ATTACHMENT C

SAFETY ASSESSMENT

FOR THE

INDIAN POINT UNIT 2

TRANSITION TO WESTINGHOUSE 15x15 OFA FUEL

WITH EXTENDED BURNUP CAPABILITY

CONSOLIDATED EDISON COMPANY
INDIAN POINT UNIT 2
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1.0 INTRODUCTION AND SUMMARY

Indian Point Unit 2 has been operating with a Westinghouse 15x15 low-parasitic (LOPAR) fueled core. For Cycle 10 (expected to commence the summer of 1989) and subsequent cycles, it is planned to refuel with Westinghouse 15x15 optimized fuel assembly (OFA) regions. As a result, core loadings would range from approximately a 35% OFA-65% LOPAR mixed core to eventually an all OFA fueled core. The 15x15 OFA fuel has design features similar to 15x15 LOPAR fuel. The major design difference is the use of 7 middle Zircaloy grids for the OFA fuel versus 7 middle Inconel grids for LOPAR fuel (see Table 2.1). The end grids for both designs continue to be made of Inconel. The methodology used for the 15x15 OFA fuel design has been generically approved by the NRC via their review of the "Reference Core Report - 17x17 Optimized Fuel Assembly," WCAP-9500-A (Reference 1).

The OFA fuel to be used in Indian Point Unit 2 will incorporate some features of the Westinghouse VANTAGE 5 (Reference 2) fuel design, namely the reconstitutable top nozzle and rod and assembly modifications to allow extended burnup. It is also planned to remove thimble plugging devices from the core. Cycle 10 analyses utilized the Improved Thermal Design Procedure (ITDP), (Reference 3), the THINC-IV design code (References 4 and 5) and the WRB-1 DNB correlation (Reference 6) computer codes for DNB related accidents, as well as the PAD 3.4 fuel performance model (Reference 7).

Non-LOCA accidents which could potentially be affected by the OFA reload and ITDP methodology were reviewed. LOCA Hydraulic Forces (Reference 8) were analyzed using the MULTIFLEX (Reference 9) computer program. BASH (Reference 10) and BART (Reference 11) computer codes performed the Large Break LOCA analysis and the NOTRUMP (Reference 12) code performed the Small Break LOCA analysis.

This safety assessment is to serve as a reference safety evaluation/analysis report for the region-by-region reload transition from the present Indian Point Unit 2 LOPAR-fueled core to an all OFA-fueled core. This report examines the differences between the OFA and LOPAR Westinghouse fuel assembly

designs and evaluates the effect of these differences on the cores during the transition to an all OFA core. This safety assessment utilizes the standard reload design methods described in Reference 13 and will be used as a basic reference document in support of future Indian Point Unit 2 Reload Safety Evaluations (RSEs) for OFA reloads. The safety assessment also considers the Transition Core Effects described in Chapter 18 of Reference 1. Sections 2.0 through 5.0 of the safety assessment summarize the Mechanical, Nuclear, Thermal and Hydraulic, and Accident Evaluations respectively. Where applicable, LOPAR and OFA testing information is included in the evaluation sections. Enclosures 1 and 2 contain the non-LOCA and LOCA analyses respectively. Enclosure 3 contains the Significant Hazards Evaluation for the Technical Specification changes proposed in Attachment A..

The safety analyses/evaluations bound the following full power conditions: 2758 MWt core power, $F\Delta H$ of 1.62, F_Q of 2.32, 2250 psia primary system pressure, 567.7°F vessel average temperature, and 322,800 gpm RCS thermal design flow. These conditions are conservatively bounded by conditions used in Section 5.0 accident evaluation/analyses to justify safe operation with up to 25% plugging in any steam generator.

Consistent with the Westinghouse standard reload methodology (Reference 13) for analyzing reloads, parameters are chosen to maximize the applicability of these evaluations for future cycles. The objective of subsequent cycle specific RSEs will be to verify that applicable safety limits are satisfied based on the evaluation/analysis established by this safety assessment.

The first 15x15 OFA reloads started irradiation in 1983 with D. C. Cook Unit 1 followed by Turkey Point Unit 3 and Zion Unit 1 in 1984 and Indian Point Unit 3 in 1985. In 1988, twenty-seven plants operated with at least one region of OFA fuel, nine plants were operating with OFA regions in their third cycle of operation. Operational experience of OFA performance is given in Reference 14.

The results of evaluation/analyses and tests described herein lead to the following conclusions:

1. The Westinghouse OFA reload fuel assemblies for Indian Point Unit 2 are mechanically and hydraulically compatible with the current LOPAR fuel assemblies, control rods, and reactor internals interfaces. Both fuel assemblies satisfy the current design bases.

2. Changes in the nuclear characteristics due to the transition from LOPAR to OFA fuel will be within the range normally seen from cycle to cycle due to fuel management effects.

3. All or some of the thimble tube plugging devices may be removed from the core.

4. Plant operating limitations given in the Technical Specifications will be satisfied with the proposed changes noted in Attachment A. These are applicable to cores containing any combination of Westinghouse 15x15 OFAs and LOPAR assemblies, including a full OFA core. The plant can safely operate at conditions up to 2758 MWt core power with steam generator tube plugging levels up to 25% in any steam generator.

5. A reference is established upon which to base Westinghouse reload safety evaluations for future reloads.

2.0 MECHANICAL EVALUATION

2.1 Introduction

This section evaluates the mechanical design and compatibility of the extended burnup 15x15 OFA and core components with the current LOPAR assembly during the transition through mixed-fuel cores to an all OFA core. The OFAs have been designed to be compatible with the LOPAR assemblies, reactor internals interfaces, and fuel handling and refueling equipment. The OFA design dimension as shown in Figure 2.1 are essentially equivalent to the LOPAR design (Figure 2.2) from an exterior assembly envelope and reactor internals interface standpoint.

Based on the evaluation of the OFA/LOPAR design differences and hydraulic test results, it is concluded that the two designs are mechanically compatible with each other. The OFA and fuel rod design bases remain unchanged from that used for the previous reload core which utilized only LOPAR fuel assemblies. As such, compliance with the "Acceptance Criteria" of the Standard Review Plan (SRP, NUREG 0800) Section 4.2 Fuel System design was fully demonstrated.

The fuel has been designed according to the Westinghouse fuel performance model (Reference 7) and the clad flattening model (Reference 15). The fuel rod internal pressure design bases (Reference 16) are satisfied for all fuel regions.

Two mechanical features of the VANTAGE 5 design incorporated for the transition are the Reconstitutable Top Nozzle (RTN) and extended burnup capability. In addition all or some portion of the thimble plugging devices will be removed from the core.

Table 2.1 provides a comparison of the LOPAR and extended burnup OFA design parameters.

2.2 Mechanical Compatability of Fuel Assemblies

Table 2.1 presents a comparison of the OFA and LOPAR fuel assemblies, and Figures 2.1 and 2.2 shows a comparison of the LOPAR and OFA fuel assemblies respectively. A comparison between the OFA and LOPAR assembly designs shows a change in guide thimble and instrumentation tube diametral dimensions, a change from the seven middle LOPAR inconel grids to OFA Zircaloy grids, a change to the removable top nozzle (RTN) with modified holddown springs, a change to a low profile removable bottom nozzle, and the increased fuel rod and assembly length for extended burnup. Based on the evaluation of the design differences it is concluded that the two designs are mechanically compatible with each other.

2.2.1 Grid Assemblies

The top and bottom grids of the OFA assembly are the same as the Inconel grids of the LOPAR fuel assembly. The seven intermediate OFA Zircaloy grids have thicker and wider straps than LOPAR Inconel grids (see Table 2.1) to compensate for differences in material strength properties. Impact tests that have been performed to obtain the dynamic strength data verify that the Zircaloy grid strength at reactor operating conditions is structurally acceptable. Zircaloy grids maintain their integrity during applicable loading conditions.

2.2.2 Guide Thimble and Instrumentation Tubes

The 15x15 OFA assembly guide thimbles are similar in design to their counterparts in the LOPAR fuel assemblies except for an ID and OD reduction above the dashpot. The diameter reduction is due to a reduced grid cell. This results from use of thicker grid strap material for the Zircaloy grids in order to maintain grid strength as close to the original Inconel design as possible. The reduced cross-sectional area has been evaluated and determined to be acceptable for all duty load and postulated accident conditions. Below the dashpot the OFA and LOPAR assembly guide thimble dimensions are identical.

The OFA guide thimble tube ID continues to provide an adequate nominal diametral clearance for the control rods and other core components. However, due to the reduced annular clearance, the time for control rod insertion into the dashpot assumed for accident analyses is increased for the OFA assemblies. This difference was determined from conservative analytical calculations. The increase in rod drop time required accident reanalysis as described in Section 5.0

The OFA instrumentation tube also has the same diametral decrease compared to the LOPAR instrumentation tube. There is sufficient diametral clearance for the instrumentation thimble to traverse the tube.

2.2.3 Top Nozzle and Holddown Springs

The fuel assembly top nozzle for the OFA assembly differs from the current design in two ways: A groove is provided in each thimble thru-hole in the nozzle plate to facilitate removal, and the nozzle adapter plate is thinner than the LOPAR non-removable top nozzle. Additional details of this design feature, the design basis and the evaluation of the reconstitutable top nozzle are given in Section 2.3.2 in Reference 2.

The OFA top nozzle uses a 3-leaf holddown spring instead of the LOPAR assembly 4-leaf spring. The 3-leaf spring design has been successfully used in the 17x17 OFA design and other 15x15 LOPAR assemblies. Lift force holddown spring evaluations show that all spring criteria are met by the 3-leaf spring design.

2.2.4 Bottom Nozzle

The 15x15 OFA low profile bottom nozzle assembly is shorter when compared to the existing LOPAR assembly to allow for fuel rod growth. The difference in length is due to the OFA's nozzle plate being thinner and the bottom nozzle legs being shorter.

The OFA bottom nozzle retains the reconstitutable feature found on the LOPAR design. The design bases and evaluation of the low profile bottom nozzle are given in Section 2.3.1 in Reference 2.

2.2.5 Extended Burnup Capability

The extended burnup capability is achieved by using the RTN and low profile nozzle to allow more room for rod growth and a longer fuel rod plenum to accommodate fission gas release. The basis for designing to extended burnup is contained in Reference 17.

2.2.6 Fuel Rods

Nominal fuel rod parameters are shown in Table 2.1 for the OFA extended burnup and LOPAR designs. Both fuel rod designs retain the nominal pellet stack height of 144 inches; however, the OFA fuel rod length increases by 0.385 inches due to an increased gas plenum for additional fission gas releases to higher burnups. Several changes within the fuel rod shall be implemented on this OFA reload. These include the use of longer plenum springs and reduced length pellets.

2.3 Fuel Rod Performance

Fuel rod performance for all fuel rod designs is shown to satisfy the Standard Review Plan (SRP), (NUREG-0800), fuel rod design bases on a region by region basis. These bases are applicable to all fuel rod designs, including the Westinghouse LOPAR and Extended Burnup OFA fuel designs.

There is no effect from a fuel rod design standpoint due to having fuel with more than one type of geometry simultaneously residing in the core during the transition cycles. The mechanical fuel rod design evaluation for each region incorporates all appropriate design features of the region, including any changes to the fuel rod or pellet geometry from that of previous fuel regions.

Fuel performance evaluations are completed for each fuel region to demonstrate that the design criteria will be satisfied for all fuel rod types in the core under the planned operating conditions. Any changes from the plant operating conditions originally considered in the mechanical design of a fuel region (for example, a power uprating or an increase in the peaking factors) are addressed for all affected fuel regions. Fuel rod design evaluations are

performed using the NRC approved models in References 7 and 15, to demonstrate that the SRP fuel rod design criteria (including the rod internal pressure design basis in Reference 16) will be satisfied. The fuel is designed to operate so that clad flattening will not occur.

2.4 Rod Bow

The rod bow magnitude of the 15x15 OFAs is predicted to be less than that of the 15x15 LOPAR assemblies. The OFAs will have reduced grid forces (due to Zircaloy grid) and the same fuel tube thickness-to-diameter ratio (t/d) as the standard assembly, which should tend to decrease OFA rod bow compared to LOPAR fuel. For a given burnup, the magnitude of rod bow gap closure for the OFA is conservatively taken to be the same as that applied to the 15x15 LOPAR fuel assembly. This is consistent with the NRC approved method of scaling gap closure for different fuel assemblies described in Reference 18.

2.5 Fuel Rod Wear

The wear of a fuel rod is dependent on both the support provided by the assembly skeleton and the flow environment to which it is subjected. Of concern is the existence of crossflow caused by the difference in axial pressure distribution of the OFA and LOPAR fuel assemblies due to different grids. Hydraulic flow tests were performed to verify the compatibility of the two assemblies. The results of the tests show that no significant OFA or LOPAR fuel rod wear occurs due to the small amount of crossflow between fuel assemblies.

The above conclusions on rod integrity have also been supported by analytical results. The analysis accounted for rod vibrations caused by both axial and cross flows and the effect of potential fuel rod to grid gaps.

Both tests and analytical results therefore confirm that no fuel clad failures will occur due to fuel rod fretting.

2.6 Loading on Fuel Assemblies

2.6.1 Introduction

LOCA hydraulic forces analysis provides LOCA hydraulic forcing functions to verify the structural integrity of the core components for the proposed 15x15 OFA fuel reload and at a NSSS power level of 3083.4 Mwt (core power of 3071.4 Mwt), at a range of RCS fluid temperatures. The LOCA hydraulic forcing functions were generated for the Accumulator branch line in the cold leg and the Pressurizer Surge line in the hot leg. The LOCA hydraulic forces analysis takes advantage of the elimination of large primary pipe ruptures to reduce some of the expected increase in the magnitude of the peak forces which may occur due to the range of primary temperatures analyzed. In performing the LOCA hydraulic forcing function analysis it was assumed that leak before break technology has been approved as the design basis for Indian Point Unit 2 (See pending submittal Reference 8).

2.6.2 Method of Analysis

The method of analysis, to determine the LOCA hydraulic forcing functions for the 15x15 OFA reload lower RCS temperatures, considered the Accumulator branch line and the Pressurizer Surge line breaks. The computer codes used to evaluate the postulated LOCAs were MULTIFLEX 1.0, LATFORC, FORCE2, and THRUST. MULTIFLEX (Reference 9) is used to calculate the thermal hydraulics of the reactor coolant system due to a postulated LOCA. LATFORC uses the pressure distribution in the downcomer annulus region calculated by MULTIFLEX to determine the lateral hydraulic forcing functions on the reactor vessel, core barrel and the thermal shield. FORCE2 uses the pressure transient in the reactor vessel calculated by MULTIFLEX to calculate the vertical forces on the vessel internals and core components.

2.6.3 Results

Results of the LOCA hydraulic forcing functions have shown that eliminating large pipe ruptures and analyzing reactor coolant branch line breaks reduce the LOCA hydraulic forcing functions for the range of coolant temperatures at a NSSS power level up to 3083.4 Mwt. An evaluation has shown that the LOCA hydraulic forcing functions from a double-ended guillotine break used in some of the current structural integrity analyses (Reference 19, 20 and 23) are still more limiting than the branch line LOCA hydraulic forcing functions at the 3083.4 Mwt reduced temperature conditions. The LOCA hydraulic forcing functions generated for the Accumulator branch line and Pressurizer Surge line breaks have been used as input to determine the structural integrity of the core components.

An evaluation of fuel assembly structural integrity considering the lateral effects of a LOCA accident has been performed. A comparison of the maximum grid impact forces during the accident with experimental data obtained from grid impact tests at operating temperature results in adequate safety margins. Analyses of the transition and all OFA cores show that the Zircaloy and Inconel grids will not result in permanent set deformation and grid buckling due to the impact forces. The stresses in the OFA components resulting from LOCA induced deflections are within acceptable limits.

The structural characteristics of the OFA assembly are compatible with those of the LOPAR fuel assembly. The basic design layout and their boundary constraints in the core are identical. These two fuel assembly designs are essentially dynamically equivalent.

2.7 Thimble Plug Removal Evaluation

Coincident with implementation of the Indian Point Unit 2 OFA transition, Consolidated Edison plans to remove all or some portion of the thimble plugging devices from the core. This includes the removal of thimble plugs from the OFA assemblies, LOPAR assemblies, and all new core component clusters (burnable absorbers and sources).

Thimble plugging devices are currently utilized in the Indian Point Unit 2 core to limit the core bypass flow. All guide thimble tubes that are not under RCC locations or are not equipped with sources and burnable absorbers currently have thimble plugs inserted in them.

The mechanical design evaluation of the removal of thimble plugging devices addressed fuel rod fretting wear, control rod wear, seismic and LOCA loadings and reactor internals structural adequacy. Based on the assessment of the impact of the thimble plug removal on system and component structural adequacy and core plant safety, it is concluded that it is acceptable to remove these devices from the Indian Point Unit 2 core.

TABLE 2.1

Comparison of OFA with Extended Burnup and LOPAR Assembly Design

Parameter	15x15 OFA (Extended Burnup) Assembly Design	15x15 LOPAR Fuel Assembly Design
*Fuel Assembly Length, in.	159.975	159.710
*Fuel Rod Length, in.	152.235	151.85
Assembly Envelope, in.	8.424	8.424
Compatible with Core Internals	Yes	Yes
Fuel Rod Pitch, in.	0.563	0.563
Number of Fuel Rods/Assembly	204	204
Number of Guide Thimbles/Assembly	20	20
Number of Instrumentation Tube/Assembly	1	1
Compatible w/Movable In-Core Detector System	Yes	Yes
Fuel Tube Material	Zircaloy-4	Zircaloy-4
Fuel Rod Clad OD, in.	0.422	0.422
Fuel Rod Clad Thickness, in.	0.0243	0.0243
Fuel/Clad Gap, mil	7.5	7.5
Fuel Pellet dia. in.	0.3659	0.3659
Guide Thimble Material	Zircaloy-4	Zircaloy-4
*Guide Thimble OD, in.	0.533	0.546
Guide Thimble Wall Thickness, in	0.017	0.017
*Structural Material - Seven Intermediate Grids	Zircaloy-4	Inconel
Structural Material - Two End Grids	Inconel	Inconel
*Grid Inner Strap Thickness, mil	26 (Zircaloy Grid)	13.5 (Inconel Grid)
*Grid Outer Strap Thickness, mil	32 (Zircaloy Grid)	20.5 (Inconel Grid)

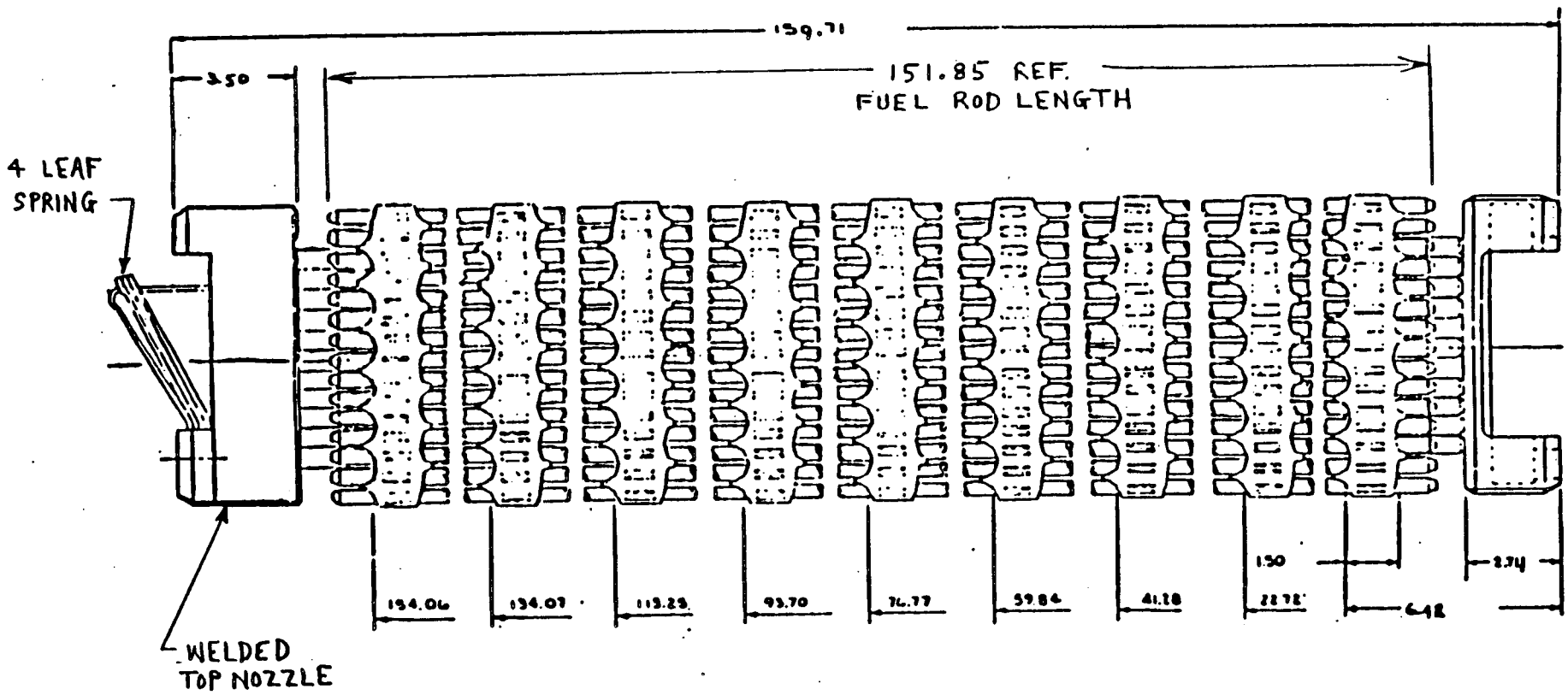
*Note: OFA design change compared to LOPAR fuel assembly.

TABLE 2.1 (Continued)

Comparison of OFA with Extended Burnup and LOPAR Assembly Design

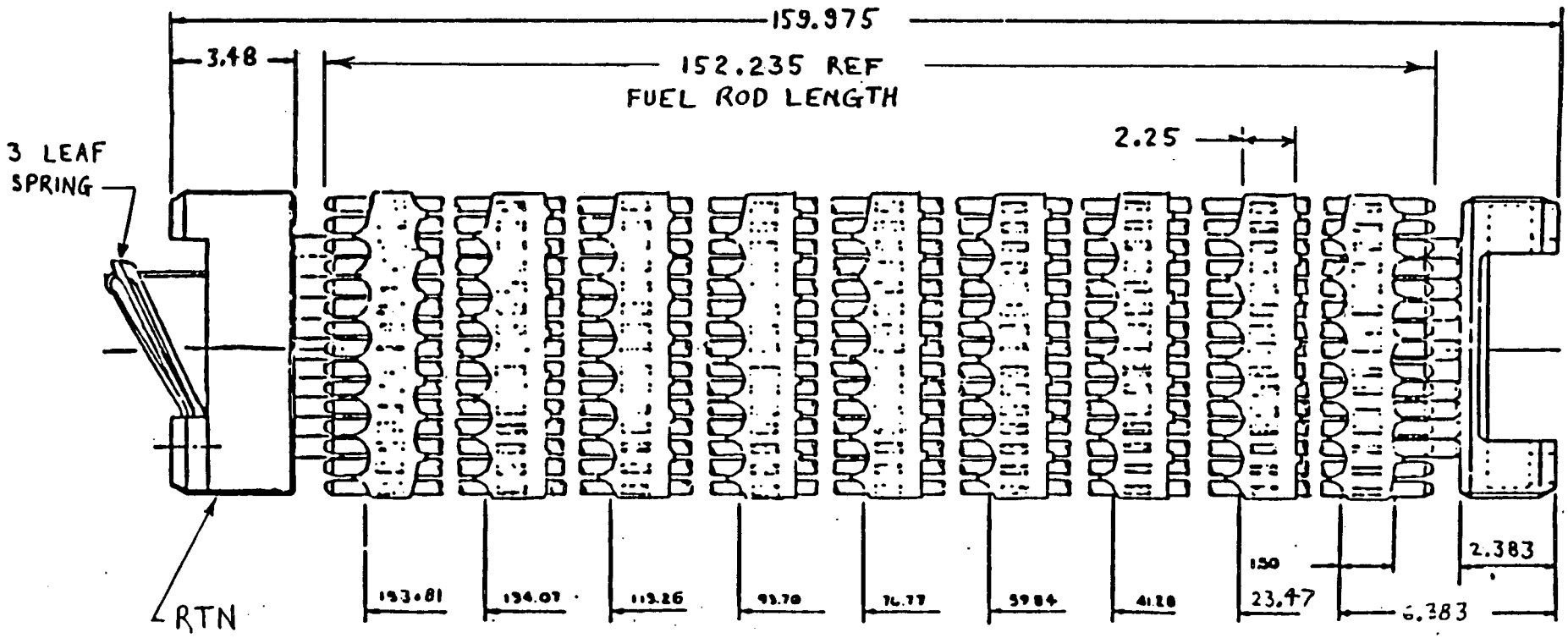
Parameter	15x15 OFA (Extended Burnup) Assembly Design	15x15 LOPAR Fuel Assembly Design
*Grid Height, inch Valley-to-Valley for Straps	2.25	1.50
*Instrumentation Tube ID, in.	.499	.512
*Top Nozzle Holddown Springs	3-leaf	4-leaf
*Top Nozzle	Reconstitutable	Welded
*Bottom Nozzle	Low Profile	Std. Height

*Note: OFA design change compared to LOPAR fuel assembly.



DIMENSIONS IN INCHES

Figure 2.1 LOPAR Fuel Assembly



DIMENSIONS IN INCHES

Figure 2.2 OFA Fuel Assembly

3.0 NUCLEAR EVALUATION

An evaluation to determine changes in core characteristics for cores of 1/3 OFA, 2/3 OFA and all OFA in comparison to the characteristics of all LOPAR cores has been made using the calculational methods described in Reference 13 with the addition of the Westinghouse Advanced Nodal Code (ANC) Reference 21. The four cores used in this evaluation are identical in fuel burnup, fuel enrichment, and fuel region arrangement. The only changes between the different cores is the type of fuel (OFA or LOPAR) in each region. This evaluation thus excludes the effects of fuel management which would mask the OFA effects.

This evaluation shows that overall physics characteristics for the all OFA, all LOPAR, and the transition cores are similar. The core which has physics characteristics most different from an all LOPAR core is the 1/3 OFA core. The results also show that when the core is all OFA fuel, its physics characteristics (with the exception of reactivity) are very similar to an all LOPAR core.

Further evaluations were done for the all OFA and 1/3 OFA cores in comparison to the all LOPAR core. The 2/3 OFA core was not included since its characteristics are bounded by the 1/3 OFA and all OFA cores. These results show that changes in physics behavior which will occur due to the transition from LOPAR to OFA will be within the range normally seen from cycle-to-cycle due to fuel management effects. The transition from LOPAR to OFA fuel will not result in changes from the current nuclear design bases given in the FSAR (Reference 22 and 23).

4.0 THERMAL AND HYDRAULIC EVALUATION

The Improved Thermal Design Procedure (ITDP) (Reference 3) and the THINC-IV (Reference 4 and 5) computer code are used for evaluation of both the standard and optimized fuel assemblies. The WRB-1 (Reference 6) DNB correlation is used in the 15x15 OFA analyses. The 15x15 LOPAR (STD) fuel analyses continue to use the W-3, L-grid (Reference 24) correlation.

The Departure from Nuclear Boiling Ratio (DNBR) correlation limits are 1.24 for the LOPAR fuel and 1.17 for the OFA. The thermal hydraulic design of this core is analyzed for operation at 3071.4 MWt core power which envelopes the current rated power of 2758 MWt. Thermal-Hydraulic Design Parameters are given in Table 4.1.

The design method employed to meet the DNB design basis is the Improved Thermal Design Procedure (ITDP) which has been approved by the NRC. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability with a 95 percent confidence level that the minimum DNBR will be greater than or equal to the correlation limit DNBR for the limiting power rod. Plant parameter uncertainties are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR limit, establishes a DNBR value which must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses for DNB related transients are performed using values of input parameters without uncertainties. For this application, the minimum required design DNBR values are 1.32 for OFA and 1.38 for LOPAR for thimble coldwall cells (three fuel rods and a thimble tube) and 1.33 for OFA and 1.44 for LOPAR for typical cell (four fuel rods).

In addition to the above considerations, specific plant DNBR margin has been considered in the analysis. In particular, the DNBR values of 1.47 and 1.52, for thimble and typical cells respectively, were employed in the safety analyses of the LOPAR fuel. A safety DNBR limit of 1.52 for both typical and thimble cells is used in design of the OFA. The DNBR margin between the DNBRs used in the safety analyses and the design DNBR values is broken down as

follows. A fraction of the margin is utilized to accommodate the transition core DNBR penalty (3.5%) for the OFA and the appropriate fuel rod bow DNBR penalty (based on the methodology described in References 18, 25-27) for both fuel types which is less than 1.0%. The existing margin between the design and safety analysis DNBR limits also includes DNBR margin reserved for flexibility in the design.

The LOPAR and OFA designs have been tested and shown to be hydraulically compatible.

The phenomenon of fuel rod bowing, as described in Reference 18, must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. The safety analysis of Indian Point 2 maintains sufficient margin between the safety analysis limit DNBRs and the design limit DNBRs to accommodate full-flow and low-flow DNBR penalties identified in Reference 25, which are applicable to 15x15 OFA analysis utilizing the WRB-1 DNB correlation, and 15x15 LOPAR fuel using the W-3 L-grid correlation.

The transition core DNB methodology given in Reference 1 has been approved by the USNRC. Using this methodology, transition cores are analyzed as if they were full-core OFAs, applying the 3.5% DNBR transition core penalty. This penalty is included in the safety analysis limit DNBR such that sufficient margin over the design limit DNBR will exist to accommodate the transition core DNBR penalty and the appropriate rod bow DNBR penalty.

The main effect of thimble plug removal is the increase in core bypass flow. This increase has been incorporated into the non-LOCA and LOCA safety analyses that have been performed in support of implementation of the 15x15 OFA transition. (See Sections 5.0 and Enclosures 1 and 2).

The fuel temperatures for use in safety analysis calculations for the OFA fuel are not significantly different from those for the LOPAR fuel. These small differences do not adversely affect the safety analysis calculations which are performed using the Westinghouse fuel performance code (Reference 7). Based upon the above discussion, the transition from LOPAR to OFA fuel will continue to meet currently licensed core thermal design safety criteria.

TABLE 4.1
Thermal and Hydraulic Design Parameters

ITDP Parameters		Design Value
Reactor Core Heat Output, Mwt		3071.4
	10^6 BTU/hr	10,483
Heat Generated in Fuel, %		97.4
Core Pressure, psia		2280
Radial Power Distribution		
	LOPAR	1.56[1+0.3(1-P)]
	OFA	1.59[1+0.3(1-P)]
Minimum DNBR at Nominal Conditions		
	Typical Flow Channel	2.41 LOPAR
		2.45 OFA
	Thimble Flow Channel	2.13 LOPAR
		2.33 OFA
Design Limit DNBR	Typical Flow Channel	1.44 LOPAR
		1.33 OFA
	Thimble Flow Channel	1.38 LOPAR
		1.32 OFA
Safety DNBR for Design		
	Typical Flow Channel	1.52 LOPAR
		1.52 OFA
	Thimble Flow Channel	1.47 LOPAR
		1.52 OFA
DNB Correlation	LOPAR	W-3, L-Grid
	OFA	WRB-1
HFP Nominal Coolant Conditions		
Vessel Minimum Measured Flow Rate (MMF) (Including Bypass),	10^6 lbm/hr	124.38
	GPM	330,000
Vessel Thermal Design Flow Rate (TDF) (Including Bypass)	10^6 lbm/hr	121.72
	GPM	322,800
Nominal Vessel/Core Inlet Temp, °F		
	based on TDF	547.7
	based on MMF	548.3
Vessel Average Temp, °F		579.7
Vessel Outlet Temp, °F		
	based on TDF	611.7
	based on MMF	611.1

HFP Nominal Coolant Conditions		Design Value
Average Temperature Rise in Vessel, °F		
	based on TDF	64.0
	based on MMF	62.8
Heat Transfer		
Active Heat Transfer Surface Area, ft^2		52,100
Average Heat Flux,	BTU/hr-ft^2	196,000
Average Linear Power,	kw/ft	6.35

5.0 ACCIDENT EVALUATION

5.1 NON-LOCA ACCIDENTS

5.1.1 Introduction and Summary

This section summarizes the effects of the complete transition of Indian Point Unit 2 from Westinghouse 15x15 LOPAR fuel to Westinghouse 15x15 OFA fuel on the FSAR Chapter 14 Non-LOCA Accident Analyses. The methods used for accident evaluation are described in Reference 13 and are the same as those applied to previous Indian Point Unit 2 reloads. To be consistent with the current Indian Point Unit 2 licensing basis accident analyses, the analyses and evaluations performed for the complete transition to OFA also incorporate the most bounding effects of 25% uniform steam generator tube plugging.

The Indian Point Unit 2 licensing basis as reported in the FSAR (Reference 22) includes analyses or evaluations of thirteen (13) Non-LOCA accidents. These accidents are:

a. Uncontrolled Control Rod Withdrawal from a Subcritical Condition

b. Uncontrolled Rod Cluster Control Assembly Withdrawal at Power

c. Rod Cluster Control Assembly Drop

d. Chemical and Volume Control System Malfunction

e. Loss of Reactor Coolant Flow

f. Startup of an Inactive Reactor Coolant Loop

g. Loss of External Electrical Load

h. Loss of Normal Feedwater

i. Reduction in Feedwater Enthalpy Incident

j. Excessive Load Increase Incident

k. Loss of All AC Power to the Station Auxiliaries

l. Rupture of a Steam Pipe

m. Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

All of the above Non-LOCA accidents which could potentially be affected by the OFA reload have been reviewed and include consideration for the following design and/or Technical Specification changes:

1. The control rod scram time to the dashpot is increased (as discussed in Section 2.2.2) from 1.8 seconds to 2.4 seconds. This increased drop time primarily affects the fast reactivity transients.

2. For events reanalyzed, the analyses were conservatively performed assuming a nominal thermal power level of 3071.4 MWt with a 579.7°F vessel average temperature and thermal design flow rate of 322,800 gpm. These will bound Cycle 10 operation at the current nominal thermal power of 2758 MWt. This change has the largest affect on transients that are limiting at full power conditions. Fuel temperature based on the revised PAD computer code fuel thermal safety model (Reference 28) were also included for events reanalyzed.

3. As discussed in Section 4.0, the OFA transition includes implementation of the Improved Thermal Design Procedure (ITDP) using both the WRB-1 and W-3 L-grid DNB correlations for OFA and LOPAR fuels respectively, an increase in the nuclear enthalpy rise hot channel factor $F^N_{\Delta H}$

limit from 1.55 to 1.62 and removal of all or some portion of the thimble tube plugging devices. A conservative set of core thermal safety limits, overtemperature delta-T and overpower delta-T setpoints were generated that are applicable to the above conditions and are valid for both transition and full OFA cores.

4. A change in the shutdown margin during refueling to reflect a Keff = 0.95 was assumed for the OFA transition and full OFA cores.

The results of the FSAR Chapter 14 Non-LOCA accident reanalyses and evaluations are contained in Enclosure 1. Based on the plant operating limitations given in the Technical Specifications and the proposed Technical Specification changes described in Attachments A & B to this application, the results show that the transition from 15x15 LOPAR to 15x15 OFA fuel, including the aforementioned design changes, can be accommodated with margin to the applicable FSAR safety limits. Therefore, it is concluded that Indian Point Unit 2 can be safely operated with cores containing any combination of Westinghouse 15x15 OFA and LOPAR assemblies.

5.2 LOCA ACCIDENTS

5.2.1 Large Break LOCA

5.2.1.1 Description of Analysis Assumptions for 15x15 OFA Fuel

The large break Loss-of-Coolant (LOCA) analysis for Indian Point Unit 2 applicable to a full core of Optimized Fuel Assembly (OFA) core was performed using the NRC approved 1981 Evaluation Model with BART/BASH (Reference 10) for a NSSS power level of 3083.4 Mwt. The analysis was performed for a spectrum of Moody discharge coefficients (0.4, 0.6 and 0.8) based on a limiting double-ended guillotine break of the RCS cold leg. The analysis was also performed to allow a range of reactor vessel temperatures (vessel average temperature between 549°F and 579.7°F) to bound anticipated future changes in this operating parameter. Other pertinent analysis assumptions include: increase NSSS power of 3083.4 MWt, 25% uniform steam generator tube plugging, 90% of Thermal Design Flow, a hot channel enthalpy rise factor of 1.62 and thimble plug removal.

For the 15x15 fuel array, the fuel rod diameter is the same for the OFA and LOPAR fuel rods. For this reason, the primary difference between LOPAR and OFA fuel, in terms of the large break LOCA analysis, is in the hydraulic effects of differences in size and material in the fuel assembly grids.

The large break LOCA analysis performed and reported in Enclosure 2 assumed a full core of OFAs. The OFA design is hydraulically similar to the LOPAR design which it replaces, demonstrating only small differences (<5%) in total assembly hydraulic resistance.

When assessing the effect of transition cores on the large break LOCA analysis, it must be determined whether the transition core can have a greater calculated peak clad temperature (PCT) than a complete core of either the

LOPAR assembly design or the OFA design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch. This hydraulic resistance mismatch exists only for transition cores and is the only unique difference between a complete core of either fuel type and the transition core.

Evaluation of hydraulic mismatch of less than 10% has shown an insignificant effect on blowdown cooling during a LOCA. The SATAN-IV code models the cross flows between the average core flow channel (N-1 fuel assemblies) and the hot assembly flow channel (one fuel assembly) during blowdown. To better understand the large break LOCA blowdown transient phenomena, conservative blowdown fuel clad heat-up calculations have been performed to determine the clad temperature effect on the new fuel design for mixed core configurations. The effect was determined by reducing the axial flow in the hot assembly at the appropriate elevations to simulate the effects of the transition core hydraulic mismatch. In addition, the Westinghouse blowdown evaluation model was modified to account for grid heat transfer enhancement during blowdown for this evaluation. The results of this analysis have shown that no peak clad temperature penalty is observed during blowdown for the mixed core. Therefore, it is not necessary to perform a blowdown calculation for OFA transition core configurations because the Evaluation Model blowdown calculation performed for the full OFA core is conservative and bounding.

Since the overall hydraulic resistance between the two types of fuel is similar and the rods are the same size, only the cross flows during core reflood due to different grid designs needs to be evaluated. The LOCA analysis uses the BASH computer code (Reference 10) which utilizes the BART code (Reference 11) to calculate the reflood transient. A detailed description of the use of the BASH code is given in Enclosure 2. Fuel assembly design specific analyses have been performed with a version of the BART computer code (Reference 11) which accurately models mixed core cases during reflood. Westinghouse specific core designs, including 14x14, 15x15 and 17x17 standard to OFA transition core cases, were analyzed. The various

fuel assembly specific transition core analyses resulted in peak clad temperature increases of up to 10°F for the core axial elevations which bound the PCT. Therefore, the maximum PCT penalty possible for the transition core is 10°F. Once a full core of OFA is achieved, the large break LOCA analysis will apply without the crossflow penalty.

5.2.1.2 Methods of Analysis

The methods of analysis including code descriptions and assumptions are described in detail in Enclosure 2.

5.2.1.3 Results

A more detailed presentation of the analytical results including tabular and plotted results for all analyzed cases is contained in Enclosure 2.

5.2.1.4 Conclusions

For breaks up to and including the doubled ended severance of a reactor coolant pipe, the emergency core cooling system will meet the acceptance criteria of 10CFR50.46 at nuclear core power levels of 2758 MWt thermal and 3071.4 MWt thermal. That is:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed one percent of the total amount of zircaloy in the reactor.

3. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time. That is required to remove the heat from the long-lived reactivity in the core.

The time sequence of events for all breaks analyzed is shown in Table 14.3-4 contained in Enclosure 2.

The large break LOCA analysis for Indian Point Unit 2, utilizing the BASH model, resulted in a peak clad temperature of 2039°F at a NSSS power level of 3083.4 Mwt for the limiting break case ($C_D = 0.4$ at a high vessel average temperature under minimum safeguards assumptions) at a total peaking factor of 2.32. The maximum local metal-water reaction was 5.54 percent, and the total metal-water reaction was less than 0.3 percent for all cases analyzed. The clad temperature turned around at a time when the core geometry is still amenable to cooling. Criteria 5 is addressed separately in a specific evaluation for each reload cycle.

The small impact of crossflow for transition core cycles is conservatively evaluated to be no greater than 10°F, which is easily accommodated in the margin to the 10CFR50.46 limits (i.e. transition PCT $\leq$ 2049°F).

It can be seen from the results of this large break ECCS analysis that Indian Point Unit 2 remains in compliance with the requirements of 10CFR50.46.

5.2.2 Small Break LOCA

5.2.2.1 Description of Analysis/Assumptions

The small break loss-of-coolant accident (LOCA) was analyzed using axial power shapes consistent with the peaking factor limits assumed for the transition cores at reactor core power of 3071.4 MWt thermal and 25% steam generator tube plugging. The NRC approved NOTRUMP small break ECCS Evaluation Model was employed to analyze a spectrum of cold leg break sizes (4 in., 6 in. and 8 in. equivalent diameter). Enclosure 2 contains a full description of the conditions and assumptions utilized for the small break LOCA analysis.

The small break LOCA analysis, which was performed prior to the large break analysis for 15x15 OFA, modeled a full core of 15x15 LOPAR fuel. The rod diameter for the 15x15 OFA and 15x15 LOPAR are the same and the hydraulic resistances between the 15x15 OFA and LOPAR designs are very similar. An evaluation was performed to address the effects of hydraulic differences between the two types of fuel on the small break LOCA analysis. Based on previous evaluations of the effects of hydraulic resistance differences on small break LOCA results with NOTRUMP and on the specifics of the 15x15 LOPAR and 15x15 OFA fuel designs, the evaluation concluded that the NOTRUMP small break LOCA analysis assuming a full core of 15x15 LOPAR remains applicable to the 15x15 OFA design.

When assessing the effect of transition cores on the small break LOCA analysis, it must be determined whether the transition core can have a greater calculated peak clad temperature (PCT) than either a complete core of the LOPAR assembly design or the OFA design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch. This hydraulic resistance mismatch exists only for transition cores and is the only unique difference between a complete core of either fuel type and the transition core.

The NOTRUMP computer code (Reference 12) is used to model the core hydraulics during a small LOCA event. Only one core flow channel is modeled in the NOTRUMP code since the core flow during a small break is relatively slow and this provides enough time to maintain flow equilibrium between fuel assemblies (i.e, cross flow). Therefore, hydraulic mismatch is not a factor for small break. Thus, it is not necessary to perform a small break evaluation for transition cores, and it is sufficient to reference the small break LOCA analysis for the 15x15 LOPAR fuel for LOPAR, OFA and the transition core.

5.2.2.2 Method of Analysis

The methods of analysis including code descriptions and assumptions are described in detail in Enclosure 2.

5.2.2.3 Results

A more detailed presentation of the analytical results including tabular and plotted results for all analyzed cases is contained in Enclosure 2.

5.2.2.4 Conclusions

In the event of a Small Break LOCA the emergency core cooling system provides adequate core protection up to core power level of 3071.4 MWt thermal by satisfying the acceptance criteria for 10CFR50.46. That is:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed one percent of the total amount of zircaloy in the reactor.

3. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time. This is required to remove the heat from the long-lived reactivity in the core.

The time sequence of events for all breaks analyzed is shown in Table 14.3-13 contained in Enclosure 2.

The small break LOCA analysis for the Indian Point Unit 2 Plant, utilizing the NOTRUMP model, resulted in a peak clad temperature of 1218.5°F for the limiting break case and limiting operating temperature condition for an F_Q envelope based on a total peaking factor of 2.32 and a hot channel enthalpy rise factor of 1.65 (which conservatively bounds 1.62). The maximum local metal-water reaction was 0.08 percent, and the total metal-water reaction was

less than 0.3 percent for all cases analyzed. The clad temperature excursion was calculated to reverse at a time when the core geometry would still be amenable to cooling. Criteria 5 is addressed separately in a specific evaluation for each reload cycle.

Mixed core hydraulic resistance mismatch is not a significant factor for small break LOCA analysis. Therefore, it is not necessary to perform any additional evaluations for small break transition cores, and it is sufficient to reference the small break LOCA analysis for the 15x15 LOPAR fuel for LOPAR, OFA and the transition core.

6.0 REFERENCES

1. Davidson, S. L.,; Iorii, J. A., "Reference Core Report - 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

2. Davidson, S. L. and Kramer, W. R.; (Ed.) "Reference Core Report VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985.

3. Chelemer, H. et. al., "Improved Thermal Design Procedure," WCAP-8567, July 1975.

4. Hochreiter, L. E., Chelemer, H., Chu, P. T., "THINC IV, An Improved Program For Thermal Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.

5. Hochreiter, L. E., "Application of the THINC IV Program to PWR Design," WCAP-8054, October 1973.

6. Motley, F. E., et. al., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8262-P-A and WCAP-8763, July 1985.

7. Weiner, R. A., et. al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851, June 1985.

8. S. B. Braun to Document Control Desk dated May 23, 1988; Leak Before Break.

9. K. Takeuchi, et. al., MULTIFLEX 1.0 - A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic Structure System Dynamics, WCAP-8708-P-A (proprietary), Westinghouse Electric Corporation, September 1977.

10. Kabadi, J. N., et. al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266, Revision 2, August 1986, (Westinghouse Proprietary).

11. Young, M., et. al., "BART-1A: A Computer Code for the Best Estimate Analysis of Reflood Transients," WCAP-9561-P-A, 1984 (Westinghouse Proprietary).

12. Lee, N., Rupprecht, S. D., Schwarz, W. R., Tauche, W. D., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-A (Non-Proprietary) August 1985.

13. Davidson, S. L., (Ed.) et. al., "Westinghouse Reload Safety Evaluation Methodology, WCAP-9272-P-A, July 1985.

14. Skaritka, J., "Operational Experience with Westinghouse Cores (up to December 31, 1986)," WCAP-8183-Rev. 15, July 1987.

15. George, R. A., (et. al.), "Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381 (Non-Proprietary), July 1974.

16. Risher, D. H., (et. al.), "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8963-P-A (Proprietary), August 1978.

17. Davidson, S. L. (Ed.) et. al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

18. Skaritka, J., (Ed.) "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1 (Proprietary), July 1979.

19. G. J. Bohm, Indian Point Unit No. 2 Reactor Internals Mechanical Analysis for Blowdown Exitation, WCAP-7332-L (proprietary), WCAP-7822 (non-proprietary, Westinghouse Electric Corporation, February 1970.

20. Analysis of Reactor Coolant System for Postulated Loss-of-Coolant Accident: Indian Point 3 Nuclear Power Plant, WCAP-9117 (proprietary), Westinghouse Electric Corporation, June 1977.

21. Liu, Y. S., et. al., "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, September 1986.

22. Indian Point Unit 2 Final Safety Analysis Report, Docket No. 50-247.

23. Letter from W. J. Cahill (Con. Edison) to V. Stello (NRC), Indian Point Unit 2, Docket No. 50-247, June 15, 1978.

24. Motley, F. E., Cadek, F. F., "Application of Modified Spacer Factor to L-Grid Typical and Cold Wall Cell DNB"; WCAP-7988, Oct. 1972.

25. "Partial Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1" letter, E. P. Rahe, Jr. (Westinghouse) to J. R. Miller (NRC), NS-EPR-2515, dated October 9, 1981; "Remaining Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1" letter, E. P. Rahe, Jr. (Westinghouse) to J. R. Miller (NRC), NS-EPR-2572, dated March 16, 1982.

26. Letter from C. O. Thomas (USNRC) to E. P. Rahe, Jr. (W), dated December 29, 1982 - Acceptance for Referencing of Licensing Topical Report, WCAP-8691(P)/WCAP-8692(NP).

27. Letter C. Berlinger (NRC) to E. P. Rahe, Jr. (Westinghouse), "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty," June 18, 1986.

28. Leech, W. J. et. al., "Revised PAD Code Thermal Safety Model," WCAP-8720, Addendum 2, October 1982.

Enclosure 1
to
Attachment C

Limiting Non-LOCA Analysis

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
September 30, 1988

Non-LOCA Accident Analysis

A.1 INTRODUCTION

This section evaluates the effects of the complete transition of Indian Point Unit 2 from Westinghouse 15x15 LOPAR fuel to Westinghouse 15x15 OFA fuel on the FSAR Chapter 14 Non-LOCA Accident Analyses. The methods used for accident evaluation are described in Reference 1 and are the same as those applied to previous Indian Point Unit 2 reloads. To be consistent with the current Indian Point Unit 2 licensing basis accident analyses, the analyses and evaluations performed for the complete transition to OFA also incorporate the most bounding effects of 25% uniform steam generator tube plugging.

The Indian Point Unit 2 licensing basis as reported in the FSAR (Reference 2) includes analyses or evaluations of thirteen (13) Non-LOCA accidents. These accidents are:

a. Uncontrolled Control Rod Withdrawal from a Subcritical Condition

b. Uncontrolled Rod Cluster Control Assembly Withdrawal at Power

c. Rod Cluster Control Assembly Drop

d. Chemical and Volume Control System Malfunction

e. Loss of Reactor Coolant Flow

f. Startup of an Inactive Reactor Coolant Loop

g. Loss of External Electrical Load

h. Loss of Normal Feedwater

i. Reduction in Feedwater Enthalpy Incident

j. Excessive Load Increase Incident

k. Loss of All AC Power to the Station Auxiliaries

l. Rupture of a Steam Pipe

m. Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

All of the above Non-LOCA accidents which could potentially be affected by the OFA reload have been reviewed and include consideration for the following design and/or Technical Specification changes:

1. The control rod scram time to the dashpot is increased (as discussed in Section 2.2.2) from 1.8 seconds to 2.4 seconds. This increased drop time primarily affects the fast reactivity transients.

2. For events reanalyzed, the analyses were conservatively performed assuming a nominal thermal power level of 3071.4 MWt with a 579.7°F vessel average temperature and thermal design flow rate of 322,800 gpm. These will bound Cycle 10 operation at the current nominal thermal power of 2758 MWt. This change has the largest affect the transients that are limiting at full power conditions. Fuel temperature based on the revised PAD computer code fuel thermal safety model (Reference 3) were also included for events reanalyzed.

3. As discussed in Section 4.0, the OFA transition includes implementation of the Improved Thermal Design Procedure (ITDP) using both the WRB-1 and W-3 L-grid DNB correlations for OFA and LOPAR fuels respectively, an increase in the nuclear enthalpy rise hot channel factor $F^N_{\Delta H}$ limit from 1.55 to 1.62 and removal of all or some portion of the thimble tube plugging devices. A conservative set of core thermal safety limits, overtemperature

delta-T and overpower delta-T setpoints were generated that are applicable to the above conditions and are valid for both transition and full OFA cores.

4. A change in the shutdown margin during refueling to reflect a K_{eff}= 0.95 was considered for the OFA transition and full OFA cores.

A.2 ACCIDENTS REANALYZED

A.2.1 General

In the process of assessing the impact of the OFA transition and the associated design changes discussed in Section A.1, only the most limiting Non-LOCA accidents which are affected by the changes associated with the OFA transition were reanalyzed. These transients include; 1) the Uncontrolled Rod Cluster Control Assembly Withdrawal at Power, 2) the Rod Cluster Control Assembly Drop, 3) the Chemical and Volume Control System Malfunction, 4) the Loss of Reactor Coolant Flow, 5) the Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection), and 6) the Uncontrolled Control Rod Withdrawal from a Subcritical Condition. The remaining FSAR Non-LOCA Accidents were evaluated for the OFA transition and associated design changes and are discussed in Section A.4.

Transients affected by the increased rod drop time are the fast reactivity transients for which the protection system responds by tripping the reactor within a few seconds after the transient begins. The transients that fall into this category are the Uncontrolled RCCA Withdrawal from a Subcritical Condition, Loss of Reactor Coolant Flow (including Locked Rotor), and RCCA Ejection. In addition to the Loss of Flow and Locked Rotor transients which are impacted by the implementation of ITDP and the increase in the $F^N_{\Delta H}$, other transients affected include those which are are DNB limited or rely on the Overtemperature Delta-T protection logic to trip the reactor. The limiting events in this category include Uncontrolled Rod Cluster Control Assembly Withdrawal at Power and the Rod Cluster Control Assembly Drop. The only event impacted by a change in the shutdown margin during refueling to

reflect a K_{eff}=0.95 is the Chemical and Volume Control System Malfunction which affects the dilution during refueling. All the above events have been reanalyzed and are discussed in section A.3.

A.2.2 Computer Codes Used

Summaries of the principal computer codes used in the non-LOCA transient analyses are given below.

A.2.2.1 FACTRAN

FACTRAN calculates the transient temperature distribution in a cross-section of a metal clad UO2 fuel rod and the transient heat flux at the surface of the clad using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, density). The code uses a fuel model which simultaneously exhibits the following features:

a. A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents.

b. Material properties which are functions of temperature and sophisticated fuel-to-clad gap heat transfer calculation.

c. The necessary calculations to handle post-departure from nucleate boiling (DNB) transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the fuel.

FACTRAN is further discussed in Reference 4.

