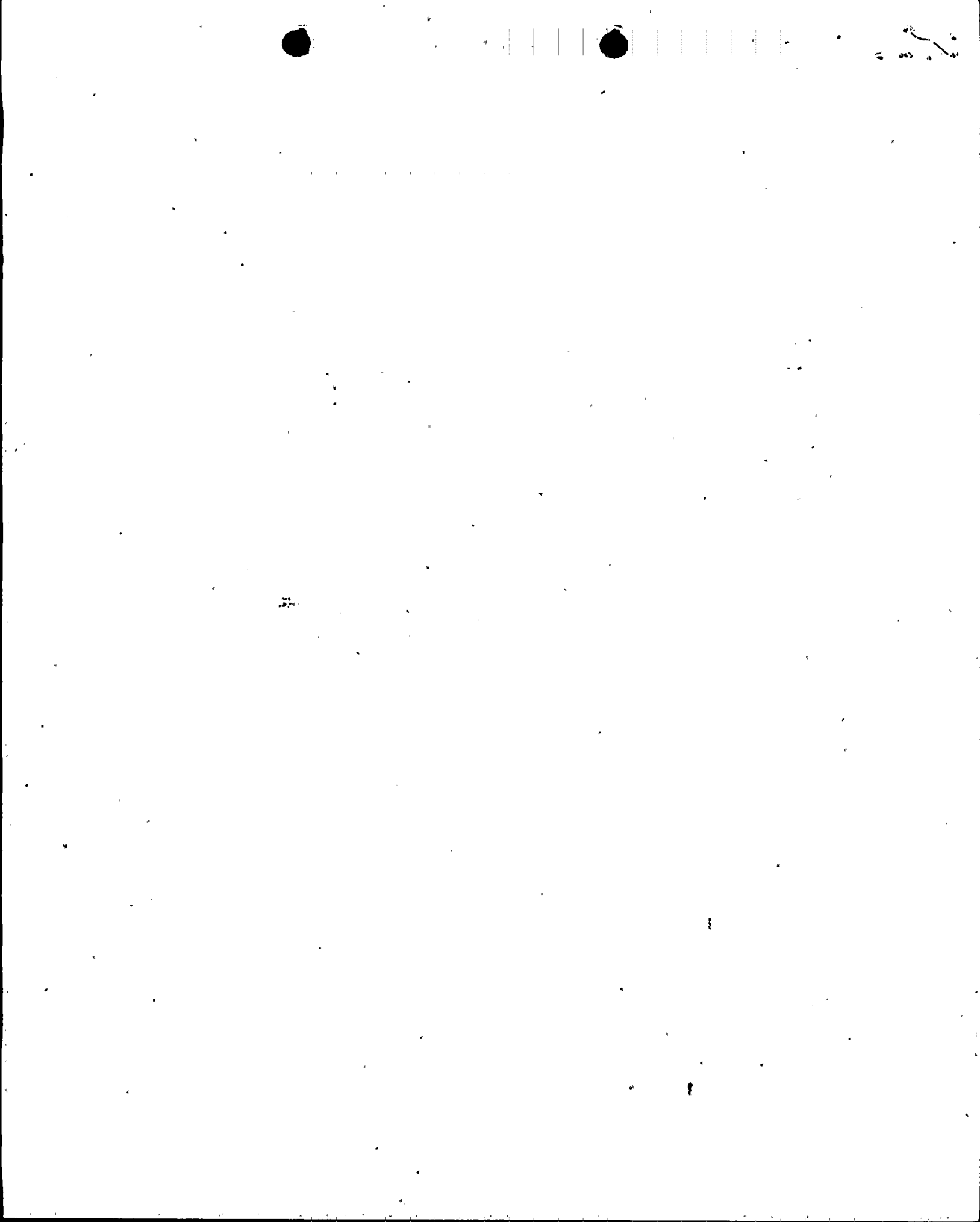
WESTINGHOUSE ELECTRIC CORPORATION

G. J. Murray, Project Engineer
Florida Power & Light Project

GJM:rst



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Electric Corporation

Water Reactor
Divisions

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Mr. C. O. Woody, Manager
Power Resources, Nuclear
Florida Power and Light Company
P.O. Box 529100
Miami, FL 33152

November 20, 1981

P.O. 93000-85525
G.O. MI-28622

Dear Mr. Woody:

**FLORIDA POWER AND LIGHT COMPANY
TURKEY POINT UNITS 3 AND 4
Reactor Vessel Thermal Shock
Effect of BIT Removal**

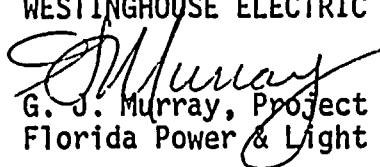
In a recent telecon, S. K. Mathavan, of Florida Power and Light, requested Westinghouse's assessment of the impact on reactor vessel thermal shock as a result of the functional removal of the Boron Injection Tank (BIT).

Westinghouse has reviewed this issue and determined that removal of the BIT will not affect the fracture mechanics analysis results for the Turkey Point Units 3 and 4 reactor vessel beltlines under postulated severe thermal shock transient events. Elimination of the present 155°F boron water injected early in an accident will not affect flaw initiation and crack arrest values which occur in the long-term portion of the applied transients. In fact, the latest Turkey Point analyses for the large loss-of-coolant accident event consider a step change in temperature from normal operating temperatures to a refueling water storage temperature of 39°F at time = 0.0 for analytical efficiency.

If you have any further questions on this subject, please do not hesitate to contact us.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION


G. J. Murray, Project Engineer
Florida Power & Light Project

GJM:rst

cc: C. O. Woody, FP&L
H. Paduano, FP&L
H. E. Yaeger, FP&L Turkey Point Site
J. K. Hays, FP&L Turkey Point Site
J. J. Quinn, W Turkey Point Site
E. V. Rutledge, W Miami Sales
R. L. Whitney, MNC-529

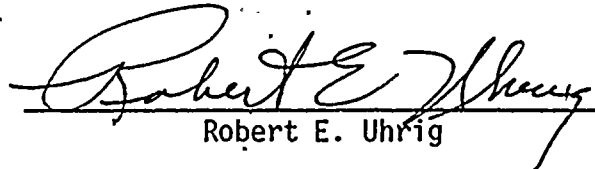


STATE OF FLORIDA)
)
COUNTY OF DADE) ss.

Robert E. Uhrig, being first duly sworn, deposes and says:

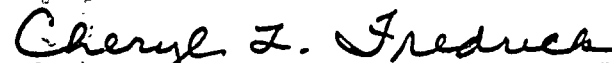
That he is Vice President of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this said document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said Licensee.


Robert E. Uhrig

Subscribed and sworn to before me this

30 day of November, 19 81


NOTARY PUBLIC, in and for the County of Dade,
State of Florida

My commission expires:
Notary Public, State of Florida at Largo
My Commission Expires October 30, 1983
Bonded thru Maynard Bonding Agency