A.2.2.2 LOFTRAN

The LOFTRAN program is used for transient response studies of a pressurized water reactor (PWR) system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), and the pressurizer. The pressurizer heaters, spray, relief valves, and safety valves

are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The reactor protection system is simulated to include reactor trips on high neutron flux, overtemperature delta-T, overpower delta-T, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure control. The ECCS, including the accumulators, is also modeled.

LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core limits. The core limits represent the limit value of the DNBR as calculated for the typical and thimble cells.

LOFTRAN is further discussed in Reference 5.

A.2.2.3 TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which was patterned after steady-state codes used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided; e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power and fuel temperatures.

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution. TWINKLE is further described in Reference 6.

A.2.2.4 THINC

The THINC-IV computer program is used to perform thermal-hydraulic calculations. The THINC-IV code is used to calculate coolant density, mass velocity, enthalpy, void fractions, static pressure, and DNBR distributions along flow channels within a reactor core under all expected operating conditions. The THINC-IV code is described in detail in References 7 and 8, including models and correlations used.

A.3 REANALYZED ACCIDENT DESCRIPTIONS

The following sections contain the detailed descriptions of the reanalyzed accidents, which were reanalyzed employing assumptions of (a) 15x15 OFA fuel, (b) a core power level of 3071.4 MWt thermal and (c) 25% steam generator tube plugging. In all cases the applicable FSAR acceptance criteria are satisfied.

A.3.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power

A.3.1.1 Introduction

Uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, in order to avert damage to the fuel clad, the Reactor Protection System is designed to terminate any such transient before the DNBR falls below the safety analysis limit values.

A.3.1.2 Method of Analysis

The transient is analyzed by the LOFTRAN Code (Reference 5) which is discussed in Section A.2.2.2.

The analyses were performed considering the transition to OFA fuel at a nominal core power of 3071.4 MWt which conservatively bounds the licensed nominal core power of 2758 MWt. The analyses also considered the other design changes associated with the OFA transition as discussed in Section A.1.

The Improved Thermal Design Procedure (Reference 9) was used in the analysis so the initial conditions for power, RCS pressure, and Tavg are at the nominal values. In performing the analysis, the following assumptions are made to assure bounding results are obtained for all possible normal operational conditions:

1. Reactivity Coefficients - Two cases are analyzed.

   a. Minimum Reactivity Feedback. A least negative moderator density coefficient of reactivity is assumed, corresponding to the beginning of core life. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed.

   b. Maximum Reactivity Feedback. A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed.

2. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors; the delays for trip actuation are assumed to be the maximum values.

3. The trip reactivity is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position.

4. A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the two control banks having the maximum combined worth at maximum speed.

5. A range of power levels between 10% and 100% power are considered.

A.3.1.3 Results

Figures A.3-1 through A.3-3 show the transient response for a rapid RCCA withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in Tavg and pressure result and margin to DNB is maintained.

The transient response for a slow RCCA withdrawal from full power is shown in Figures A.3-4 through A.3-6. Reactor trip on Overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is greater than the safety analysis limit values.

Figure A.3-7 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for minimum and maximum reactivity feedback. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and Overtemperature ΔT channels. The minimum DNBR is never less than the safety analysis limit values.

Figures A.3-8 and A.3-9 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60 and 10 percent power, respectively, for minimum and maximum reactivity feedback. The results are similar to the 100 percent power case, except as the initial power is decreased, the range over which the Overtemperature ΔT trip is effective is increased. In all cases the DNBR does not fall below the safety analysis limit value.

The shape of the curves of minimum DNB ratio versus reactivity insertion rate in the reference figures is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Referring to Figure A.3-8, for example, it is noted that:

For reactivity insertion rates above ~20 pcm/sec reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to significant increase in heat flux or water temperature with resultant high minimum DNB ratios during the transient. As reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux. Minimum DNBR during the transient thus decreases with decreasing insertion rate.

2. The Overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant loop ΔT exceeds a setpoint based on measured Reactor Coolant System average temperature and pressure.

3. For reactivity insertion rate below ~20 pcm/sec the Overtemperature ΔT trip terminates the transient.

   For reactivity insertion rates between ~20 pcm/sec and ~2 pcm/sec the effectiveness of the Overtemperature ΔT trip increases (in terms of increased minimum DNBR) due to the fact that with lower insertion rates the power increase rate is slower, the rate of rise of average coolant temperature is slower and the system lags and delays become less significant.

4. For reactivity insertion rates less than ~2 pcm/sec, the rise in the reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening of these valves, which act as an additional heat load on the Reactor Coolant System, sharply decreases the rate of increase of Reactor Coolant System average temperature.

For transients initiated from higher power levels (for example, see Figure A.3-7) the effect described in item 4 above, which results in the sharp peak in minimum DNBR at approximately 2 pcm/sec, does not occur since the steam generator safety valves are not actuated prior to trip.

Figures A.3-7, A.3-8, and A.3-9 illustrate minimum DNBRs calculated for minimum and maximum reactivity feedback.

Since the RCCA withdrawal at power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant, and the core heat flux lags behind the neutron flux response. Due to this lag, the peak core heat flux does not exceed 118 percent of its nominal value (i.e., the high neutron flux trip setpoint assumed in the analysis).

For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the Overtemperature ΔT reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 118 percent of its nominal value.

Since DNB does not occur at any time during the RCCA withdrawal at power transient, the ability of the primary coolant to remove heat from the fuel rod is not reduced.

The calculated sequence of events for this accident is shown on Table A.3-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

A.3.1.4 Conclusions

Under the core conditions as analyzed, the high neutron flux and Overtemperature ΔT trips provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than the safety analysis limit values.

A.3.2 Rod Cluster Control Assembly Drop

A.3.2.1 Introduction

The dropped RCCA accident is initiated by a single electrical or mechanical failure which causes any number and combination of rods from the same group of a given bank to drop to the bottom of the core. Protection is provided by an automatic turbine runback and an automatic rod withdrawal block. The acceptance criterion for this event is that no fuel failures occur. This is verified by demonstrating that the departure from nucleate boiling ratio (DNBR) remains above the limit value for the plant.

A dropped RCCA causes an initial reduction in nuclear power which corresponds to the reactivity worth of the rod. In addition, a turbine runback to a preset power level is actuated by either a rod-on-bottom signal or a one-out-of-four negative flux rate signal. These signals also actuate the block of automatic rod withdrawal.

For purposes of the analysis, a single or multiple dropped RCCA occurrence which causes a reduction in core power to a value greater than the turbine power at the runback setpoint is called a "dropped rod". The multiple dropped RCCAs may be any number and combination of rods from the same group of a given bank. With sufficient reactivity feedback the core power will tend to match the turbine load and the plant will stabilize at the runback setpoint. However, if there is no moderator reactivity feedback, power will stabilize at the level corresponding to that caused by the dropped rod. Primary reactor power will be greater than turbine power, resulting in a heatup of the primary coolant. The steam generator safety valves will open to accommodate the mismatch between reactor and turbine power. If the combined relieving capacity of the turbine and steam generator safety valves is sufficient to match the reactor power, the plant will stabilize in this condition. However, the primary system heatup may be too great in which case an overtemperature delta-T reactor trip will terminate the event.

A multiple dropped RCCA occurrence which causes a reduction in core power to a value less than the turbine power at the runback setpoint is called a "dropped bank". A dropped bank causes a reduction in core power to a value less than the turbine power at the runback setpoint. With maximum reactivity feedback the core power will increase to match the turbine load and the plant will stabilize at the runback setpoint.

A.3.2.2 Method of Analysis

The dropped rod transient was analyzed using the current Westinghouse methodology for turbine runback plants. The transient following a dropped rod is simulated by using the LOFTRAN computer code (Ref. 5). Two cases, dropped rod and dropped bank were considered in the analysis. The assumptions for the dropped rod case are:

1. Minimum moderator reactivity feedback corresponding to beginning of core life (0 MTC).

2. Least negative Doppler temperature coefficient.

The assumptions used for the dropped bank analysis are:

Maximum moderator reactivity feedback corresponding to the end-of-life core condition (0.54 $\Delta k/g/cc$ MDC).

2. Most negative Doppler temperature coefficient

The analyses were performed considering the transition to OFA fuel at a nominal core power of 3071.4 MWt which conservatively bounds the licensed nominal core power of 2758 MWt.

The Improved Thermal Design Procedure was used in the analysis so the initial conditions for power, RCS pressure, and Tavg are at the nominal values. In addition, a turbine runback setpoint (nominally, 86% $\pm$ 4% uncertainty) of 90% is conservatively assumed.

A.3.2.3 Results

Figures A.3-10 through A.3-12 show the reactor coolant system response to a dropped rod with a worth of 100 pcm. A reactor trip on overtemperature delta-T occurs at approximately 227 seconds. The trip is caused by the power mismatch between the reactor and turbine which causes a heatup of the RCS. Figures A.3-13 through A.3-15 show the response to a dropped bank with a worth of 400 pcm. Nuclear power and core heat flux stabilize at levels corresponding to the turbine runback setpoint plus uncertainty. After approximately 50 seconds, all of the plant parameters depicted in Figures A.3-13 through A.3-15 have reached equilibrium values. For all cases analyzed, the DNBR was calculated to be greater than the limit value.

A.3.2.4 Conclusion

Based on the above DNBR results, the analysis demonstrates that the DNBR criterion are met, and therefore, dropped RCCAs do not lead to conditions that cause core damage and that all applicable safety criteria are satisfied for this event.

A.3.3 Chemical and Volume Control System Malfunction

A.3.3.1 Introduction

Reactivity can be added to the core with the chemical and volume control system by feeding reactor makeup water into the reactor coolant system via the reactor makeup control system. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the concentration of reactor coolant makeup water to that existing in the coolant at the time. The chemical and volume control system is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

There is only a single, common source of dilution water to the blender from the primary water makeup system; inadvertent dilution can be readily terminated by isolating this single source. The operation of the primary water makeup pumps that take suction from the primary water storage tank (PWST) provides the non-borated supply of makeup water to the blender. The boric acid from the boric acid storage tank(s) is blended with the reactor makeup water in the blender, and the composition is determined by the preset flow rates of boric acid and reactor makeup water on the reactor makeup control. The operator must switch from the automatic makeup mode to the dilute mode and move the start-stop switch to start, or, alternatively, the boric acid flow controller could be set to zero. Since these are deliberate actions, the possibility of inadvertant dilution is very small. In order for this dilution water to be added to the reactor coolant system, the charging pumps must be running in addition to the primary water makeup pumps. Also, any diluted water introduced into the volume control tank (VCT) must pass through the charging pumps to be added to the reactor coolant system.

Thus, the rate of addition of diluted water to the reactor coolant system from any source is limited to the capacity of the charging pumps. This addition rate is 294 gpm for all three charging pumps. This is the maximum delivery rate based on a pressure drop calculation comparing the pump curve with the system resistance curve. Normally, only one charging pump is operating while the others are on standby.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of pumps in the chemical and volume control system. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

Boron dilution during refueling, startup, and power operation are considered in this analysis.

A.3.3.2 Method of Analysis and Results

A.3.3.2.1 Dilution During Refueling

During refueling the following conditions exist:

1. One residual heat removal pump is normally running except during short time periods as allowed by the technical specifications.

2. The chemical and volume control system and/or safety injection system are aligned so that there is at least one flow path to the core for boric acid injection when there is fuel in the reactor, as required by the Technical Specifications.

3. The minimum boron concentration of the refueling water is at least 2000 ppm or higher to maintain a shutdown of at least 5 percent $\Delta k/k$ with all control rods in; periodic sampling ensures that this concentration is maintained.

4. Neutron sources are installed in the core and detectors connected to instrumentation giving audible count rates are installated outside or within the reactor vessel to provide direct monitoring of the core.

A minimum water volume in the reactor coolant system of 3497 ft3 is considered. This corresponds to the volume necessary to fill the reactor vessel above the nozzles to ensure mixing via the residual heat removal loop. The maximum dilution flow of 294 gpm and uniform mixing are also considered.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High count rate is alarmed in the reactor containment and the main control room. The count-rate increase is proportional to the multiplication factor.

The boron concentration must be reduced from 2000 ppm to approximately 1390 ppm before the reactor will go critical. This would require more than 30 minutes. This is ample time for the operator to recognize the audible high count-rate signal and isolate the reactor makeup source by closing valves and stopping the primary water makeup pumps, and/or charging pumps.

A.3.3.2.2 Dilution During Startup

In this mode, the plant is being taken from one long-term mode of operation, Hot Standby, to another, Power Operation. Typically, the plant is maintained in the Startup mode only for the purpose of startup testing at the beginning of each cycle. During this mode of operation rod control is in manual. All normal actions required to change power level, either up or down, require operator initiation. Conditions assumed for the analysis are:

1. Dilution flow is the maximum capacity of the charging pumps, 294 gpm.

2. A minimum RCS water volume of 8360 ft^3. This corresponds to the active RCS volume taking into account 25% uniform steam generator tube plugging minus the pressurizer and the reactor vessel upper head.

3. The initial boron concentration is assumed to be 1800 ppm, which is a conservative maximum value for the critical concentration at the condition of hot zero power, rods to insertion limits, and no Xenon.

4. The critical boron concentration following reactor trip is assumed to be 1550 ppm, corresponding to the hot zero power, all rods inserted (minus the most reactive RCCA), no Xenon condition. The 250 ppm change from the initial condition noted above is a conservative minimum value.

This mode of operation is a transitory operational mode in which the operator intentionally dilutes (borates) and withdraws control rods to take the plant critical. During this mode, the plant is in manual control with the operator required to maintain a high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution (boration) and subsequently manually withdraw the control rods, a

process that takes several hours. The Technical Specifications require that the operator assure that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source range reactor trip nominally set at 5.0 E5 cps after receiving P-6 from the intermediate range. Too fast a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the source range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor.

However, in the event of an unplanned approach to criticality or dilution during power escalation while in the Startup mode, the plant status is such that minimal impact will result. The plant will slowly escalate in power to a reactor trip on the power range neutron flux - high, low setpoint (nominal 25 percent power). From initiation of the event, there are greater than 15 minutes available for operator action prior to return to criticality.

A.3.3.2.3 Dilution at Power

In this mode, the plant may be operated in either automatic or manual rod control. Conditions assumed for the analysis are:

1. Dilution flow is the maximum capacity of the charging pumps, 294 gpm.

2. A minimum RCS water volume of 8360 ft3. This corresponds to the active RCS volume (with 25% uniform steam generator tube plugging) minus the pressurizer and reactor vessel upper head.

3. The initial boron concentration is assumed to be 1800 ppm, which is a conservative maximum value for the critical concentration at the condition of hot full power, rods to insertion limits, and no Xenon.

4. The critical boron concentration following reactor trip is assumed to be 1450 ppm, corresponding to the hot zero power, all rods inserted (minus the most reactive RCCA), no Xenon condition. The 350 ppm change from the initial condition noted above is a conservative minimum value.

With the reactor in automatic rod control, the power and temperature increase from boron dilution results in insertion of the control rods and a decrease in the available shutdown margin. The rod insertion limit alarms (LOW and LOW-LOW settings) alert the operator at least 32 minutes prior to criticality. This is sufficient time to determine the cause of dilution, isolate the reactor water makeup source, and initiate boration before the available shutdown margin is lost.

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach the Overtemperature ΔT trip setpoint resulting in a reactor trip. The boron dilution transient in this case is essentially the equivalent to an uncontrolled RCCA bank withdrawal at power. The maximum reactivity insertion rate for a boron dilution is conservatively estimated to 3.0 pcm/sec, which is within the range of insertion rates analyzed. Thus, the effects of dilution prior to reactor trip are bounded by the uncontrolled RCCA bank withdrawal at power analysis (Section A.3.1). Following reactor trip there are grater than 15 minutes prior to criticality. This is sufficient time for the operator to determine the cause of dilution, isolate the reactor water makeup source, and initiate boration before the available shutdown margin is lost.

A.3.3.3 Conclusions

Because of the procedures involved in the dilution process requiring operator action, an erroneous dilution is considered very unlikely. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to changes in dilution are slow enough to allow the operator to determine the cause of the addition and take corrective action before shutdown margin is lost.

A.3.4 Loss of Reactor Coolant Flow

A.3.4.1 Introduction

As demonstrated in the FSAR, Section 14.1.6, the most severe loss of flow transient is caused by the simultaneous loss of electric power to all four reactor coolant pumps. This transient was reanalyzed to determine the effect of the increased rod drop time and other design changes associated with the OFA transition on the minimum DNBR reached during the incident.

A.3.4.2 Method of Analysis

The analyses were performed assuming implementation of the transition to OFA fuel, and a nominal core power of 3071.4 MWt, which conservatively bounds the currently licensed nominal core power of 2758 MWt.

The methods and assumptions used in the analyses are consistent with those employed in the FSAR with the exception that the Improved Thermal Design Procedure (ITDP) was used. These assumptions include:

1. Full power initial operating conditions, nominal value of power; nominal steady state pressure and maximum steady state average programmed temperature.

2. Highest value (absolute) of Doppler Power coefficient and zero moderator temperature coefficient.

3. Time from loss of power to all pumps to the initiation of control rod assembly motion (under voltage reactor trip) of 1.5 seconds.

4. 4% ΔK trip reactivity from full power; and

5. A conservative value for the nominal RCP and motor inertia was used to compute the flow coastdown based on reactor coolant pump characteristics.

The flow coastdown transient was computed by the LOFTRAN code. The THINC code was used to calculate DNBR. The FACTRAN code was used to calculate heat flux. The effects of the slower rod drop time were determined for both typical and thimble cell models.

A.3.4.3 Results

Figures A.3-16 through A.3-19 show the flow coastdown, nuclear power, heat flux, and DNB ratio vs. time for a full OFA core which yielded the worst results of the cases analyzed.

The sequence of events and summary of results is given in Table A.3-2.

A steady-state THINC analysis of the limiting point in the transient verified that the DNBR remains above the limiting value.

A.3.4.4 Conclusions

The increase in rod drop time and other design changes associated with the OFA transition do not result in violation of the DNBR limit value for the complete loss of flow transient. Loss of a single pump with all loops in service has also been analyzed and results have shown that this transient is less severe than the complete loss of forced coolant flow.

All the safety criteria are met.

A.3.5 Locked Rotor

A.3.5.1 Introduction

This transient, which is reported in Section 14.1.6 of the FSAR, was reanalyzed to evaluate the effects of the increased rod drop time and other design changes associated with the OFA transition.

A.3.5.2 Method of Analysis

The analysis were performed, assuming implementation of the transition to OFA fuel, and a nominal core power of 3071.4 MWt which bounds the current nominal core power of 2758 MWt.

The method and assumptions used in the analysis are consistent with those employed in the FSAR, with the exception that the Improved Thermal Design Procedure was used for the Rods-In-DNB calculation.

The following effects of the Locked Rotor were investigated using the 2.4 second rod drop time:

1. Primary pressure transient.

2. Fuel clad temperature transient (This is calculated assuming film boiling in order to give the worst possible results).

3. DNB transient (for determining the percentage of rods in DNB for the offsite dose release calculations).

The following assumptions were used:

1. Initial operating conditions most adverse with respect to margin to clad temperature, RCS pressure;

   a. Power = 102% of nominal (1)
   b. Inlet Temperature = 548.3 °F (1)
   c. RCS Pressure = 2281.5 psia (2)

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(1) For the RODS-IN-DNB calculation the nominal value was used according to the Improved Thermal Design Procedure.

(2) For the RODS-IN-DNB calculation the expected value of core pressure (2280 psia) was used according to the Improved Thermal Design Procedure.

2. Highest value (absolute) of Doppler Power coefficient and zero moderator temperature coefficient.

3. 4% ΔK trip reactivity from full power.

4. For clad temperature calculation DNB is assumed to occur at time = 0.

The flow coastdown transient was computed by the LOFTRAN code. The FACTRAN code was used to calculate fuel rod temperatures and heat flux distribution. The THINC code was used to calculate DNBR. The analysis was performed without offsite power available.

A.3.5.3 Results

Under the conditions used in the analysis, peak reactor coolant pressure was determined to be 2524 psia, and peak clad temperature 1671 °F. Figures A.3-20 through A.3-23 show the core flow coastdown, nuclear power, reactor coolant pressure, and fuel clad temperature transients respectively. The sequence of events and summary of results is given in Table A.3-3.

The most limiting case yields no rods in DNB.

A.3.5.4 Conclusions

The 2.4 seconds rod drop time and other design changes associated with OFA can be accommodated by existing margins with regard to the Locked Rotor transient. The peak pressure of 2524 psia is below the maximum allowable value of 2750 psia and the peak clad temperature of 1671°F is well below the maximum (hot spot) average clad temperature limit of 2200°F. The safety criteria and dose release limits are not exceeded.

A.3.6 Rod Ejection

A.3.6.1 Introduction

This accident, which is reported in Section 14.2.6 of the FSAR, was reanalyzed to assure fuel rod enthalpy, melt and clad temperature criteria would not be violated by the increased rod drop time associated with OFA transition.

A.3.6.2 Method of Analysis

Methods and assumptions used in the analysis were consistent with those employed in the FSAR. The effects of rod ejection were investigated assuming the following.

1. A conservative value of trip rod worth is used assuming a stuck rod in addition to the ejected rod.

2. Initial Power = HZP or 1.02 x nominal HFP.

3. Initial Pressure = 2206.5 psia.

4. Initial Coolant Average Temperature = 586.2°F.

5. Initial Fuel Temperature = 2400°F.

The Rod Ejection accident transient was simulated using the TWINKLE and FACTRAN computer codes. Four conditions were analyzed: EOL-HFP, EOL-HZP, BOL-HFP, BOL-HZP. Additional detailed information on the Rod Ejection accident is included in FSAR Section 14.2.6.

A.3.6.3 Results

The results are presented in Table A.3-4. Figures A.3-24 and A.3-25 show the nuclear power and fuel rod temperature transients for the EOL-HFP and EOL-HZP cases, respectively.

A.3.6.4 Conclusions

All safety criteria are satisfied when the increased rod drop time is assumed.

A.3.7 Uncontrolled RCCA Withdrawal from a Subcritical Condition

A.3.7.1 Introduction

This accident is described in Section 14.1.1 of the FSAR. The nuclear power response is characterized by a very fast rise terminated by the reactivity effect of the negative doppler fuel temperature coefficient. Subsequently, the rods fall into the core, which terminates the transient and prevents the nuclear power from increasing again at a slower rate. Even though the peak thermal flux is limited by the doppler coefficient and not reactor trip, this transient was reanalyzed to determine the effect of increasing rod drop time and other design changes associated with the OFA transition on the minimum DNBR reached during the accident.

A.3.7.2 Method of Analysis

The analysis was performed assuming the implementation of the transition to OFA fuel at a nominal core power of 3071.4 MWt, which conservatively bounds the currently licensed nominal core power of 2758 MWt.

The methods and the assumptions used in the analysis are consistent with those employed in the FSAR. These assumptions include:

1. The reactor is assumed to be just critical at hot zero power (no load) Tavg (547 °F).

2. The rate of reactivity insertion is based on the assumption of two consecutive control banks moving with 100% overlap, at the maximum speed in the maximum worth region.

3. Only two reactor coolant pumps are assumed to be in operation.

4. Conservatively low (absolute value) Doppler Power Coefficient at BOC and zero moderator temperature coefficient.

5. Maximum fraction of delayed neutron.

6. Minimum steady state pressure.

7. Most adverse combination of instrument and setpoint errors, as well as delay for trip signal actuation and control rod assembly release, were taken into account.

The neutron flux transient was calculated using the TWINKLE computer code. The FACTRAN code was used to calculate heat flux and hot spot temperature. The THINC code was used to calculated DNBR.

A.3.7.3 Results

Figures A.3-26 through A.3-29 show the nuclear power, heat flux, fuel average temperature, and clad temperature at the hot spot versus time for a full OFA core which yields the worst results.

The sequence of events and summary of the results are provided in Table A.3-5.

A steady-state THINC analysis of the limiting point in the transient verified that the DNBR remains above the limiting value.

A.3.7.4 Conclusions

For the accidents analyzed in this section, the increase in rod drop time and the other design changes associated with the OFA transition do not result in a violation of the DNBR limit value. Also, fuel and clad temperature remain well below the limit values. Based on this, it is concluded that all the applicable safety criteria for the Uncontrolled RCCA Withdrawal from Subcritical Condition are met.

A.4 ACCIDENTS NOT REANALYZED

As discussed in Section A.2, only the most limiting Non-LOCA accidents which are affected by the changes associated with the OFA transition were reanalyzed. The remaining less limiting non-LOCA FSAR accidents are discussed below.

A.4.1 Start-up of an Inactive Reactor Coolant Loop

This accident is described in Section 14.1.7 of the FSAR. As stated in the FSAR, operation of the plant with an inactive loop causes reversed flow through the inactive loop because there are no isolation valves or check valves in the reactor coolant loops. If the reactor was operated at power in this condition, there would be a decrease in the coolant temperature in that loop (in comparison with the active loops) and subsequent restart of the idle reactor coolant pump, without bringing the loop temperature closer to the average temperature, would result in the injection of cold water into the core. This cooler water would cause a rapid reactivity increase.

However, Technical Specification 3.1 requires that all 4 reactor coolant pumps be operating for reactor power operation and precludes operation with an inactive loop (except for testing or repair and not to exceed the time specified). This event was originally included in the FSAR licensing basis when operation with a loop out of service was considered. Based on the current Technical Specifications which prohibit at power operation with a loop of service as indicated above and, based on the proposed changes to the Technical Specification as presented in Section 6.0 of this report which deletes all references to Three Loop Operation, it is concluded that this event should also be deleted from the current FSAR licensing basis. Elimination of this event from the FSAR licensing basis is similar to what was previously done for FSAR Section 14.1.3 entitled Incorrect Positioning of Part-Length Rods, as a result of changes in plant operations.

A.4.2 Loss of External Electrical Load

This accident, as described in Section 14.1.8 of the FSAR, may result from an abnormal variation in network frequency, or an accidental opening of the main breaker from the generator, which fails to cause a turbine trip but causes a rapid large load reduction by the turbine governor control.

The plant is designed to accept a 50 percent step loss of load without actuating a reactor trip by actuation of the automatic steam bypass (steam dump) system. In the event the steam bypass valves fail to open following a large load loss, the steam-generator safety valves are actuated and the reactor may be tripped by the high pressurizer pressure signal or the high pressurizer level signal. The steam-generator shell-side pressure and reactor coolant temperatures increase rapidly. The pressurizer safety valves are sized to protect the reactor coolant system against overpressure without taking credit for the steam bypass system.

The most likely source of a complete loss of load on the NSSS is a trip of the turbine generator. In this case, there is a direct reactor trip signal derived from the turbine autostop oil pressure. Reactor coolant temperatures and pressure do not increase if the steam bypass and pressurizer pressure control systems are functioning properly. However, the plant behavior is also evaluated for a complete loss of load from full power without a direct reactor trip. This is primarily to show the adequacy of the pressure-relieving devices and that no core damage occurs. The pressure-relieving capacities of the reactor coolant and steam systems are designed to ensure the safety of the plant without requiring the automatic rod control, pressurizer pressure control, and/or steam bypass control systems.

As indicated in the preceding discussion, the Loss of External Electrical Load event is analyzed to ensure the safety of the plant primarily against the possibility. of overpressure conditions. As described in detail in the FSAR, the analysis for this event conservatively considers cases with initial conditions, reactivity feedback conditions, and control systems such that margin to core protection limits is minimized in order to demonstrate acceptable plant safety under these most extreme conditions. The transition

from LOPAR to OFA fuel and the design changes associated with this transition as discussed in Section A.1 will not significantly impact the FSAR analysis results for this event. A review of the FSAR results show that the 0.6 second increase in the rod drop time will not lead to overpressure conditions. With respect to DNB, the conclusions of the FSAR will also remain valid for the changes associated with the OFA transition. The FSAR results show that the DNBR increases from initiation of the event in all cases analyzed and that the minimum DNBR is sufficiently high at initial full power conditions to offset the effect of the design changes on minimum DNBR.

Therefore, it is concluded that the FSAR conclusions for the Loss of External Electrical Load event remain valid for the OFA transition and the associated design changes.

A.4.3 Loss of Normal Feedwater / Loss of All AC Power to the Station Auxiliaries

The Loss of Normal Feedwater and Loss of All AC Power to the Station Auxiliaries (Station Blackout) events are described in detail in FSAR Sections 14.1.9 and 14.1.12, respectively. The analyses are performed for these events to show that the auxiliary feedwater system is adequate to remove the stored and residual heat to prevent water relief through the pressurizer safety or relief valves. The only OFA transition design change that could affect the results of the FSAR analysis of this event is the increase in the rod drop time. However, increasing the rod drop time by 0.6 seconds results in an insignificant increase in the heat load that must be removed over the duration of the transient. Therefore, it is concluded that the FSAR conclusions remain valid for this event.

A.4.4 Reduction in Feedwater Enthalpy Incident

This accident is described in Section 14.1.10 of the FSAR. A reduction in feedwater enthalpy is another means of increasing core power above full power. Such increases are attenuated by the thermal capacity in the secondary plant and in the reactor coolant system. The overpower-overtemperature protection (nuclear overpower and delta-T trips) prevent any power increase that could lead to a DNBR less than the DNBR limit.

The extreme example of excess heat removal by the feedwater system considered in the FSAR analysis is transients associated with the accidental opening of the feedwater bypass valve which diverts flow around the low-pressure feedwater heaters. For this event, there is a sudden reduction in inlet feedwater temperature to the steam generator. The increased subcooling of the secondary side will create a greater load demand on the primary side which can lead to reactor trip conditions.

As described in detail in the FSAR, the analysis for this event considers two cases. The first case assumes that the feedwater enthalpy reduction occurs without automatic rod control (e.g., plant in manual control) and with a zero moderator coefficient since this represents a condition where the plant has the least inherent transient feedback capability. The second case assumes automatic rod control with a large negative moderator coefficient. In both cases the event is assumed to occur for hot full power initial conditions.

The FSAR results for the first case indicate that the core power level remains essentially constant at full power, a small increase in delta-T results, and that the pressure decreases relatively slow throughout the event. A reactor trip on low pressurizer pressure would occur for this event around 160 seconds and there is considerable margin to the DNBR limit under these conditions.

For the second case, which is much more limiting than the first case, a slow increase in core power occurs which reduces the rate of decrease in the coolant average temperature and pressurizer pressure. Without a reactor trip actuation modeled, steady-state conditions are reached at approximately 115 % power with significant margin to the DNBR limit.

Core protection for slow increases in core power such as the second case is provided by the combination of the overpower-overtemperature protection which has been shown to be adequate for the OFA transition design changes in conjunction with the reanalysis of the Uncontrolled Rod Cluster Control Assembly Withdrawal at Power as described in Section A.3.1. Furthermore, as indicated in the FSAR for the Reduction in Feedwater Enthalpy Incident, the second case, with automatic rod control assumed, is much more limiting than the case without automatic rod control. This is of significance since the actual plant control configuration for Indian Point Unit 2 is such that automatic rod withdrawal capability has been physically disabled.

Based on the above, it is concluded that with the OFA transition design changes, all applicable design criteria will be met for the Reduction in Feedwater Enthalpy Incident and the conclusions of the FSAR for this transient remain valid.

A.4.5 Excessive Load Increase Incident

This accident is described in Section 14.1.11 of the FSAR. An excessive load increase incident is defined as a rapid increase in steam generator steam flow causing a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10 percent step-load increase and a 5 percent per minute ramp load increase without a reactor trip in the range of 15 to 100 percent full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system. The purpose of the analysis presented in the FSAR is to show that all applicable design criteria are met without a reactor trip while allowing for a 10 percent step-load and a 5 percent per minute ramp load increase.

As described in detail in the FSAR, the analysis for this event considers two cases. The first case assumes no automatic rod control (e.g., plant in manual control) and the second case assumes the plant in automatic rod control. In both cases the event is assumed to occur for hot full power initial conditions and with a zero moderator coefficient which represents a condition where the plant has the least inherent transient capability.

The FSAR results for the first case indicate that the core power level remains essentially constant at full power, a small increase in delta-T results, and that the pressure decreases relatively slow throughout the event. A reactor trip on low pressurizer pressure would occur for this event around 150 seconds and there is considerable margin to the DNBR limit under these conditions.

For the second case, which is more limiting than the first case, an increase in core power occurs which reduces the rate of decrease in the coolant average temperature and pressurizer pressure. Without a reactor trip actuation modeled, steady-state conditions are reached at 110% power after a peak of approximately 112 % power. During this transient there is significant margin to the DNBR limit.

As was discussed in Section A.4.4 for the Reduction in Feedwater Enthalpy Incident, core protection for slow increases in core power is provided by the combination of the overpower-overtemperature protection which has been shown to be adequate for the OFA transition design changes in conjunction with the reanalysis of the Uncontrolled Rod Cluster Control Assembly Withdrawal at Power as described in Section A.3.1. As indicated in the FSAR for the Excessive Load Increase Incident, the second case, with automatic rod control assumed, is more limiting than the case without automatic rod control. Again, as previously stated in Section A.4.4, this is of significance since the actual plant control configuration for Indian Point Unit 2 is such that automatic rod withdrawal capability has been physically disabled.

Based on the above, it is concluded that with the OFA transition design changes, all applicable design criteria will be met for the Excessive Load Increase Incident and the conclusions of the FSAR for this transient remain valid.

A.4.6 Rupture of a Steam Pipe

The steamline break accident is described in Section 14.2.5 of the FSAR in detail. For the evaluation of core response due to a steamline break, the FSAR licensing basis analysis considers the transient conditions resulting from various steamline break cases, all initiated from hot zero power

conditions. At these initial conditions, the control rods are inserted in the core and any increase in rod drop time or change to reactor trip setpoints associated with the OFA transition design changes are irrelevant and, thus, will not impact the FSAR results. At hot zero power conditions, the increase in the full power $F^N_{\Delta H}$ from 1.55 to 1.62 is also inconsequential. Therefore, the only possible impact of the OFA transition on the steamline break core response analysis is that associated with the small increase in core bypass flow due to thimble plug removal and, for the OFA fuel, the implementation of the WRB-1 DNB correlation. An evaluation of these changes show that DNB criteria is met for the steamline break core response cases.

For containment pressure response due to mass and energy releases during a steamline break, hot full power initial conditions are assumed in the analysis presented in FSAR Section 14.2.5.6. An increase in the rod drop time by 0.6 seconds would result in an insignificant increase in the overall mass and energy released to containment over the duration of the long transient, and, therefore, would have an insignificant impact on the peak containment pressure. The other design changes associated with the OFA transition do not impact the steamline break containment pressure response analysis.

Based on the above, it is concluded that the OFA transition and associated design changes will not alter the conclusions of the FSAR for the Rupture of a Steam Pipe. Therefore, the FSAR conclusions for this transient remain valid.

A.5 REFERENCES

1. Davidson, S. L., (Ed.) et. al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

2. Indian Point Unit 2 Final Safety Analysis Report, Docket No. 50-247.

3. Leech, W. J. et. al., "Revised PAD Code Thermal Safety Model," WCAP-8720, Addendum 2, October 1982.

4. H. G. Hargrove, "FACTRAN - A Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June 1972.

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

6. D. H. Risher, Jr. and R. F. Barry, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary), WCAP-8028-A (Non-Proprietary), January 1975.

7. Hochreiter, L. E., Chelemer, J., Chu, P. T. "THINC IV, An Improved Program for Thermal Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.

8. Hochreiter, L. E., "Application of the THINC IV Program to PWR Design," WCAP-8054, October 1973.

9. Chelemer, H. et. al., "Improved Thermal Design Procedure," WCAP-8567, July 1975.

TABLE A.3-1

TIME SEQUENCE OF EVENTS
FOR
UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER

Accident	Event	Time (sec)
Uncontrolled RCCA bank withdrawal at power		
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (80 pcm/sec)	0
	Power range high neutron flux high trip point reached	1.7
	Rods begin to fall into core	2.2
	Minimum DNBR occurs	2.4
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (1 pcm/sec)	0
	Overtemperature ΔT reactor trip signal initiated	109.5
	Minimum DNBR occurs	111.1
	Rods begin to fall into core	111.5

TABLE A.3-2

TIME SEQUENCE OF EVENTS

COMPLETE LOSS OF FLOW

Event	Time (Seconds)
All the pumps begin to coastdown	0.
Reactor coolant pump undervoltage trip point reached at	0.
Rods begin to fall	1.5
Maximum clad temperature occurs	1.8
Minimum DNBR occurs	1.9
Maximum RCS pressure occurs	3.0

SUMMARY OF THE RESULTS

COMPLETE LOSS OF FLOW

Maximum Reactor Coolant System Pressure .....2322.
Maximum Clad Avg Temperature (°F) .....677.
Maximum Peak Fuel C/L Temp. (°F).....2562.

TABLE A.3-3

TIME SEQUENCE OF EVENTS

LOCKED ROTOR EVENT - HOT SPOT

Event	Time (Seconds)
Rotor in one pump seizes	0.
Reactor low flow trip point reached at	0.1
Rods begin to fall	1.1
Maximum RCS pressure occurs	3.2
Maximum clad temperature occurs	3.5

SUMMARY OF THE RESULTS

LOCKED ROTOR EVENT - HOT SPOT

Maximum Reactor Coolant System Pressure (psia) .... 2524.
Maximum Clad Avg Temperature (°F) .....1671.
Maximum Peak Fuel C/L Temp. (°F) .....3710.
% Zirconium Reacted .....0.2 %

TABLE A.3-4

SUMMARY OF ROD EJECTION ANALYSIS PARAMETERS AND RESULTS

Accident Parameters	Time in Cycle			
	Beginning	Beginning	End	End
Initial Power, % Rated Power	0	102	0	102
Ejected Rod Worth, % $\Delta k/k$	.65	.17	.80	.20
Delayed Neutron Fraction (b_{eff})	.0050	.0050	0.0040	0.0040
FQ during Event	12.0	6.8	20.0	7.1
Results-Rod/Drop Time = 2.4 Secs				
Max. Fuel Centerline Temperature (°F)	2764	*	3606.	*
Max. Clad Average Temperature (°F)	1840	2165.	2525.	2114
Max. Fuel Enthalpy (Btu/lb)	174.2	303.8	247.5	296.5

*Less than 10% fuel centerline melt at fuel rod hot spot.

TABLE A.3-5

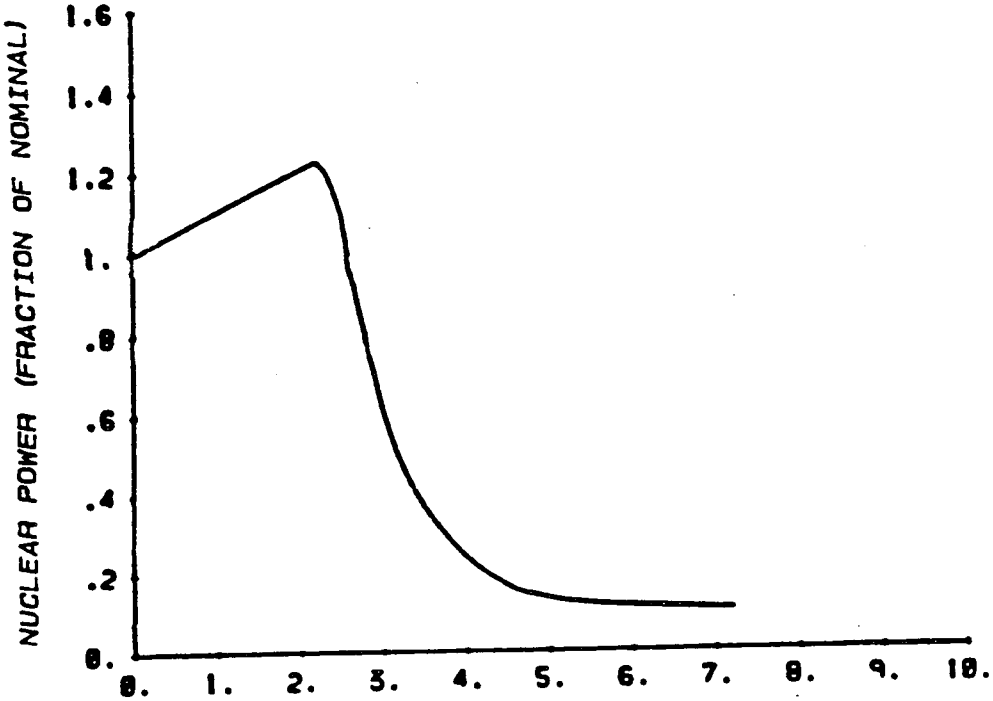
UNCONTROLLED RCCA WITHDRAWAL FROM A SUBCRITICAL CONDITION

TIME SEQUENCE OF EVENTS

Event	Time (Seconds)
Start of the accident	0.0
High Neutron Flux Reactor Trip Setpoint (Low setting) reached	9.2
Rods begin to fall	9.7
Minimum DNBR occurs	11.2
Peak Clad Average Temperature occurs	11.5
Peak Fuel Average Temperature occurs	11.8
Peak Fuel Centerline Temperature occurs	12.5

SUMMARY OF THE RESULTS

Peak Clad Average Temperature (°F).....677.
Peak Fuel Average Temperature (°F).....1818.
Peak Fuel Centerline Temperature (°F).....2152.



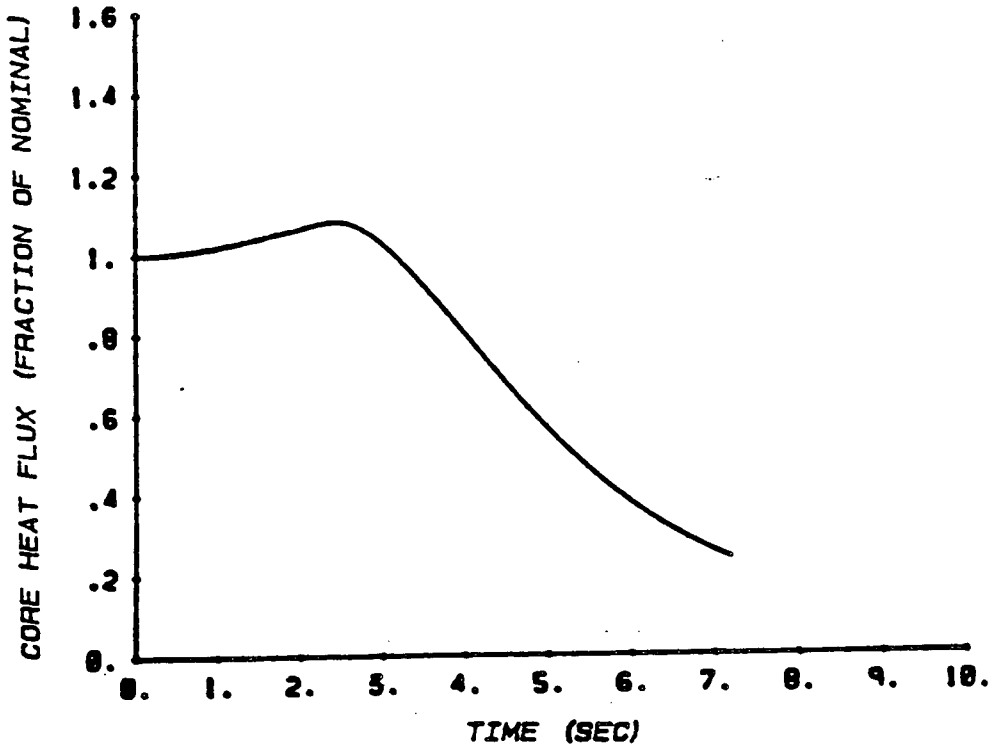


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FIGURE A.3-1

UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (80 PCM/SEC WITHDRAWAL RATE)

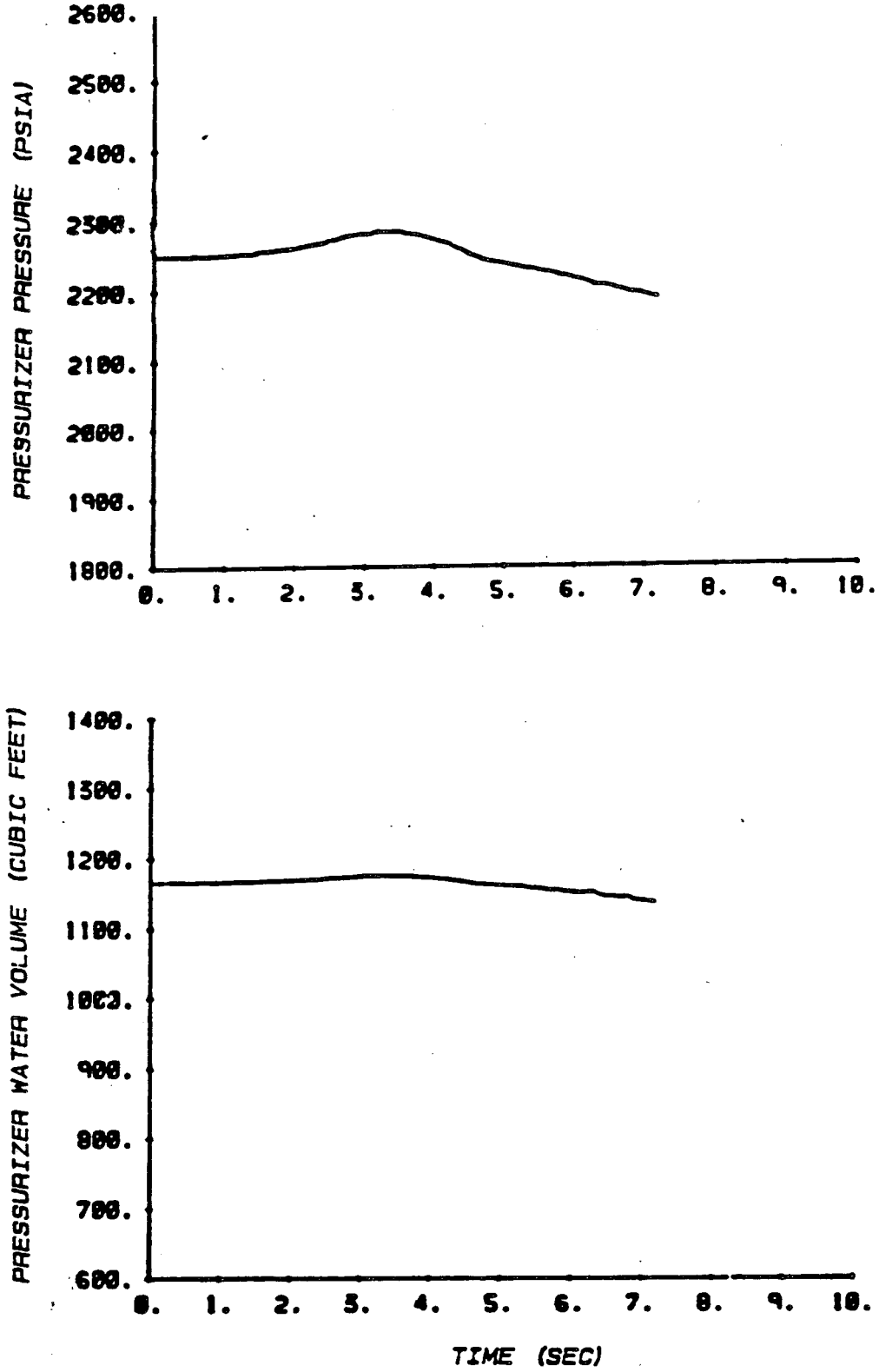

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FIGURE A.3-2

UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (80 PCM/SEC WITHDRAWAL RATE)

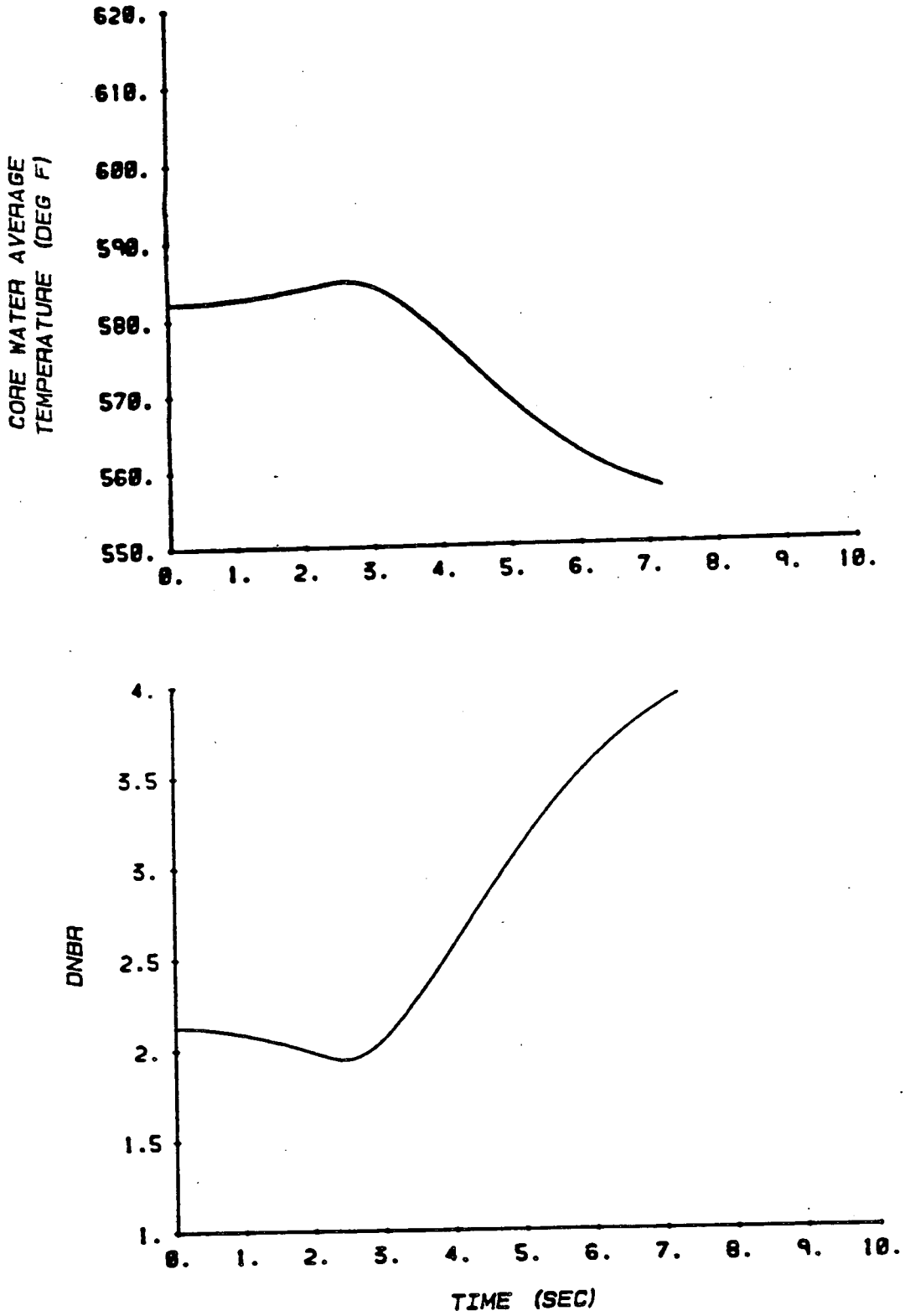

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FIGURE A.3-3

UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (80 PCM/SEC WITHDRAWAL RATE)

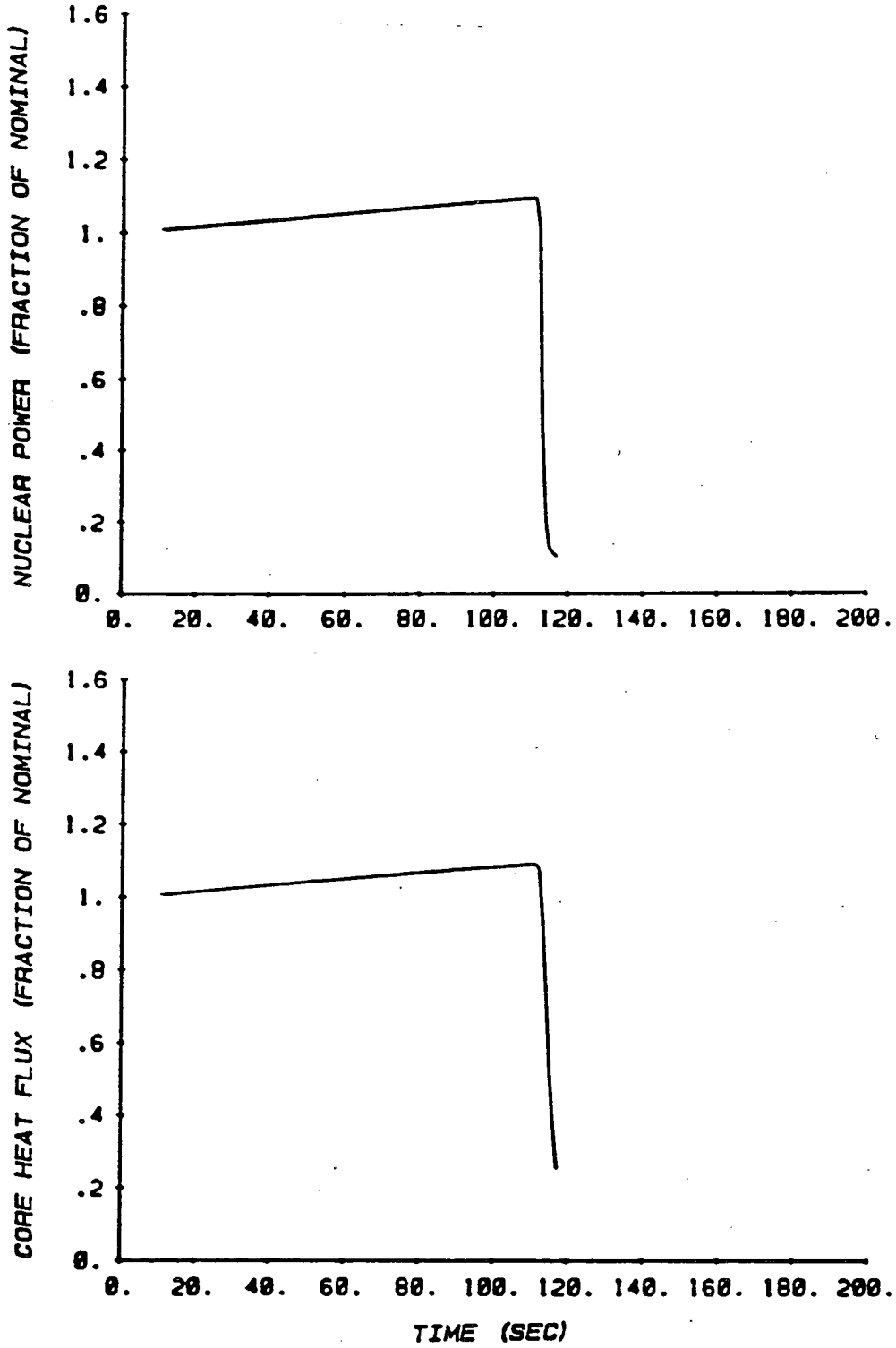

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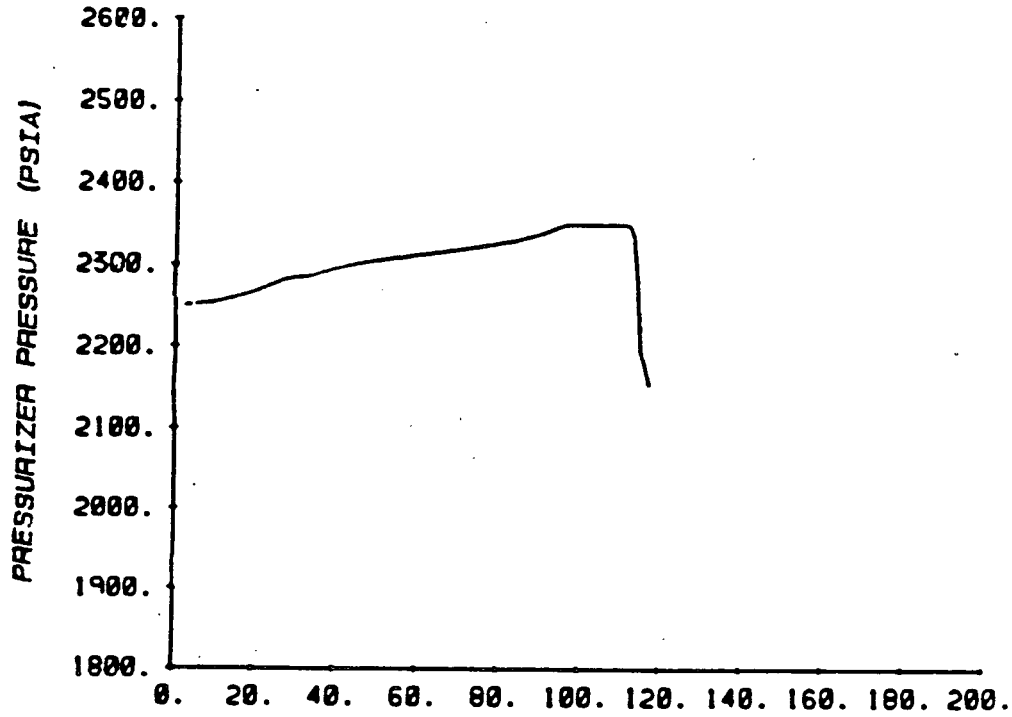
FIGURE A.3-4

UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (1 PCM/SEC WITHDRAWAL RATE)



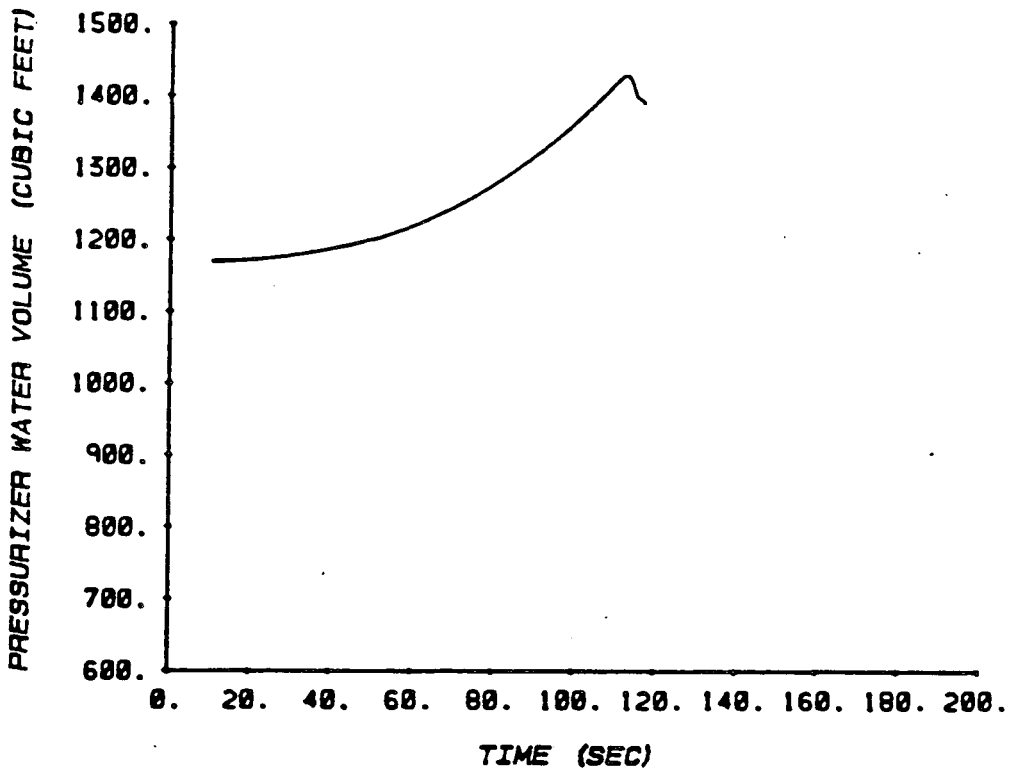


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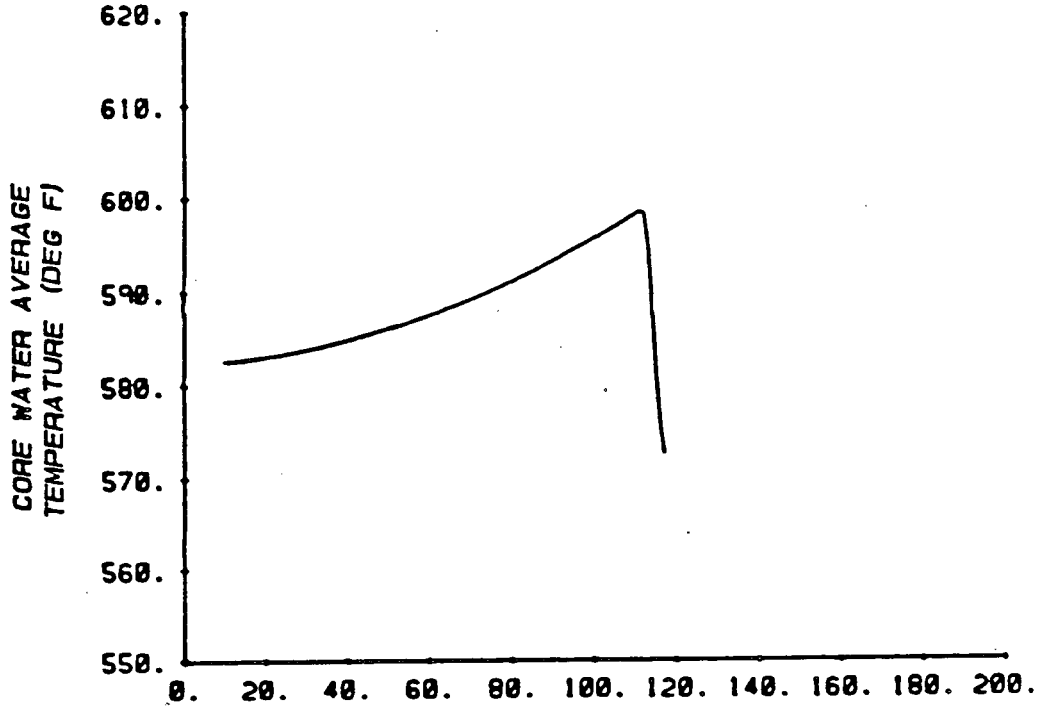
FIGURE A.3-5

UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (1 PCM/SEC WITHDRAWAL RATE)



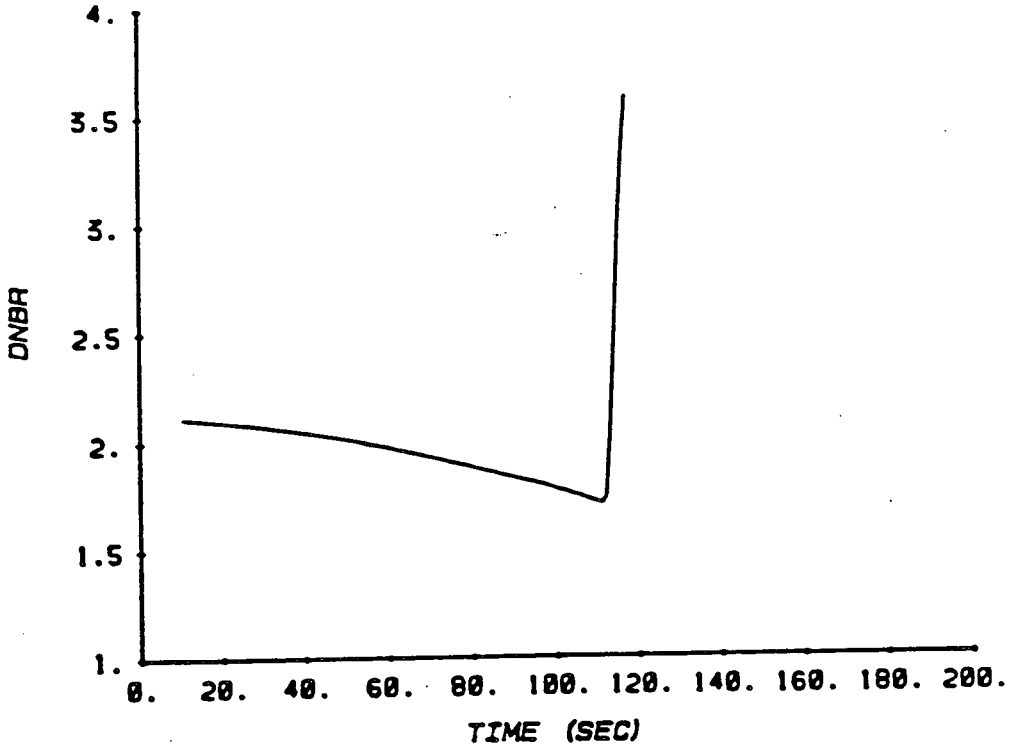


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FIGURE A.3-6

UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (1 PCM/SEC WITHDRAWAL RATE)

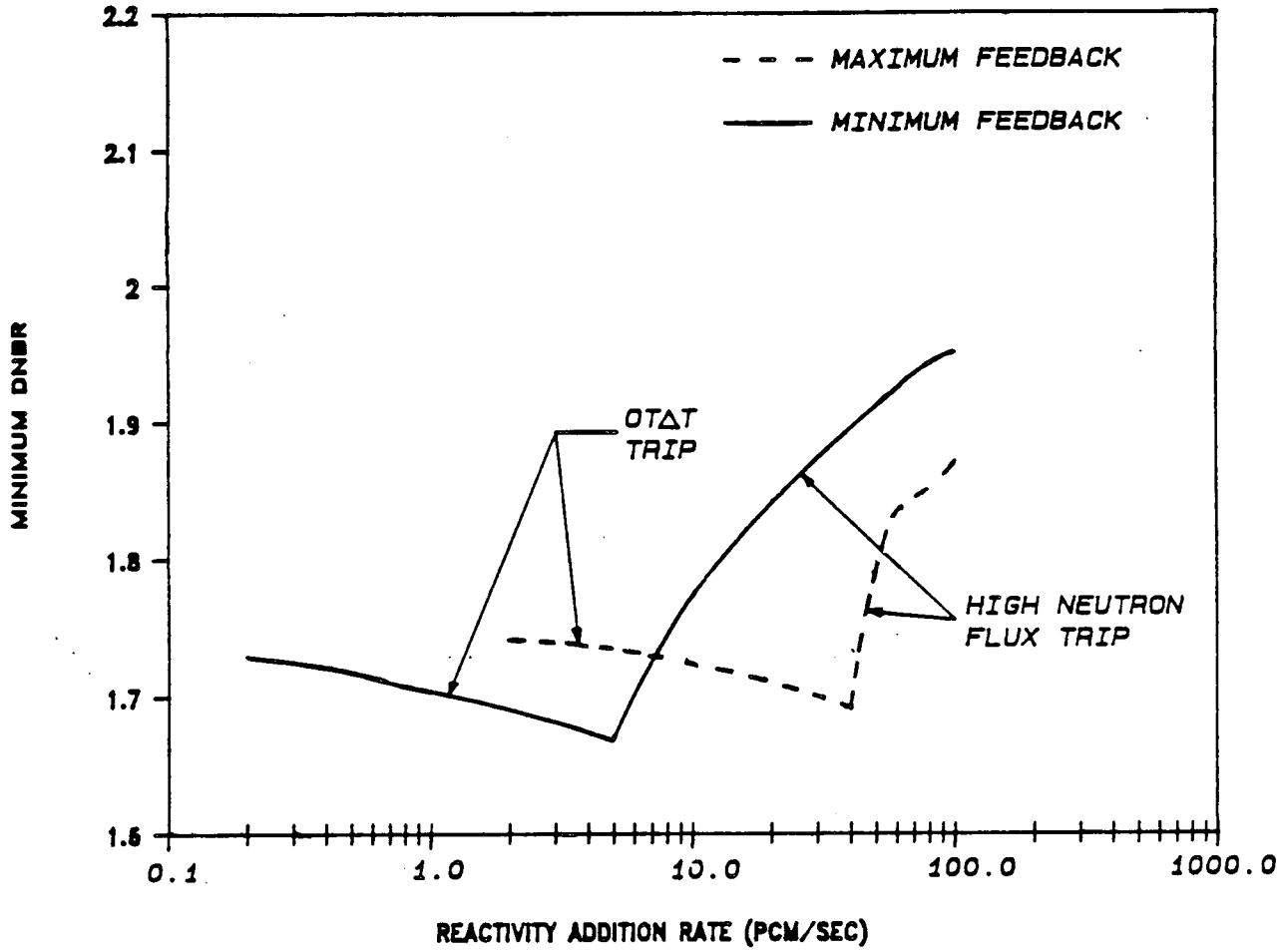

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FIGURE A.3-7

MINIMUM DNBR VERSUS REACTIVITY INSERTION RATE, ROD WITHDRAWAL FROM 100 PERCENT POWER

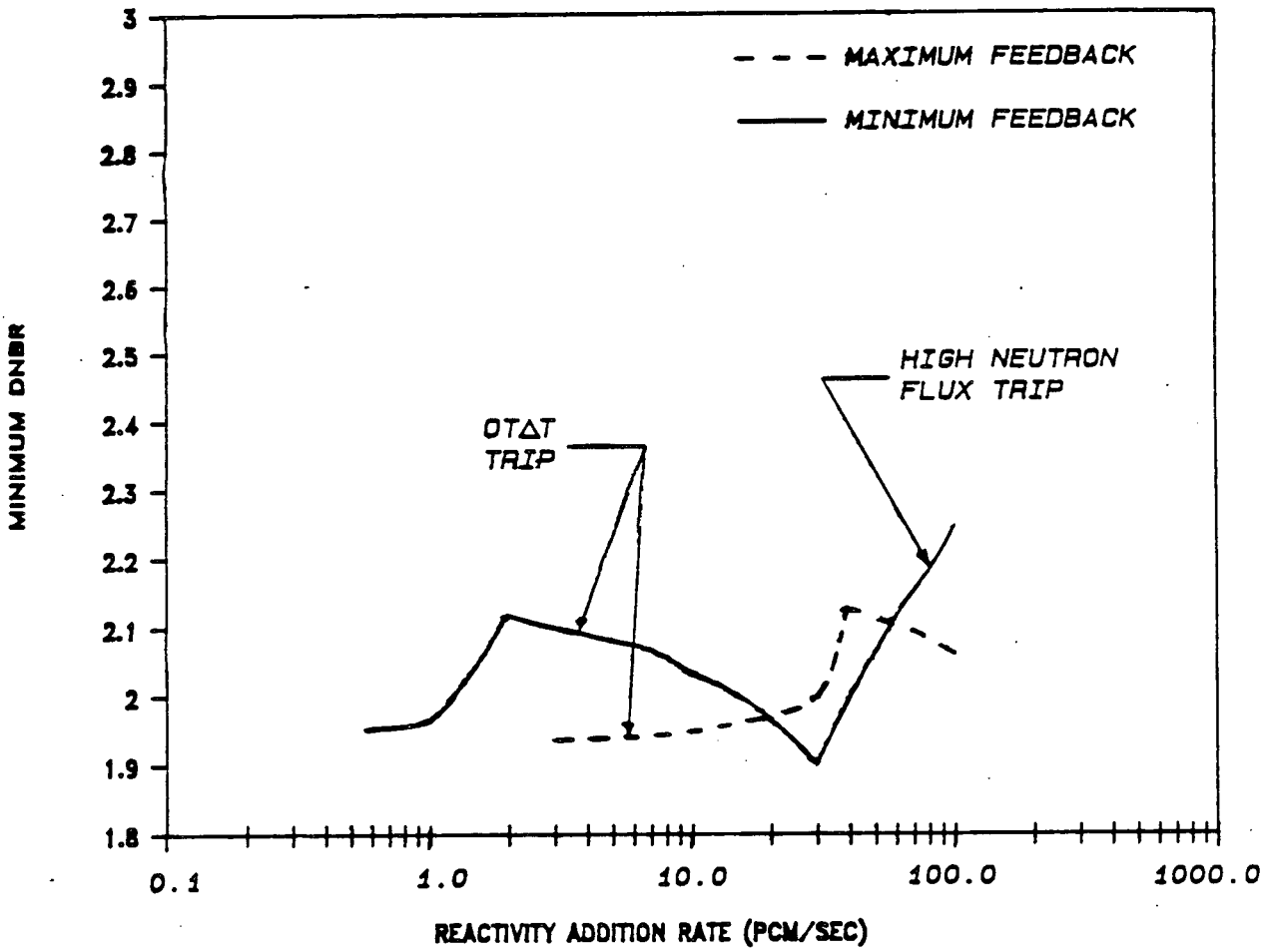

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FIGURE A.3-8

MINIMUM DNBR VERSUS REACTIVITY INSERTION RATE, ROD WITHDRAWAL FROM 60 PERCENT POWER

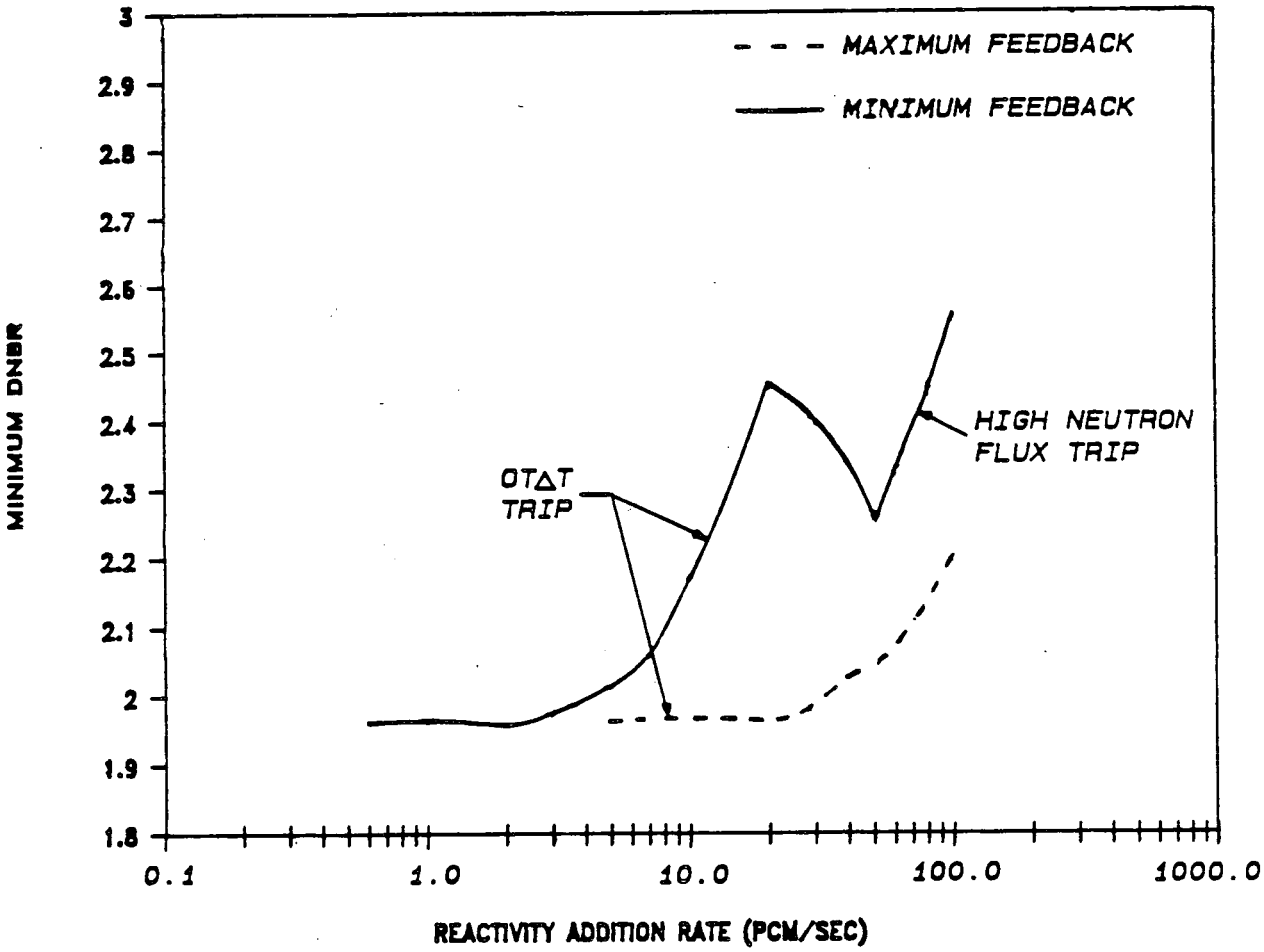

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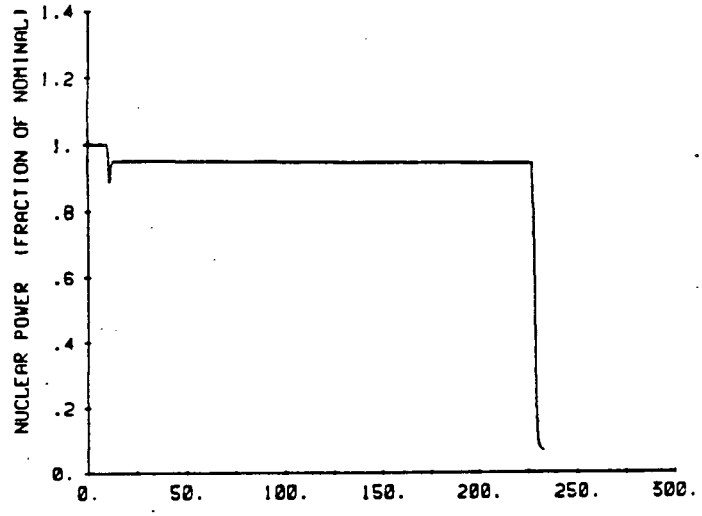
FIGURE A.3-9

MINIMUM DNBR VERSUS REACTIVITY INSERTION RATE, ROD WITHDRAWAL FROM 10 PERCENT POWER



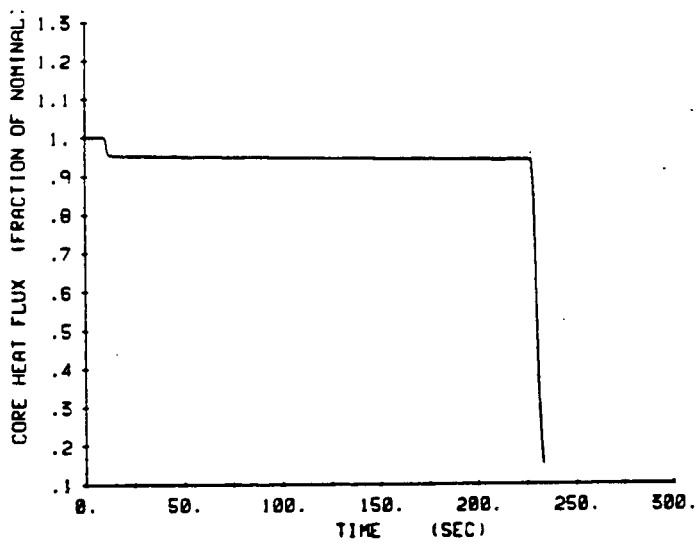


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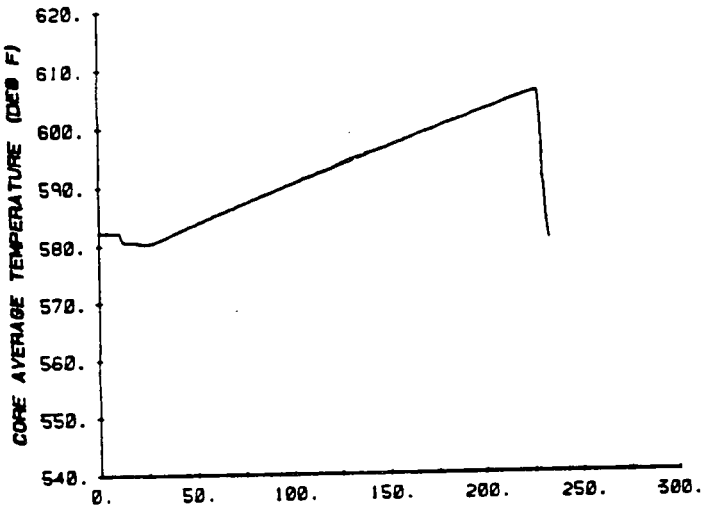
FIGURE A.3-10

DROPPED ROD INCIDENT
NUCLEAR POWER AND CORE HEAT FLUX
FOR DROPPED RCCA OF WORTH = 100 PCM



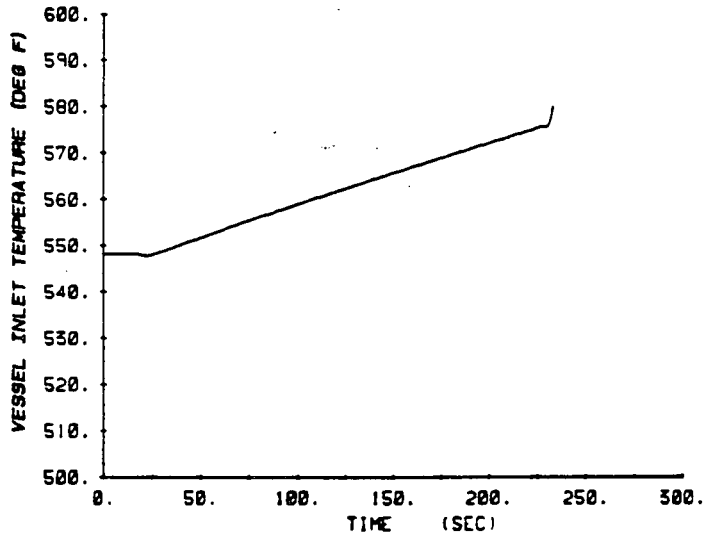


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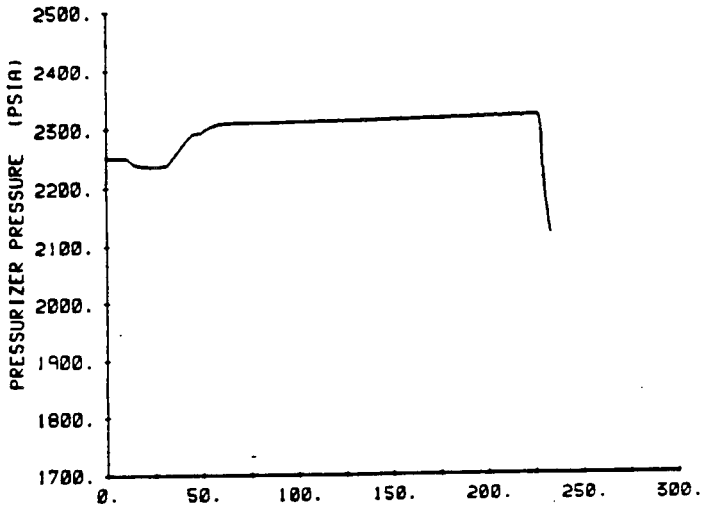
FIGURE A.3-11

DROPPED ROD INCIDENT
CORE AVERAGE AND INLET TEMPERATURE
FOR DROPPED RCCA OF WORTH = 100 PCM



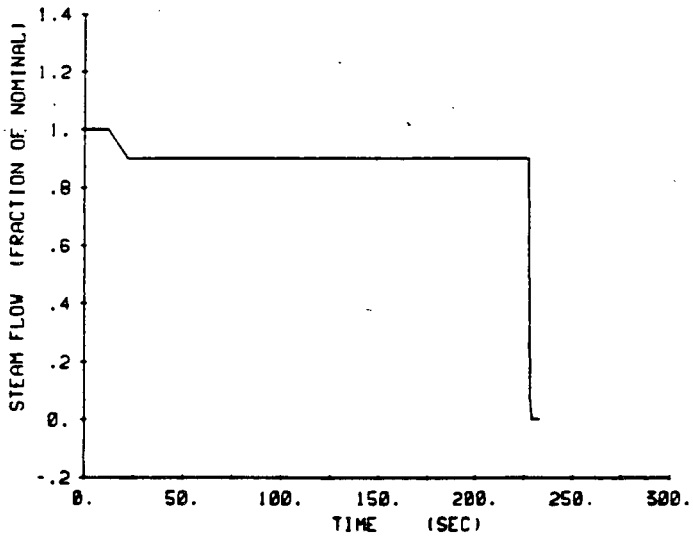


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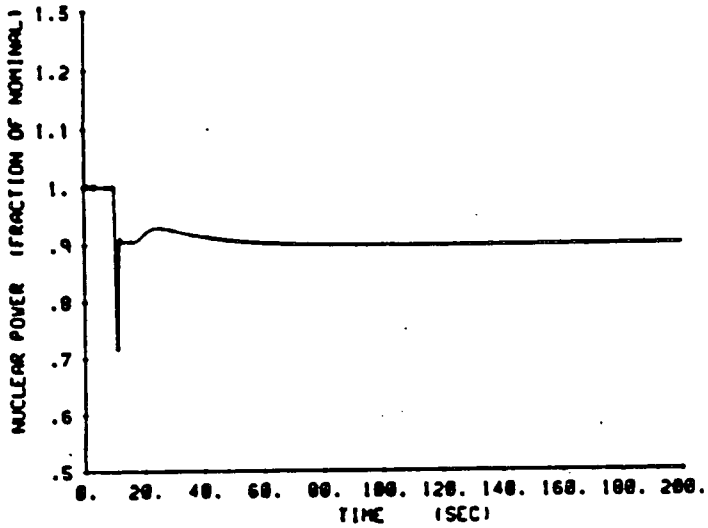
FIGURE A.3-12

DROPPED ROD INCIDENT
PRESSURIZER PRESSURE AND PLANT
STEAM FLOW FOR DROPPED RCCA
WORTH OF 100 PCM



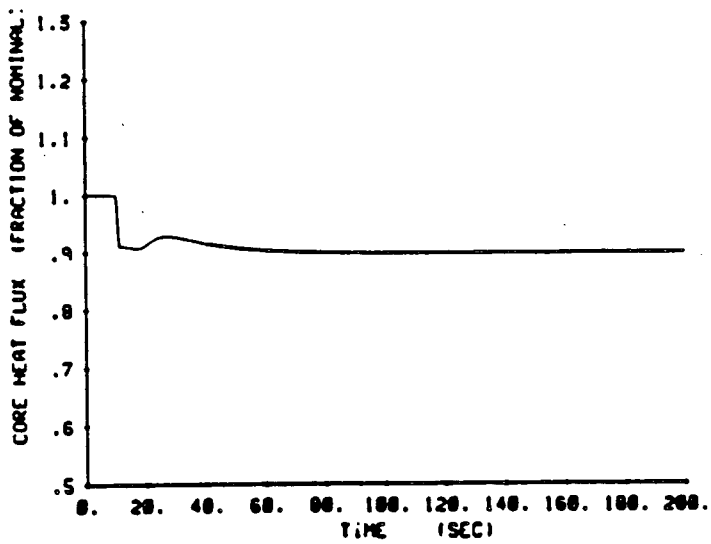


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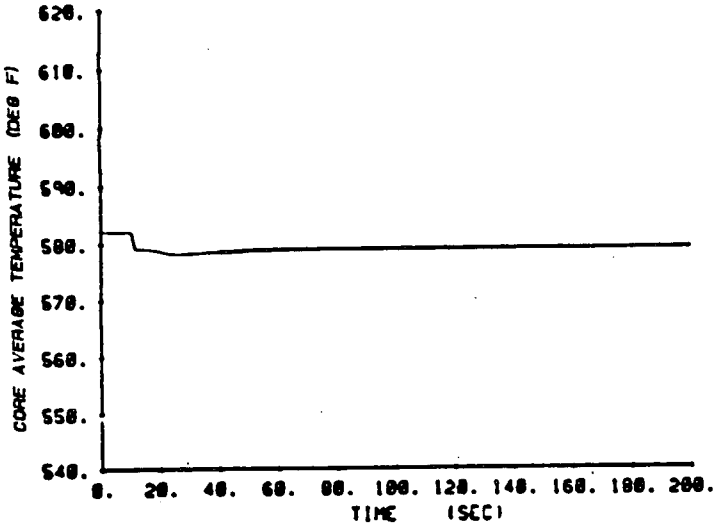
FIGURE A.3-13

DROPPED ROD INCIDENT
NUCLEAR POWER AND CORE HEAT FLUX
FOR DROPPED RCCA OF WORTH = 400 PCM



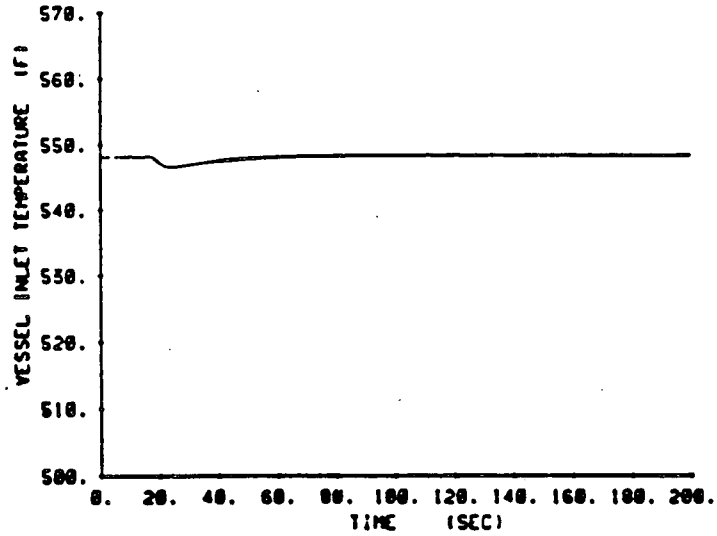


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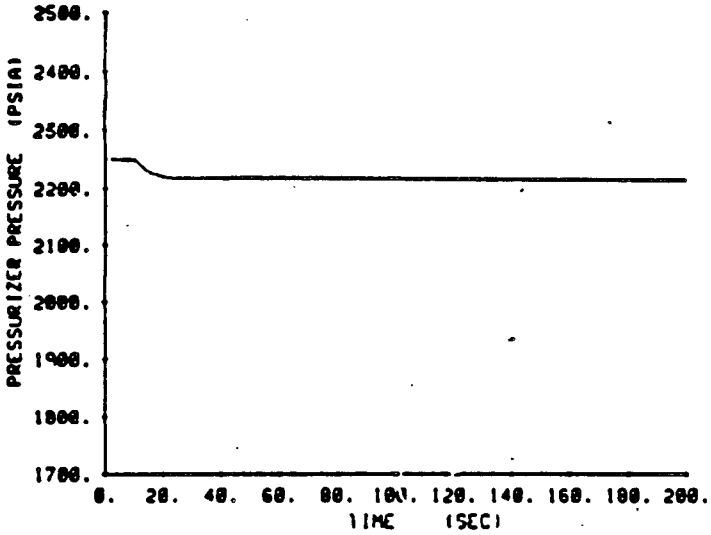
FIGURE A.3-14

DROPPED ROD INCIDENT
CORE AVERAGE AND INLET TEMPERATURE
FOR DROPPED RCCA OF WORTH = 400 PCM



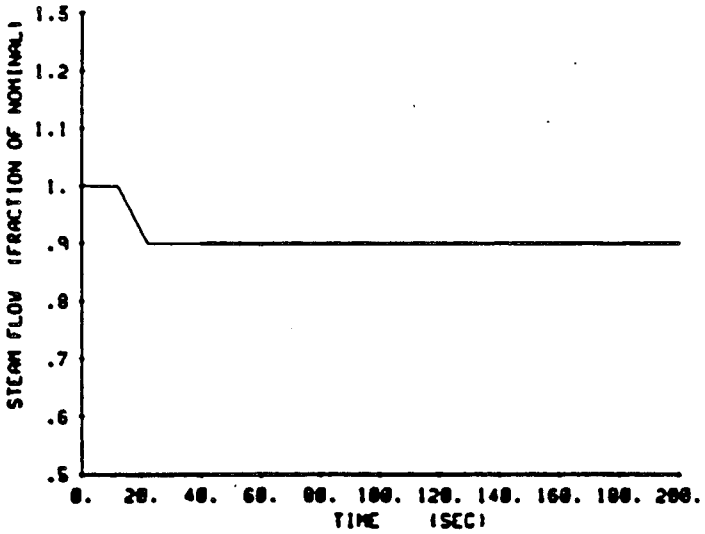


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FIGURE A.3-15

DROPPED ROD INCIDENT
PRESSURIZER PRESSURE AND PLANT
STEAM FLOW FOR DROPPED RCCA
WORTH OF 400 PCM

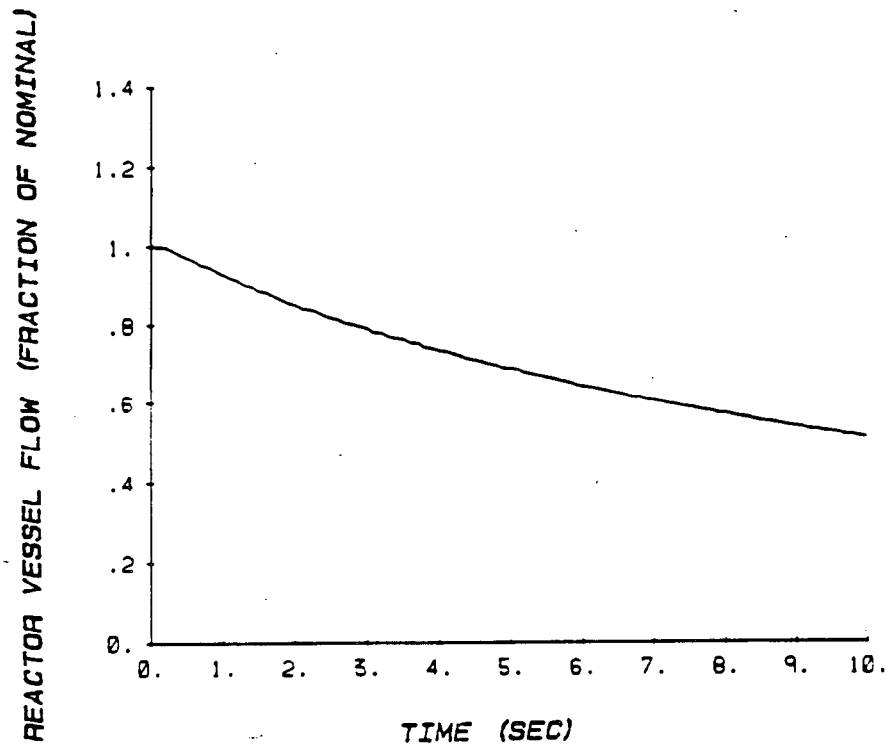

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FIGURE A.3-16

FOUR-PUMP LOSS OF FLOW INCIDENT
REACTOR VESSEL FLOW VS. TIME

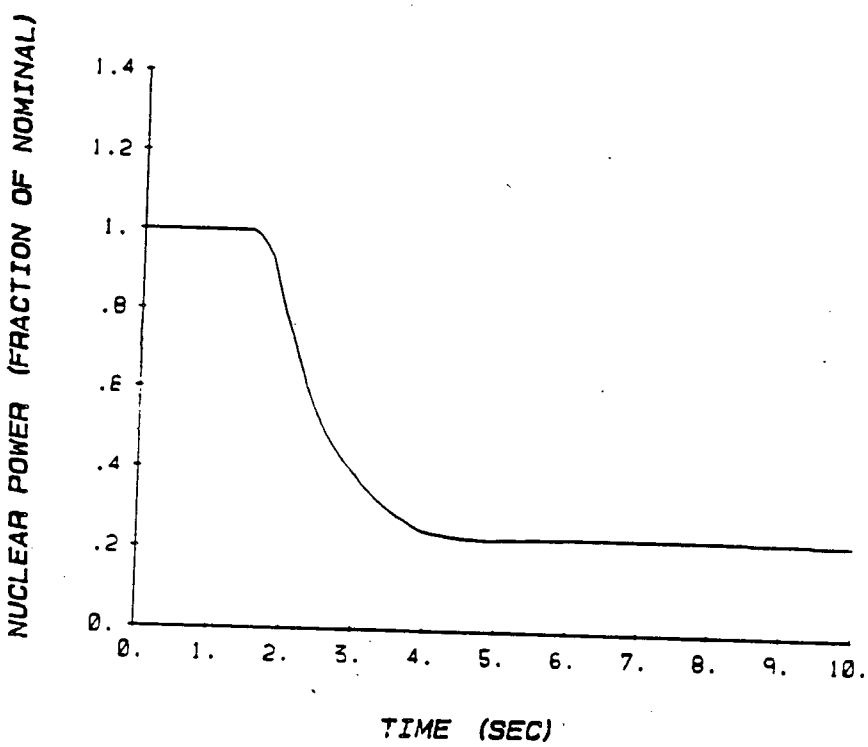

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FIGURE A.3-17

FOUR-PUMP LOSS OF FLOW INCIDENT
NUCLEAR POWER VS. TIME

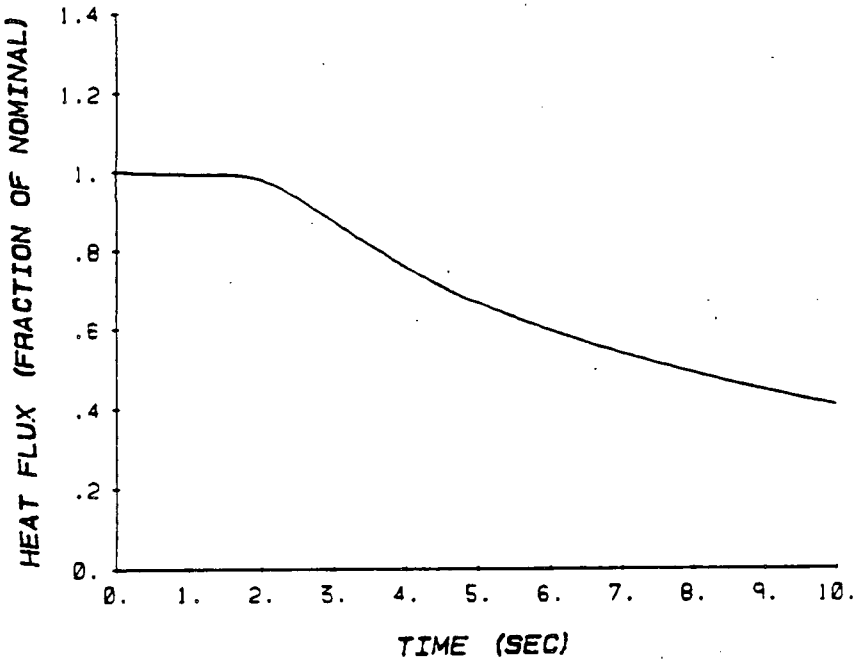

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FIGURE A.3-18

FOUR-PUMP LOSS OF FLOW INCIDENT
HEAT FLUX VS. TIME

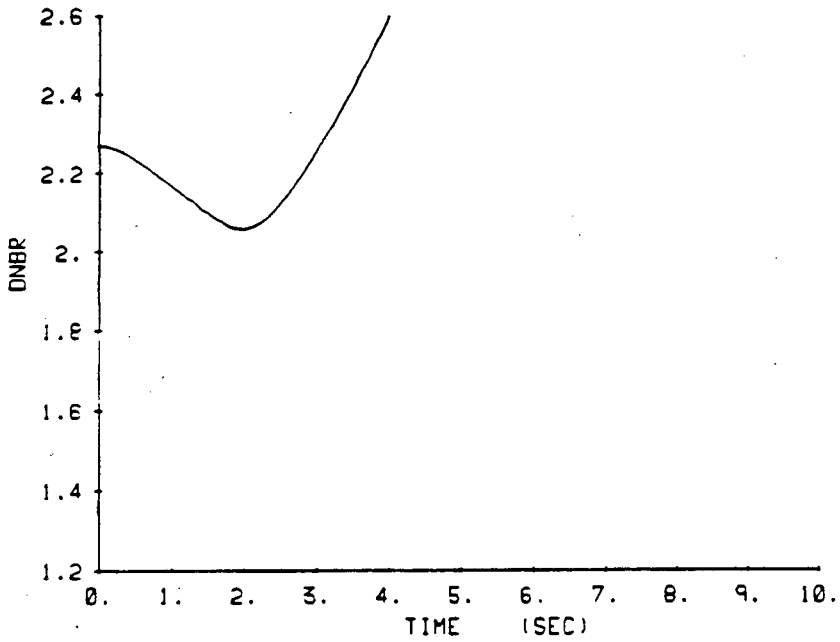

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FIGURE A.3-19

FOUR-PUMP LOSS OF FLOW INCIDENT
DNBR VS. TIME

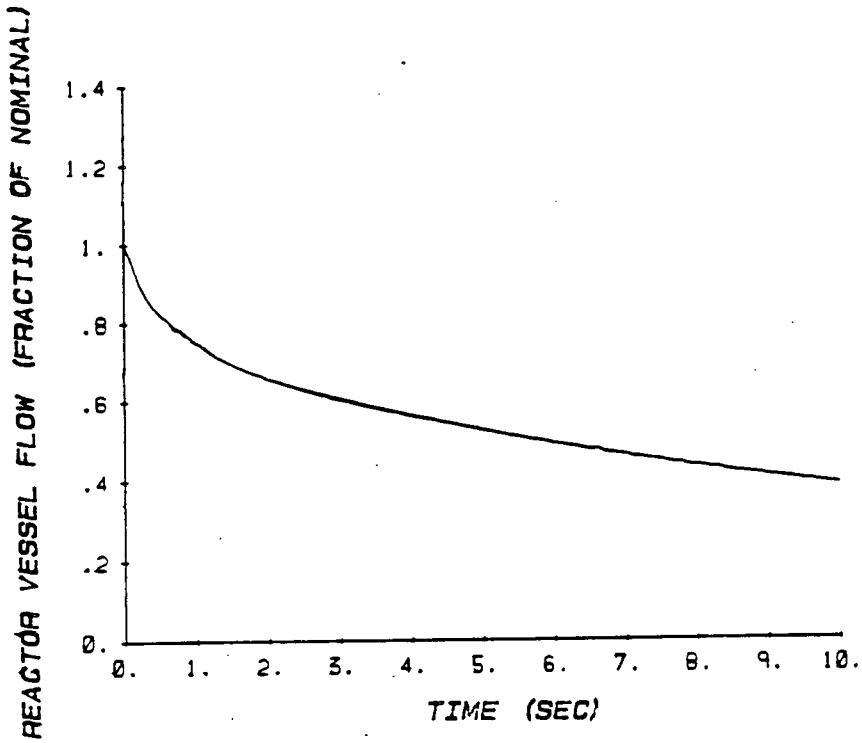

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FIGURE A.3-20

LOCKED ROTOR INCIDENT
REACTOR VESSEL FLOW VS. TIME

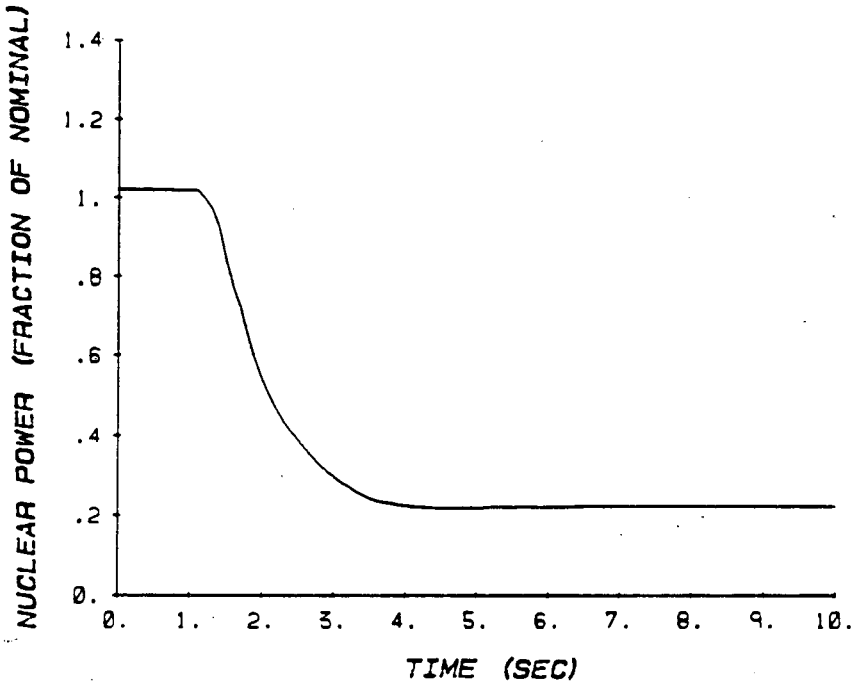

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FIGURE A.3-21

LOCKED ROTOR INCIDENT
NUCLEAR POWER VS. TIME

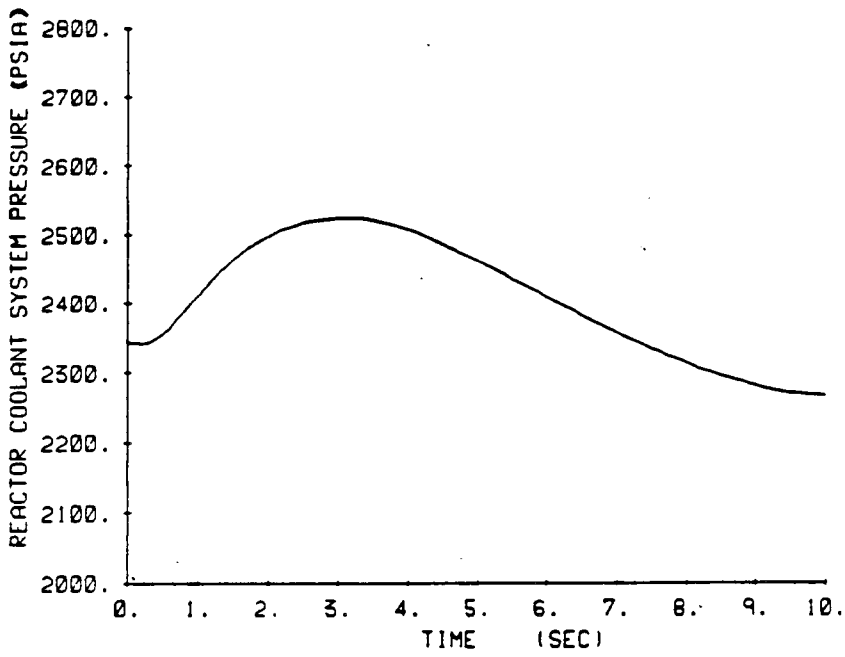

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FIGURE A.3-22

LOCKED ROTOR INCIDENT
RCS PRESSURE VS. TIME

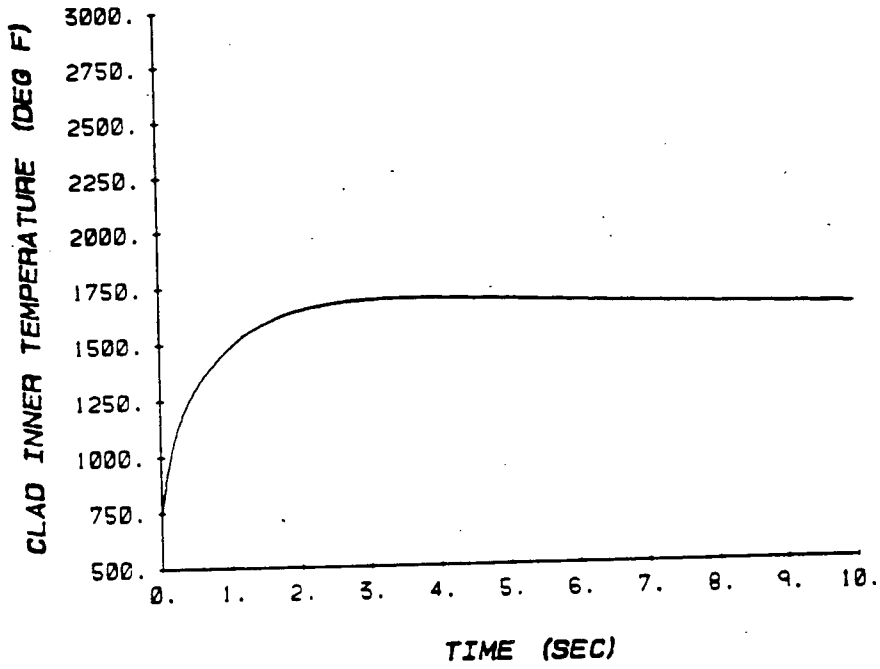

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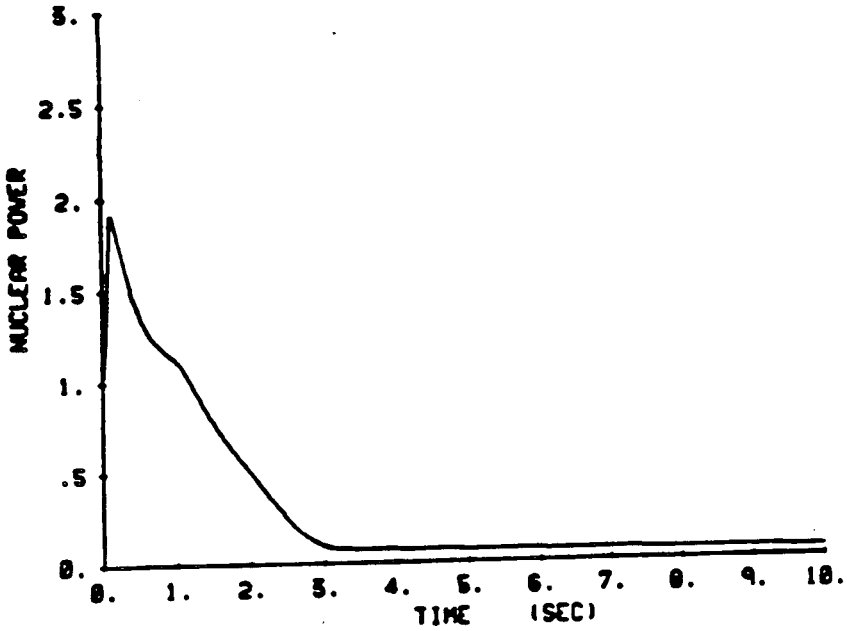
FIGURE A.3-23

LOCKED ROTOR INCIDENT
CLAD INNNER TEMPERATURE VS. TIME



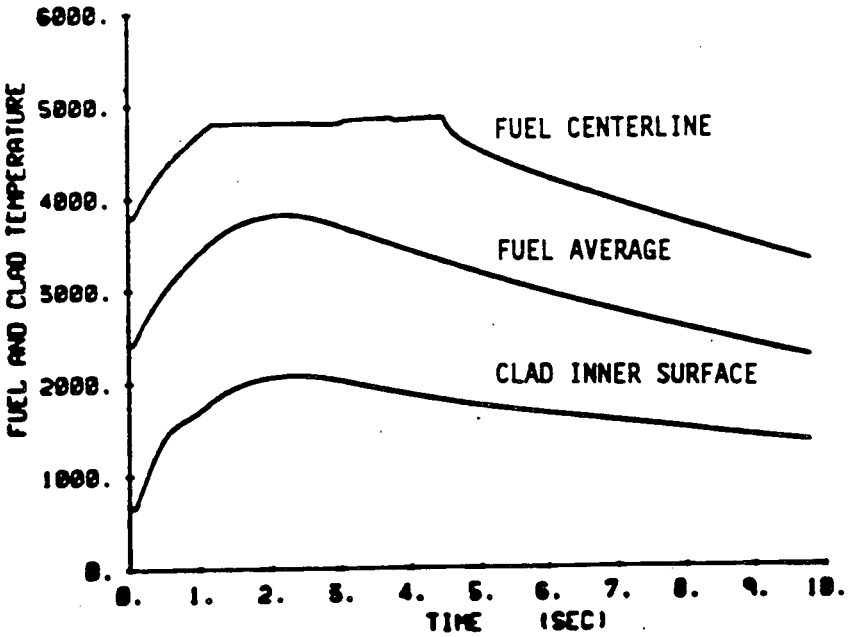


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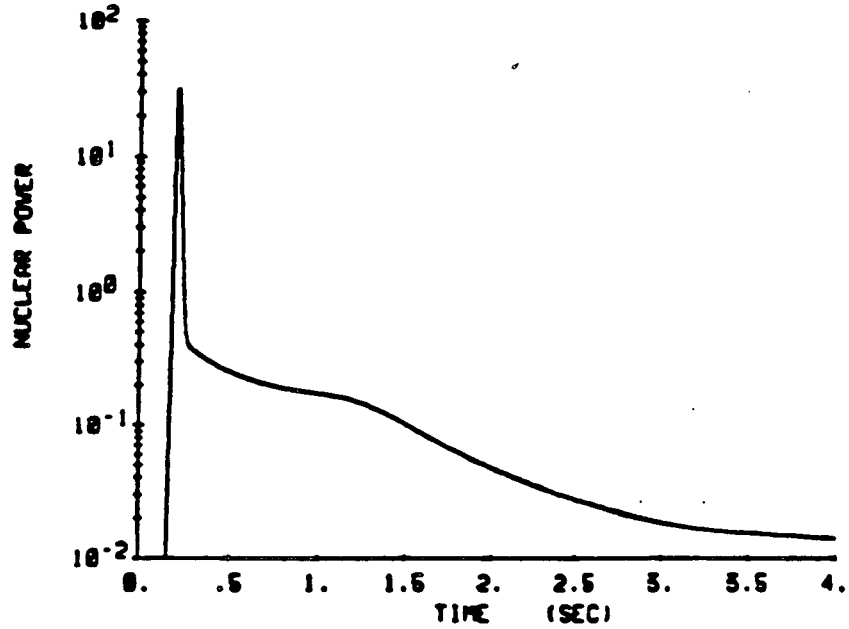
FIGURE A.3-24

ROD EJECTION INCIDENT
INITIAL CONDITIONS: EOL - FULL POWER
FRACTION OF NUCLEAR POWER AND TEMPERATURE



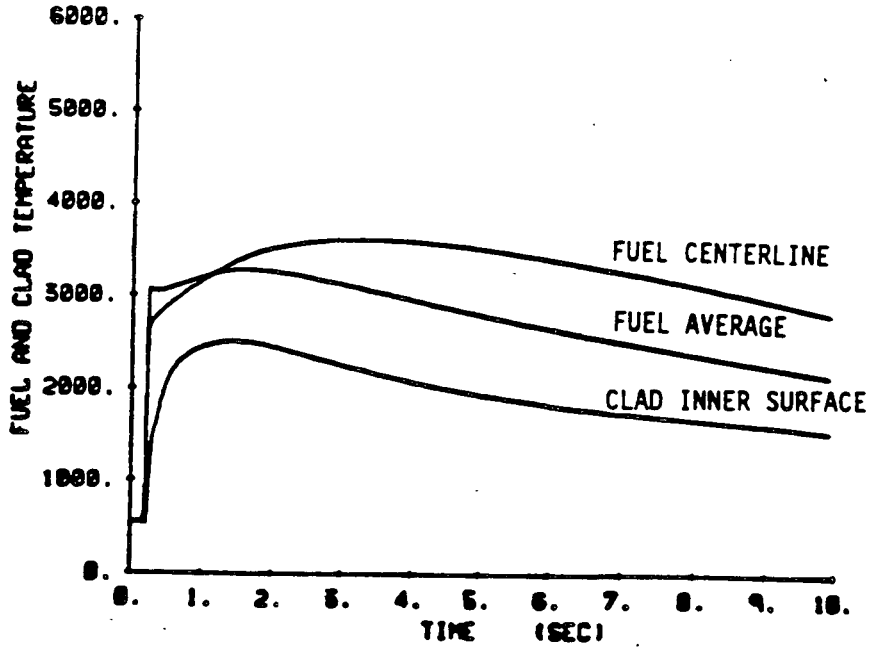


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FIGURE A.3-25

ROD EJECTION INCIDENT
INITIAL CONDITIONS: EOL - HOT ZERO POWER
FRACTION OF NUCLEAR POWER AND
TEMPERATURE

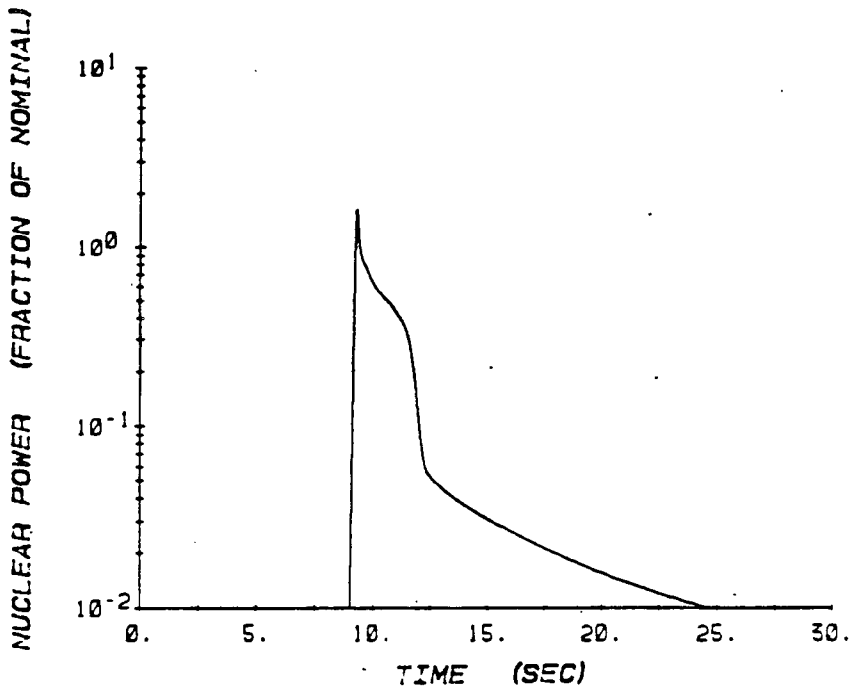

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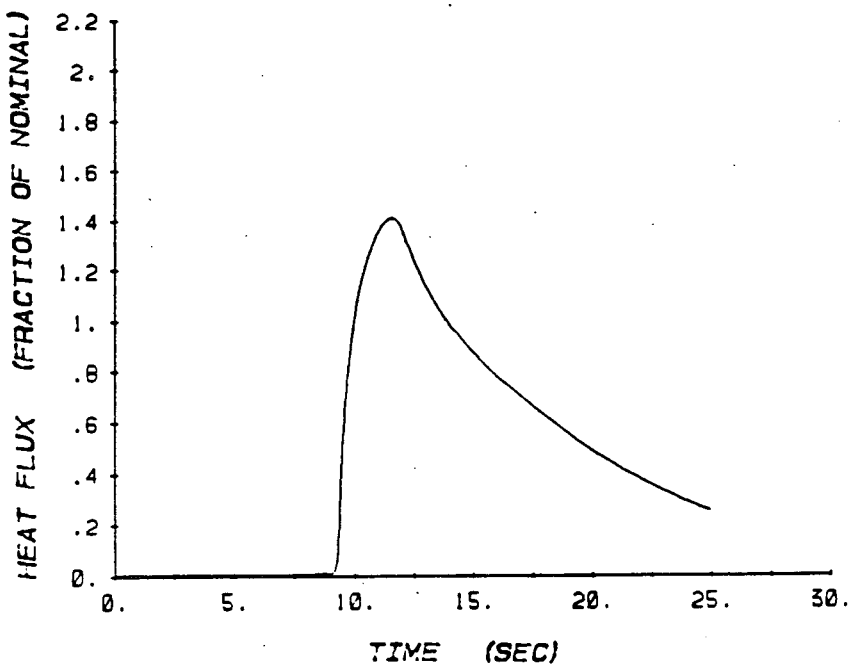
FIGURE A.3-26

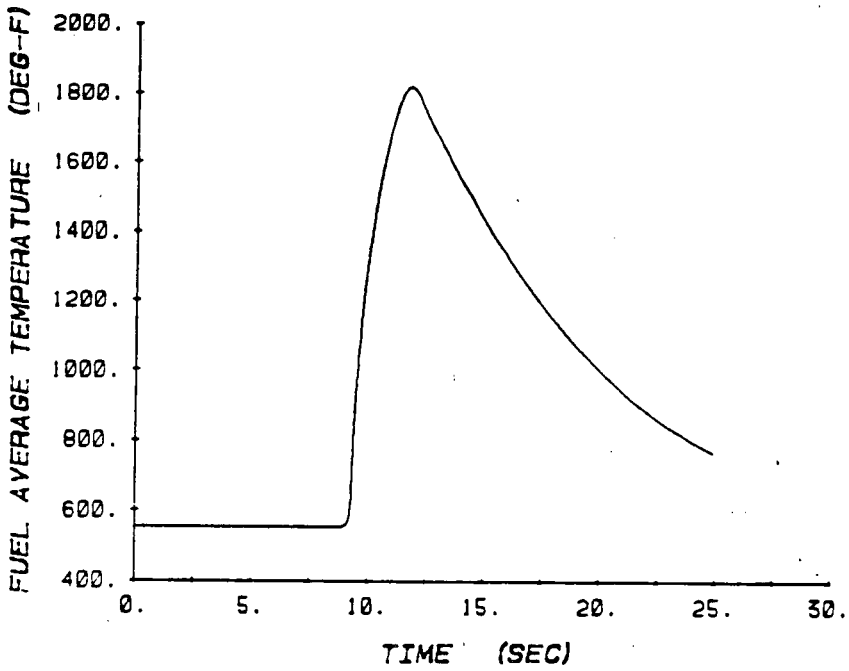
UNCONTROLLED RCCA WITHDRAWAL
FROM A SUBCRITICAL CONDITION
NUCLEAR POWER VS TIME



CONSOLIDATED EDISON CO.
INDIAN POINT 2

FIGURE A.3-27

UNCONTROLLED RCCA WITHDRAWAL
FROM A SUBCRITICAL CONDITION
HEAT FLUX VS TIME
AT HOT SPOT



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INDIAN POINT 2

FIGURE A.3-28

UNCONTROLLED RCCA WITHDRAWAL
FROM A SUBCRITICAL CONDITION
FUEL AVERAGE TEMPERATURE VS TIME
AT HOT SPOT

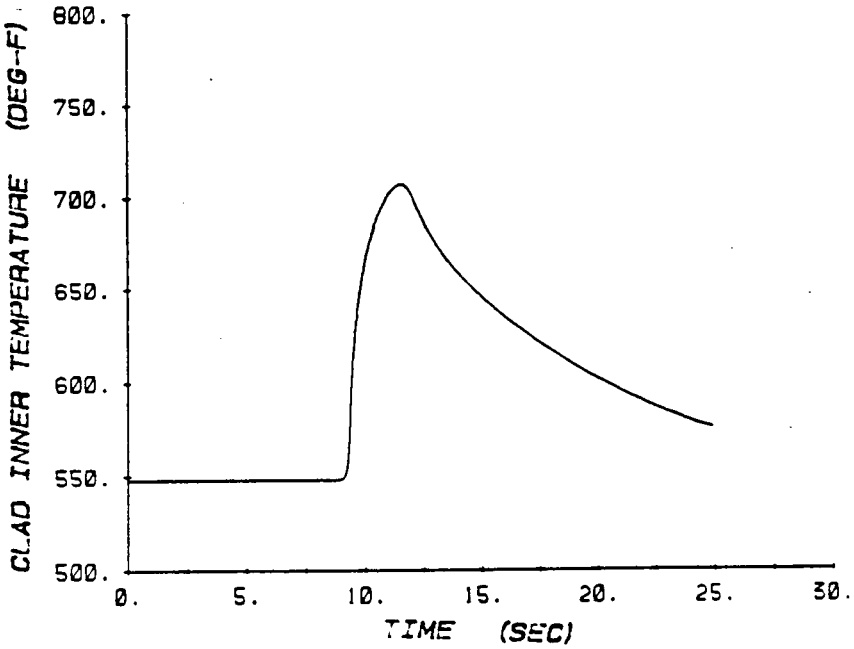

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FIGURE A.3-29

UNCONTROLLED RCCA WITHDRAWAL
FROM A SUBCRITICAL CONDITION
CLAD INNER TEMPERATURE VS TIME
AT HOT SPOT

Enclosure 2
to
Attachment C

LOCA Analysis

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
September 30, 1988

The following provides a brief summary of the Large Break (1981 EM with BART/BASH) and Small Break (NOTRUMP EM) LOCA analyses performed for Indian Point Unit 2. These analyses demonstrate the continued conformance of Indian Point Unit 2 to the Acceptance Criteria of 10CFR50.46. Section I provides a summary of the Large Break LOCA analysis. Section II provides a summary of the Small Break LOCA analysis.

Section I: Large Break LOCA

Introduction

This section reports the results of an analysis which was performed to demonstrate that the Indian Point Unit 2 Nuclear Plant conforms to the requirements of Title 10 of the Code of Federal Regulations, Part 50, Section46 (10CFR50.46, Ref. 1) in accordance with Appendix K (10CFR50, Appendix K) for Large Break Loss-of-Coolant Accidents (LOCA), and to allow operation of Indian Point Unit 2 at power levels up to 3083.4 MWt NSSS power, an increased hot channel enthalpy rise factor (F-DELTA-H) of 1.62 and transition to Westinghouse Optimized Fuel Assemblies (OFA) is assured. The analyses cover a full range of reactor vessel operating temperatures from 549°F to 579.7°F (T_{ave}).

Background and Assumptions

Pursuant to a proposed increase in authorized power level, an increase in F-DELTA-H from 1.55 to 1.62, and a transition to Westinghouse 15x15 OFA fuel, a Large Break LOCA analysis was performed for Indian Point Unit 2 to demonstrate the continued conformance of this unit to the Acceptance Criteria of 10CFR50.46. The analysis assumed a core thermal power level of 102% of 3071.4 MWt Core Power, an F-DELTA-H of 1.62 and 25% uniform Steam Generator Tube Plugging (SGTP). The core was assumed to consist of 193 assemblies of Westinghouse 15x15 Optimized (OFA) fuel.

Method of Analysis

The analysis was performed using the Westinghouse BASH Evaluation Model (Ref. 2) for a full spectrum of break discharge coefficients (C_E = 0.4, 0.6 and 0.8) at conditions consistent with a reactor vessel average temperature of 579.7°F (high T_{ave}). The limiting discharge coefficient was then reanalyzed at conditions consistent with a reactor vessel average temperature of 549°F (low T_{ave}).

The Westinghouse BASH Emergency Core Cooling System (ECCS) Large Break Evaluation Model was developed to determine the Reactor Coolant System (RCS) response to design basis large break LOCAs and consists of the SATAN-VI, WREFLOOD, COCO, BASH and LOCBART computer codes. The SATAN-VI code (Ref. 3) was used to generate the blowdown portion of the transient. The WREFLOOD code (Ref. 4) was used to calculate the refill portion of the transient. The COCO code (Ref. 5) operates interactively with the WREFLOOD code to evaluate the containment pressure response. The BASH code (Ref. 2) was used to calculate the system hydraulics during the reflood phase and cladding thermal analyses were performed with the LOCBART code. The LOCBART code (Ref. 2) is a synthesis of the LOCTA-IV (Ref. 6) and BART (Ref. 7) codes and uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the SATAN-VI, WREFLOOD and BASH codes as input. The hydraulic analyses and core thermal transient analyses assumed 102 percent of core thermal power. The pumped ECCS injection was assumed to be delivering to the RCS 25 seconds after the generation of a safety injection signal. The 25-second delay includes time required for diesel start-up and loading of the ECCS pumps onto the emergency buses. Minimum safeguards Emergency Core Cooling System capability and operability has been assumed in the full spectrum analysis. However, since the assumption of maximum safeguards has proven to be more limiting for some Westinghouse four-loop, non-upper-head injection, non-burst node limited plants, a maximum safeguards case for the most limiting discharge coefficient at the more limiting operating temperature condition was also performed.

Results

The transient was considered to be terminated when the hot rod clad average temperature "turned around" (i.e. - hot rod clad average temperature began to decline) indicating that the peak clad temperature had been reached. The analysis determined a peak clad temperature of 2039°F for the limiting discharge coefficient (C_D = 0.4) assuming minimum safeguards at the higher reactor vessel average temperature with a total peaking factor (F_Q) of 2.32. (Note: Applying the 10°F PCT penalty for the transition core reloads, the limiting case PCT will be 2049°F until an all OFA core loading is achieved.) The calculated Peak Clad Temperature for the limiting discharge coefficient (C_D = 0.4) at the lower operating temperature condition was 1919°F. The calculated peak clad temperature for the C_D = 0.6 and C_D = 0.8 cases (analyzed at the higher operating temperature) were 1940°F and 1749°F, respectively. The maximum safeguards case for the most limiting discharge coefficient (C_D = 0.4) and limiting operating temperature (High T_{have}) resulted in a peak clad temperature of 2015°F. This demonstrates the minimum safeguards case to be most limiting. The final peak clad temperatures for all the cases were determined to be below the 2200°F Acceptance Criteria limit established by 10CFR50.46.

Conclusions

For breaks up to and including the double-ended severance of a reactor coolant pipe, the emergency core cooling system will, under the operating conditions outlined above, satisfy the acceptance criteria as presented in 10CFR50.46. That is:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed one percent of the total amount of zircaloy in the reactor.

3. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time. This is required to remove the heat from the long-lived reactivity in the core.

The Large Break LOCA analysis for Indian Point Unit 2, utilizing the BASH model, resulted in a peak clad temperature of 2039°F for the limiting break case at a total peaking factor of 2.32 (2049°F for the OFA/LOPAR transition core reloads). The maximum local metal-water reaction was 4.94 percent, and the total metal-water reaction was less than 0.3 percent for all cases analyzed. The clad temperature turned around at a time when the core geometry is still amenable to cooling. Criteria 5 is addressed separately in a specific evaluation for each reload cycle.

A more complete description of the Large Break LOCA analysis and results is contained in the revisions to the Indian Point Unit 2 FSAR Chapter 14.3 which are attached.

Section II: Small Break LOCA

Introduction

This section reports the results of an analysis which was performed to demonstrate that Indian Point Unit 2 conforms to the requirements of Title 10 of the Code of Federal Regulations, Part 50, Section 46 (10CFR50.46, Ref. 1) in accordance with Appendix K (10CFR50, Appendix K) and Item II.K.3.30of NUREG 0737 for Small Break Loss-of-Coolant-Accidents (LOCA) for power levels of up 3083.4 MWt NSSS power, an increased hot channel enthalpy rise factor (F-DELTA-H) of 1.62 and transition to Westinghouse Optimized Fuel Assemblies for range of reactor vessel operating temperatures.

Background and Assumptions

Pursuant to a proposed increase in authorized power levels and an increase in F-DELTA-H from 1.55 to 1.62, a Small Break LOCA analysis was performed for the Indian Point Unit 2 Nuclear Plant to demonstrate the continued conformance of this unit to the Acceptance Criteria of 10CFR50.46. The analysis assumed a core thermal power level of 102% of 3071.4 MWt Core Power, an F-DELTA-H of 1.65 (which conservatively bounds 1.62) and 25% uniform Steam Generator Tube Plugging (SGTP). The core was assumed to consist of 193 assemblies of Westinghouse 15x15 LOPAR (STD) fuel, however, an evaluation has been performed to demonstrate that the results of this analysis are applicable to both the 15x15 Optimized (OFA) fuel and the transition core. The power shape used for the Small Break LOCA analysis was chosen to allow an increase in the peaking factor at the top of the core from 1.0 to 1.5.

Method of Analysis

The analysis was performed using the Westinghouse NOTRUMP Small Break Evaluation Model (Ref. 8) for a full spectrum of break sizes (4 in., 6 in. and 8 in.) at conditions consistent with a reactor vessel average temperature of 549°F. The limiting break size was then analyzed at conditions consistent with a reactor vessel average temperature of 579.7°F.

The Westinghouse NOTRUMP Emergency Core Cooling System (ECCS) Small Break Evaluation Model was developed to determine the Reactor Coolant System (RCS) response to design basis Small Break LOCAs and consists of the NOTRUMP and LOCA-IV computer codes (References 9 and respectively).

The NOTRUMP code was used to calculate the system hydraulics throughout the transient. The LOCA-IV code which calculated the cladding thermal response uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP code to determine the peak clad temperature during the Small Break LOCA. The hydraulic analyses and core thermal transient analyses assumed 102 percent of core thermal power. The pumped ECCS injection was assumed to be delivering to

the RCS 25 seconds after the generation of a safety injection signal. The 25-second delay includes time required for diesel start-up and loading of the ECCS pumps onto the emergency buses. Minimum safeguards Emergency Core Cooling System capability and operability has been assumed in the analysis.

Results

The transient was considered to be terminated when the hot rod clad average temperature "turned around" (i.e. - hot rod clad average temperature began to decline) indicating that the peak clad temperature had been reached. The analysis determined a peak clad temperature of 1218.5°F for the limiting break (6 in.) at the higher operating temperature condition consistent with an F_Q envelope based on a total peaking factor of 2.32. The calculated peak clad temperature for the 6 in. break at the lower operating temperature condition was 998.4°F. The calculated peak clad temperature for the 4 in. break was 721.2°F, while the 8 in. case demonstrated no core uncovery (no PCT reported). All peak clad temperatures for the Small Break analyses were below the values determined by the Large Break analyses and were determined to be below the 2200°F Acceptance Criteria limit established by 10CFR50.46.

Conclusions

In the event of a Small Break LOCA the emergency core cooling system provides adequate core protection, under the operating conditions outlined above, by satisfying the acceptance criteria of 10CFR50.46. That is:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed one percent of the total amount of zircaloy in the reactor.

3. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time. This is required to remove the heat from the long-lived reactivity in the core.

The Small Break LOCA analysis for Indian Point Unit 2, utilizing the NOTRUMP model, resulted in a peak clad temperature of 1218.5°F for the limiting break case and limiting operating temperature condition for an F_Q envelope based on a total peaking factor of 2.32. The maximum local metal-water reaction was 0.008 percent, and the total metal-water reaction was less than 0.3 percent for all cases analyzed. The clad temperature turned around at a time when the core geometry is still amenable to cooling. Criteria 5 is addressed separately in a specific evaluation for each reload cycle.

A more complete description of the Small Break LOCA analysis and results is contained in the revisions to the Indian Point Unit 2 FSAR Chapter 14.3 which are attached.

References

1. "Acceptance Criteria for Emergency Core Cooling System for Light Water Cooled Nuclear Power Reactors," 10CFR50.46 and Appendix K of 10CFR50. Federal Register, Volume 39, Number 3, January 4, 1974.

2. Kabadi, J. N., et. all., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266, Revision 2, August 1986, (Westinghouse Proprietary).

3. Bordelon, F. M., et. all., "SATAN-VI Program: Comprehensive Space Time Dependent Analysis of Loss of Coolant," WCAP-8302, (Proprietary) June 1974, and WCAP-8306, (Non-Proprietary), June 1974.

4. Kelly, R. D., et. all., "Calculated Model for Core Reflooding After a Loss of Coolant Accident (WREFLOOD Code)," WCAP-8170 (Proprietary) and WCAP-8171 (Non-Proprietary), June 1974.

5. Bordelon, F. M. and Murphy, E. T., "Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary) and WCAP-8326 (Non-Proprietary), June 1974.

6. Bordelon, F. M., et. all., "LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-8301, (Proprietary) and WCAP-8305, (Non-Proprietary), June 1974.

7. Young, M., et. all., "BART-1A: A Computer Code for the Best Estimate Analysis of Reflood Transients," WCAP-9561-P-A-, 1984 (Westinghouse Proprietary).

8. Lee, N., Rupprecht, S. D., Schwarz, W. R., Tauche, W. D., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-A (Non-Proprietary) August 1985.

9. Meyer, P. E., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Non-Proprietary), August 1985.

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14.3 LOSS-OF-COOLANT ACCIDENTS

14.3.1 IDENTIFICATION OF CAUSES AND FREQUENCY CLASSIFICATION

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the reactor coolant system pressure boundary. A major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered a limiting fault, an ANS Condition IV event, in that it is not expected to occur during the lifetime of the plant, but is postulated as a conservative design basis.

A minor pipe break (small break) is defined as a rupture of the reactor coolant pressure boundary with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered an ANS Condition III event in that it is an infrequent fault which may occur during the life of the plant.

The acceptance criteria for the loss-of-coolant accident are described in 10 CFR 50 Paragraph 46 (Reference 1), as follows:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time. This is required to remove the heat from the long lived radioactivity remaining in the core.

These criteria were established to provide significant margin in emergency core cooling system performance following a LOCA. Reference 2 presents a recent study in regard to the probability of occurrence of reactor coolant system pipe failures.

In all cases, small breaks (less than 1.0 ft^2) yield results with more margin to the acceptance criteria limits than do large breaks.

14.3.2 SEQUENCE OF EVENTS AND SYSTEMS OPERATIONS

Should a major break occur, the depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. A safety injection actuation signal is generated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in the core in causing a rapid reduction of power to a residual level corresponding to fission product decay heat.

2. The injection of borated water provides for heat transfer from the core, prevents excessive clad temperatures, and maintains subcriticality.

14.3.2.1 Description of Large-Break LOCA Transient

A typical sequence of events following a large-break LOCA is presented in Figure 14.3-1.

Before the break occurs, the reactor is in an equilibrium condition, that is, the heat generated in the core is being removed via the secondary system.

During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. Thereafter, the core heat transfer is based on local conditions with transition boiling, film boiling and forced convection to steam as the major heat transfer mechanisms.

The heat transfer between the reactor coolant system and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, secondary system pressure increases and the main steam safety valves may actuate to limit the pressure. Makeup water to the secondary side is automatically provided by the auxiliary feedwater system. The safety injection actuation signal isolates the steam generators from normal feedwater flow and initiates emergency flow from the auxiliary feedwater system. The secondary flow aids in the reduction of reactor coolant system pressure.

When the reactor coolant system depressurizes to 630 psia, the accumulators begin to inject borated water into the reactor coolant loops. Since the loss of offsite power is assumed, the reactor coolant pumps are assumed to trip at the inception of the accident. The effects of pump coastdown are included in the blowdown analysis.

The blowdown phase of the transient ends when the reactor coolant system pressure (initially assumed at 2280 psia) falls to a value approaching that of the containment atmosphere. Prior to, or at the end of the blowdown, some amount of injection water begins to enter the reactor vessel lower plenum. At this time (called end of bypass), refill of the reactor vessel lower plenum begins. Refill is complete when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods (called bottom of core recovery time).

The reflood phase of the transient is defined as the time period lasting from the end of refill until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the later

stage of blowdown and then beginning of reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the reactor coolant system, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The residual heat removal (low-head) and safety injection (high-head) pumps aid the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process.

Continued operation of the emergency core cooling system pumps supplies water during long-term cooling. Core temperatures would be reduced to long-term steady-state levels associated with the dissipation of residual heat generation. After the water level of the refueling water storage tank reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold-leg recirculation mode of operation in which spilled borated water is drawn from either the recirculation sump or containment sump by the recirculation or residual heat removal pumps and returned to the reactor coolant system cold legs. The containment spray pumps continue to operate drawing water from the refueling water storage tank (see Section 6.3.3) for further reduction of containment pressure. Approximately 24 hours after initiation of the LOCA, the emergency core cooling system is realigned to supply water to the reactor coolant system hot legs in order to control the boric acid concentration in the reactor vessel.

14.3.2.2 Description of Small-Break LOCA Transient

As contrasted with the large break, the blowdown phase of the small break occurs over a longer time period. Thus, for the small-break LOCA, there are three characteristic stages: (1) a gradual blowdown in which the decrease in water level is checked, (2) core uncovery, and (3) long-term recirculation. There is a slight increase in containment pressure.

14.3.3 CORE AND SYSTEM PERFORMANCE

14.3.3.1 Mathematical Model

The requirements of an acceptable emergency core cooling system evaluation model are presented in Appendix K of 10 CFR 50.[1]

14.3.3.1.1 Large-Break LOCA Evaluation Model

The analysis of a large break LOCA transient is divided into three phases: (1) blowdown, (2) refill, and (3) reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the reactor coolant system, the pressure and temperature transient within the containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. From these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

The description of the various aspects of the LOCA analysis methodology is given in References 3, 9, 56, 58, 64, 65 and 66. These documents describe the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the acceptance criteria. The SATAN-VI, WREFLOOD, COCO, and LOCTA-IV codes which are used in the LOCA analysis, are described in detail in References 4 through 7. Modifications to these codes are specified in References 9, 11 and 56. The BART code is described in References 58 and 64. Modifications to the BART code and BART methodology are described in References 59 and 60. The BASH and LOCBART codes are described in References 65 and 66. These codes are used to assess the core heat transfer geometry and to determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the reactor coolant system during blowdown, and the WREFLOOD and BASH computer codes are used to calculate this transient during the refill and reflood phases of the accident. The BART computer code is used to calculate the fluid and

heat transfer conditions in the core during reflood. The COCO computer code is used to calculate the containment pressure transient during all three phases of the LOCA analysis. Similarly, the LOCBART computer code is used to compute the thermal transient of the hottest fuel rod during the three phases.

The large break analysis was performed with the approved 1981 version of the Evaluation Model (Reference 56) with the approved version of BASH (References 65 and 66).

SATAN-VI is used to calculate the reactor coolant system pressure, enthalpy, density, and the mass and energy flow rates in the reactor coolant system, as well as energy transfer between the primary and steam-generator secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal energy flow rates that are assumed to be vented to the containment during blowdown. During blowdown, no credit is taken for rod insertion or boron content of the injected water. The core will shut down as a result of void formation. At the end of the blowdown phase, these data are transferred to the WREFLOOD code. Also at the end of blowdown, the mass and energy release rates during blowdown are transferred to the COCO code for use in the determination of the containment pressure response during this first phase of the LOCA. Additional SATAN-VI output data from the end of blowdown, including the core pressure, and the core power decay transient, are input to the LOCBART code.

With input from the SATAN-VI Code, WREFLOOD uses a system thermal-hydraulic model to determine the coolant pressure and temperature, and the quench front height during the refill phase of the LOCA. WREFLOOD also calculates the mass and energy flow addition to the containment through the break. Since the mass flow rate to the containment depends upon the core flooding rate and the local core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked.

BASH is an integral part of the ECCS evaluation model which provides a realistic thermal-hydraulic simulation of the reactor core and RCS during the reflood phase of a LOCA. Instantaneous values of accumulator conditions and safety injection flow at the time of completion of lower plenum refill are provided to BASH from WREFLOOD. BASH has been substituted for WREFLOOD in calculating transient values of core inlet flow and enthalpy for the detailed fuel rod model, LOCBART. A more detailed description of the BASH code is contained in Reference 65. The BASH code provides a sophisticated treatment of steam/water flow phenomena in the reactor coolant system during core reflood. A more dynamic interaction between the core thermal-hydraulics and system behavior is expected, and recent experiments have borne this out. In the BASH code reflood model, BART provides the entrainment rate for a given flooding rate, and then a system model determines loop flows and pressure drops in response to the calculated core exit flow. An updated inlet flow is used to calculate a new entrainment rate. This system produces a very dynamic flooding transient, which reflects the close coupling between core thermal-hydraulics and loop behavior.

The COCO code is a mathematical model of the containment. COCO is run using mass and energy releases to the containment provided by SATAN and WREFLOOD. COCO is described in detail in Reference 6.

The LOCBART code is a coupling of LOCTA-IV and BART. The LOCTA-IV code is a computer program that evaluates fuel, cladding and coolant temperatures during a LOCA. A more complete description than is presented here can be found in Reference 7. In the LOCTA detailed fuel rod model, for the calculation of local heat transfer coefficients, the empirical FLECHT correlation is replaced by the BART code. BART employs rigorous mechanistic models to generate heat transfer coefficients appropriate to the actual flow and heat transfer regimes experienced by the LOCTA fuel rods. This is considered a more dynamic realistic approach than relying on a static empirical correlation.

A schematic representation of the computer code interfaces is given in Figure 14.3-2.

14.3.3.1.2 Small Break LOCA Evaluation Model

For loss-of-coolant accidents due to small breaks less than 1 square foot, the NOTRUMP[61] computer code is used to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of flow through the break. The NOTRUMP computer code is a state-of-the-art one-dimensional general network code consisting of a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA emergency core cooling system (ECCS) evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants".

In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modelled explicitly, with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. A detailed description of the NOTRUMP code is provided in References 61 and 62.

The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident.

The small-break analysis was performed with the approved Westinghouse emergency core cooling system Small Break Evaluation Model (References 9, 14, and 18).

The NOTRUMP and LOCTA-IV computer codes are used in the analysis of loss-of-coolant accidents due to small breaks in the Reactor Coolant System. The NOTRUMP computer code is a state-of-the-art one-dimensional general network code consisting of a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flow limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA emergency core cooling system (emergency core cooling system) evaluation model was developed to determine the reactor coolant system response to design basis small break LOCAs and to address the Nuclear Regulatory Commission (NRC) concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants."

In NOTRUMP, the reactor coolant system is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy and momentum applied throughout the system. A detailed description of NOTRUMP is given in References 14 and 18.

The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant transient.

Clad thermal analyses are performed with the LOCTA-IV code,[7] which uses as input the reactor coolant system pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations.

For these analyses, the safety injection delivery considers pumped injection flow (which is depicted in Figure 14.3.3) as a function of reactor coolant system pressure. This figure represents injection flow from the high-head and low-head residual heat removal safety injection pumps as determined by performance curves degraded 5 percent from the design head. A 25-sec delay was assumed, which includes the time required for diesel startup, and loading of the safety injection pumps onto the emergency buses. (The modeling of a step from zero flow to full flow at 25 seconds is conservative with respect to ramped flow which reaches full flow at 27 seconds.) The effect of low-head safety injection pump (residual heat removal pump) flow is not considered since the shutoff head is lower than reactor coolant system pressure during the time portion of the transient. Also, minimum safeguards emergency core cooling system capability and operability have been assumed in these analyses.

Figure 14.3-4 presents the hot rod power shape used to perform the small-break analysis presented here. This power shape was chosen because it provides an appropriate distribution of power versus core height and also local power is maximized in the upper regions of the reactor core (10 to 12 ft). This power shape is skewed to the top of the core with the peak local power occurring at the 10.0-foot core elevation. This is limiting for the small-break analysis because of the core uncovery process for small breaks. As the core uncovers, the cladding in the upper elevation of the core heats up and is sensitive to the local power at that elevation. The cladding temperatures in the lower elevation of the core below the two-phase mixture height remain low. The peak clad temperature occurs above 10 ft.

A schematic representation of the computer code interfaces is given in Figure 14.3-5.

The small break analysis was performed with the Westinghouse ECCS Small Break Evaluation Model using the NOTRUMP code, approved for this use by the Nuclear Regulatory Commission in May 1985 (Reference 62).

14.3.3.2

Table 14.3-1 lists important input parameters and initial conditions used in the analysis.

The analysis was performed with the upper head fluid temperature equal to the reactor coolant system hot leg fluid temperature and a range of vessel average temperatures between 549°F and 579.7°F. In addition, the analysis includes the effects of 25 percent uniform steam generator tube plugging.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from extensive sensitivity studies.[17-19, 58-60] In addition, the requirements of 10CFR50, Appendix K, regarding specific model features were met by selecting models that provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time the loss-of-coolant accident (LOCA) occurs and include such items as the core peaking factors, the containment pressure, and the performance of the emergency core cooling system. Decay heat generated throughout the transient is also conservatively calculated.

Emergency core cooling system (ECCS) analyses normally assume minimum safeguards to determine safety injection flow. This minimizes the amount of flow to the reactor coolant system by assuming maximum line resistances, degraded ECCS pump performance, and the loss of one residual heat removal pump as the limiting single failure. However, for some Westinghouse four-loop, non-upper-head injection, non-burst node limited plants, the current nature of the Appendix K ECCS evaluation models is such that it may be more limiting to assume maximum possible ECCS flow delivery. A maximum safeguards case is therefore included in the large-break LOCA analysis. The maximum safeguards assumption includes minimum injection line resistance, enhanced ECCS pump performance and no single failure and thus results in the highest amount of flow delivered to the reactor coolant system.

The initial conditions within the containment before accident initiation and the containment heat sink data used in the analysis are given in Table 14.3-2.

14.3.3.3 Large-Break Results

From the results of the LOCA sensitivity studies,[18,19] the limiting large break was found to be the double-ended cold-leg guillotine (DECLG). Therefore, only the DECLG break is considered in the large break emergency core cooling system performance analysis. To bound a range of Reactor Vessel average temperatures, a spectrum of Moody discharge coefficients (0.4, 0.6 and 0.8) were analyzed at the conditions consistent with the upper end of the T_{ave} range. The limiting discharge coefficient was then analyzed at the conditions consistent with the lower end of the T_{ave} range. The analysis based on maximum safeguards assumptions was then performed for the limiting discharge coefficient and the limiting Reactor Vessel average temperature condition. The results of these calculations are summarized in Tables 14.3-3 and 14.3-4.

As shown in Table 14.3-3, C_D=0.4 (High T_{ave}) minimum safeguards is the limiting case. A C_D=0.4 break size is the lower limit of break sizes analyzed as DECLG breaks, since a C_D=0.2 case has no meaning for bracketing purposes and is therefore not analyzed. Thus, the limiting break size at 25 percent tube plugging has been identified (i.e., C_D=0.4 (High T_{ave}) MINSI).

Table 14.3-3 presents selected results from the hot rod thermal transient calculation. For these results, the hot spot is defined as the location of maximum peak clad temperatures and is specified for each break analyzed. The location indicated is in feet and represents the elevation above the bottom of the active fuel stack.

Table 14.3-4 presents the occurrence times for various events throughout the accident transient.

Tables 14.3-5 and 14.3-6 present reflood mass and energy releases to the containment and the broken loop accumulator mass and energy release to the containment, respectively.

Figures 14.3-6a through 14.3-52 present the transients for the principal parameters for the break sizes analyzed. The following items are noted:

Figures 14.3-6a through 14.3-14	These figures show the quality, mass velocity and the clad heat transfer coefficient for the hot spot and burst locations.
Figures 14.3-15a through 14.3-23	These figures show the core pressure, break flow, and core pressure drop. The break flow is the sum of the flowrates from both ends of the guillotine break. The core pressure drop is taken as the pressure just beyond the core outlet.
Figures 14.3-24a through 14.3-32	These figures show the clad temperature, fluid temperature and core flow. The clad and fluid temperatures are given for the hot spot and burst locations.
Figures 14.3-33a through 14.3-38	These figures show the downcomer and core water level during reflood and the flooding rate.
Figures 14.3-39a through 14.3-44	emergency core cooling system flowrates, for both accumulator and safety injection.
Figures 14.3-45a through 14.3-50	These figures show the containment pressure and core power transient.
Figures 14.3-51 and 14.3-52	These figures show the break energy release during blowdown and the containment wall condensing heat transfer coefficient for the worst break.

The results show that for reactor coolant breaks up to and including the double-ended severance of a reactor coolant pipe, the emergency core cooling system will meet the acceptance criteria of 10 CFR 50.46 (Reference 1). That is:

1. The calculated peak clad temperature does not exceed 2200°F based on a total peaking factor of 2.32.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed one percent of the total amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The cladding oxidation limits of 17 percent are not exceeded during or after quenching.

4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived reactivity remaining in the core.

14.3.3.4 Small-Break Results

As noted previously, the calculated peak clad temperature resulting from a small-break LOCA is less than that calculated for a large break. From the results of the LOCA sensitivity studies,[17] the limiting small break was found to be less thana 10-in. diameter rupture of the rector coolant system cold leg. In addition, sensitivity studies have indicated that little or no uncovering will occur for break sizes that are 2 in. or less. In order to bound a range of Reactor Vessel average temperatures, a spectrum of small break sizes were run at the conditions consistent with the low end of the T_{ave} range. The limiting break size was then analyzed at the conditions consistent with the upper end of the T_{ave} range. In addition, the small break analyses are based on a hot channel enthalpy rise factor of 1.65 (which conservatively bounds 1.62) and the analyses are applicable to both the Westinghouse 15 x 15 LOPAR and 15 x 15 OFA fuel designs (as well as mixed cores of these fuel types). The results of these analyses are summarized in Tables 14.3-12 and 14.3-13.

Figures 14.3-53a through 14.3-64 present the principal parameters of interest for the small-break emergency core cooling system analyses. For all cases analyzed, the following transient parameters are presented:

1. Reactor coolant system pressure.
2. Core mixture height.
3. Hot spot clad temperature.[a]

For the limiting breaks analyzed, the following additional transient parameters are presented:

1. Steam flow rate.
2. Core heat transfer coefficient.
3. Hot spot fluid temperature.
4. Core power after reactor trip.

The maximum calculated peak clad temperature for all small breaks analyzed is 1218.5^{o}F. These results are well below all acceptance criteria limits of 10 CFR 50.46, and in all cases are not limiting when compared to the results presented for large breaks.

(a) For the break sizes that have no core uncovery during the blowdown portion of the transient, there is no need to run the rod heat-up calculation code (LOCTA-IV). Therefore, for the 8 inch break (Figure 14.3-63 shows that the core never uncovers) there will not be a plot for hot spot clad temperature. Table 14.3-13 will also not be able to report results from the 8 inch break.

REFERENCES FOR SECTION 14.3

1. U. S. Nuclear Regulatory Commission, "Acceptance Criteria for Emergency Core Cooling System for Light Water Cooled Nuclear Power Reactors", 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, No. 3, January 4, 1974.

2. U. S. Nuclear Regulatory Commission, Reactor Safety Study - An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants, WASH-1400 (NUREG 75/104), October 1975.

3. F. M. Bordelon, H. W. Massie and T. A. Zordan, Westinghouse ECCS Evaluation Model-Summary, WCAP-8339 (nonproprietary), Westinghouse Electric Corporation, July, 1974.

4. F. M. Bordelon, et. al., SATAN-VI Program: Comprehensive Space Time Dependent Analysis of Loss of Coolant, WCAP-8302 (proprietary) June 1974, and WCAP-8306, (nonproprietary), Westinghouse Electric Corporation, June 1974.

5. R. D. Kelly, et. al., Calculational Model for Core Reflooding After a Loss of Coolant Accident (WREFLOOD Code), WCAP-8170 (proprietary) and WCAP-8171 (nonproprietary), Westinghouse Electric Corporation, June 1974

6. F. M. Bordelon, and E. T. Murphy, Containment Pressure Analysis Code (COCO), WCAP-8327 (proprietary) and WCAP-8326 (nonproprietary), Westinghouse Electric Corporation, June 1974.

7. F. M. Bordelon, et. al., LOCTA-IV Program: Loss of Coolant Transient Analysis, WCAP-8301 (proprietary) and WCAP-8305 (nonproprietary), Westinghouse Electric Corporation, June 1974.

8. DELETED

9. F. M. Bordelon, et.al., Westinghouse ECCS Evaluation Model - Supplementary Information, WCAP-8471-P-A (proprietary) and WCAP-8472-A (nonproprietary), Westinghouse Electric Corporation, April 1975.

10. Westinghouse Electric Corporation, Westinghouse ECCS Evaluation Model October 1975 Version, WCAP-8622 (proprietary), November 1975, and WCAP-8623 (nonproprietary), November 1975.

11. Letter from C. Eicheldinger, Westinghouse Electric Corporation, to D. B. Vassallo, NRC, letter number NS-CE-924 dated January 23, 1976.

12. C. Eicheldinger, Westinghouse ECCS Evaluation Model, February 1978 Version, WCAP-9220-P-A (proprietary), and WCAP-9221-A (nonproprietary), Westinghouse Electric Corporation, February 1978.

13. Letter from T. M. Anderson, Westinghouse Electric Corporation, to John Stolz, NRC, letter no. NS-TNA-1981, dated November 1, 1978.

14. Letter from T. M. Anderson, Westinghouse Electric Corporation to John Stolz, NRC, letter no. NS-TMA-2014, dated December 11, 1978.

15. T. A. Porsching, J. H. Murphy, J. A. Redfield, and V. C. Davis, FLASH-4; A Fully Implicit FORTRAN-IV Program for the Digital Stimulation of Transients in a Reactor Plant, WAPD-TM-840, Bettis Atomic Power Laboratory, March 1969.

16. DELETED

17. Westinghouse Electric Corporation, Westinghouse ECCS Evaluation Model Sensitivity Studies, WCAP-8341 (proprietary), July 1974, and WCAP-8342 (nonproprietary), July 1974.

18. H. S. Beck, et. al., Westinghouse ECCS - Four Loop (15 x 15) Sensitivity Studies With Upper Head Fluid Temperature at T Hot, WCAP-8855-A (nonproprietary), Westinghouse Electric Corporation, May 1977.

19. R. Salvatori, Westinghouse ECCS - Plant Sensitivity Studies, WCAP-8340 (proprietary) and WCAP-8356 (nonproprietary), Westinghouse Electric Corporation, July 1974.

20. S. Fabic, Computer Program WHAM for Calculation of Pressure Velocity, and Force Transients in Liquid Filled Piping Networks, Kaiser Engineers Report No. 67-49-R, November 1967.

21. J. Parmakinan, Water-Hammer Analysis, Prentiss Hall, 1955.

22. V. L. Streeter, and E. B. Wylie, Hydraulic Transients, McGraw-Hill, 1967, p. 19.

23. F. R. Zaloudek, The Critical Flow of Hot Water Through Short Tubes, HW-77594, May 1968.

24. H. K. Fauske, "The Discharge of Saturated Water Through Tubes," Chem. Eng. Progress Symp. Series Heat Transfer - Cleveland, No. 59, Vol. 61, 1965.

25. S. Fabic, Early Blowdown (Water Hammer) Analysis for Loss of Fluid Test Facility, Kaiser Engineers Report No. 65-28-RA, April 1967.

26. G. Bohm, Indian Point Unit No. 2 Reactor Internals Mechanical Analysis for Blowdown Excitation, WCAP-7822 (nonproprietary) Westinghouse Electric Corporation.

27. Westinghouse Electric Corporation, Analysis of Reactor Coolant System for Postulated Loss-of-Coolant Accident: Indian Point 3 Nuclear Power Plant, WCAP-9117 (proprietary), June 1977.

28. Letter from D. D. Kingsley, Jr., Chairman, Westinghouse Owners Group, to H. R. Denton, NRC, Letter No. 06-66, December 30, 1981.

29. T. Mayer, Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants, WCAP-10019, Westinghouse Electric Corporation, December 1981.

30. Westinghouse Electric Corporation, Creep Rupture Tests of Type 304 Stainless Steel Weldments with Central Axial Welds of Type 304 Stainless Steel at 593°C, WCAP-5890, Revision 1.

31. T. Tagami, Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1).

32. A. Kolflat and W. A. Chittenden, "A New Approach to the Design of Containment Shells for Atomic Power Plants," Proceedings of American Power Conference, 1957, pp. 651-659.

33. W. H. McAdams, Heat Transmission, 3rd Edition, McGraw-Hill Book Company, Inc., New York, 1954.

34. E. R. G. Eckert and P. M. J. Drake, Heat and Mass Transfer, McGraw-Hill Book Company, Inc., New York, 1959.

35. D. Q. Kern, Process Heat Transfer, Mc-Graw Hill Book Company, Inc., New York, 1950.

36. T. H. Chilton and A. P. Colburn, "Mass Transfer (Absorption) Coefficients Prediction from Data on Heat Transfer and Fluid Friction," Imd. Eng. Chem., No. 26, 1934, pp. 1183-1187.

37. S. Weinberg, Proc. Inst. Mech. Engr., Vol. 164, 1952, pp. 240-258.

38. W. Ranz and W. Marshall, Chem. Engr., Prog. Vol. 48, No. 3, 1952, pp. 141-146 and Vol. 48, No. 4, pp. 173-180.

39. J. Perry, Chemical Engineers Handbook, 3rd Edition, McGraw-Hill Book Company, Inc., 1950.

40. E. Eckert and J. Gross, Introduction to Heat and Mass Transfer, McGraw-Hill Book Company, Inc., 1963.

41. S. Gordon and E. H. Hart, Proc. Second Geneva Conf., Vol. 29, No. 13, 1958; alsoR. G. Sowden, Journal of Nuclear Materials, Vol. 8, 1963, pp. 81-101.

42. M. J. Bell, et. al., Investigations of Chemical Additives for Reactor Containment Sprays, WCAP-7153 (confidential), Westinghouse Electric Corporation, March 1958.

43. Letter from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: Main Steam Line Break with Continued Feedwater Addition at Indian Point 2, dated November 8, 1984.

44. W. S. Pasedag, Iodine Removal by Spray in Zion Containment, WCAP-7742, Westinghouse Electric Corporation, August 1971.

45. R. E. Adams and W. E. Browning, Jr., Removal of Radioiodine from Air Streams by Activated Charcoal, USAEC Report ORNL-2872, Oak Ridge National Laboratory, April 1, 1960.

46. R. E. Adams and W. E. Browning, Jr., Removal of Radioiodine from Air-Steam Mixtures, Report (ORNL Central Files No.) CF-60-11-39, Oak Ridge National Laboratory, November 14, 1960.

47. G. H. Prigge, Application of Activated Carbon in Reactor Containment, USAEC Report No. DP-778, E. I. du Pont de Nemours and Company, Savannah River Laboratory, September 1962.

48. R. E. Adams and R. D. Ackley, Removal of Elemental Radioiodine from Flowing Humid Air by Iodized Charcoals (Abstract), USAEC Report ORNL-TM-2040, Oak Ridge National Laboratory, November 2, 1967.

49. Connecticult Yankee Atomic Power Company, Connecticut Yankee Charcoal Filter Tests, Report CYAO-101, December 1966.

50. R. E. Adams and R. D. Ackley, Trapping of Radioactive Methly Iodine from Flowing Steam-Air, ORNL-TM-2728, Oak Ridge National Laboratory, December 1969.

51. D. F. Toner and J. S. Scott, Fission Product Release from Uranium Dioxide, Nuclear Safety, Vol. 3, No. 2, December 1961.

52. J. Belle, Uranium Dioxide: Properties and Nuclear Applications, Naval Reactors, Division of Reactor Development, U. S. Atomic Energy Commission, 1961.

53. Letter from J. D. O'Toole, Con Edison, to D. G. Eisenhut, NRC, with attachment, Subject: Control Room Habitability Studies, May 12, 1981.

54. Consolidated Edison Company, Indian Point Unit No. 3, PSAR, Suppliement 7 (with Supplement 9 page changes).

55. Kelly, et. al., Westinghouse Emergency Core Cooling System Evaluation Model for Analyzing Large LOCA's During Operation with One Loop Out of Service for Plants Without Loop Isolation Valves, WCAP-9166, February 1978.

56. Eicheldinger, C., Westinghouse ECCS Evaluation Model, 1981 Version, WCAP-9221-A (nonproprietary), Revision 1, February 1982.

58. M. Y. Young, et al., BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients, WCAP-9561-P-A (proprietary) and WCAP-9561-NP-A (non-proprietary), Westinghouse Electric Corporation, March 1984.

59. B. A. McIntyre, ADDENDUM TO BART-A1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS (SPECIAL REPORT: NS-NRC-85-3025-A), WCAP-9561-NP-A, Addendum 2 (non-proprietary), Westinghouse Electric Corporation, July 1985.

60. M. Y. Young, ADDENDUM TO: BART-A1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS (SPECIAL REPORT: THIMBLE MODELING IN WESTINGHOUSE ECCS EVALUATION MODEL), WCAP-9561-P-A, Addendum 3, Revision 1 (proprietary) and WCAP-9561-NP-A, Addendum 3, Revision 1 (non-proprietary), July 1986.

61. Meyer, P. E., NOTRUMP,A Nodal Transient Small Break and General Network Code, WCAP-10079-P-A, August 1985.

62. Lee, H., Rupprecht, S. D., Tauche, W. D., Schwarz, W. R., Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, WCAP-10054-P-A, August 1985.

63. Rupprecht, S. D., Osterrieder, R. A., Wills, M. E., Willis, J. M., Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With the NOTRUMP Code, WCAP-11145-P-A, June 1986.

64. Chiou, J. S., et. al., Models for PWR Reflood Calculations Using the BART code, WCAP-10062.

65. Kabadi, J. N., et. al., The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code, WCAP-10266, Revision 2 (proprietary), August 1986.

66. Letter C. E. Rossi (NRC) to E. P. Rahe, Jr. (W), "Acceptance for Referencing of Licensing Topical Report WCAP-10266", November 1986.

ADDITIONAL REFERENCE FOR SECTION 14.3

1. Letter from T. M. Anderson, Westinghouse Electric Corporation, to John Stolz, NRC, Letter No. NS-TMA-2030, dated January 1979.

Table 14.3-1

Input Parameters and Initial Conditions

Reactor power	102 % of 3071.4 MWt
Peak linear power	102 % of 14.71 kw/ft[a]
Peaking factor	2.32[b]
Accumulator water volume	795 ft^3 per tank
Accumulator pressure	630 psia
Number of safety injection pumps operating	2
Steam generator tube plugging level	25 % (uniform)

(a) The power distribution for the Small Break LOCA analysis is given in Figure 14.3-4.

Table 14.3-2 (Sheet 1 of 2)

Large-Break Containment Data

Net free volume	2.61×10^6 ft^3
Initial conditions	
Pressure	14.7 psia
Temperature	90°F
Refueling water storage tank temperature	40°F
Service water temperature	35°F
Outside temperature	-20°F
Spray system	
Number of pumps operating	2
Runout flow rate	3000 gpm
Actuation time	20 sec
Safeguards fan coolers	
Number of fan coolers operating	5
Fastest postaccident initiation of fan coolers	30 sec

Structural heat sinks

Thickness (in.)	Area (ft^2)
1. 0.007 paint, 0.375 steel, 54.0 concrete	45,684
2. 0.007 paint, 0.5 steel, 42.0 concrete	28,613
3. 12.0 concrete	15,000
4. 0.375 stainless steel, 12.0 concrete	10,000
5. 12.0 concrete	61,000
6. 0.5 steel	68,752
7. 0.007 paint, 0.375 steel	81,704
8. 0.25 steel	27,948
9. 0.007 paint, 0.1875 steel	69,800

Table 14.3-2 (Sheet 2 of 2)

Large-Break Containment Data

Structural heat sinks (continued)

Thickness (in.)	Area (ft^2)
10. 0.125 steel	3000
11. 0.138 steel	22,000
12. 0.0625 steel	10,000
13. 0.019 stainless steel, 1.25 insulation, 0.75 steel, 54.0 concrete	785
14. 0.019 stainless steel, 1.25 insulation, 0.5 steel, 54.0 concrete	6849
15. 0.025 stainless steel, 1.5 insulation 0.5 steel, 54.0 concrete	3816
16. 0.025 stainless steel, 1.5 insulation, 0.375 steel, 54.0 concrete	4362

Table 14.3-3

Large Break - Results

Results	DECLG[a] (C_D = 0.8) High T_{ave} MINSI	(C_D = 0.6) High T_{ave} MINSI	(C_D = 0.4) High T_{ave} MINSI	(C_D = 0.4) Low T_{ave} MINSI	(C_D = 0.4) High T_{ave} MAXSI
Peak clad temperature, °F	1749	1940	2039	1919	2015
Peak clad location, ft	6.50	5.75	6.50	5.75	6.50
Local Zr/H_2O reaction (max.), percent	1.93	4.42	5.54	3.47	4.94
Local Zr/H_2O location, ft	5.75	5.75	6.00	6.00	6.00
Total Zr/H_2O reaction, percent	<0.3	<0.3	<0.3	<0.3	<0.3
Hot rod burst time, sec	42.65	41.61	42.34	46.36	42.34
Hot rod burst location, ft	5.75	5.75	6.00	6.00	6.00

[a] DECLG = Double-Ended Cold-Leg Guillotine.

Table 14.3-4

Large Break - Time Sequence of Events

Results	DECLG[a] (C_D = 0.8) High T_{ave} MINSI	(C_D = 0.6) High T_{ave} MINSI	(C_D = 0.4) High T_{ave} MINSI	(C_D = 0.4) Low T_{ave} MINSI	(C_D = 0.4) High T_{ave} MAXSI
Accident Initiation	0.0	0.0	0.0	0.0	0.0
Reactor trip signal	0.471	0.476	0.484	0.475	0.484
Safety injection signal	0.38	0.43	0.520	0.590	0.520
Start accumulator injection	10.9	13.3	18.3	17.3	18.3
End of blowdown	26.126	30.181	37.247	38.887	37.247
Bottom of core recovery	41.607	45.981	54.852	56.950	53.779
Accumulator empty	56.187	59.558	66.654	67.391	69.074
Pumped injection	25.38	25.43	25.520	25.590	25.520
End of Bypass	26.126	30.181	37.247	38.887	37.247

[a] DECLG = Double-Ended Cold-Leg Guillotine.

Table 14.3-5

Reflood Mass and Energy Release to the Containment

(DECLG, C_D = 0.4, High T_{ave}, MINSI)[a]

Time (sec)	Mass (lbm/sec)	Energy (Btu/sec)
54.852	0.0	0.
55.652	6.208	8033.
56.552	6.033	7806.
59.000	21.114	27319.
69.000	94.131	116082.
79.000	99.808	122879.
89.000	106.928	131404.
99.000	128.053	143110.
119.000	364.910	206549.
139.000	383.529	202954.
159.000	391.790	196461.
179.000	433.547	199709.
199.000	467.008	203381.

[a] DECLG = Double-Ended Cold-Leg Guillotine.

Table 14.3-6

Broken Loop Injection Spill During Blowdown

(DECLG, C_D = 0.4, High T_{ave}, MINSI)[a]

Time (sec)	Mass (lbm/sec)	Energy (Btu/sec)	Enthalpy
0.0	0.000	0.000	59.700
2.0	2506.084	149614.244	59.700
4.0	2185.123	130451.827	59.700
6.0	1962.133	117139.338	59.700
8.0	1794.098	107107.658	59.700
10.0	1659.988	99101.263	59.700
12.0	1548.929	92471.088	59.700
14.0	1454.683	86844.590	59.700
16.0	1373.524	81999.355	59.700
18.0	1302.810	77777.755	59.700
20.0	1240.344	74048.529	59.700
22.0	1185.141	70752.936	59.700
24.0	1135.990	67818.624	59.700
26.0	1125.436	62920.464	55.908
28.0	1089.644	60556.537	55.575
30.0	1056.758	58388.939	55.253
32.0	1027.324	56443.604	54.942
34.0	150.083	1204.758	8.027
36.0	150.492	1208.043	8.027

[a] DECLG = Double-Ended Cold-Leg Guillotine.

Table 14.3-7

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Table 14.3-12

Small Break Time Sequence of Events (sec)

	4-in. ($0.087\ ft^2$)	6-in. ($0.196\ ft^2$)	8-in. ($0.349\ ft^2$)	6-in. High Tave Case
Start	0.0	0.0	0.0	0.0
Reactor trip signal	3.934	2.089	1.620	2.411
Top of core uncovered	369.55	135.97	NA	135.14
Accumulator injection begins	897.22	300.30	187.91	294.15
Peak clad temperature occurs	387.40	394.30	NA	376.23
Top of core covered	389.43	418.64	NA	444.33

Table 14.3-13

Small Break Results

	4-in. (0.087 ft^2)	6-in. (0.196 ft^2)	8-in. (0.349 ft^2)	6-in. High Tave Case
Peak clad temperature, °F	721.2	998.4	NA	1218.5
Peak clad location, ft	10.75	12.0	NA	11.75
Local Zr/H_2O reaction (max.), percent	0.0671	0.0675	NA	0.0809
Local Zr/H_2O location, ft	10.75	11.75	NA	12.0
Total Zr/H_2O reaction, percent	<0.3	<0.3	NA	0.3
Hot rod burst time, sec	—	—	NA	—
Hot rod burst location, ft	—	—	NA	—
Core power	102 percent of 3071.4 MWt			
Peak linear power, kW/ft	See Figure 14.3-4			
Accumulator Water Volume, ft^3	822 (per accumulator)[a]			

[a] Large Break analysis uses 795 ft^3 accumulator water volume. Small Break results are not sensitive to accumulator volume differences of this magnitude.

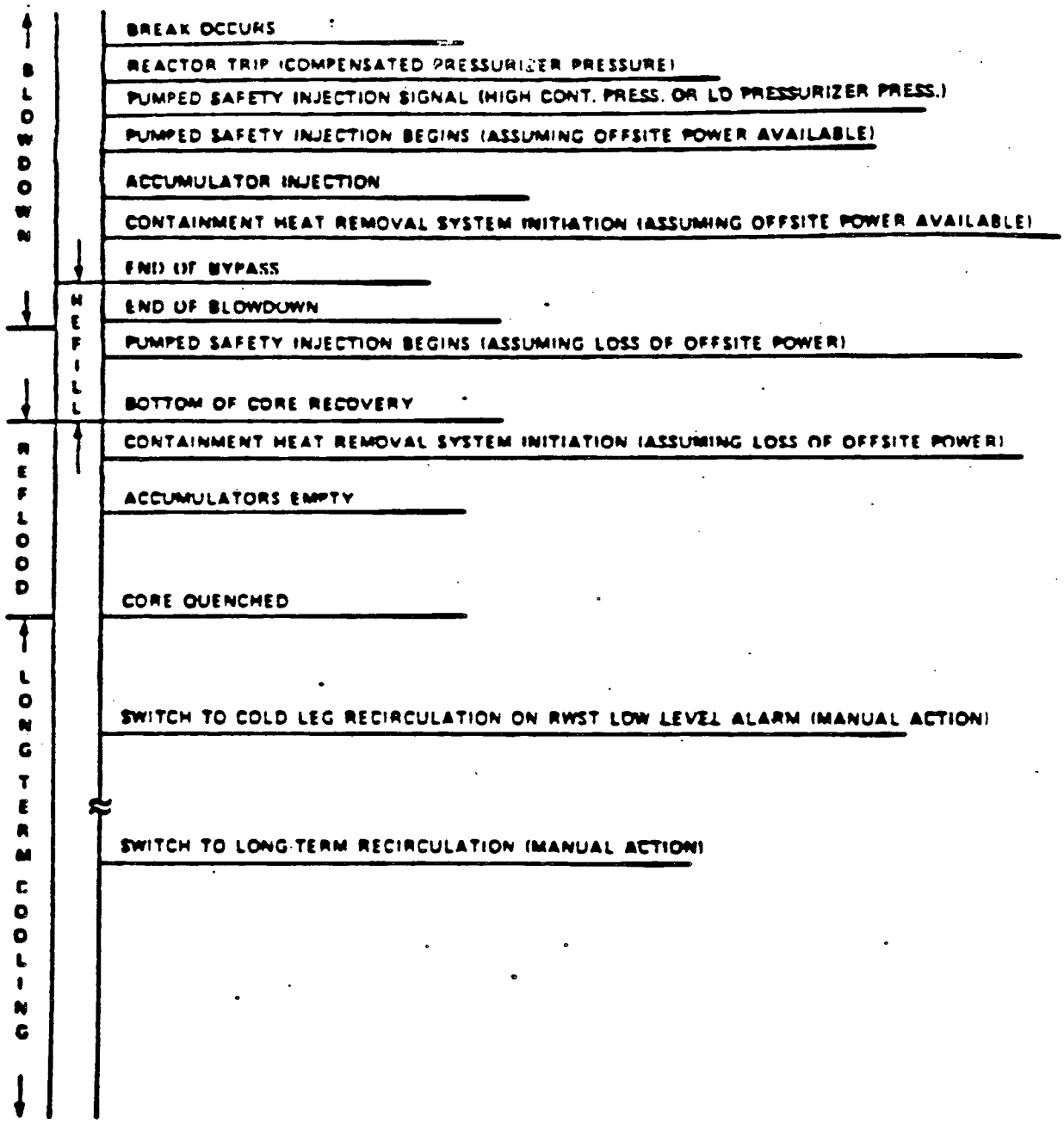

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Figure 14.3-1

Sequence of Events for Large Break Loss-of-Coolant Analysis

Revision 2

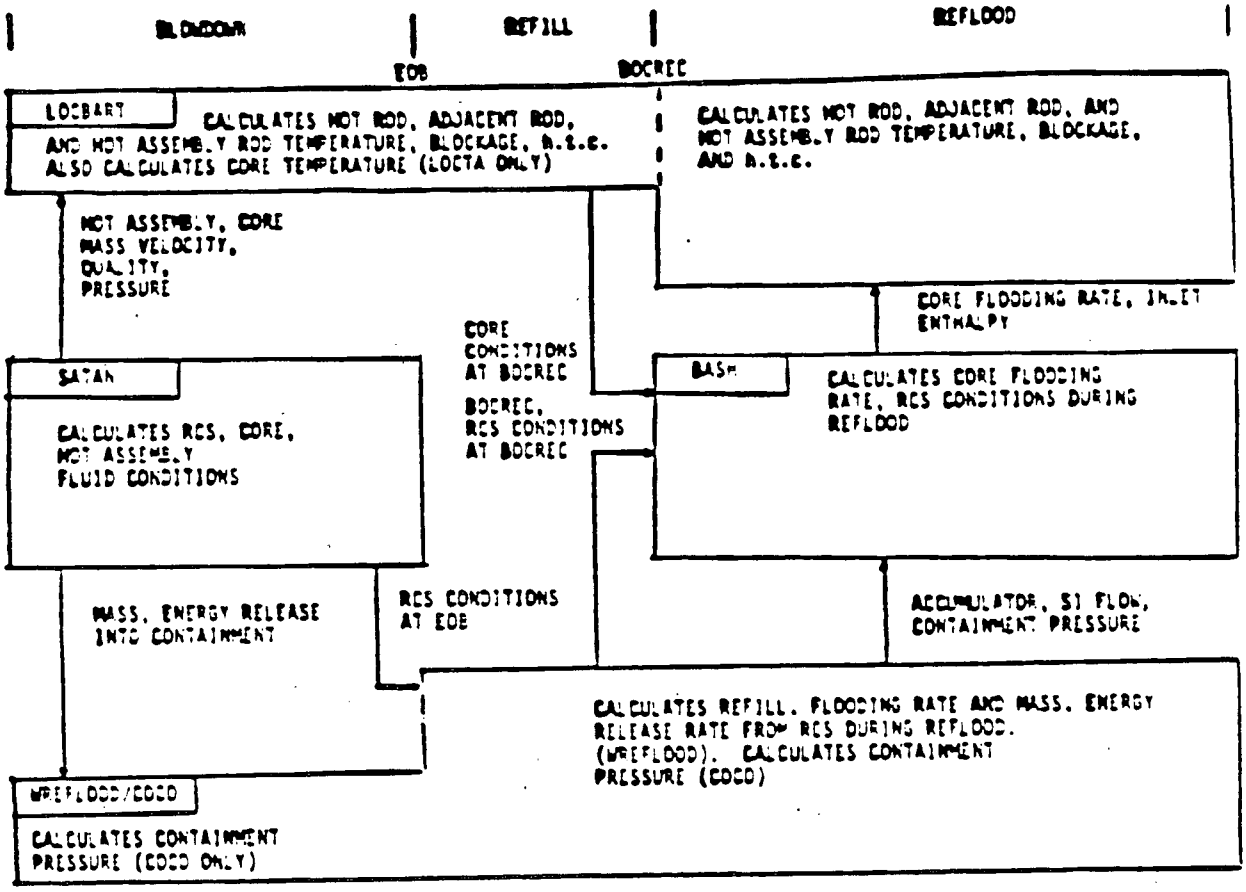

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Figure 14.3-2
Code Interface Description for
Large Break Model

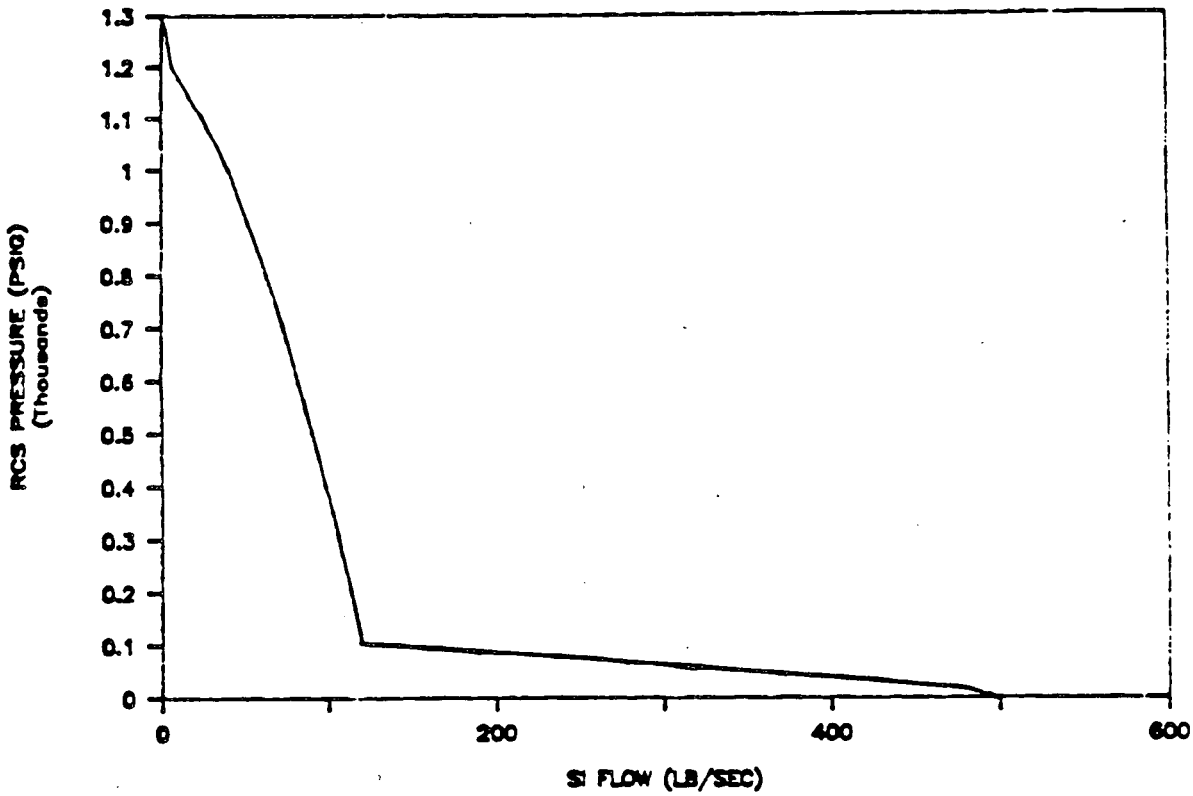

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Figure 14.3-3
RCS Pressure vs. SI Flow

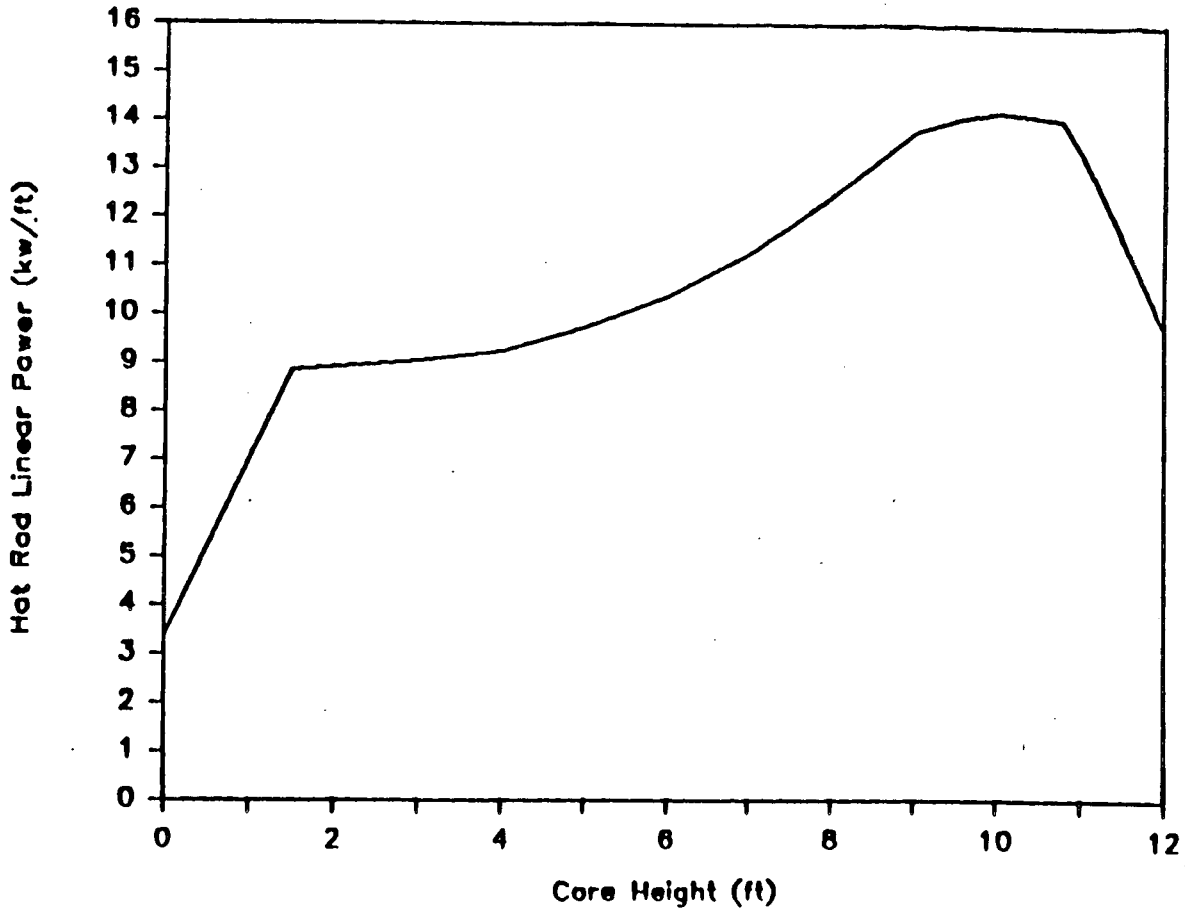

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Figure 14.3-4

Hot Rod Linear Power vs Core Height

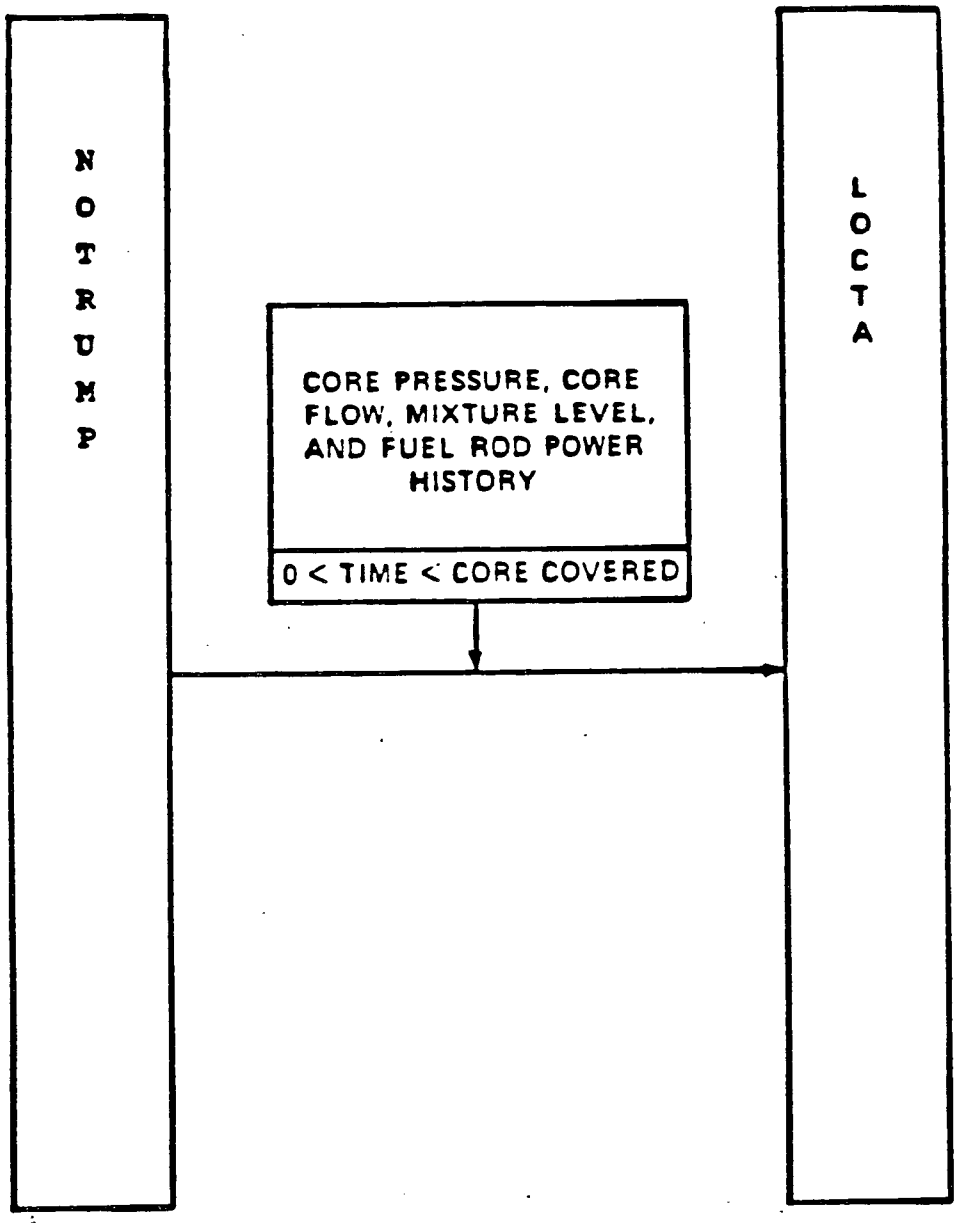

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Figure 14.3-5
Code Interface Description for Small Break Model

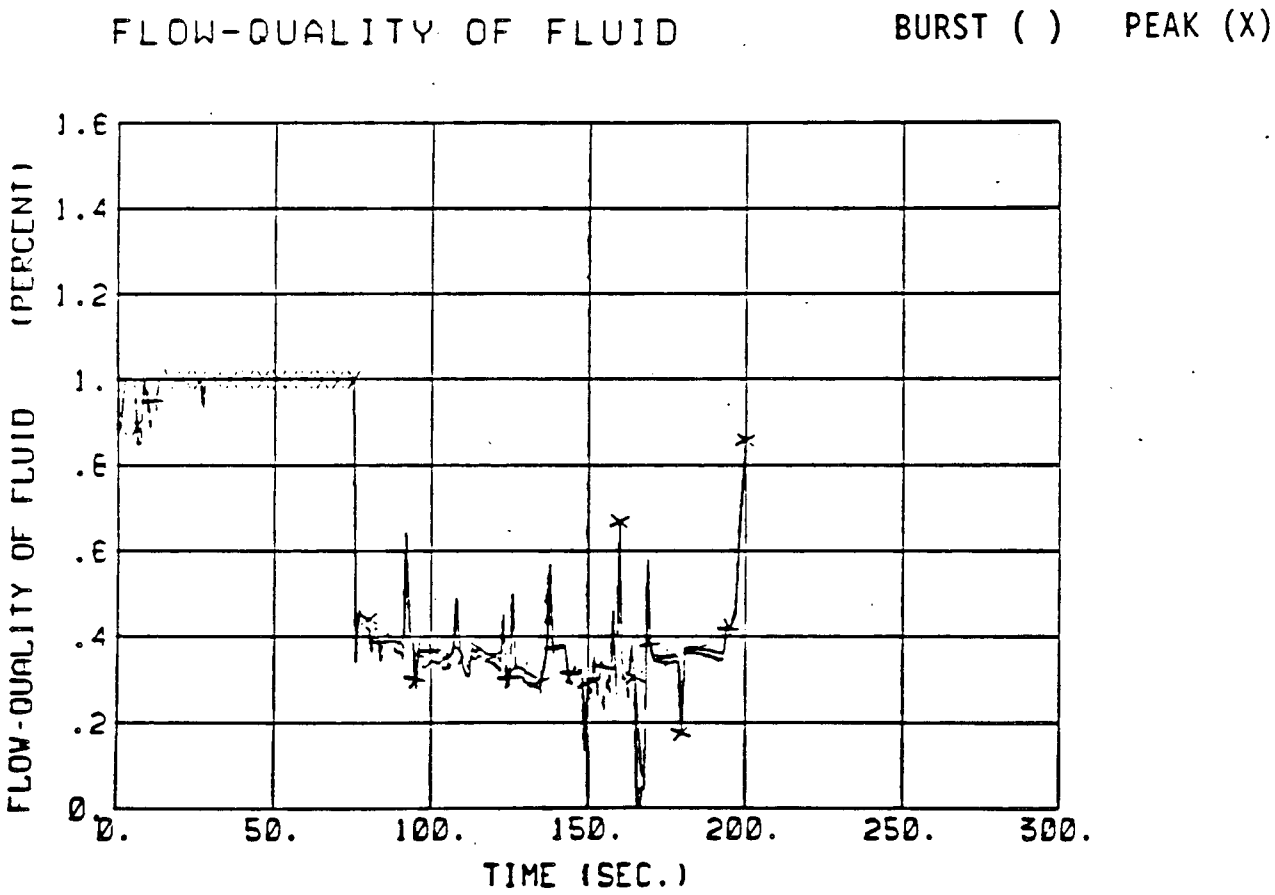

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Figure 14.3-6a
Fluid Quality-DECLG (C_D=0.4)
(High T_{ave}) MINSI

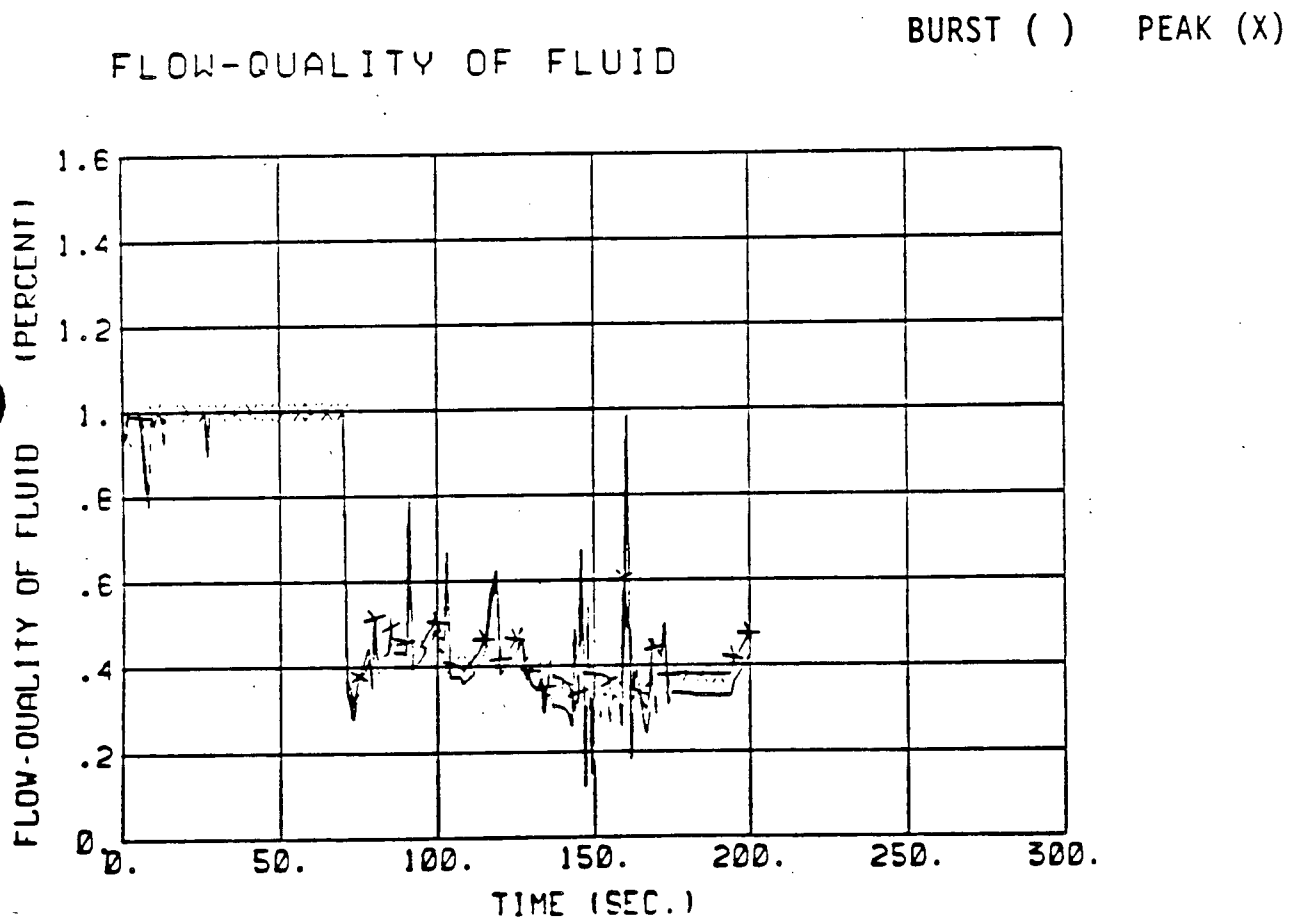

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Figure 14.3-6b
Fluid Quality-DECLG (C_D=0.4)
(Low T_{ave}) MINSI

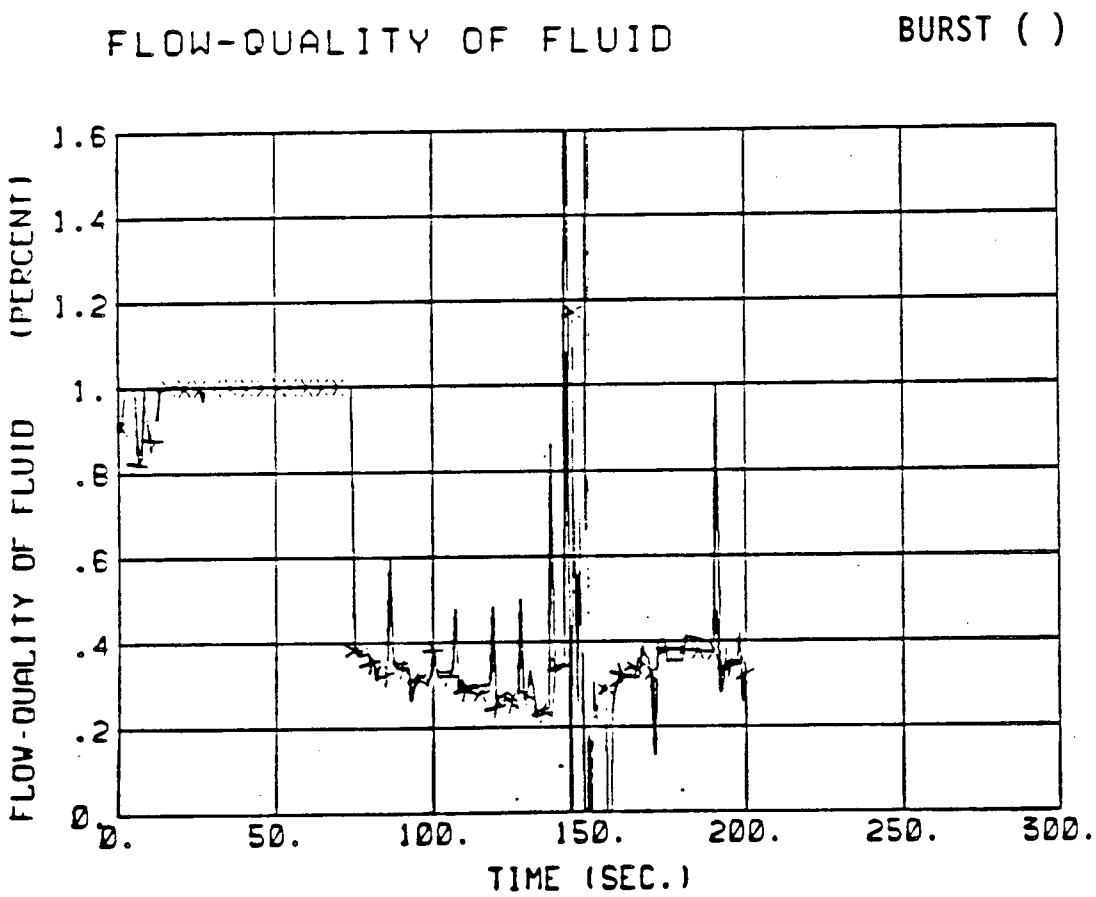

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Figure 14.3-6c
Fluid Quality-DECLG (C_D=0.4)
(High T_{ave}) MAXSI

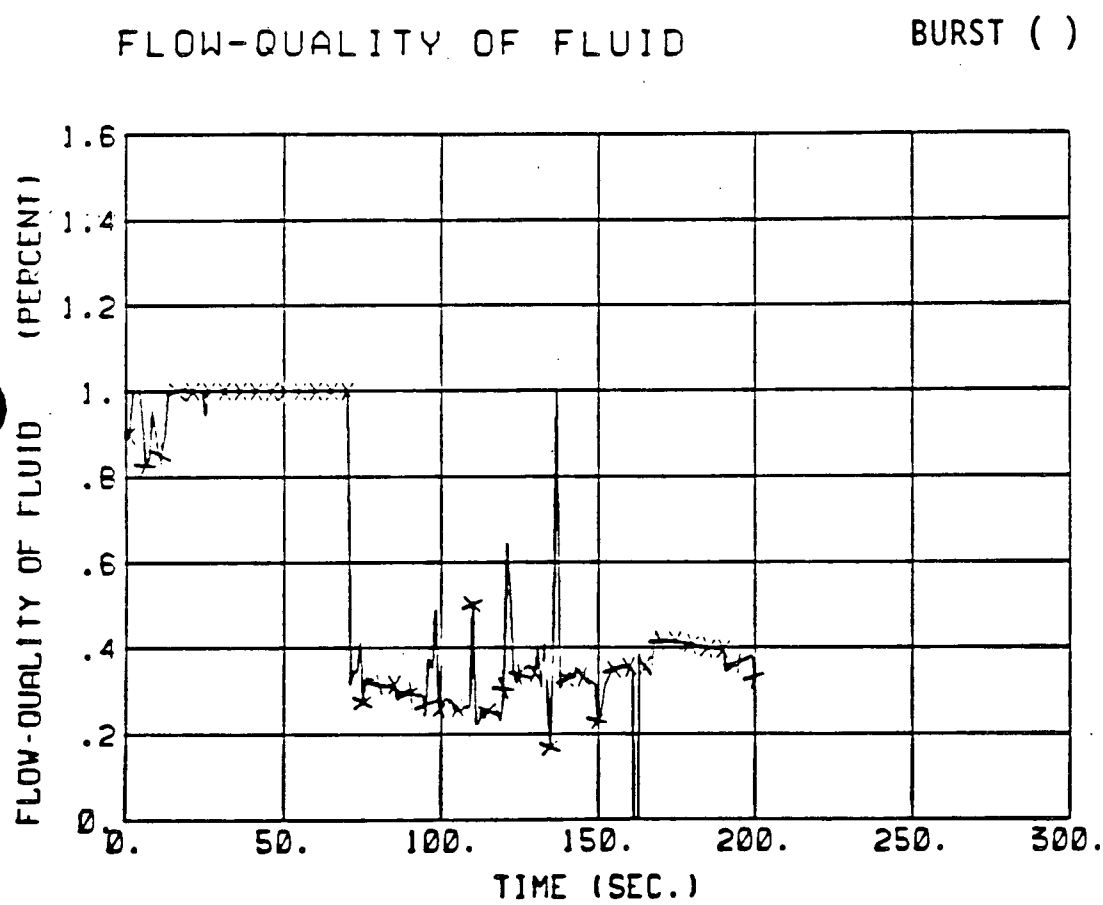

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Figure 14.3-7
Fluid Quality-DECLG (C_D=0.6)
(High T_{ave}) MINSI

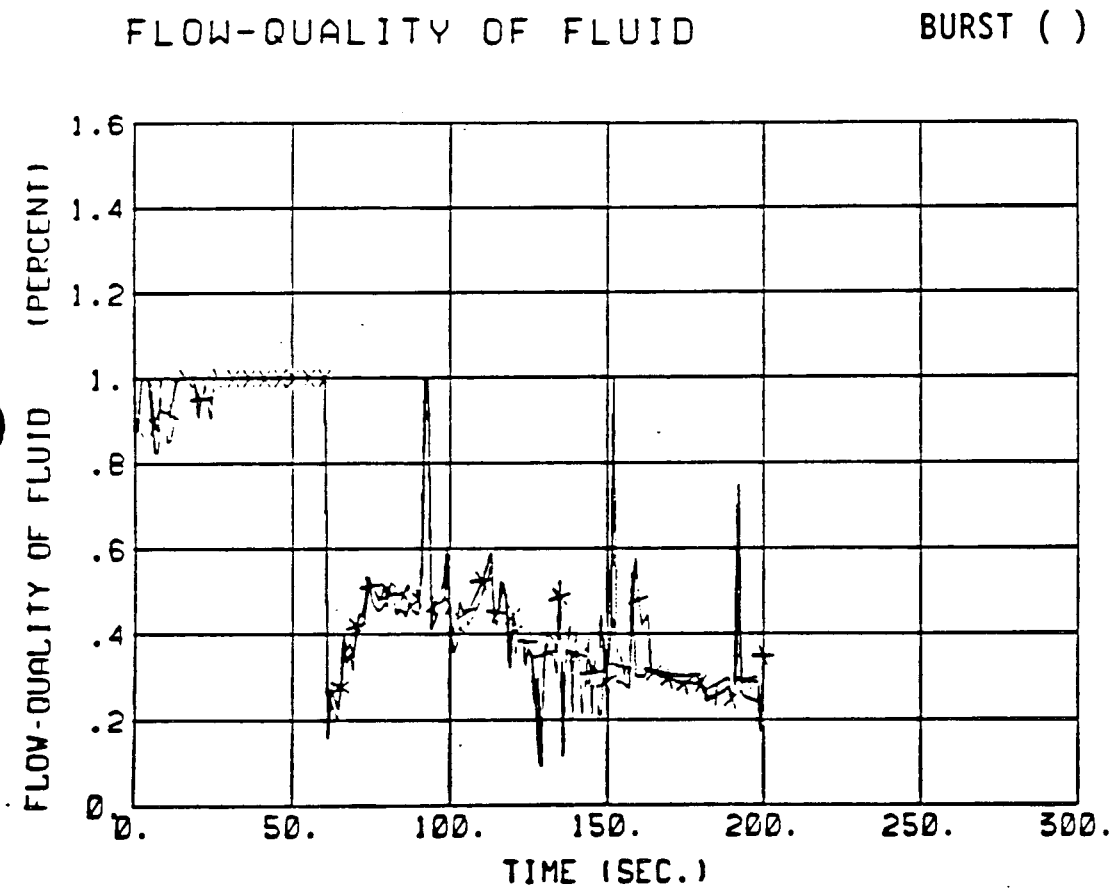

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Figure 14.3-8
Fluid Quality-DECLG (C_D=0.8)
(High T_{ave}) MINSI

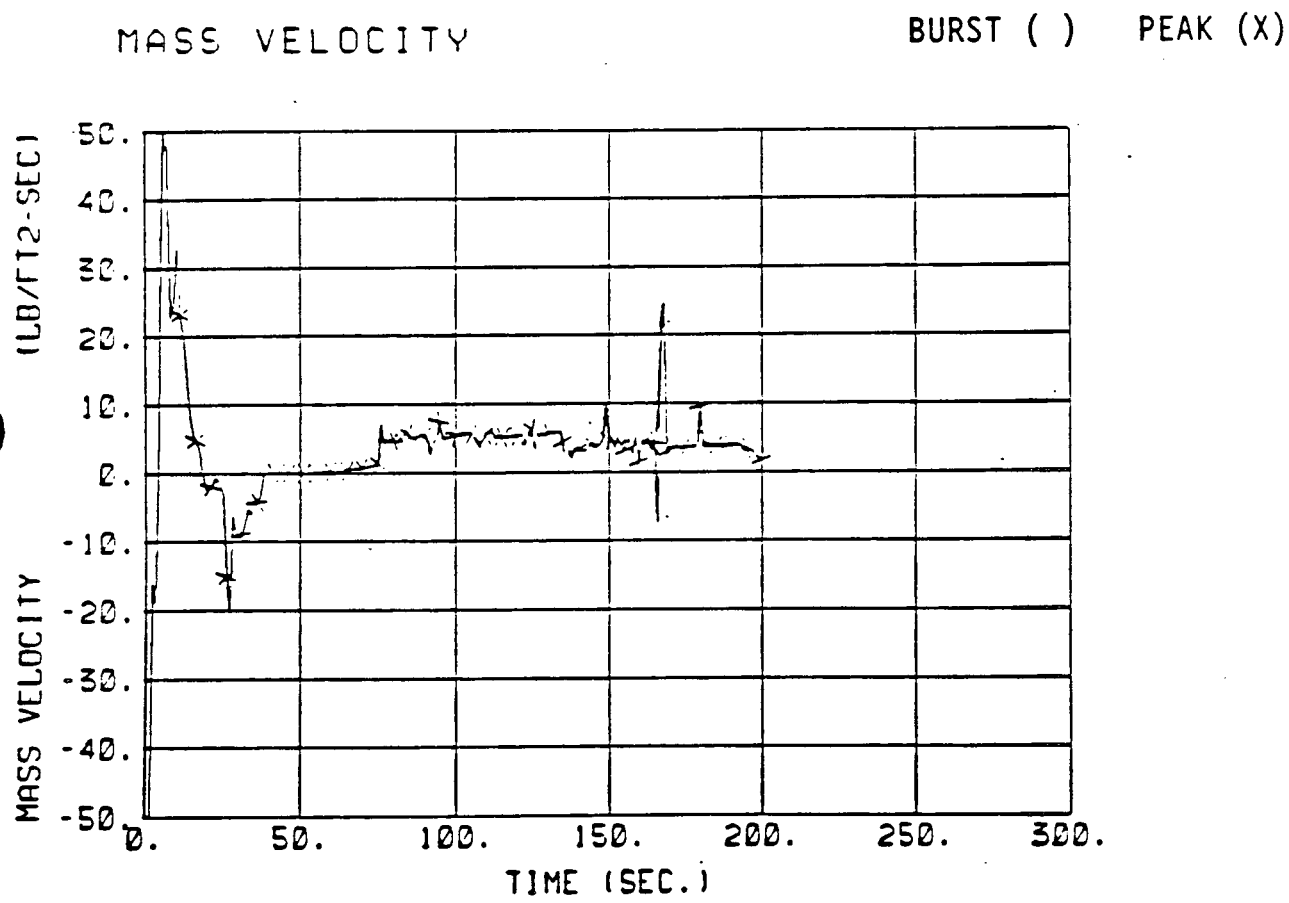

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Figure 14.3-9a
Mass Velocity-DECLG (C_D=0.4)
(High T_{ave}) MINSI

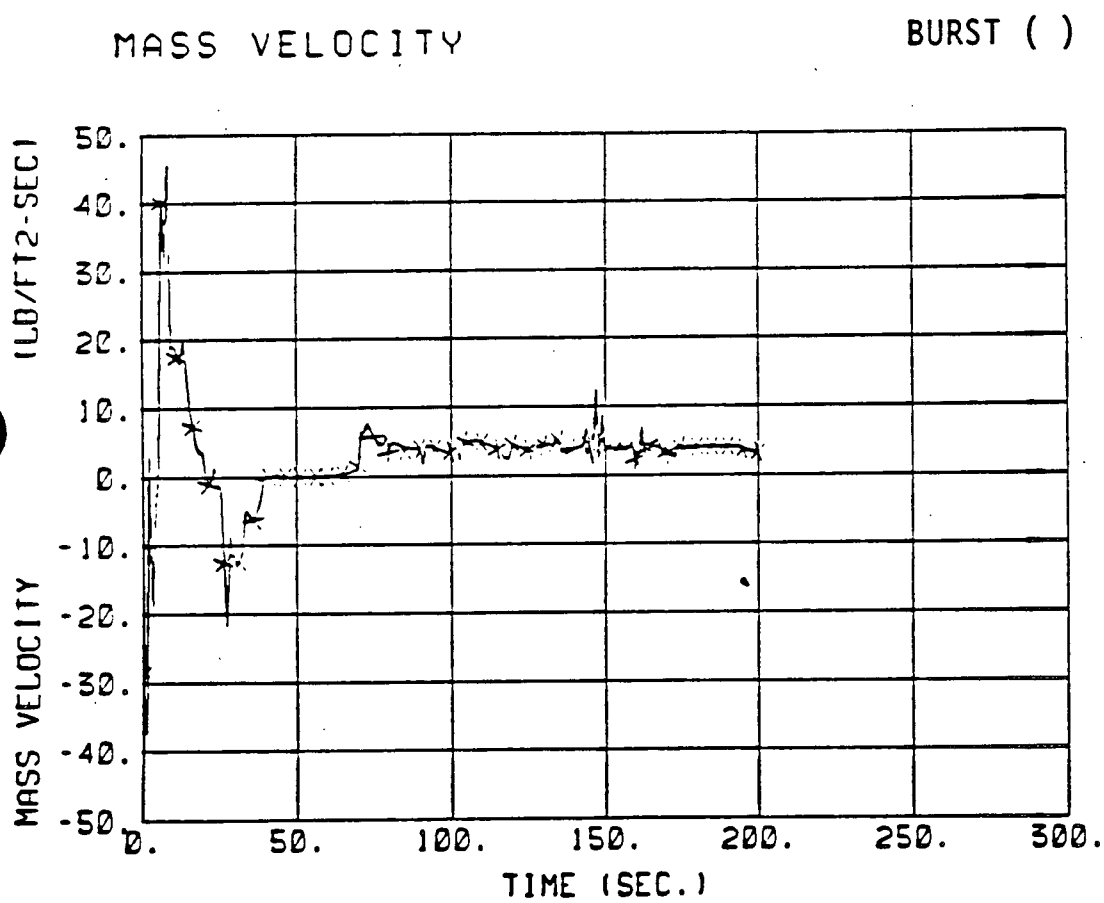

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Figure 14.3-9b
Mass Velocity-DECLG (C_D=0.4)
(Low T_{ave}) MINSI

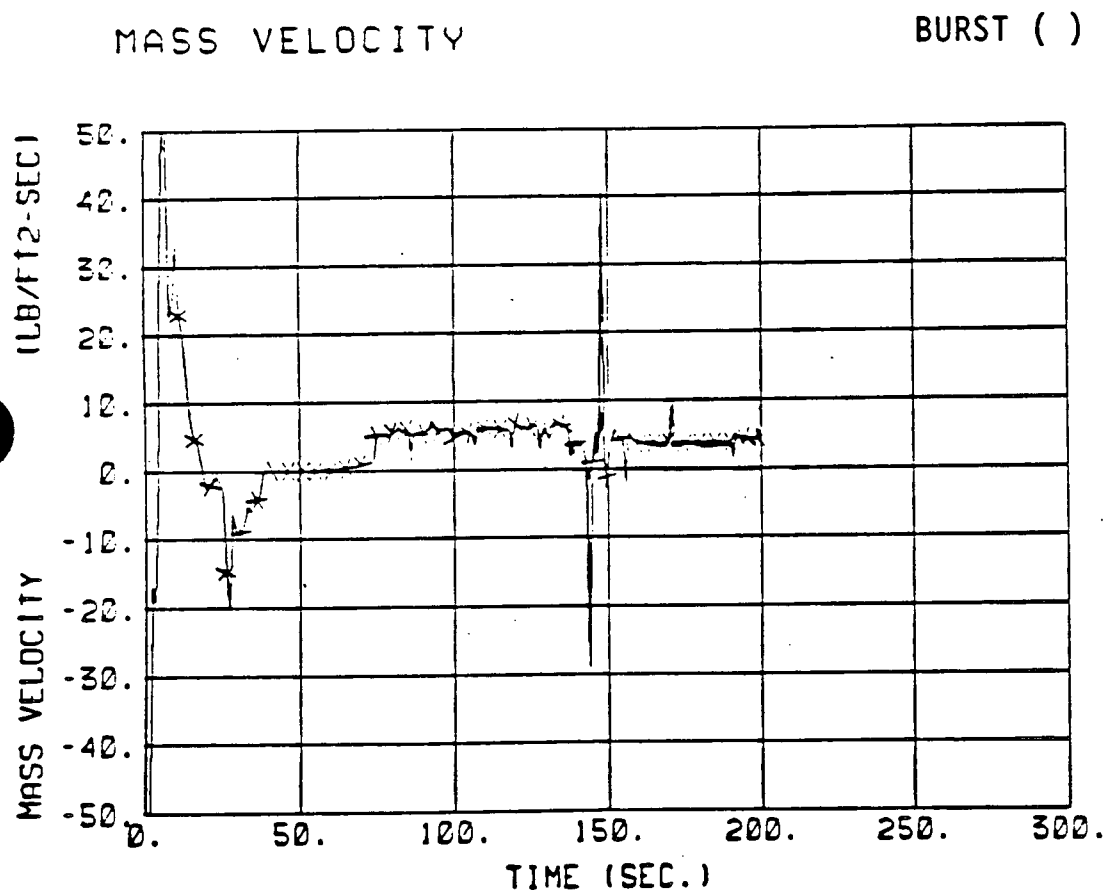

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Figure 14.3-9c
Mass Velocity- DECLG (C_D=0.4)
(High T_{ave}) MAXSI

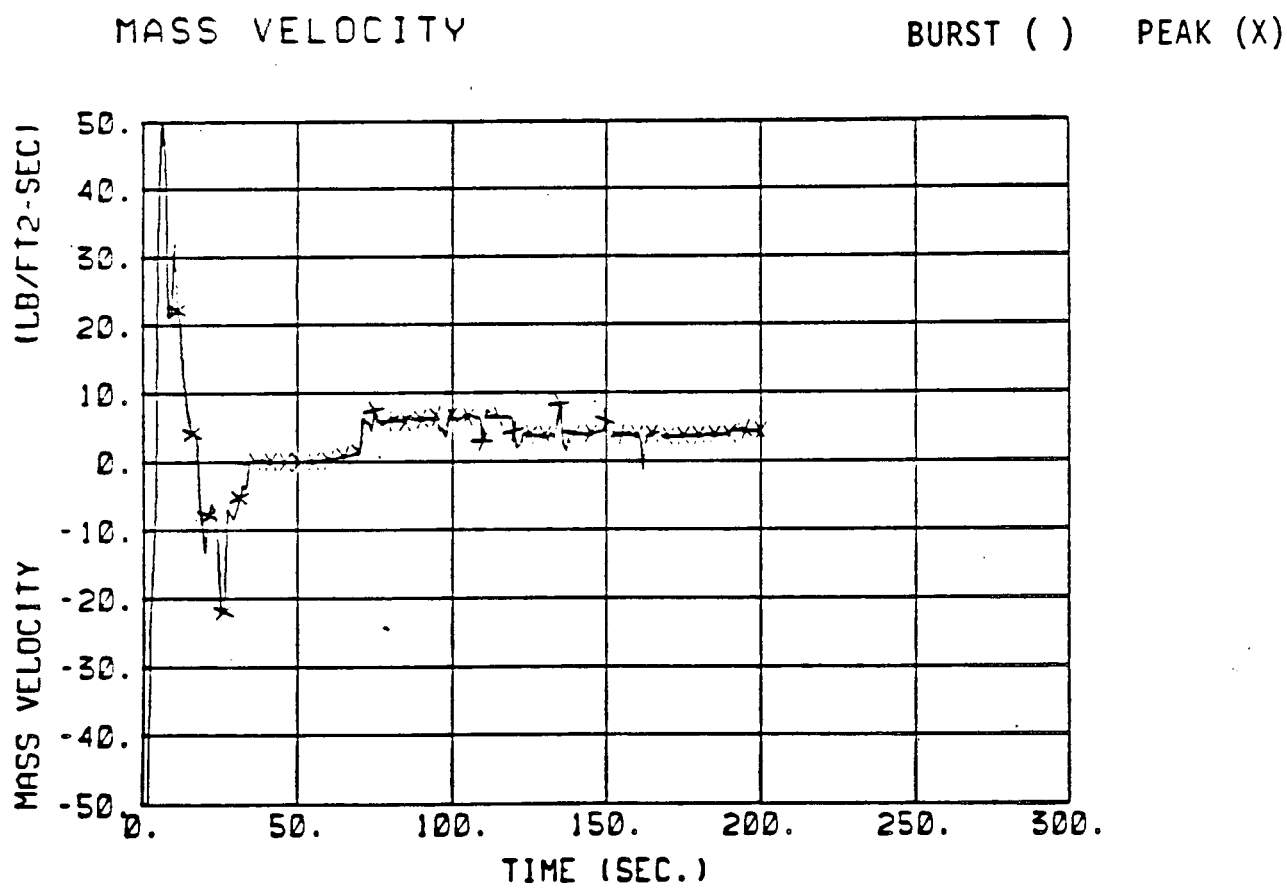

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Figure 14.3-10
Mass Velocity-DECLG (C_D=0.6)
(High T_{ave}) MINSI

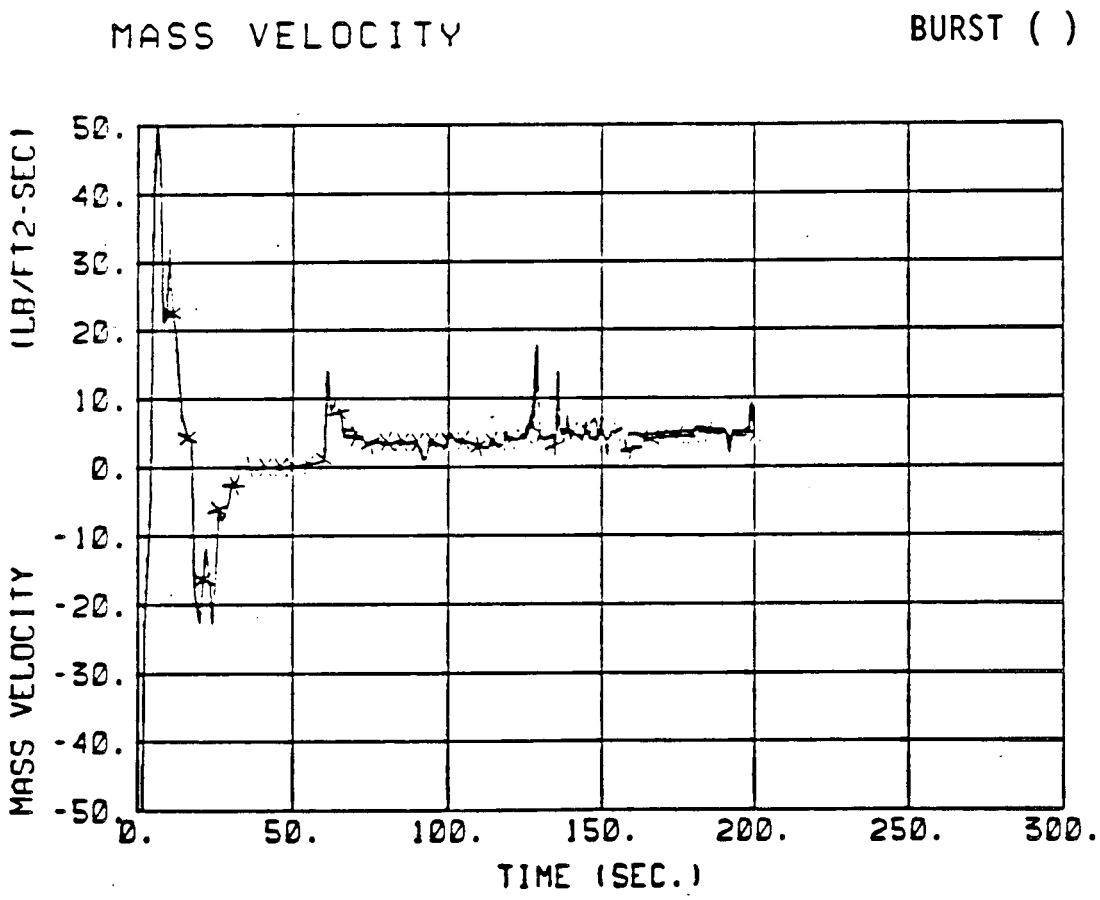

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Figure 14.3-11
Mass Velocity-DECLG (C_D=0.8)
(High T_{ave}) MINSI

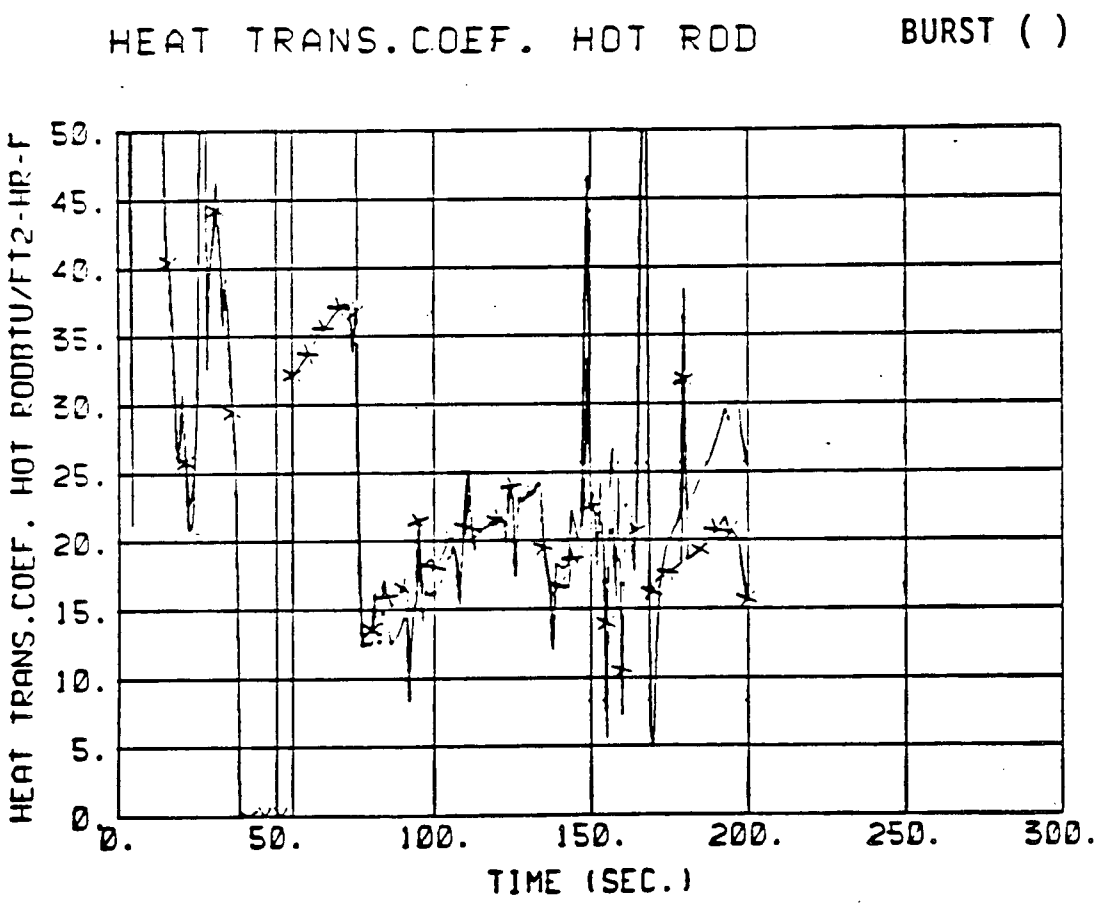

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Figure 14.3-12a
Heat Tranfer Coefficient-
DECLG (C_D=0.4)
(High T_{ave}) MINSI

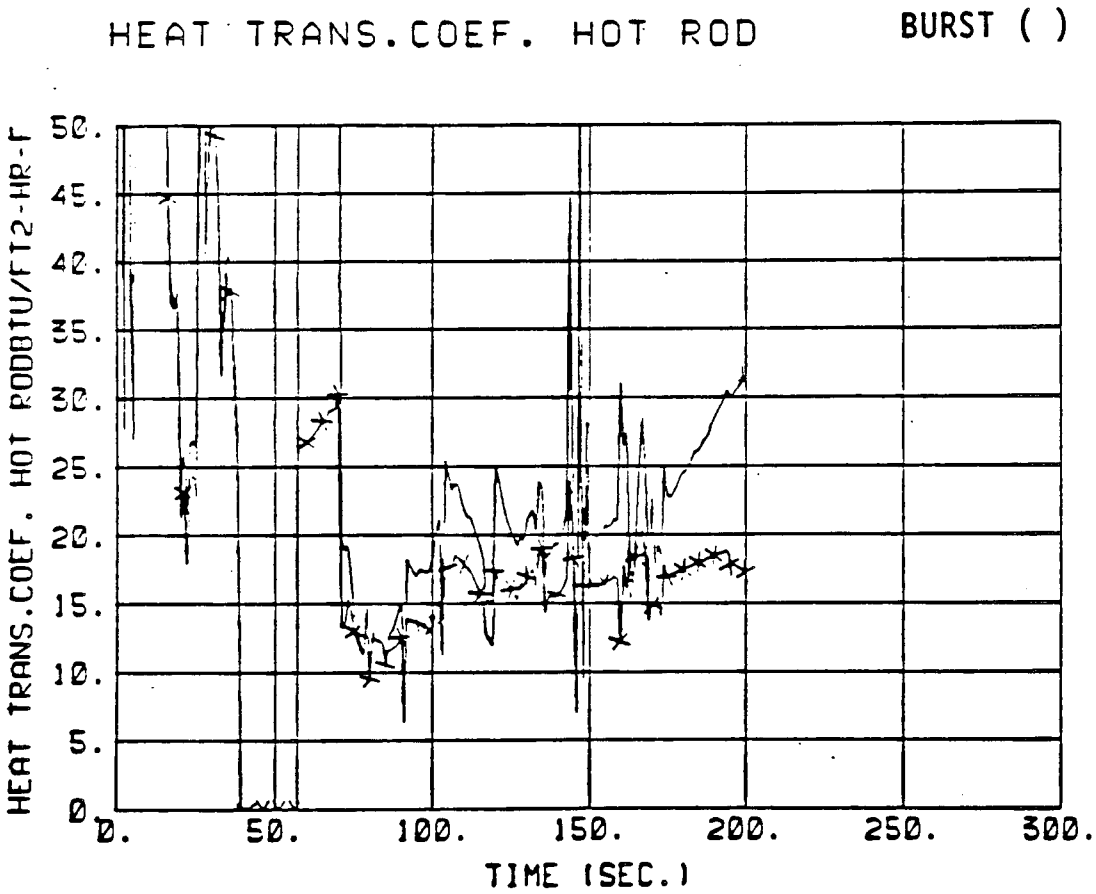

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Figure 14.3-12b
Heat Transfer Coefficient-
DECLG (C_D=0.4)
(Low T_{ave}) MINSI

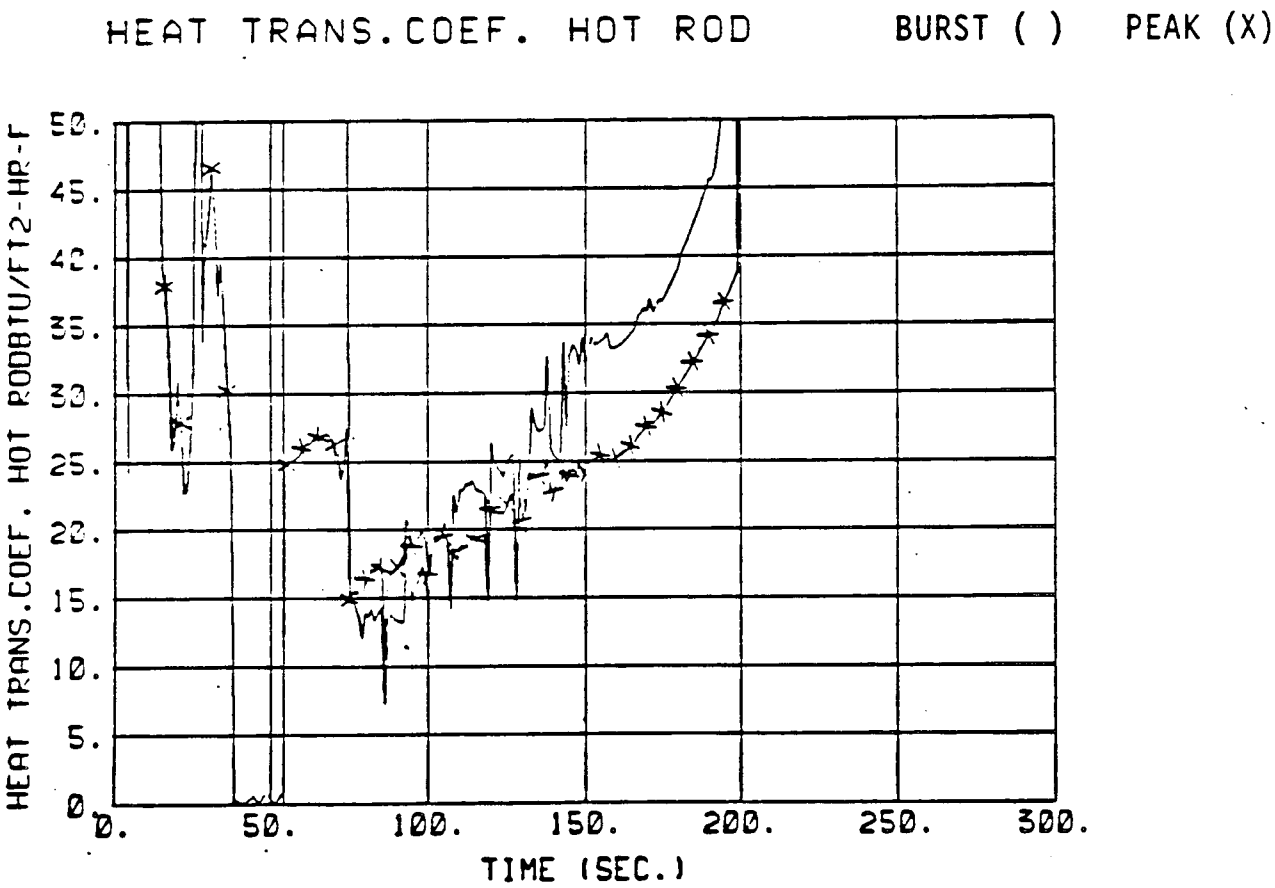

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Figure 14.3-12c
Heat Transfer Coefficient-
DECLG (C_D=0.4)
(High T_{ave}) MAXSI

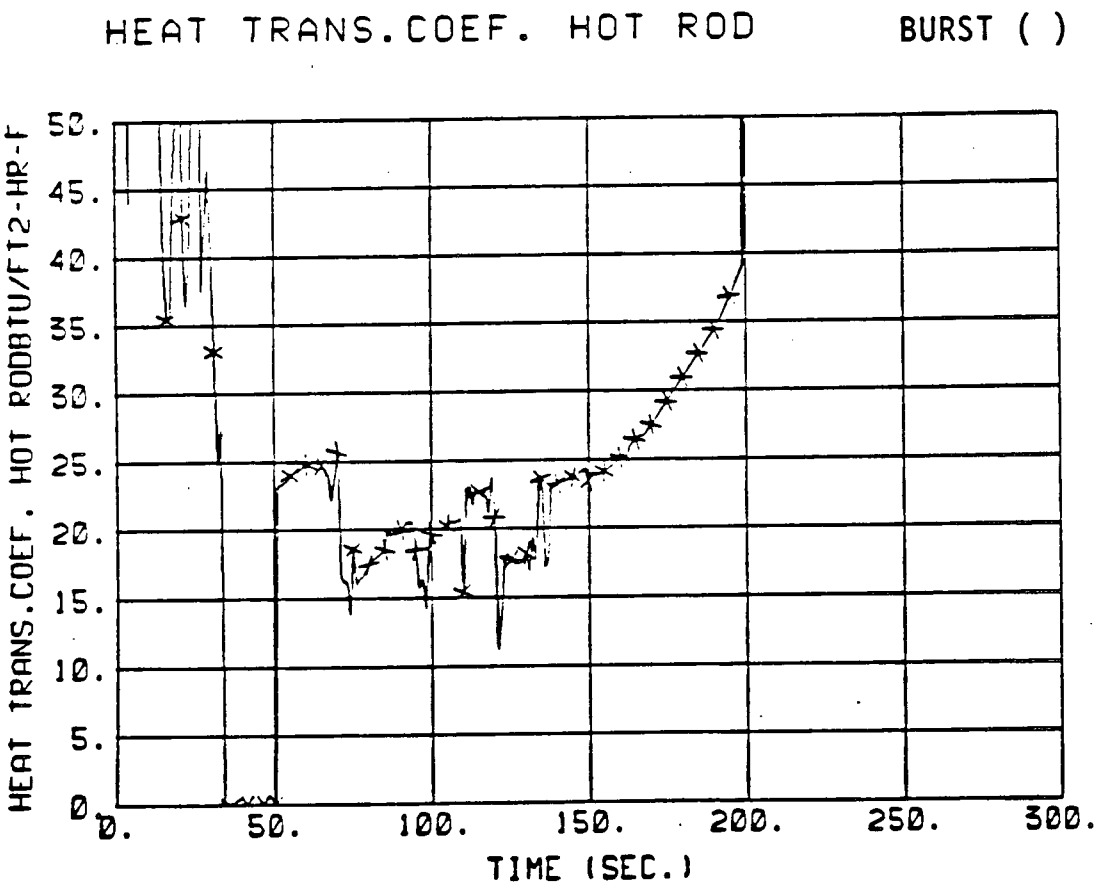

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Figure 14.3-13
Heat Transfer Coefficient-
DECLG (C_D=0.6)
(High T_{ave}) MINSI

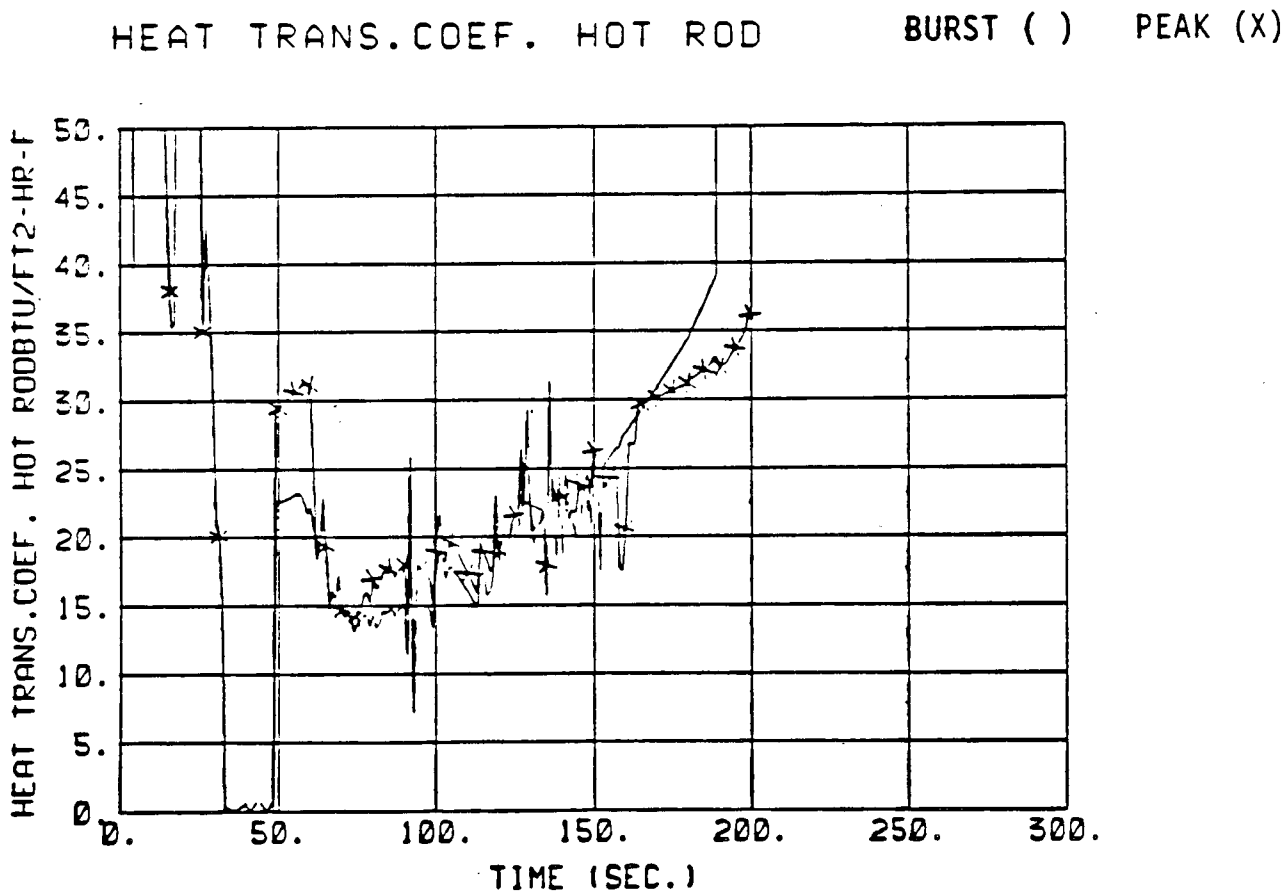

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Figure 14.3-14
Heat Transfer Coefficient-
DECLG (C_D=0.8)
(High T_{ave}) MINSI

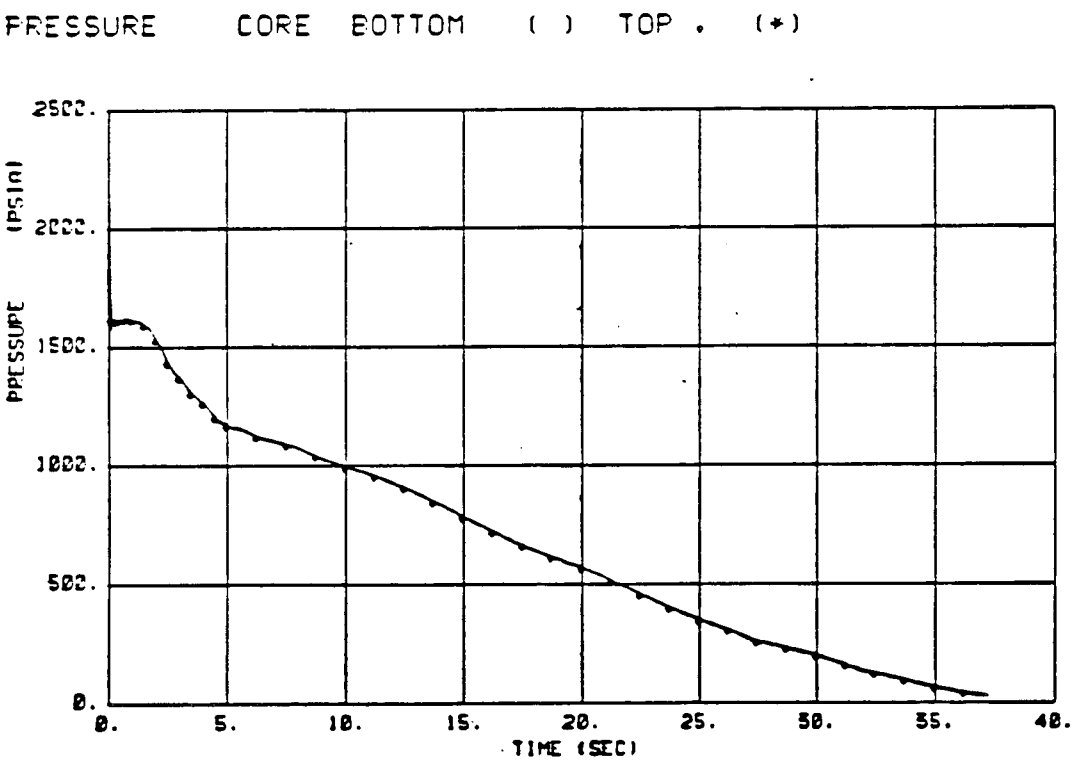

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Figure 14.3-15a
Core Pressure-DECLG (C_D=0.4)
(High T_{ave}) MINSI

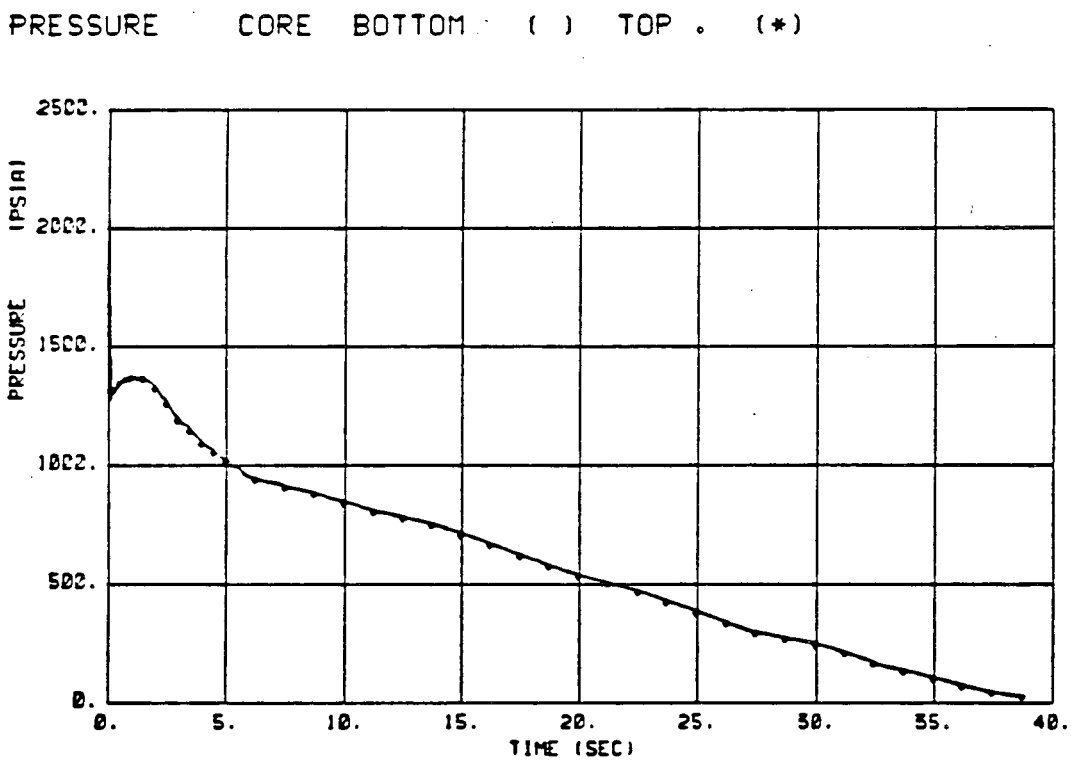

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Figure 14.3-15b
Core Pressure-DECLG (C_D=0.4)
(Low T_{ave}) MINSI

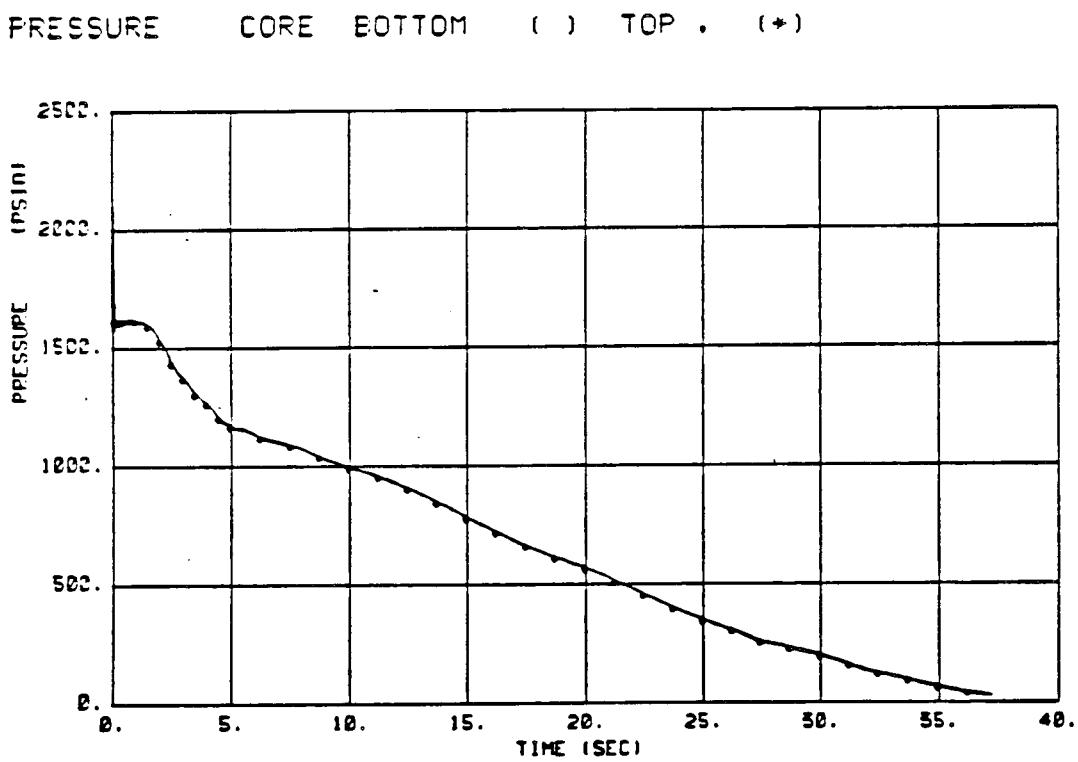

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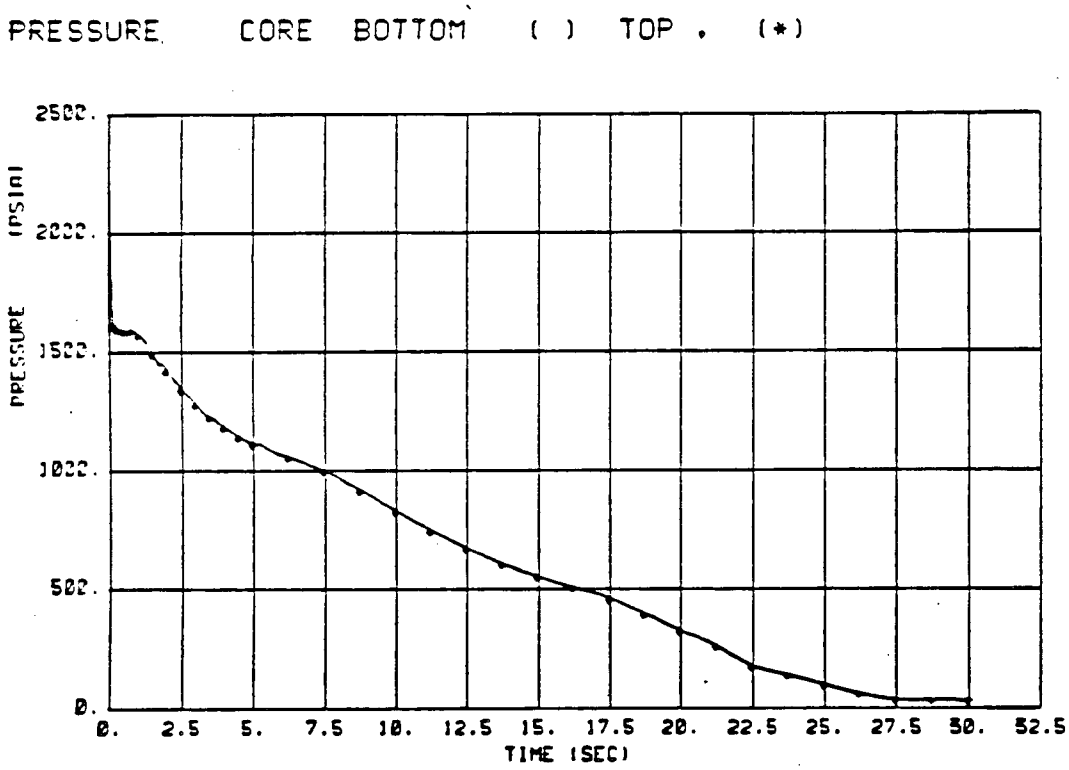
Figure 14.3-15c
Core Pressure-DECLG (C_D=0.4)
(High T_{ave}) MAXSI



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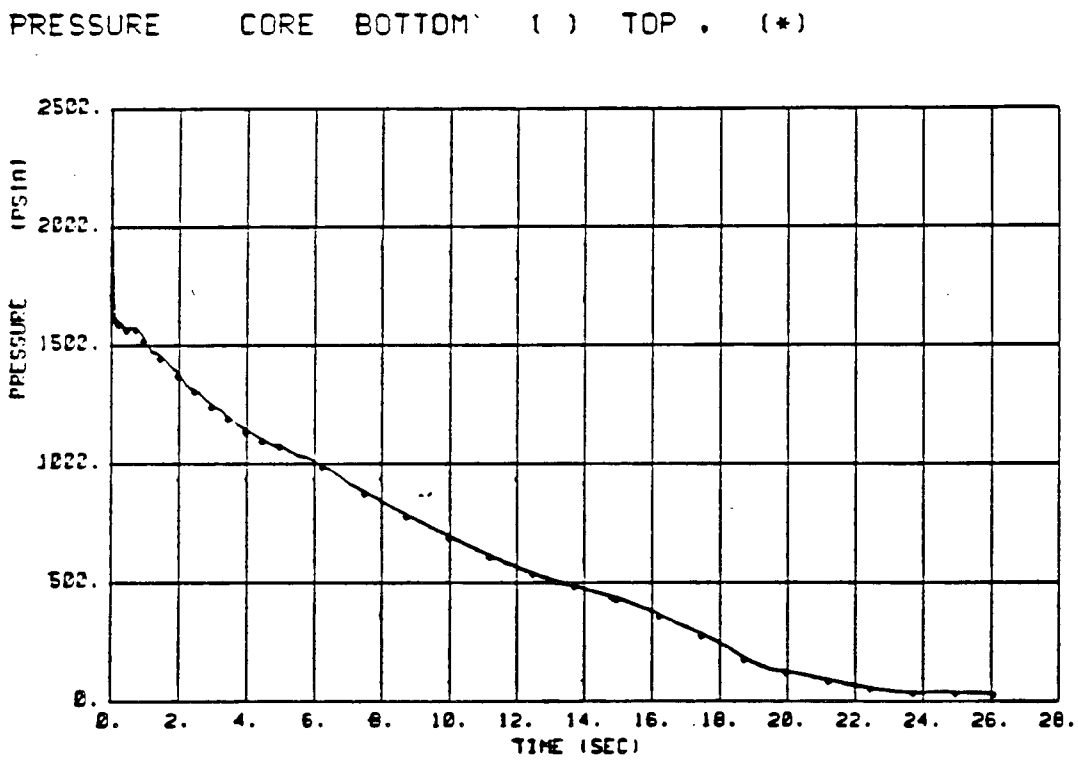
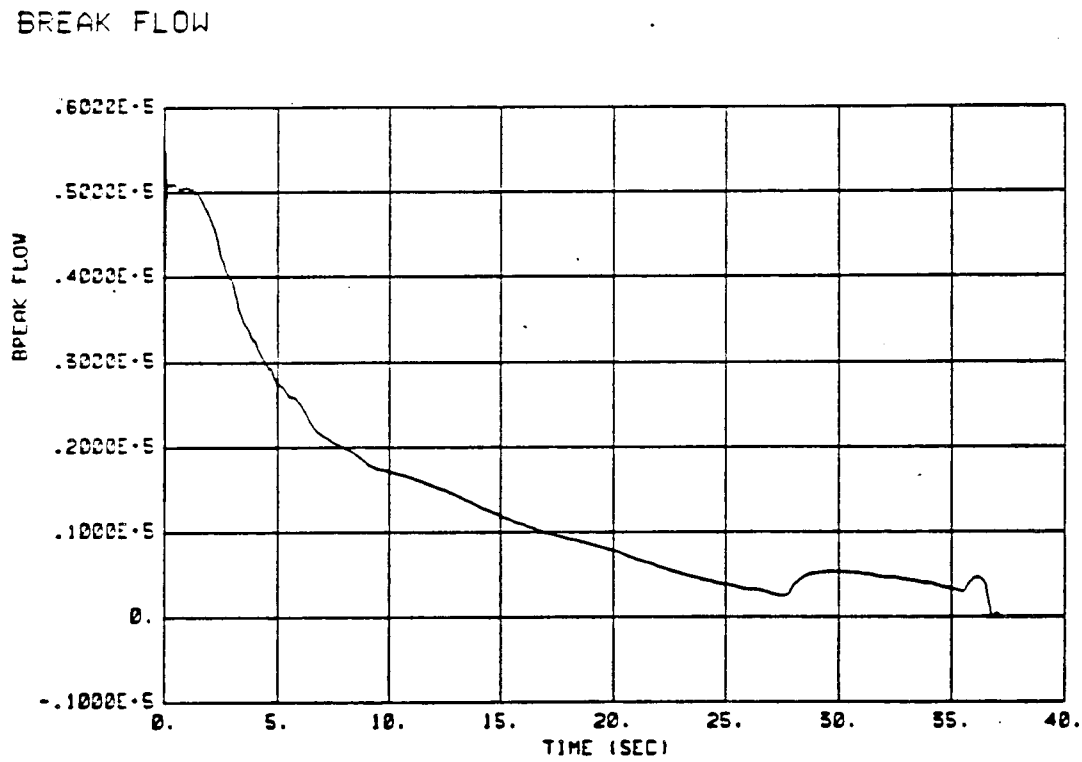
INDIAN POINT UNIT 2

Figure 14.3-16
Core Pressure-DECLG (C_D=0.6)
(High T_{ave}) MINSI



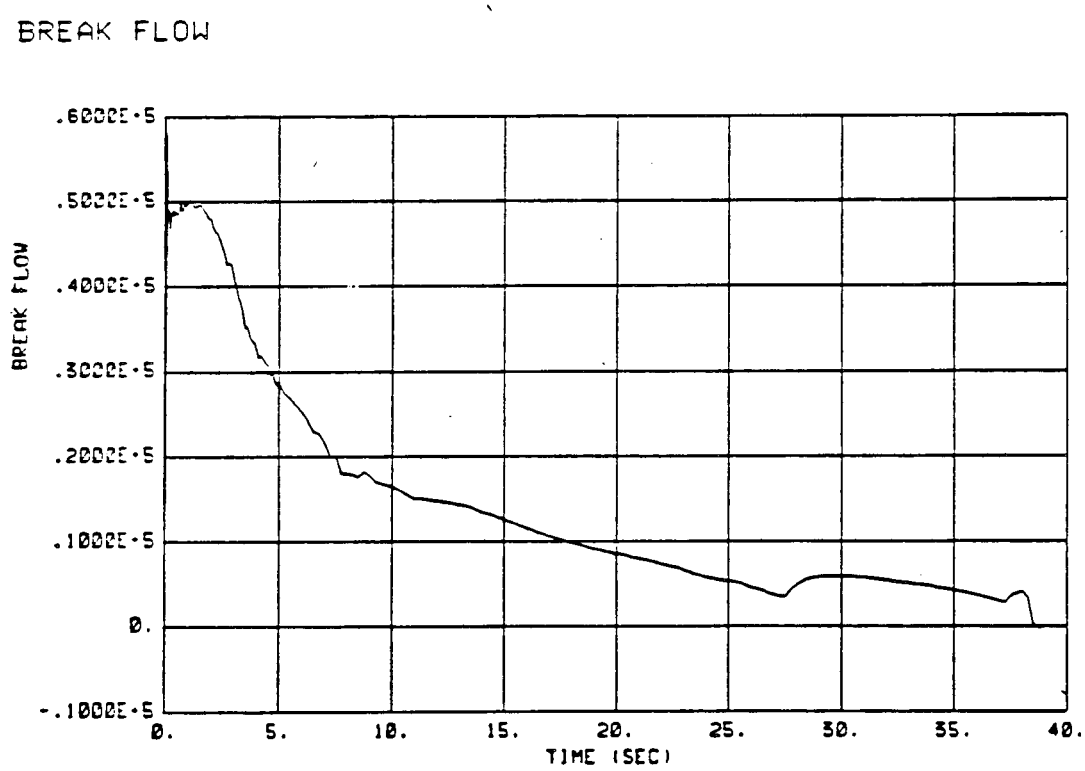
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-17
Core Pressure-DECLG (C_D=0.8)
(High T_{ave}) MINSI



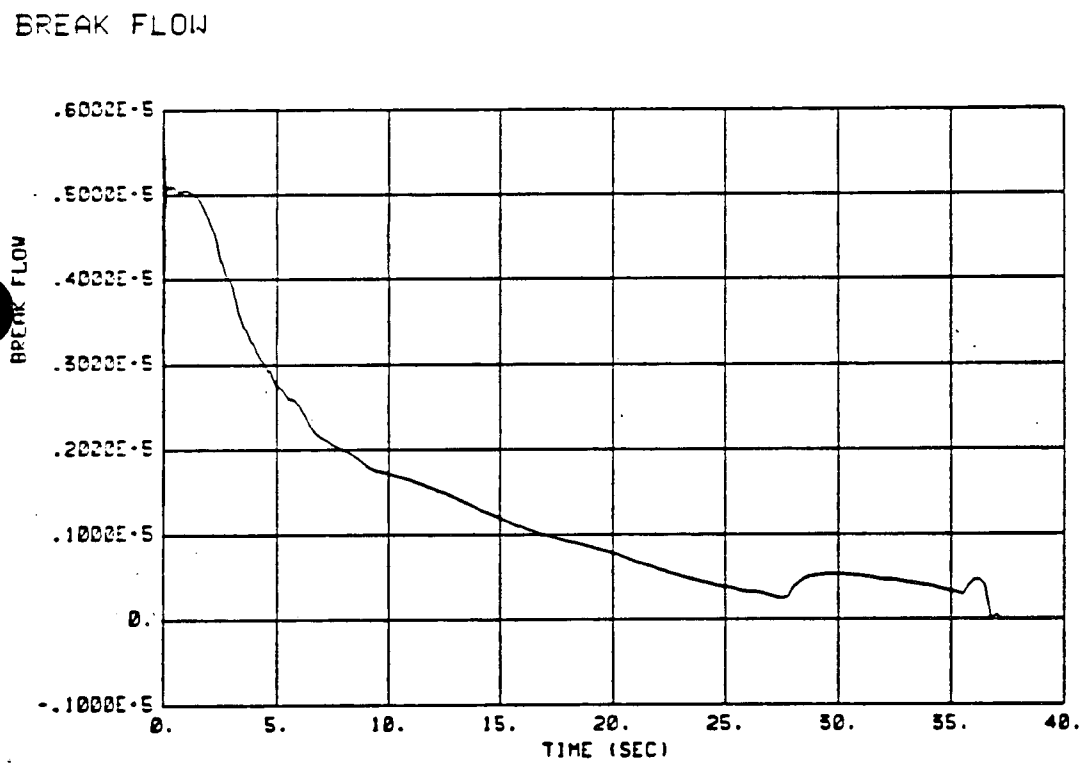
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-18a
Break Flow Rate-DECLG (C_D=0.4)
(High T_{ave}) MINSI



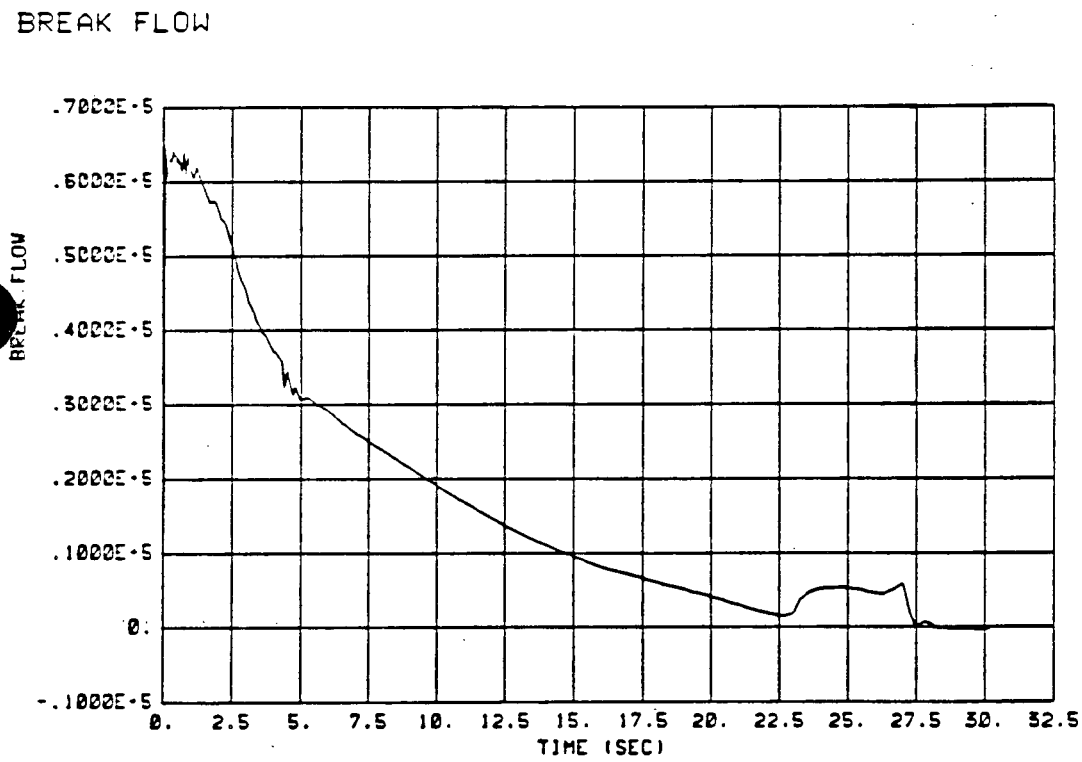
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-18b
Break Flow Rate-DECLG (C_D=0.4)
(Low T_{ave}) MINSI



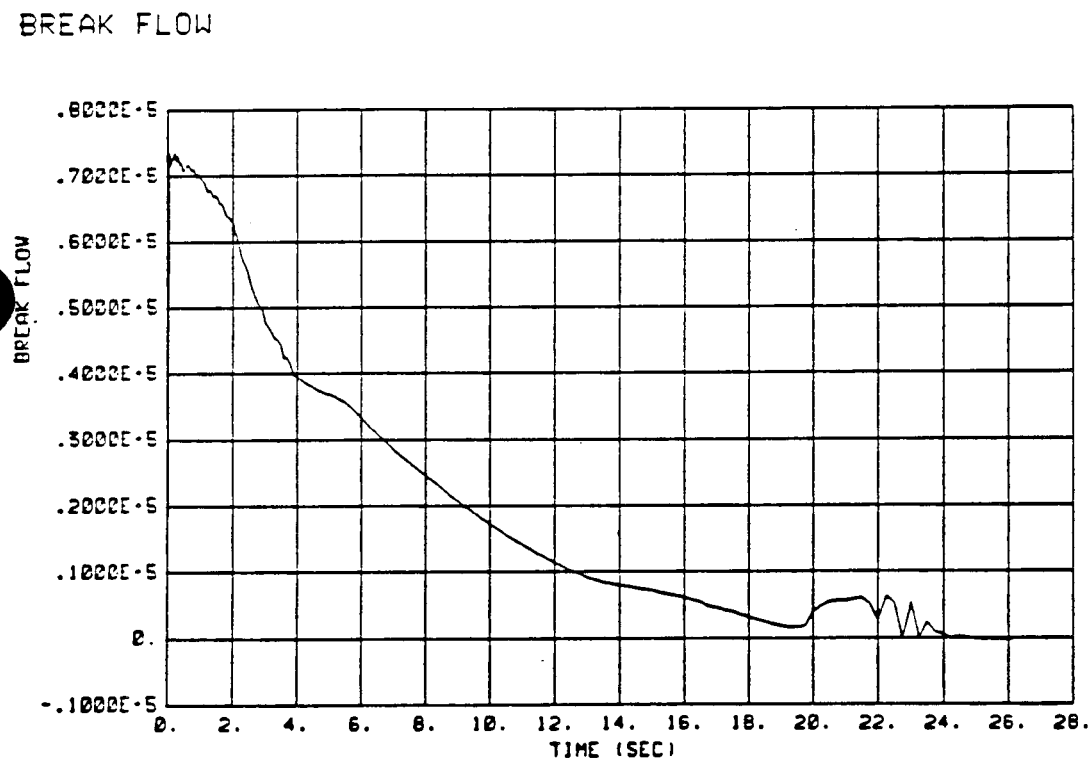
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-18c
Break Flow Rate- DECLG (C_D=0.4)
(High T_{ave}) MAXSI



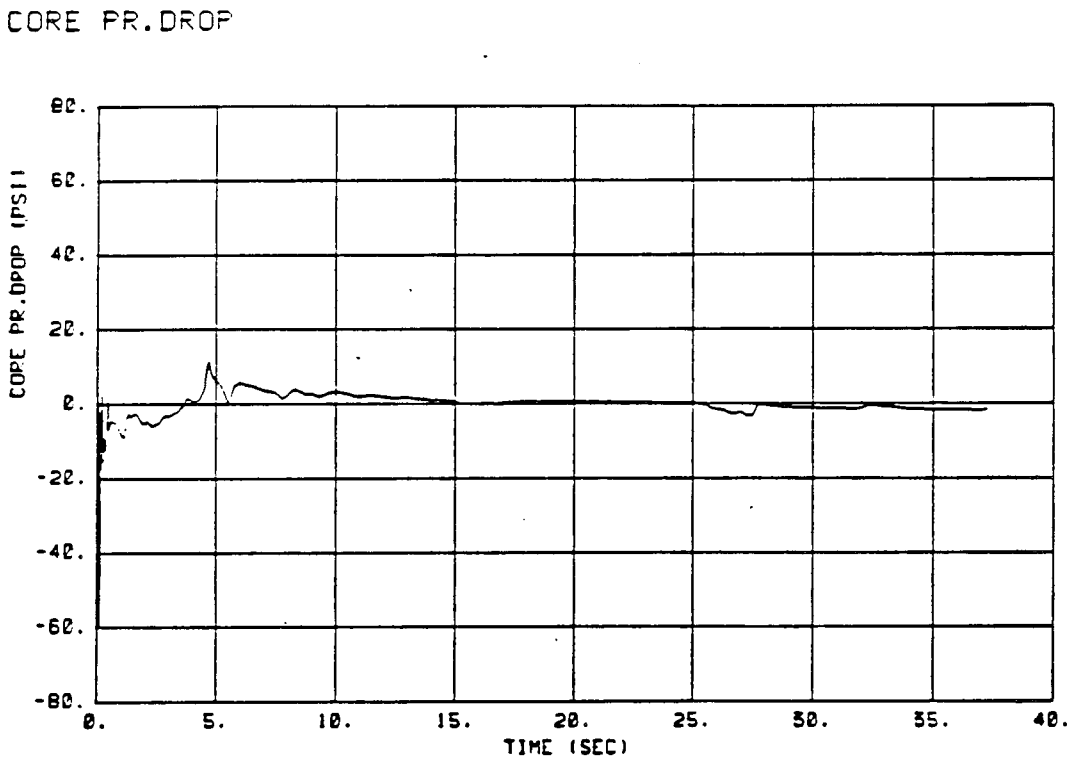
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-19
Break Flow Rate-DECLG (C_D=0.6)
(High T_{ave}) MINSI



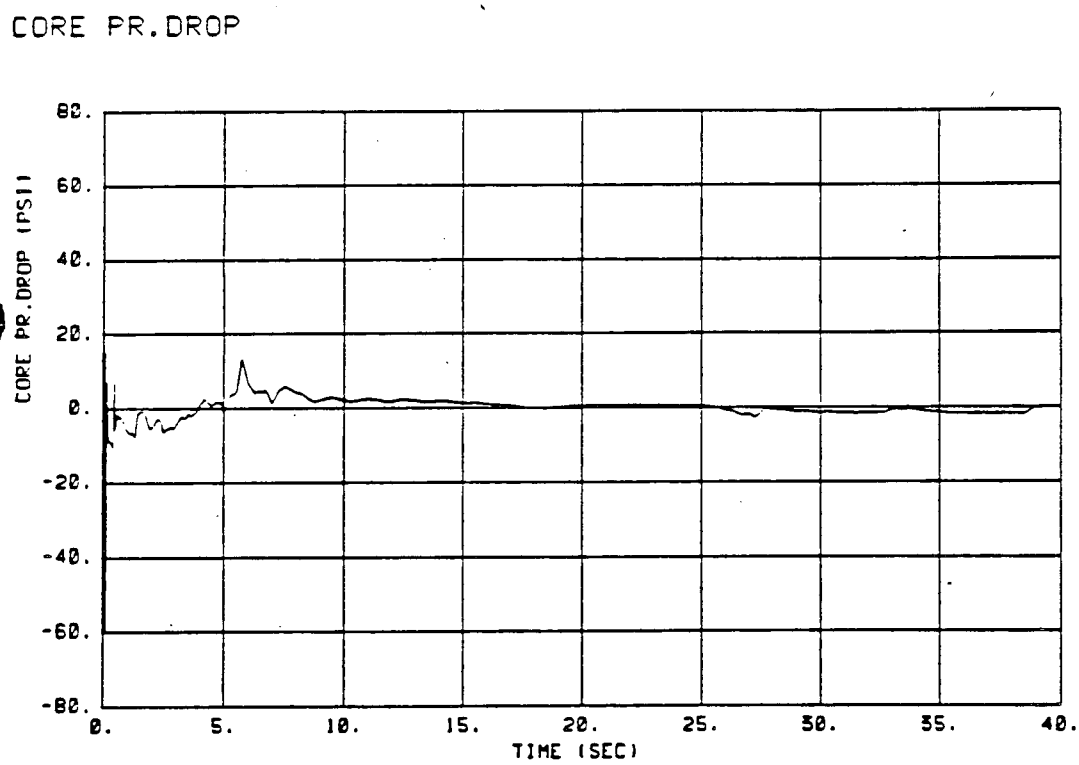
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-20
Break Flow Rate-DECLG (C_D=0.8)
(High T_{ave}) MINSI



CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-21a
Core Pressure Drop-
DECLG (C_D=0.4)
(High T_{ave}) MINSI



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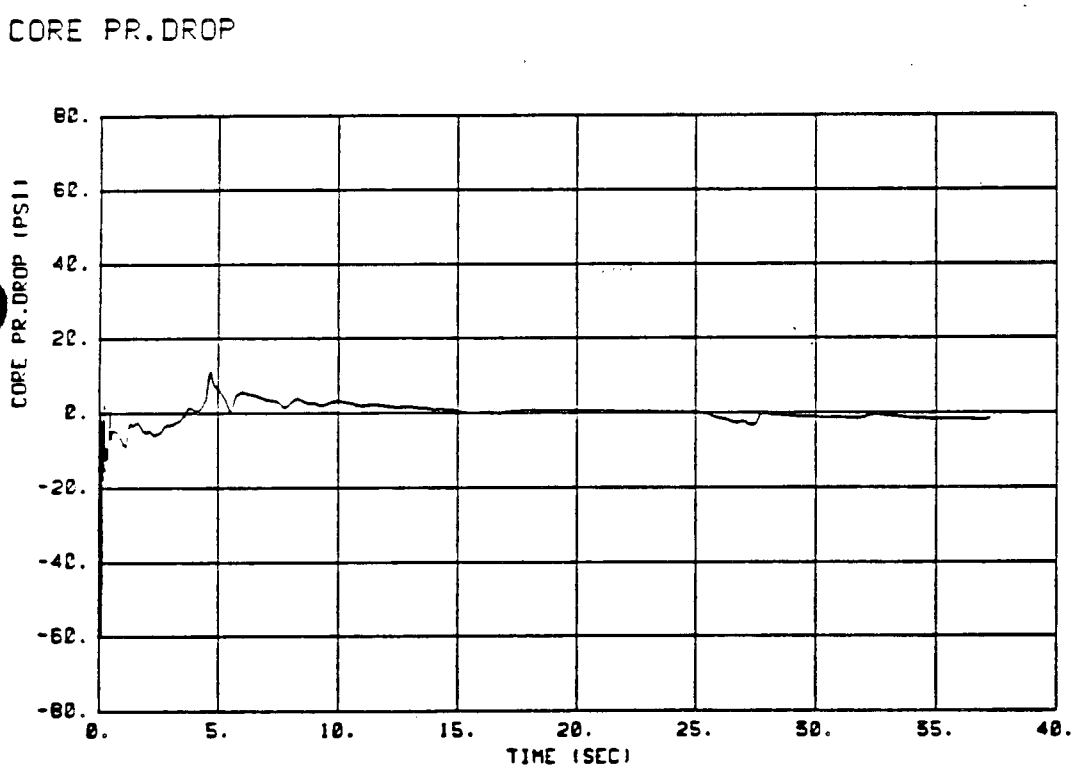
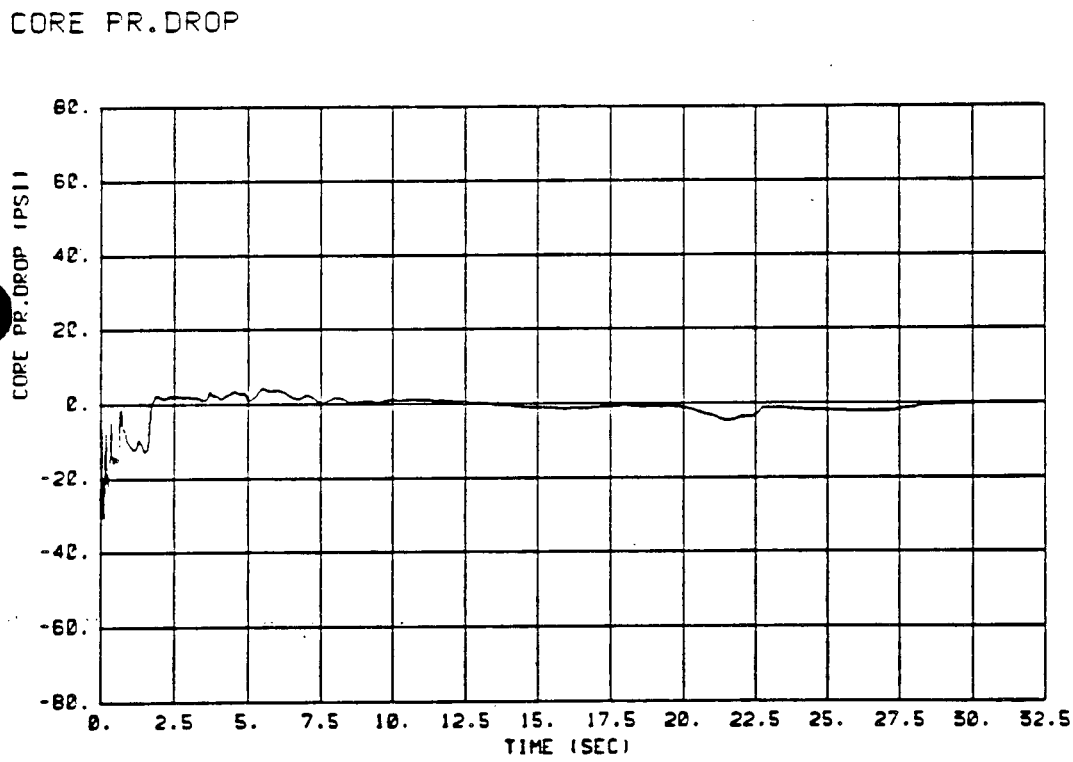
INDIAN POINT UNIT 2

Figure 14.3-21b
Core Pressure Drop-DECLG (C_D=0.4)
(Low T_{ave}) MINSI



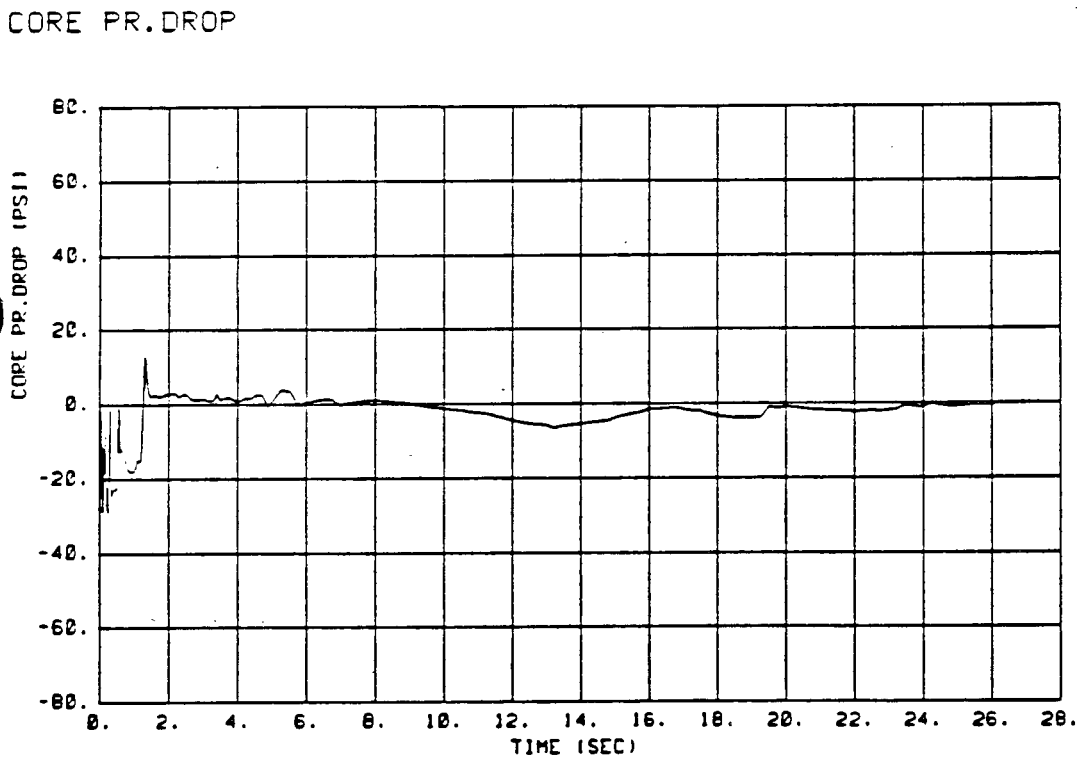
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-21c
Core Pressure Drop-
DECLG (C_D=0.4)
(High T_{ave}) MAXSI



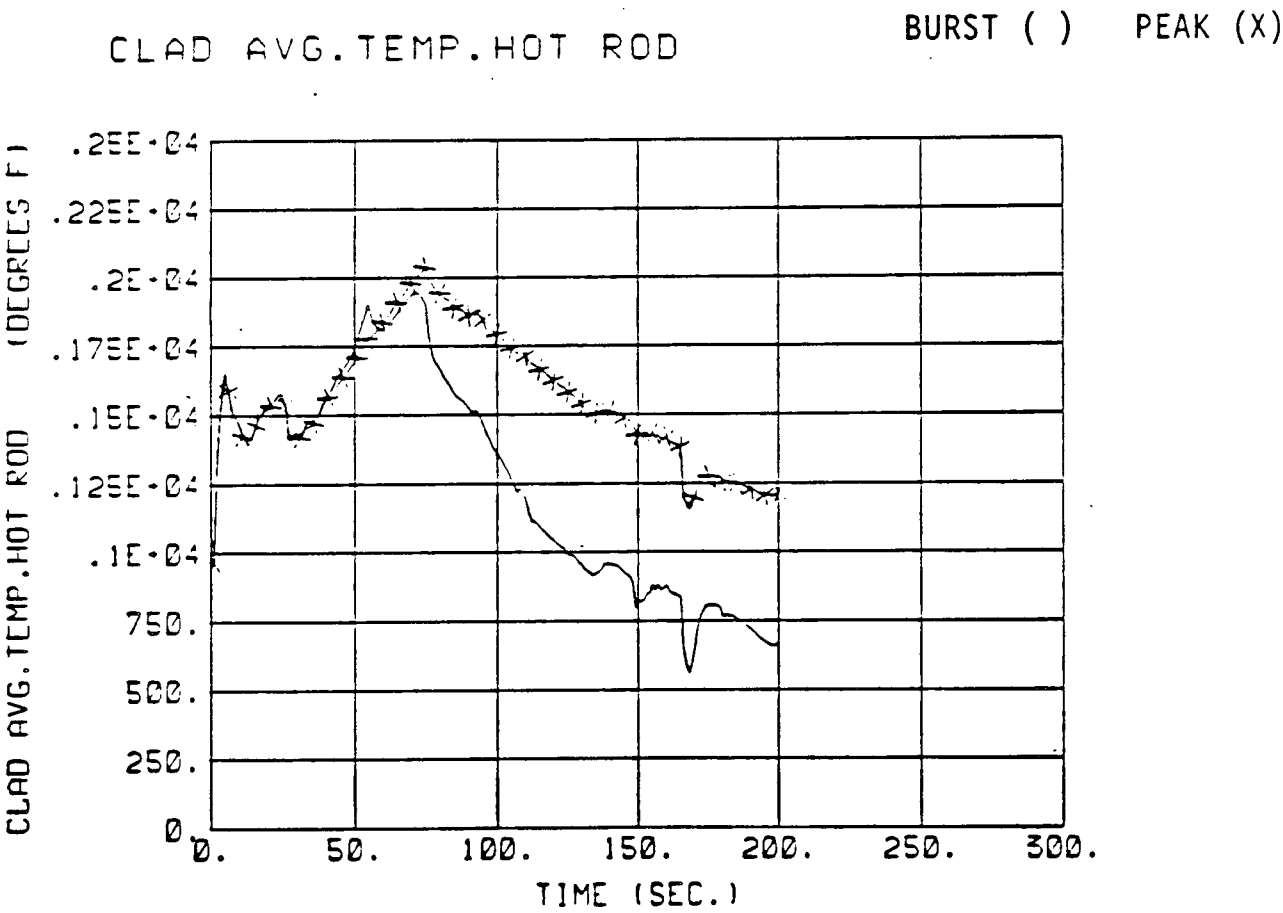
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-22
Core Pressure Drop-DECLG (C_D=0.6)
(High T_{ave}) MINSI



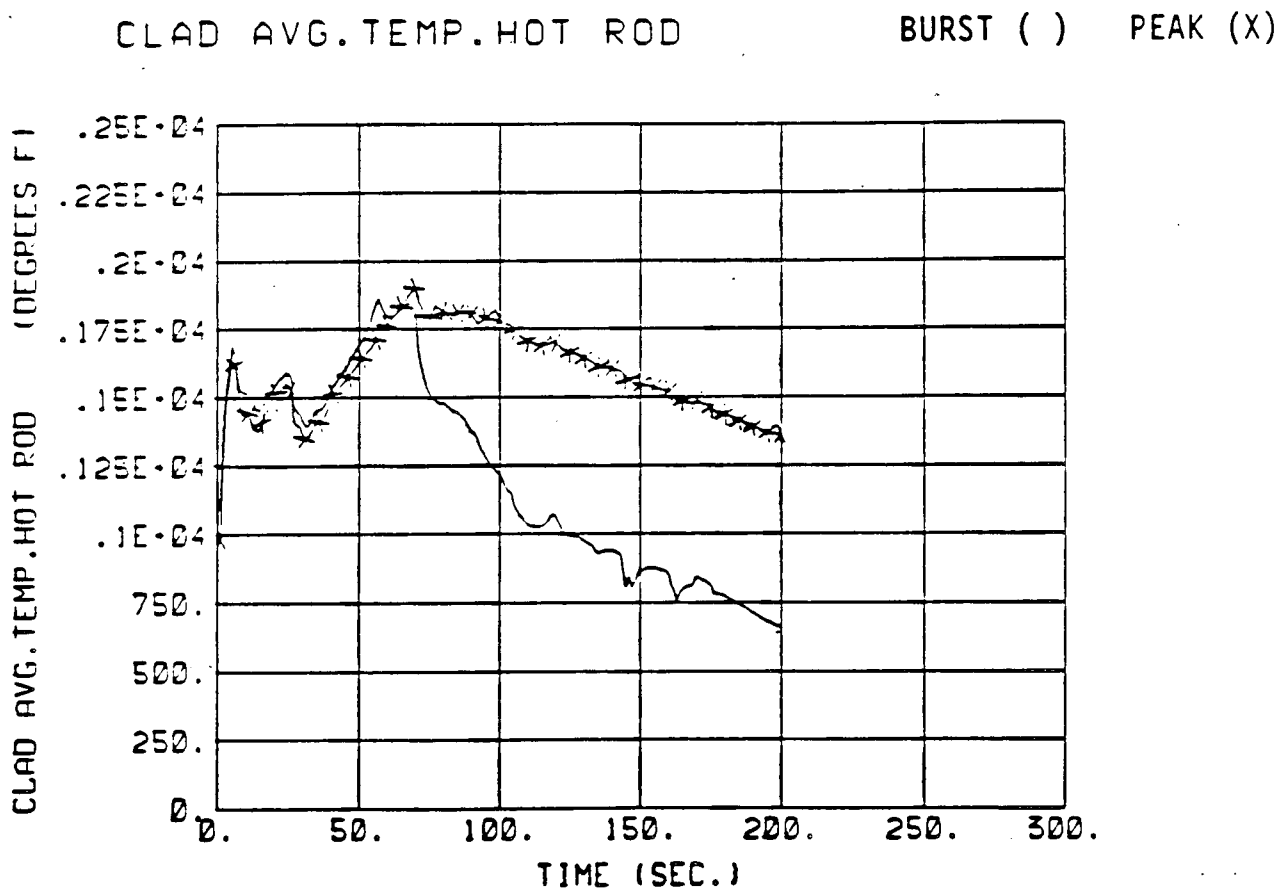
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-23
Core Pressure Drop-DECLG (C_D=0.8)
(High T_{ave}) MINSI



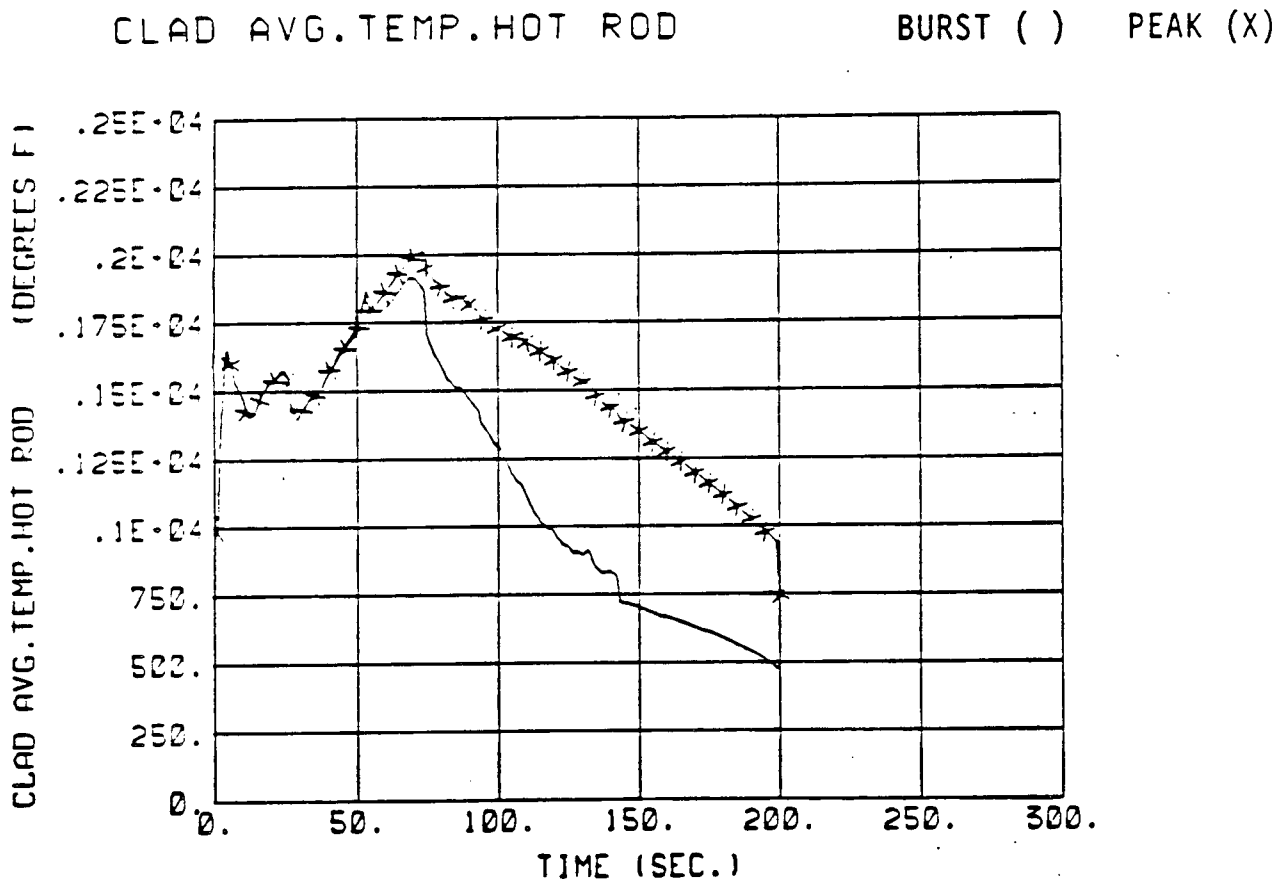
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-24a
Peak Clad Temperature-
DECLG (C_D=0.4)
(High T_{ave}) MINSI



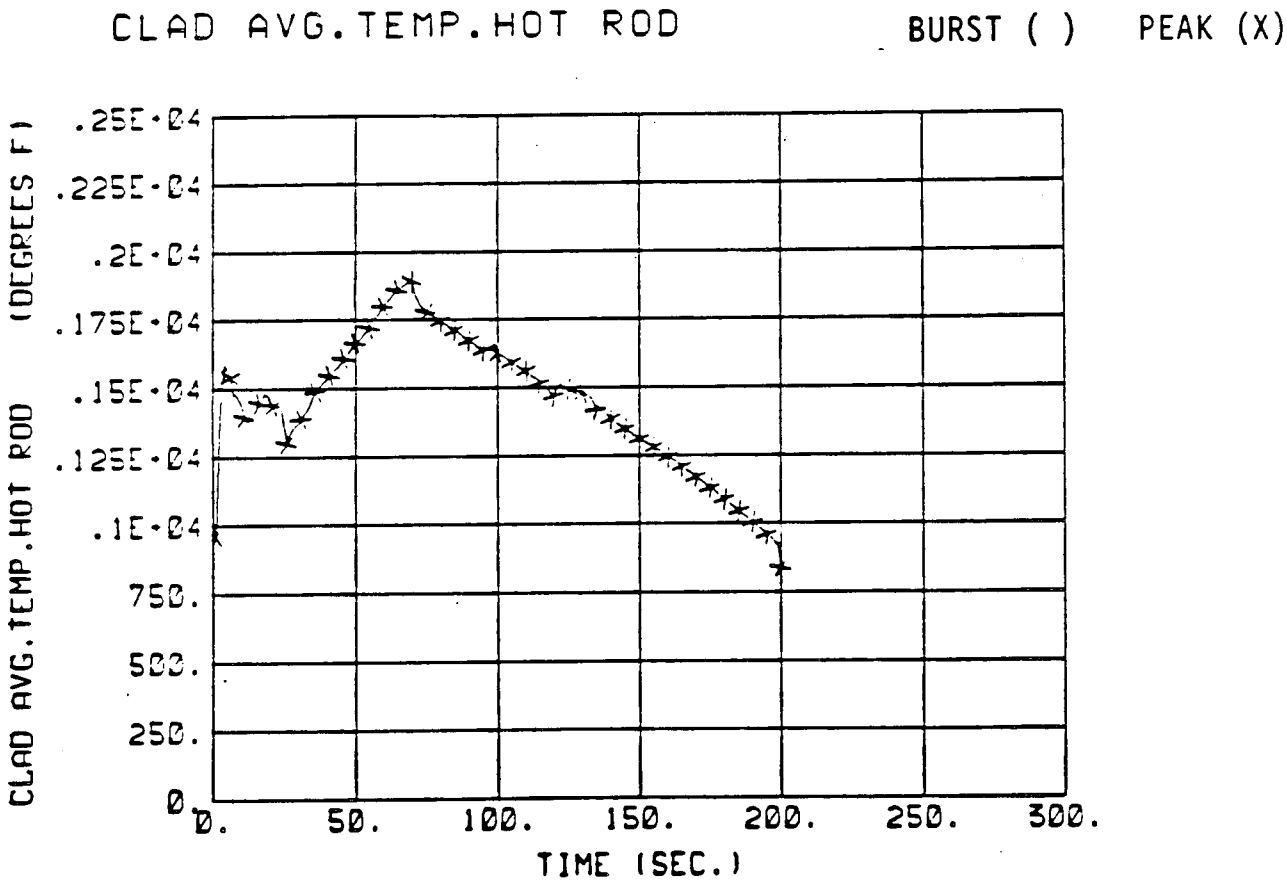
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-24b
Peak Clad Temperature-
DECLG (C_D=0.4)
(Low T_{ave}) MINSI



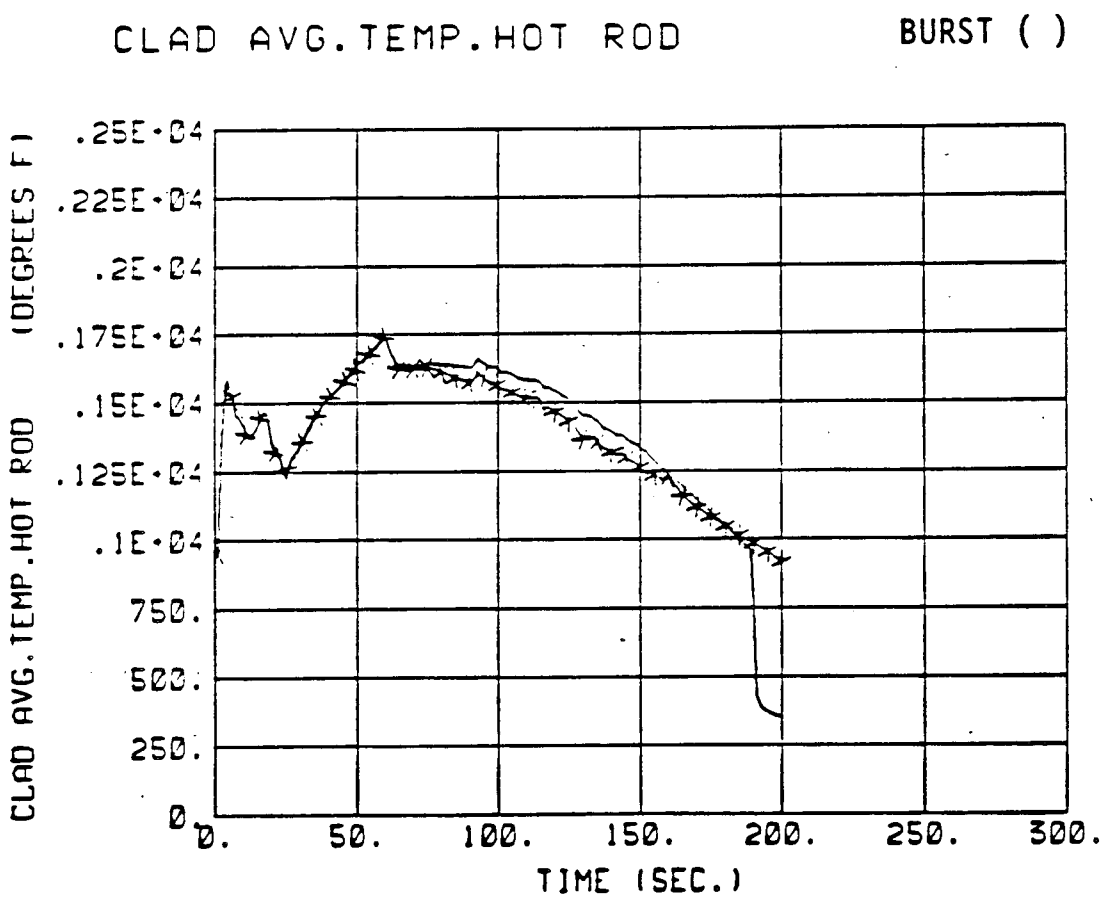
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-24c
Peak Clad Temperature-
DECLG (C_D=0.4)
(High T_{ave}) MAXSI



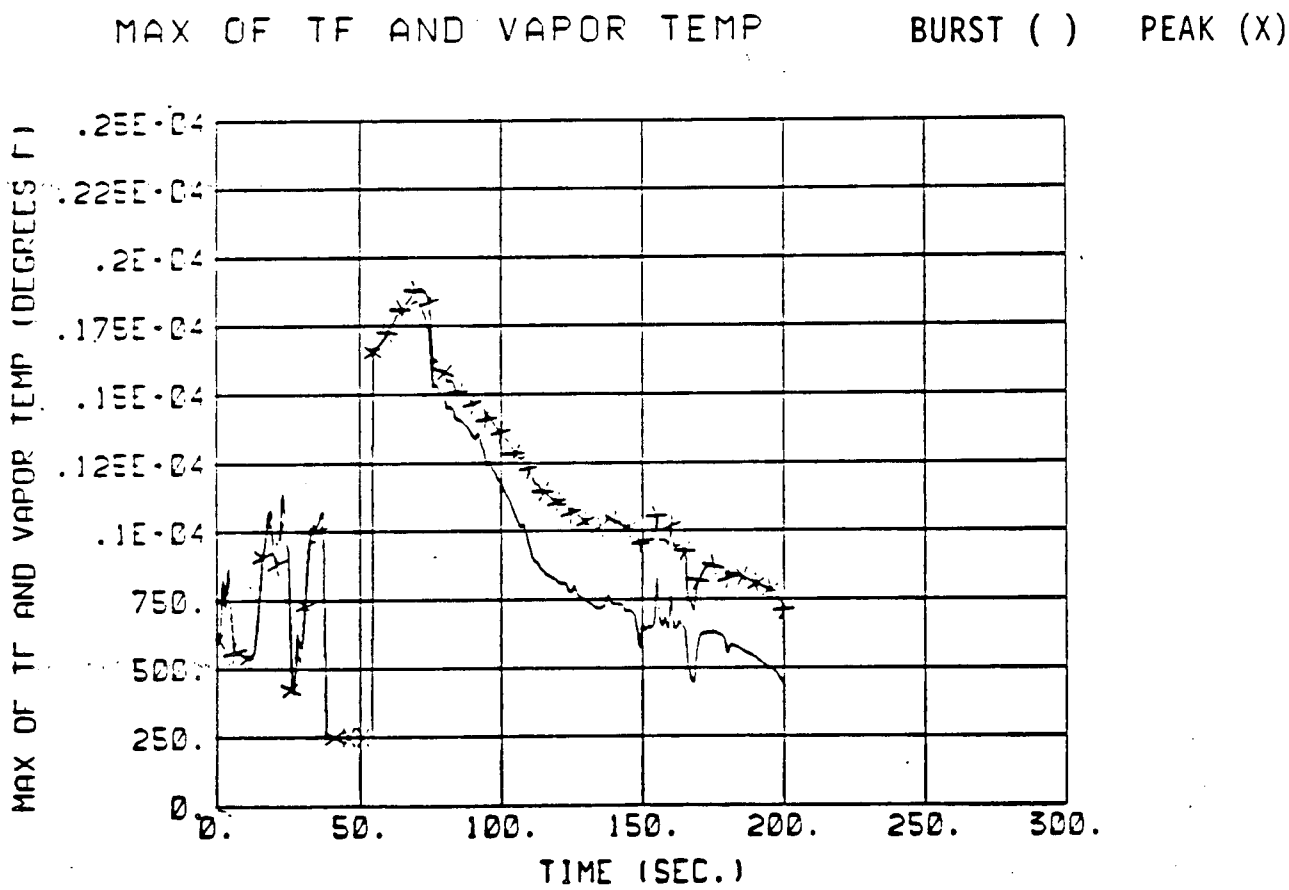
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-25
Peak Clad Temperature-
DECLG (C_D=0.6)
(High T_{ave}) MINSI



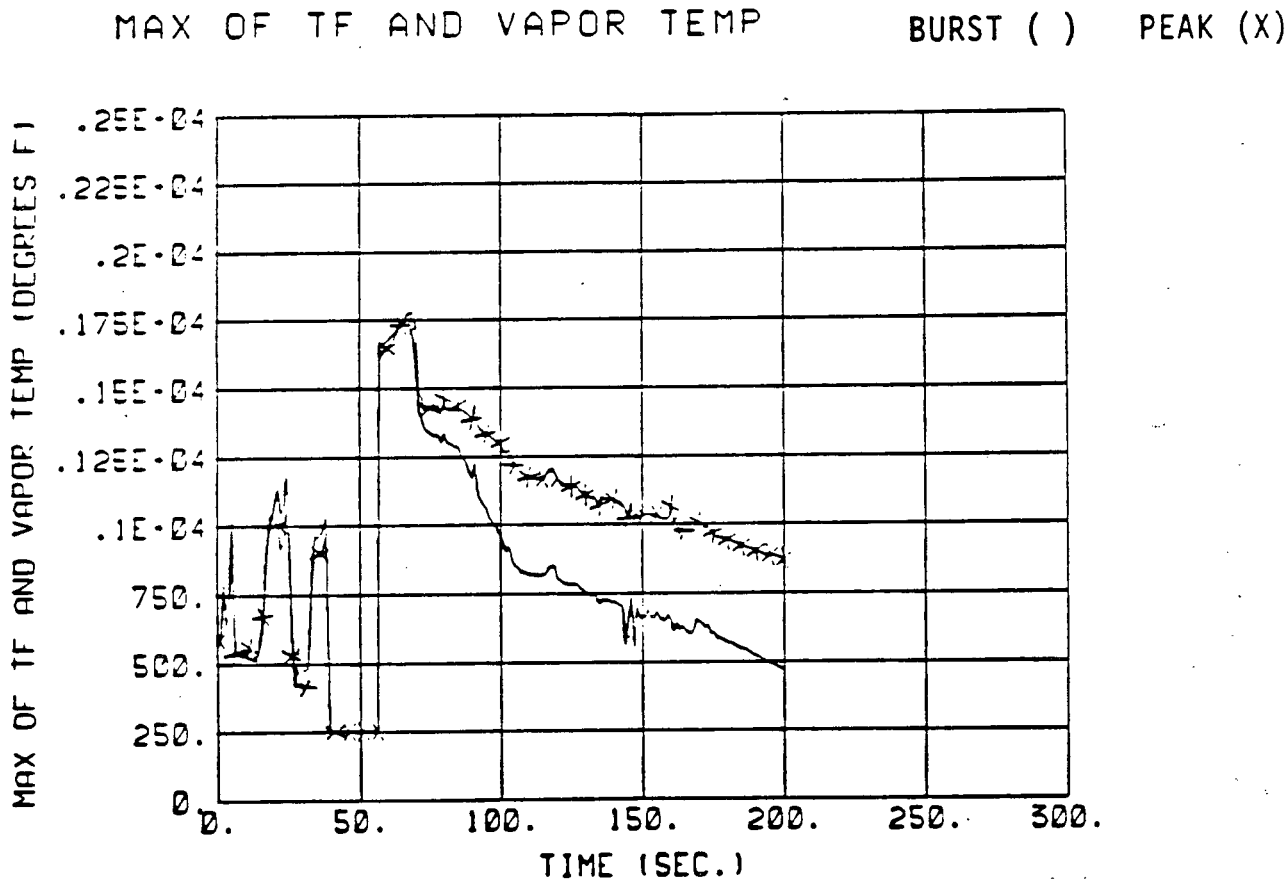
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-26
Peak Clad Temperature-
DECLG (C_D=0.8)
(High T_{ave}) MINSI



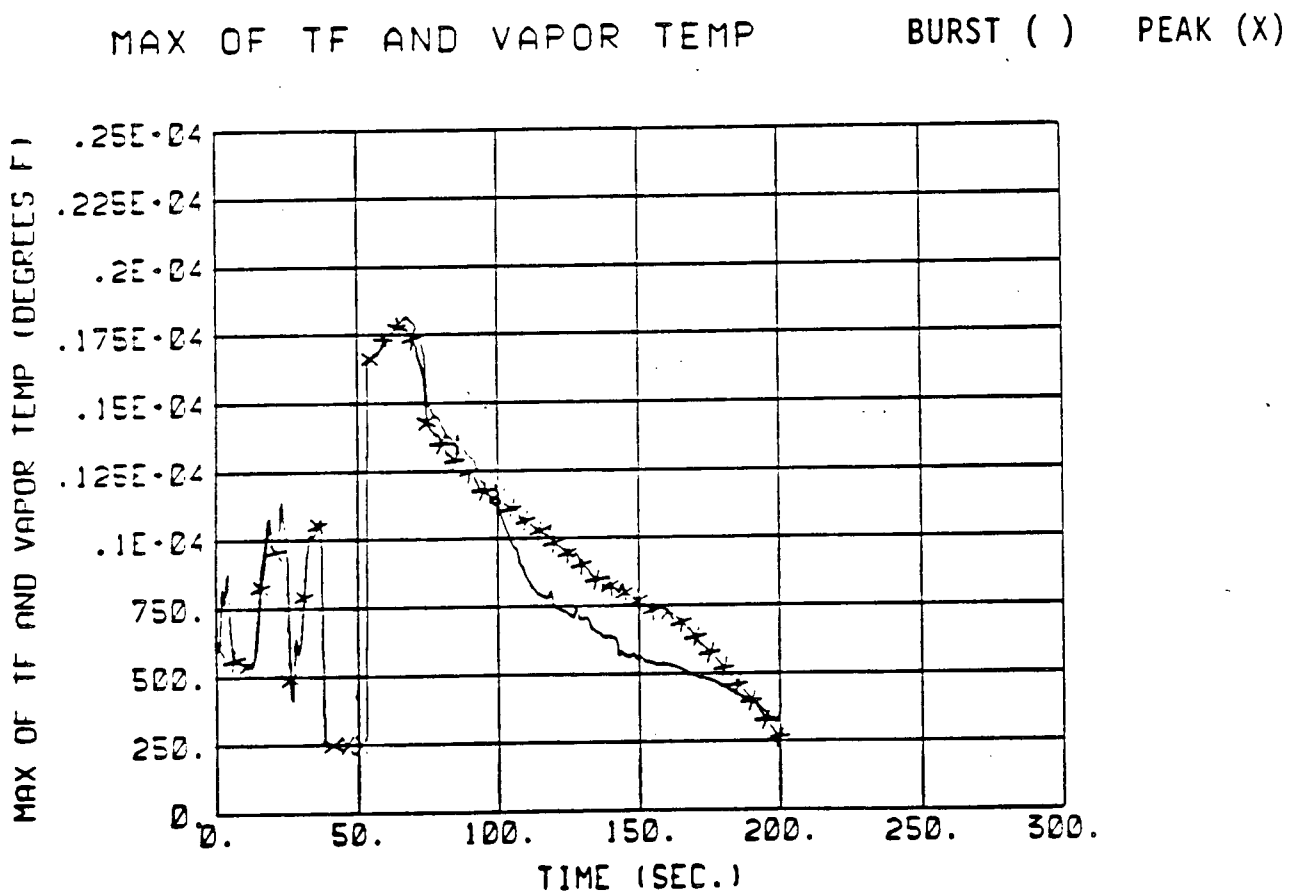
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-27a
Fluid Temperature-DECLG (C_D=0.4)
(High T_{ave}) MINSI



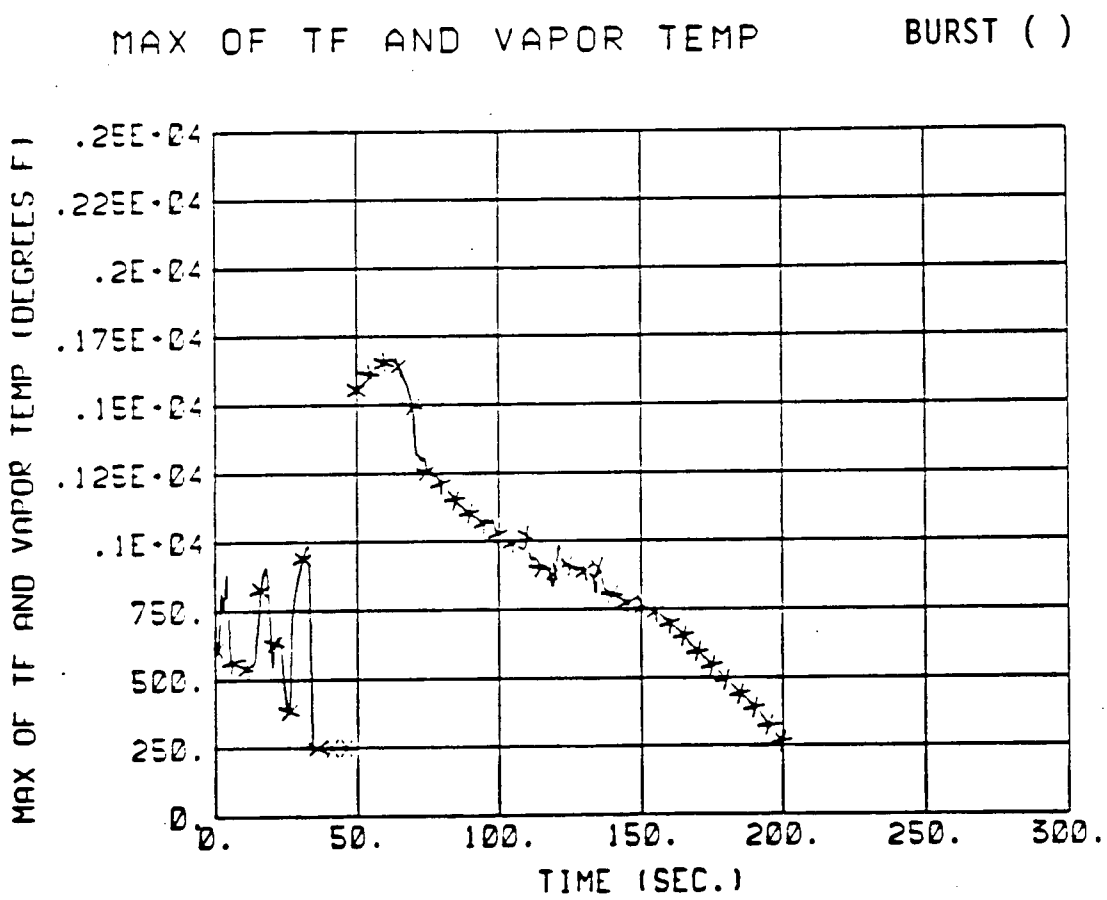
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-27b
Fluid Temperature- DECLG (C_D=0.4)
(Low T_{ave}) MINSI



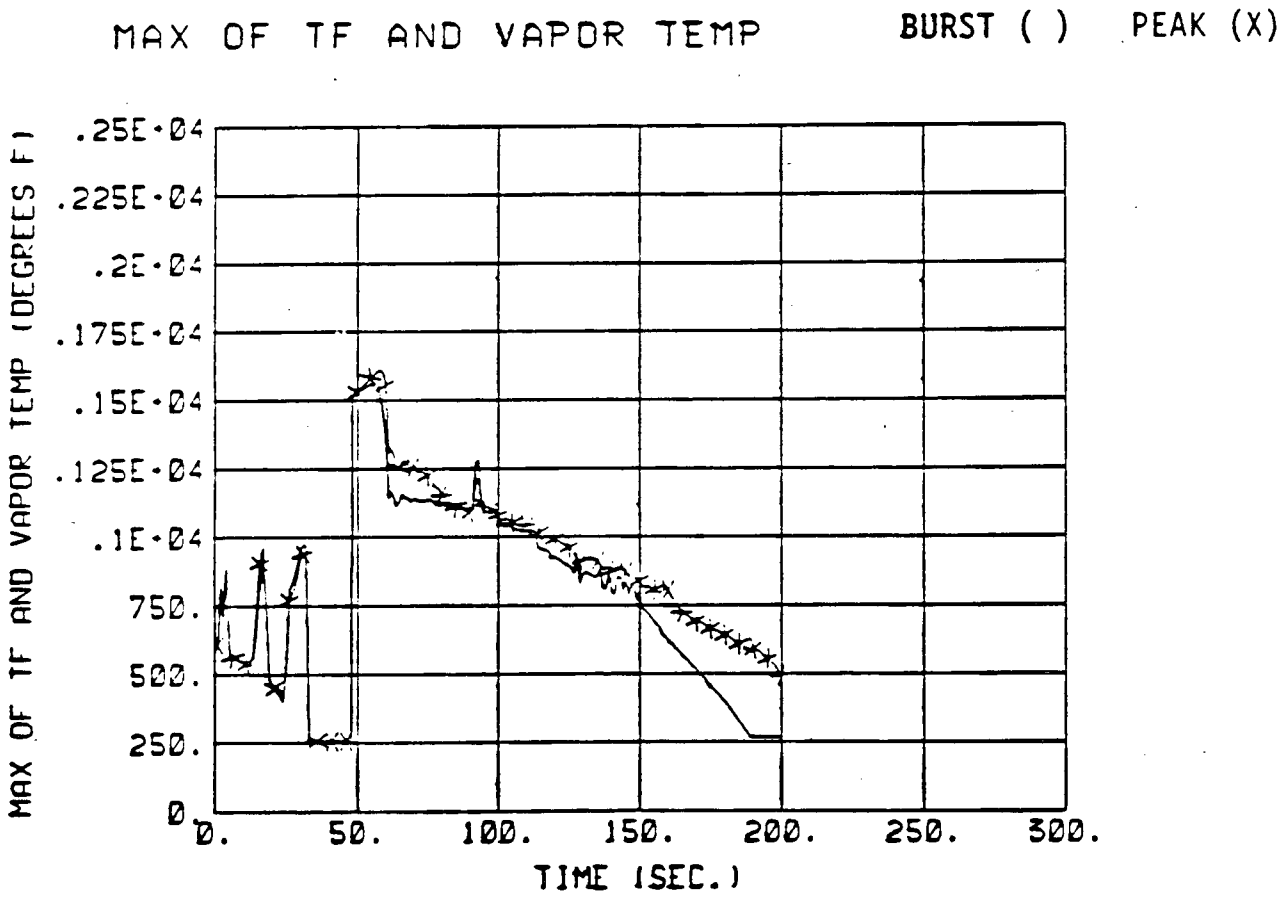
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-27c
Fluid Temperature-DECLG (C_D=0.4)
(High T_{ave}) MAXSI



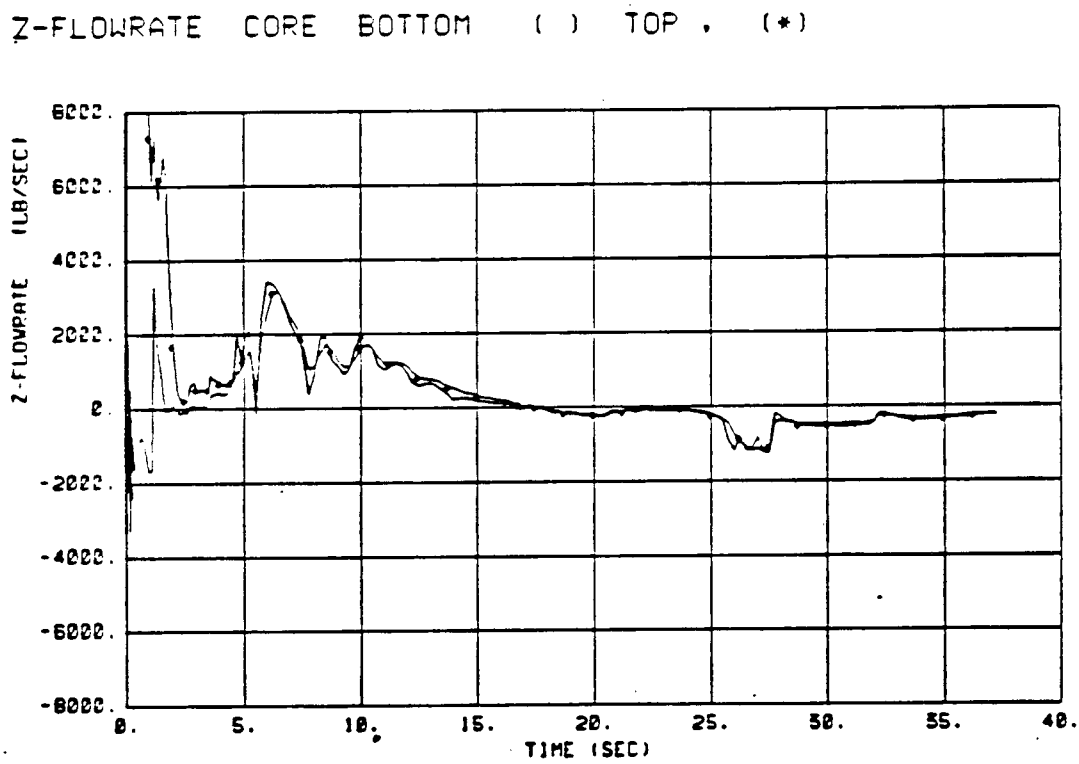
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-28
Fluid Temperature-DECLG (C_D=0.6)
(High T_{ave}) MINSI



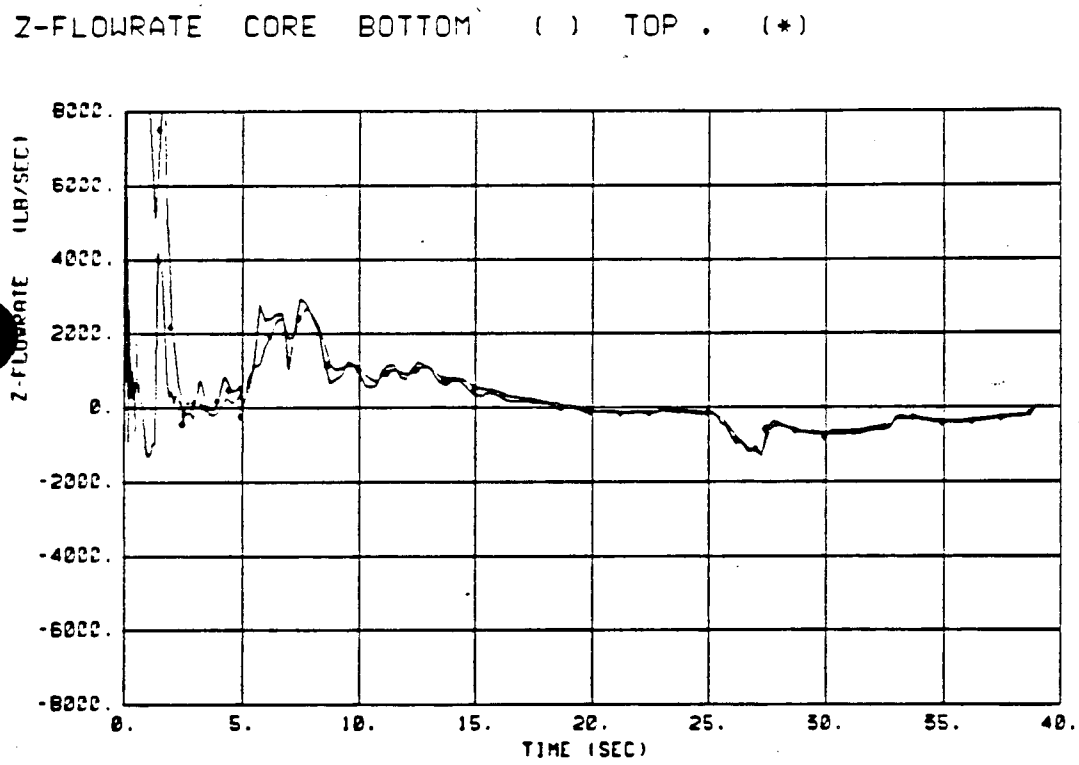
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-29
Fluid Temperature-DECLG (C_D=0.8)
(High T_{ave}) MINSI



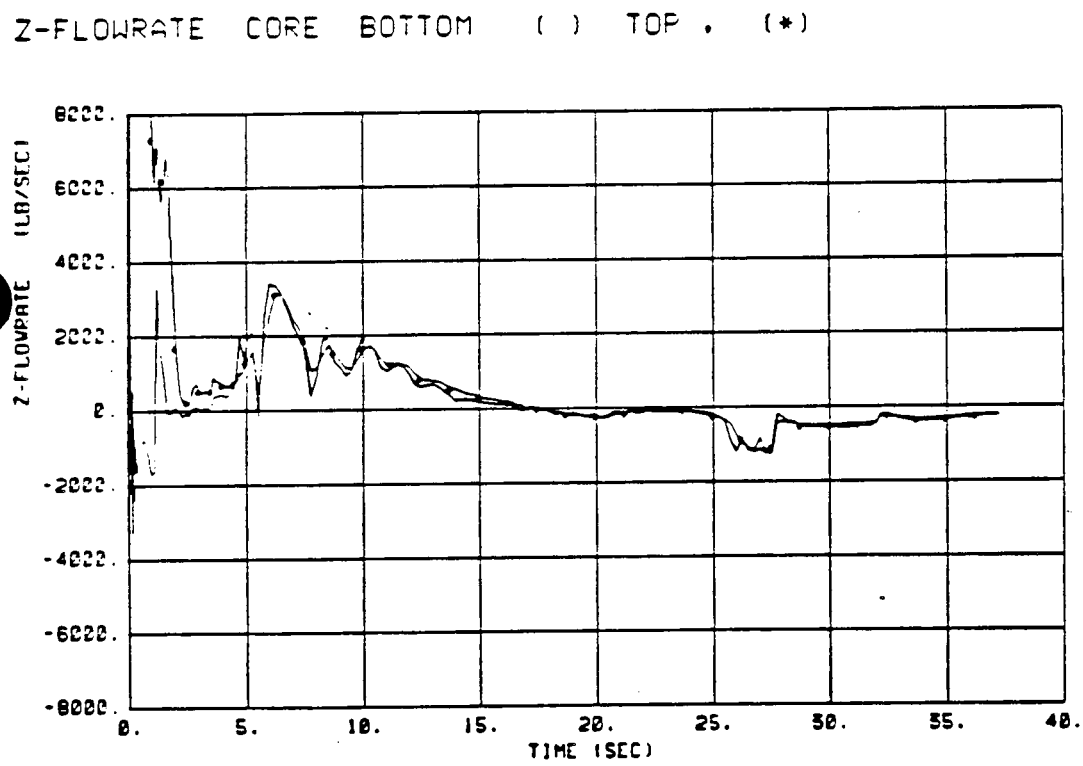
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-30a
Core Flow-Top and Bottom-
DECLG (C_D=0.4)
(High T_{ave}) MINSI



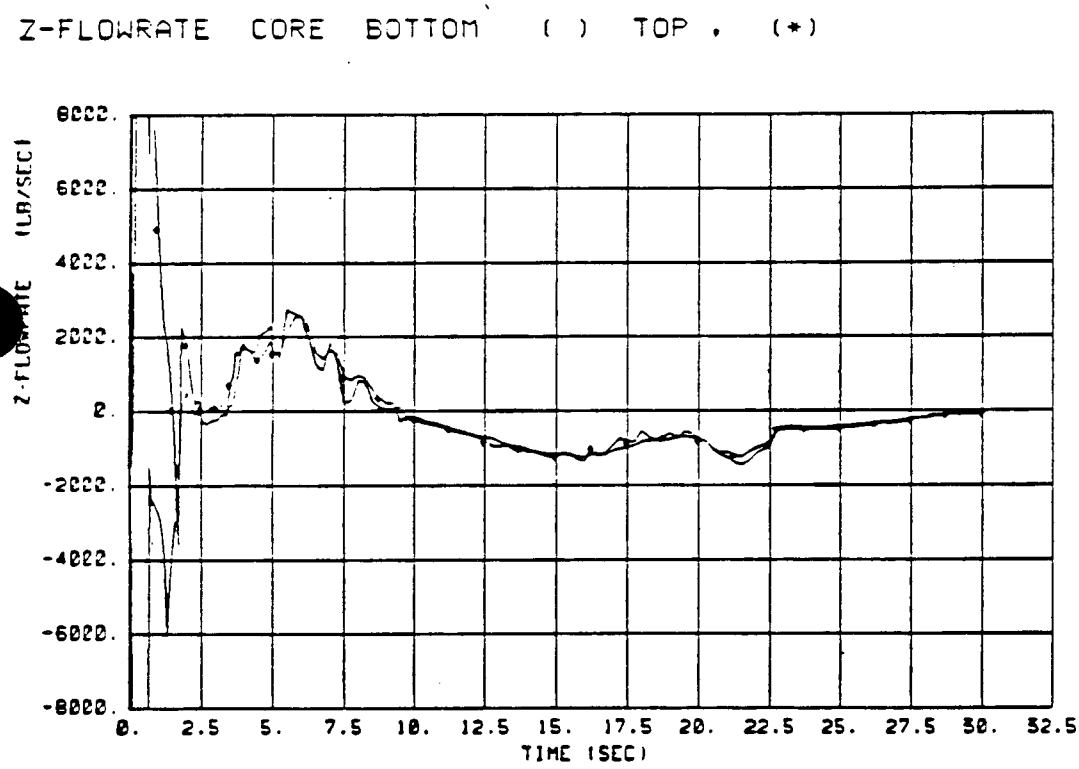
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-30b
Core Flow-Top and Bottom-
DECLG (C_D=0.4)
(Low T_{ave}) MINSI



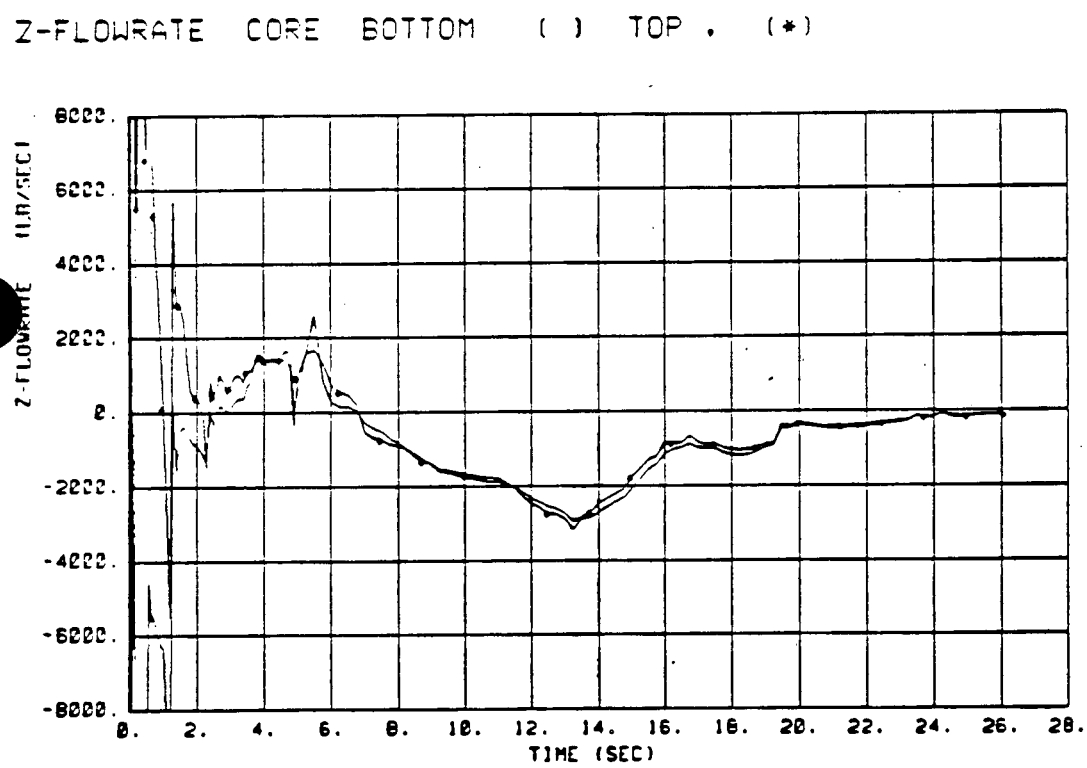
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-30c
Core Flow-Top and Bottom-
DECLG (C_D=0.4)
(High T_{ave}) MAXSI



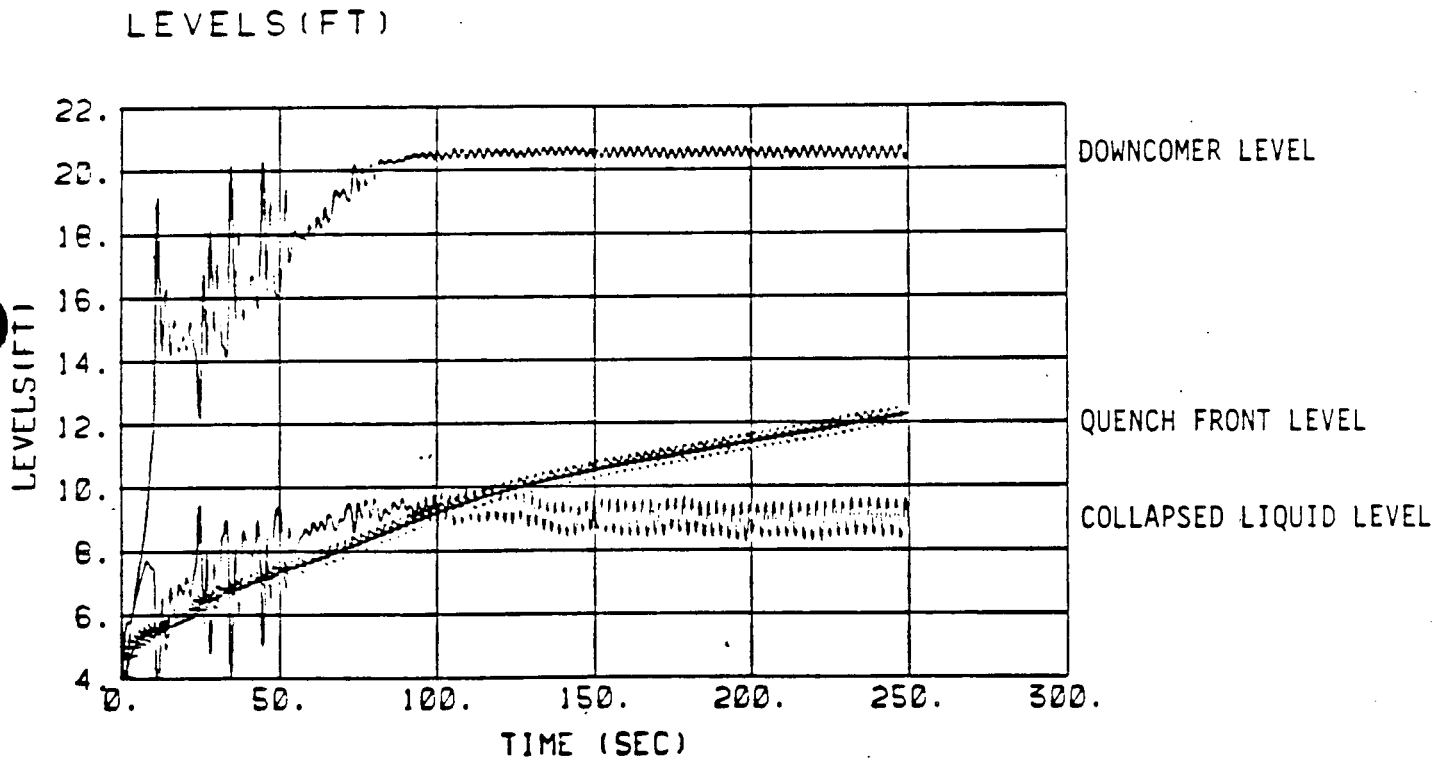
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-31
Core Flow-Top and Bottom-
DECLG (C_D=0.6)
(High T_{ave}) MINSI



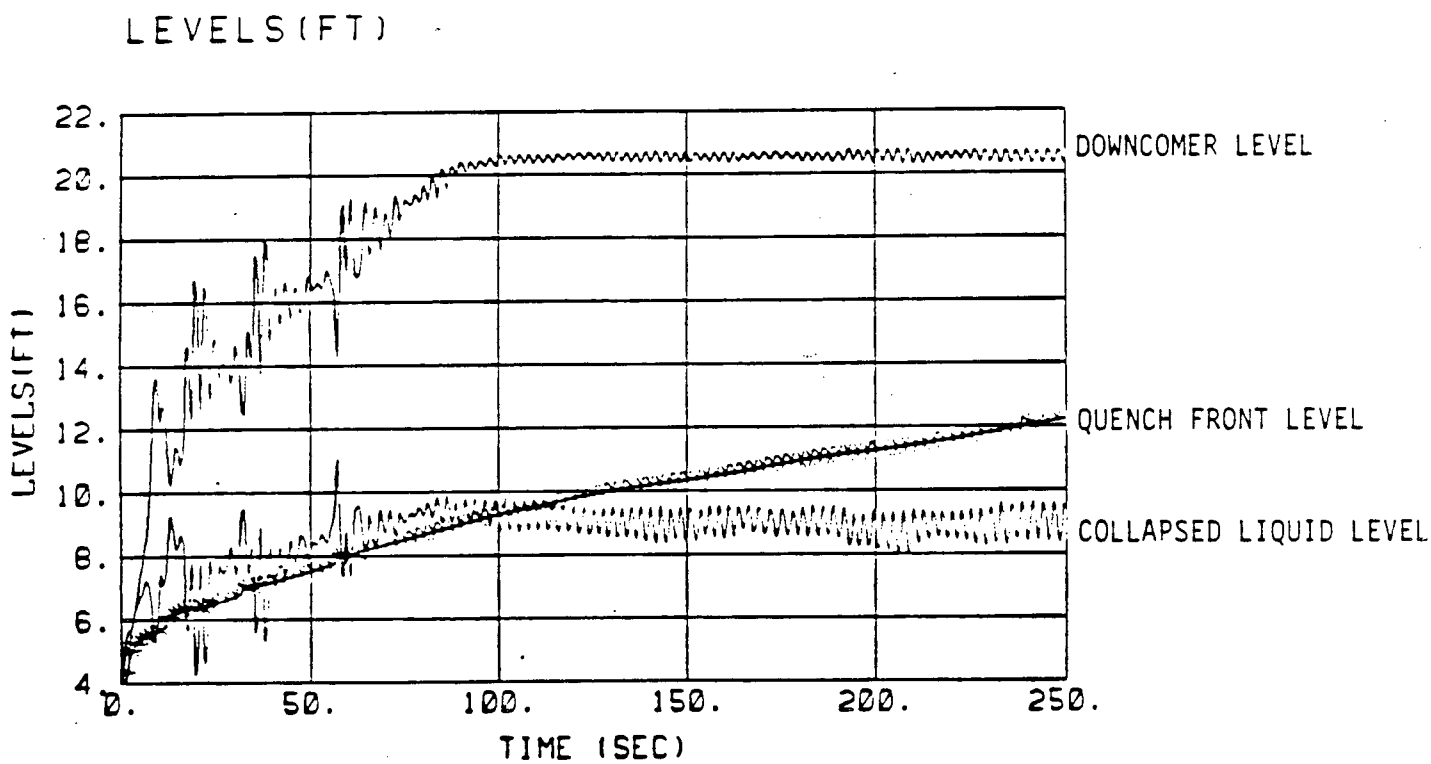
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-32
Core Flow-Top and Bottom-
DECLG (C_D=0.8)
(High T_{ave}) MINSI



CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-33a
Reflood Transient-Core and Downcomer Water Level DECLG (C_D=0.4) (High T_{ave}) MINSI



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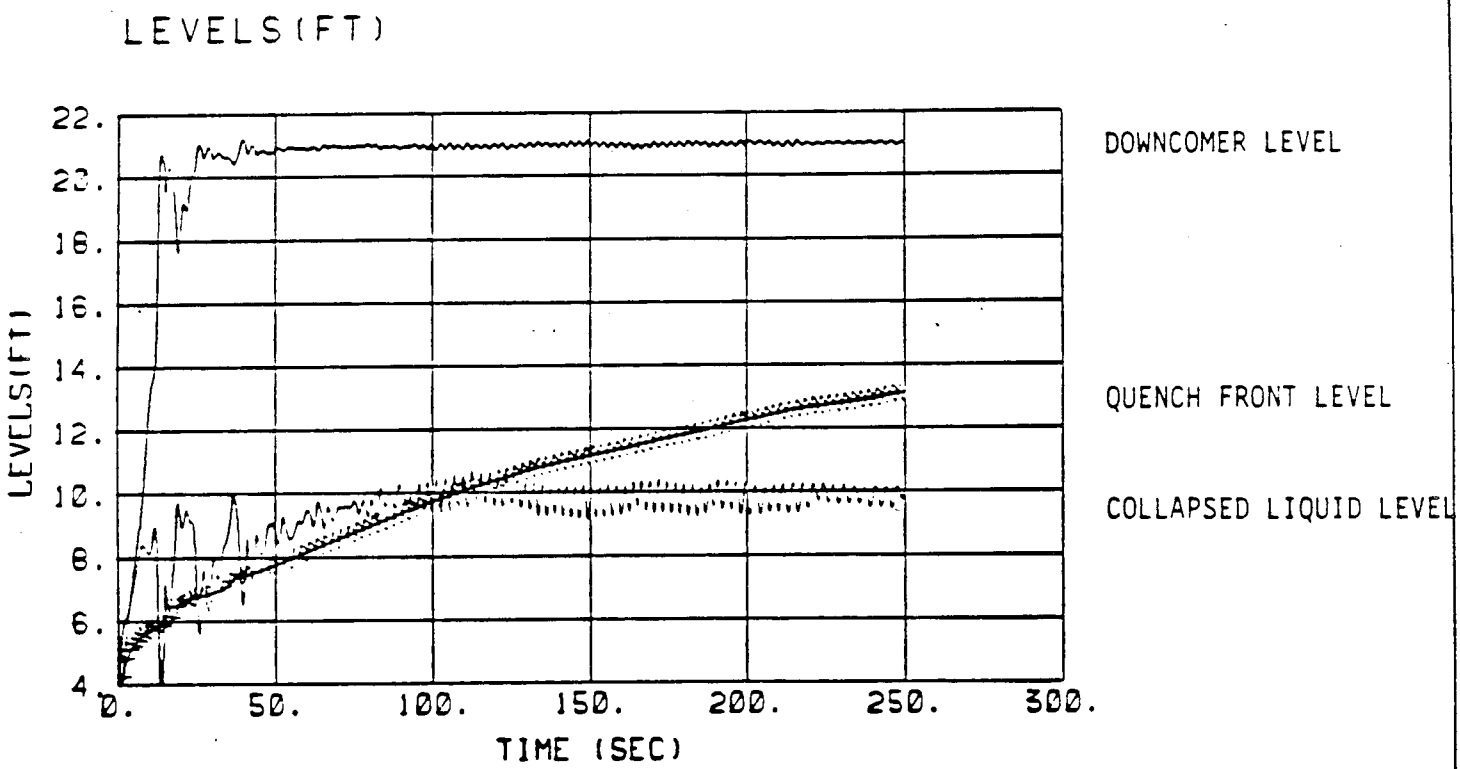
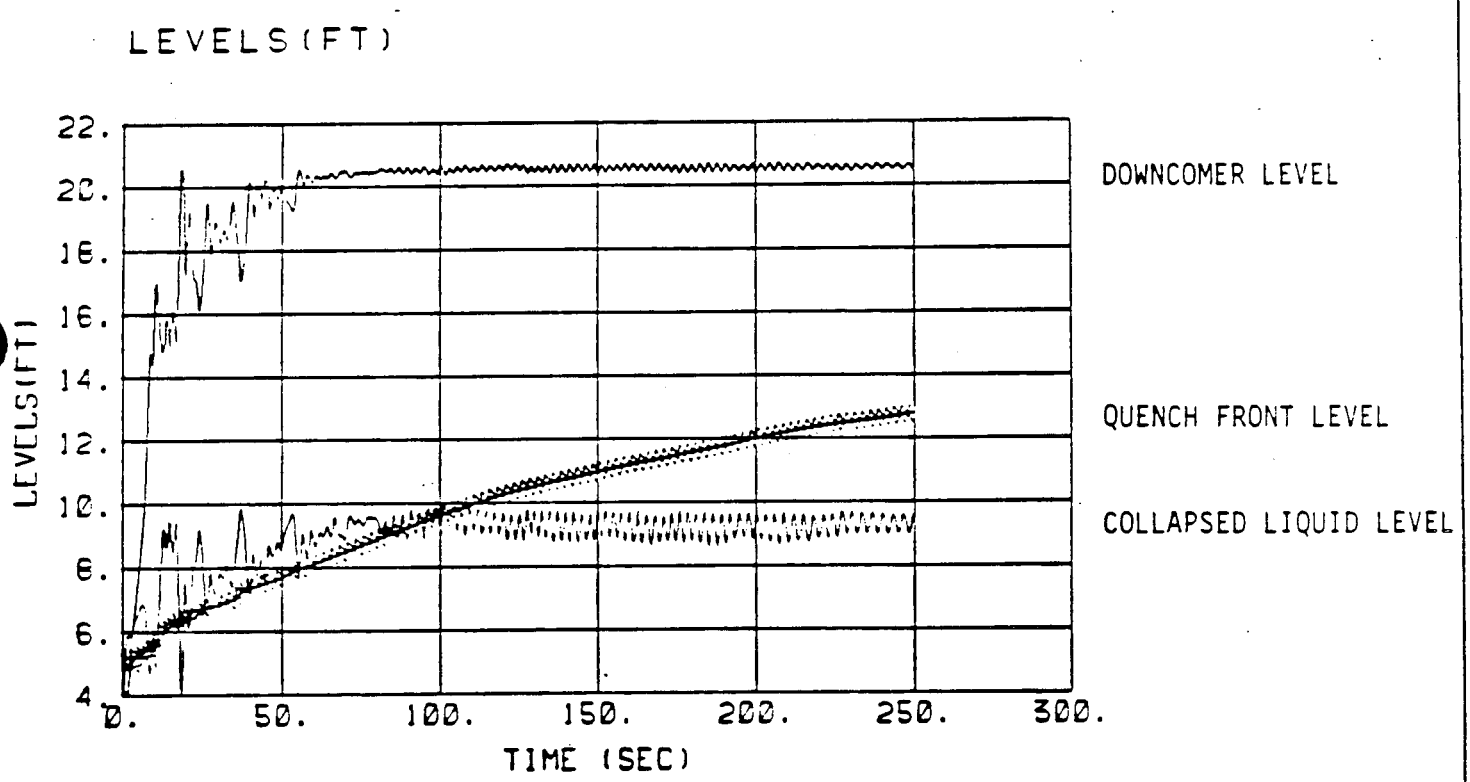
INDIAN POINT UNIT 2

Figure 14.3-33b
Reflood Transient-Core and Downcomer Water Level DECLG (C_D=0.4) (Low T_{ave}) MINSI



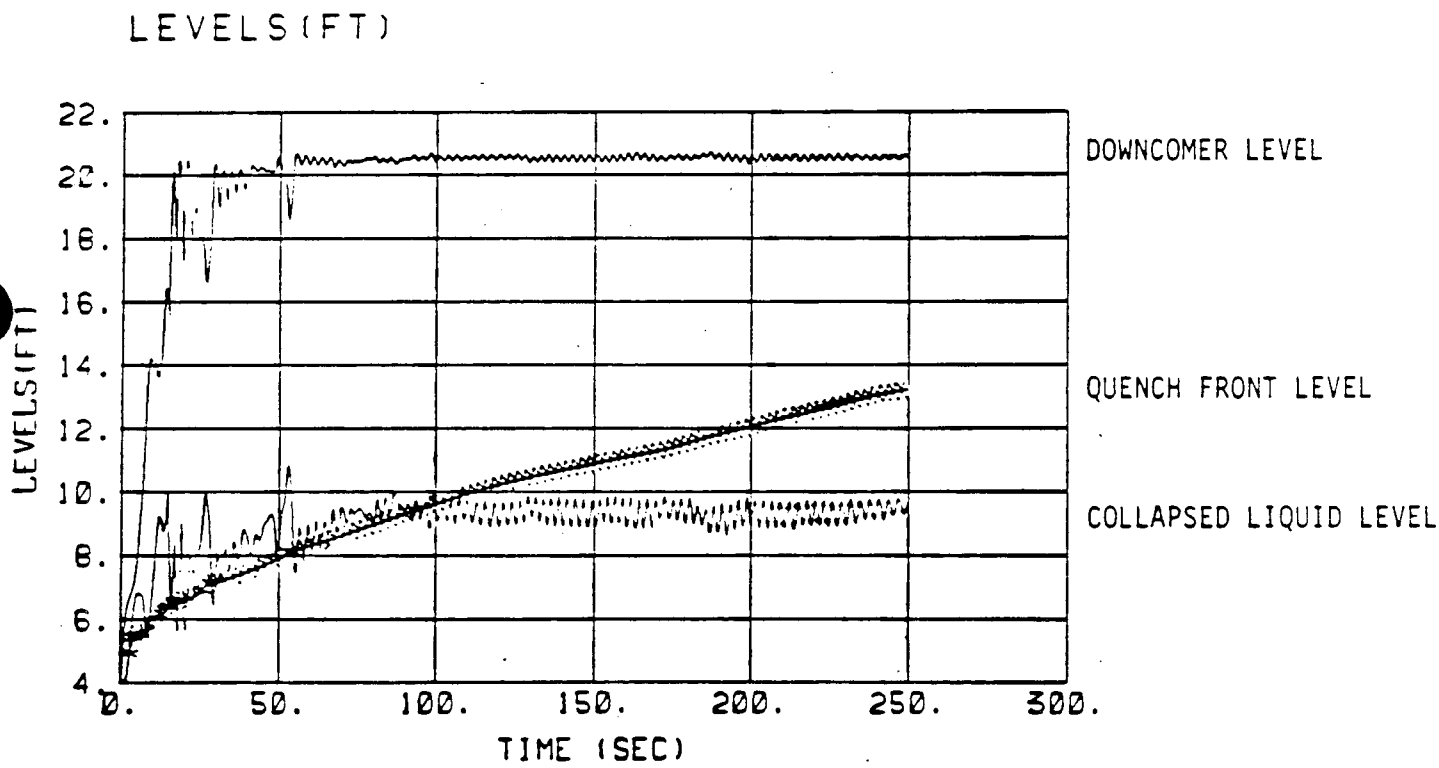
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-33c
Reflood Transient-Core and Downcomer Water Level DECLG (C_D=0.4) (High T_{ave}) MAXSI



CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-34
Reflood Transient-Core and Dowcomer Water Level DECLG (C_D=0.6) (High T_{ave}) MINSI



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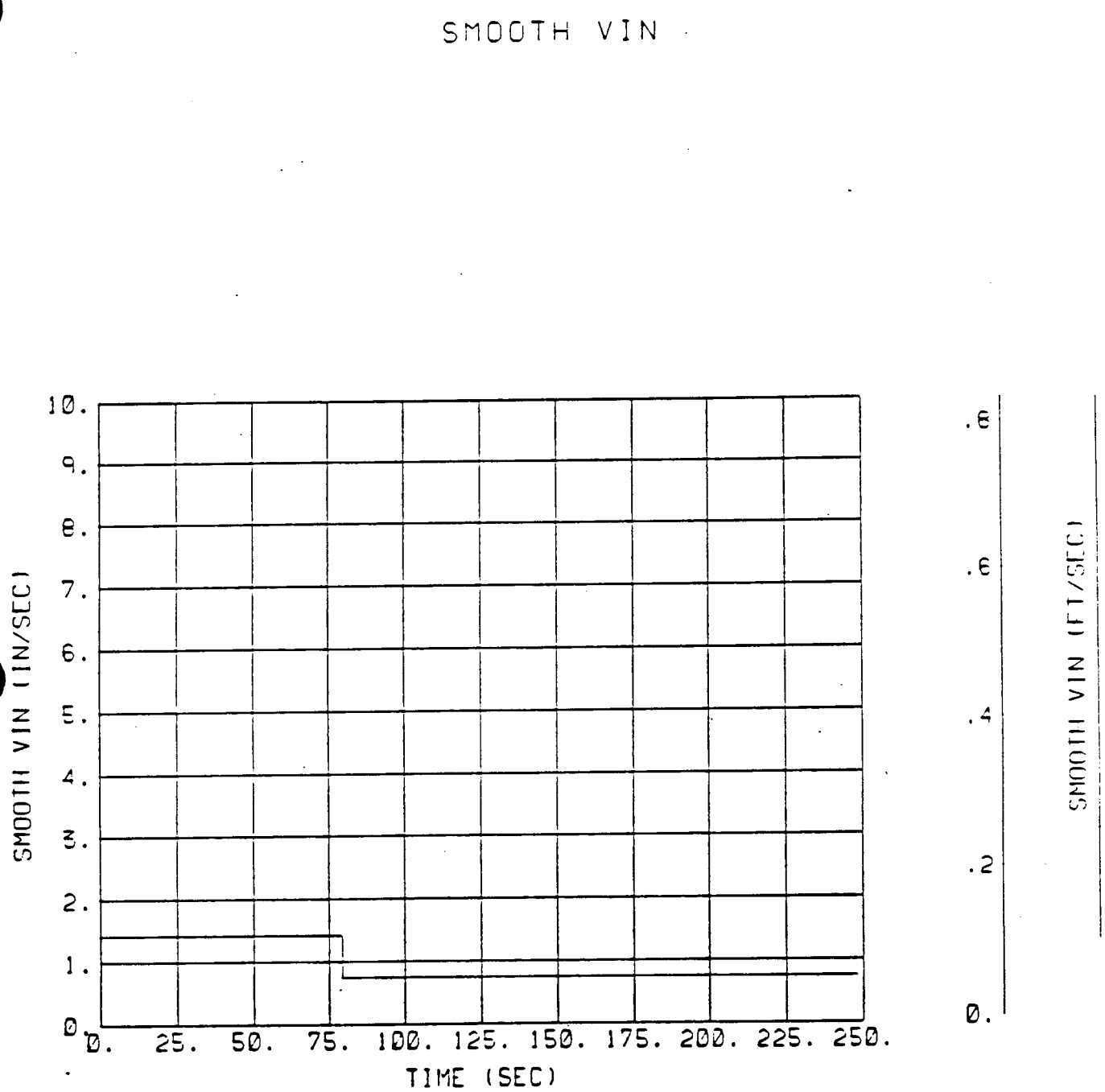
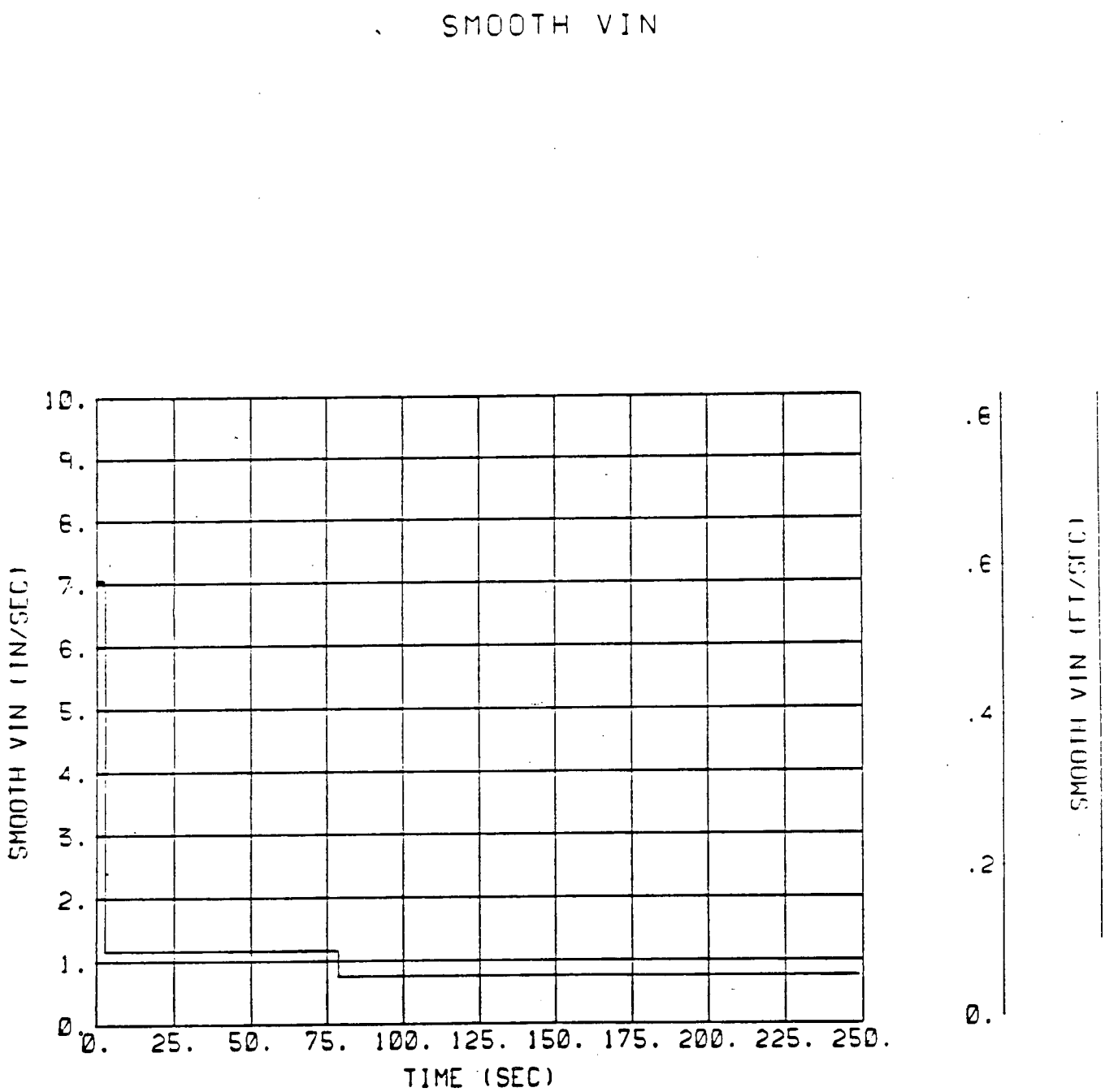
INDIAN POINT UNIT 2

Figure 14.3-35
Reflood Transient-Core and Dowcomer Water Level DECLG (C_D=0.8) (High T_{ave}) MINSI



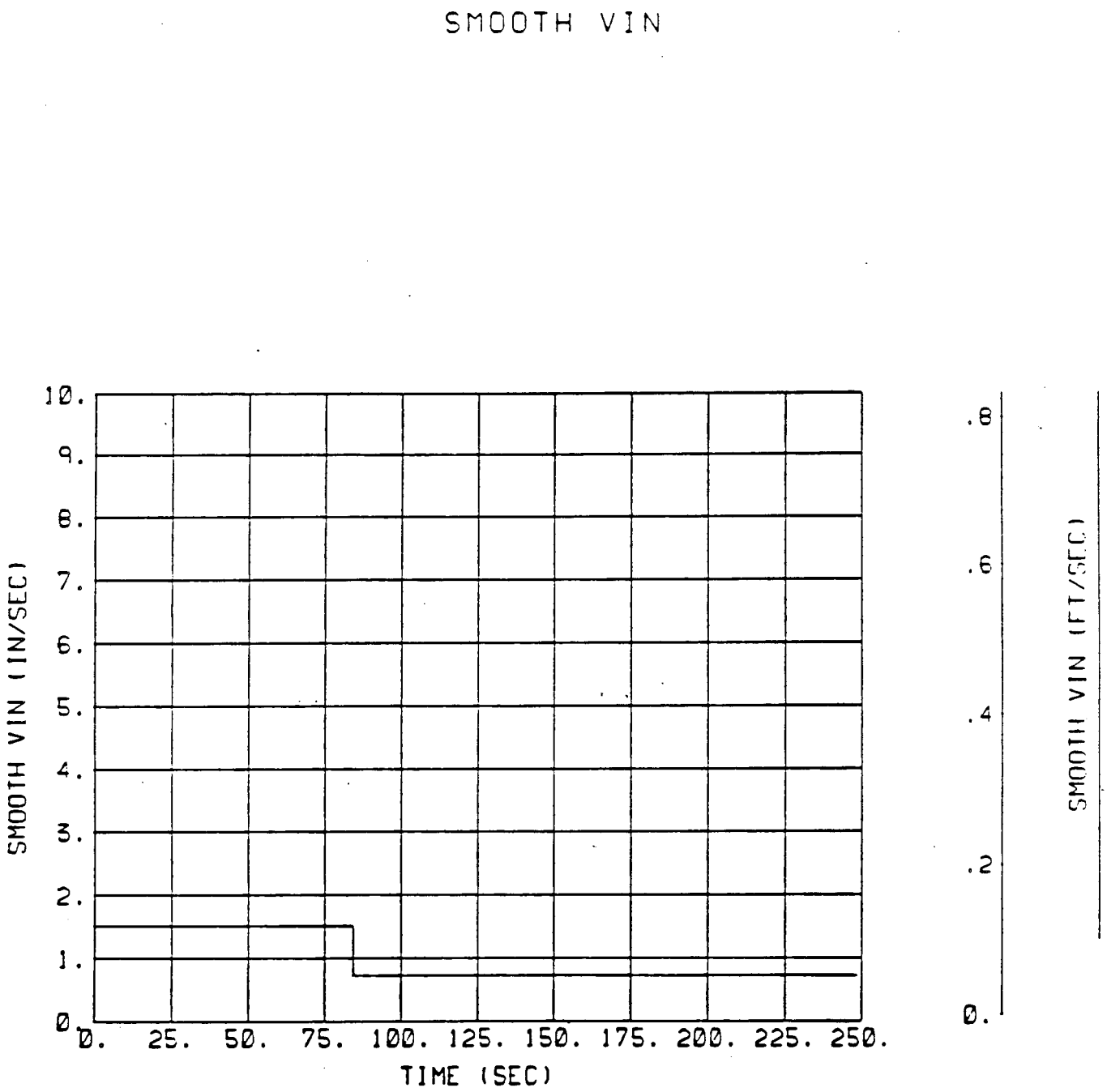
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-36a
Reflood Transient-Core Inlet Velocity
DECLG (C_D=0.4)
(High T_{ave}) MINSI



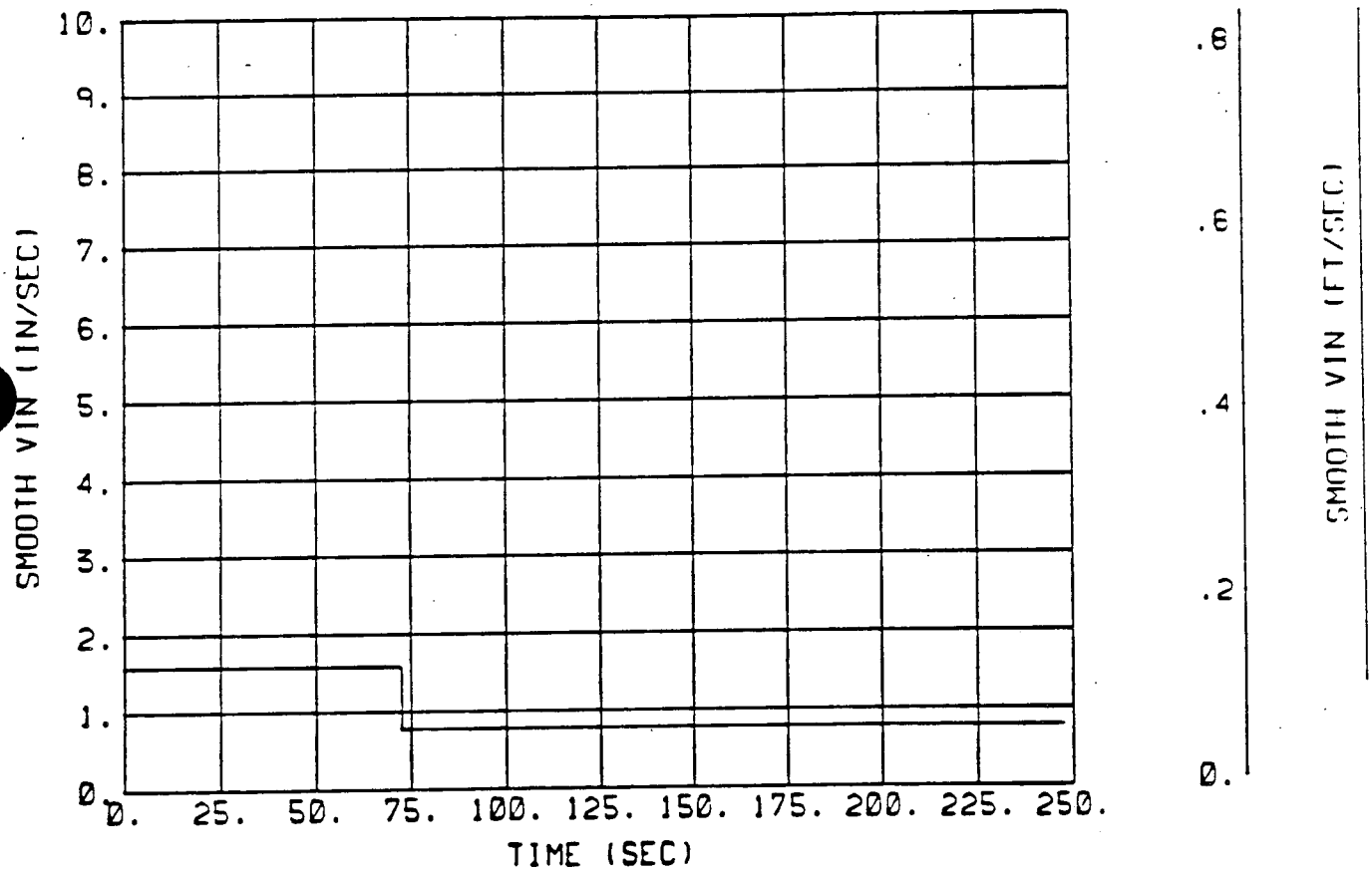
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-36b
Reflood Transient-Core Inlet Velocity
DECLG (C_D=0.4)
(Low T_{ave}) MINSI



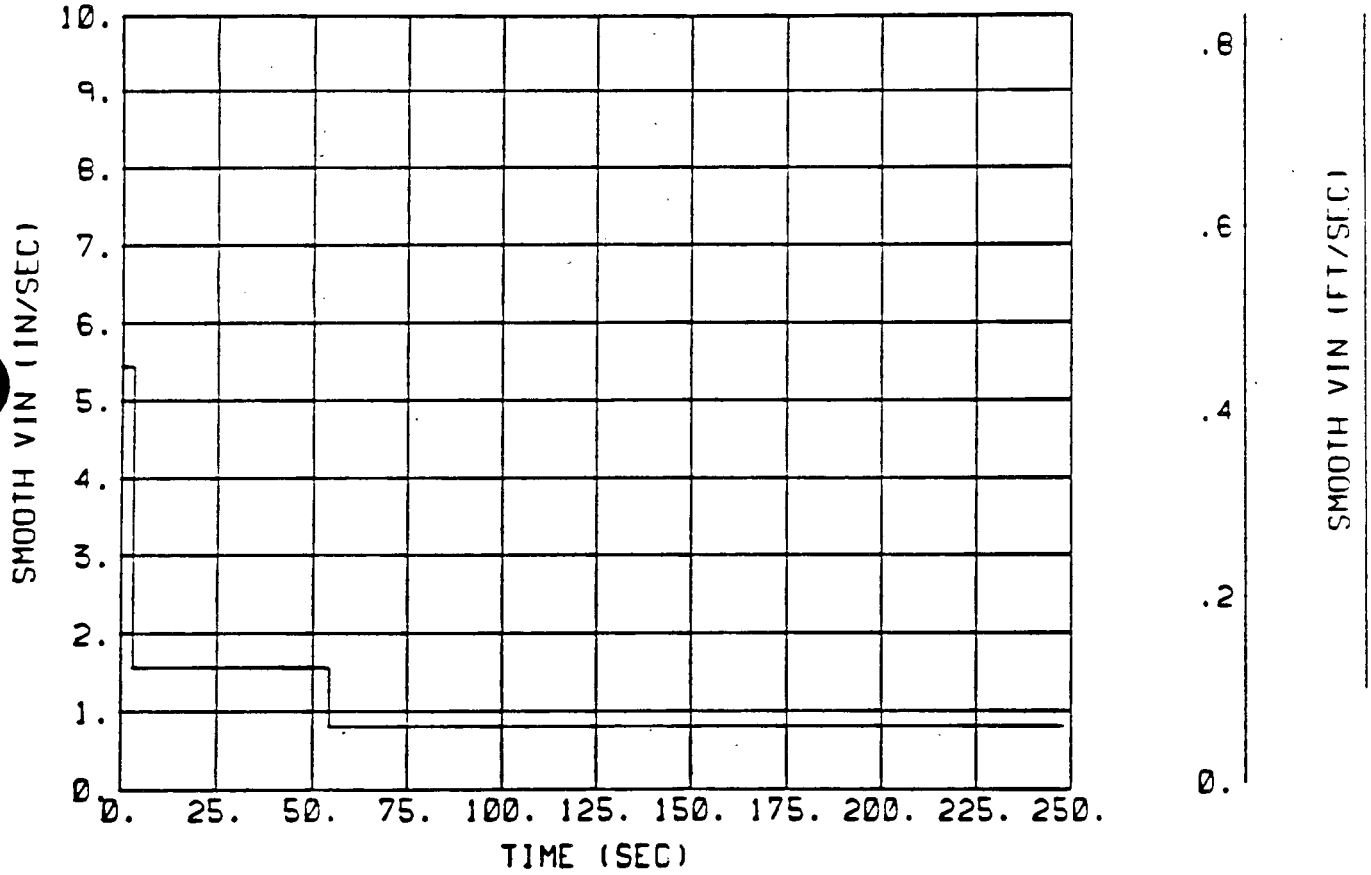
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-36c
Reflood Transient-Core Inlet Velocity
DECLG (C_D=0.4)
(High T_{ave}) MAXSI



CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-37
Core Inlet Velocity-DECLG (C_D=0.6)
(High T_{ave}) MINSI



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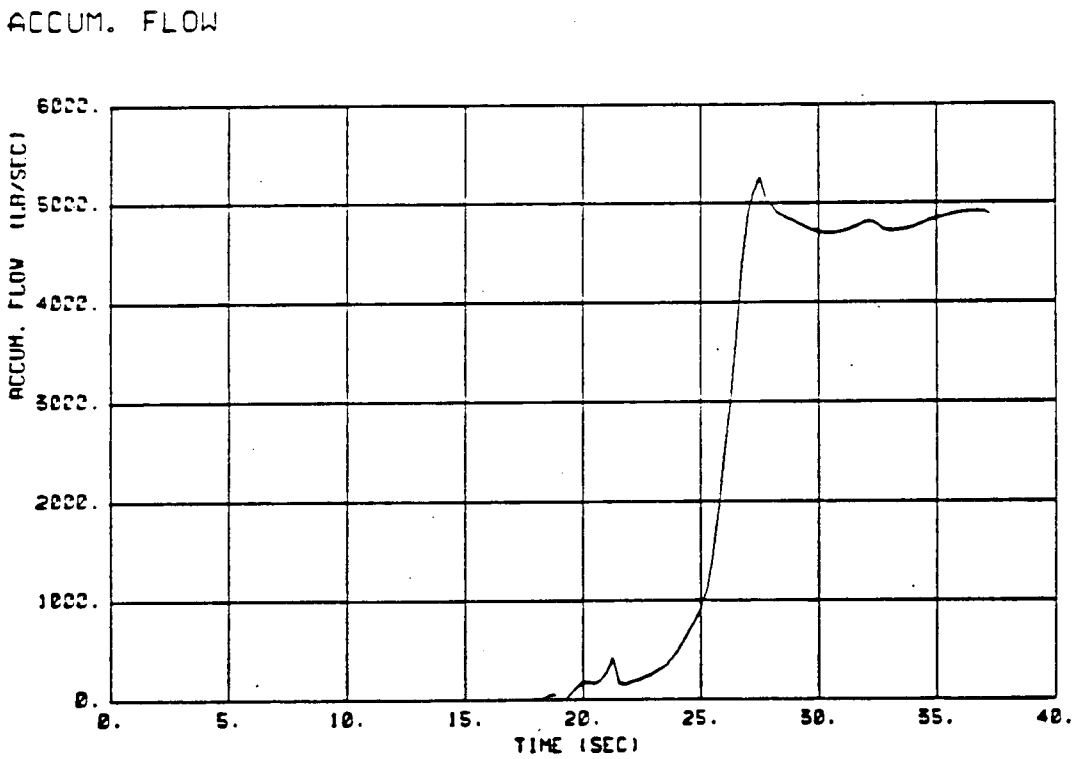
INDIAN POINT UNIT 2

Figure 14.3-38
Core Inlet Velocity-DECLG (C_D=0.8)
(High T_{ave}) MINSI



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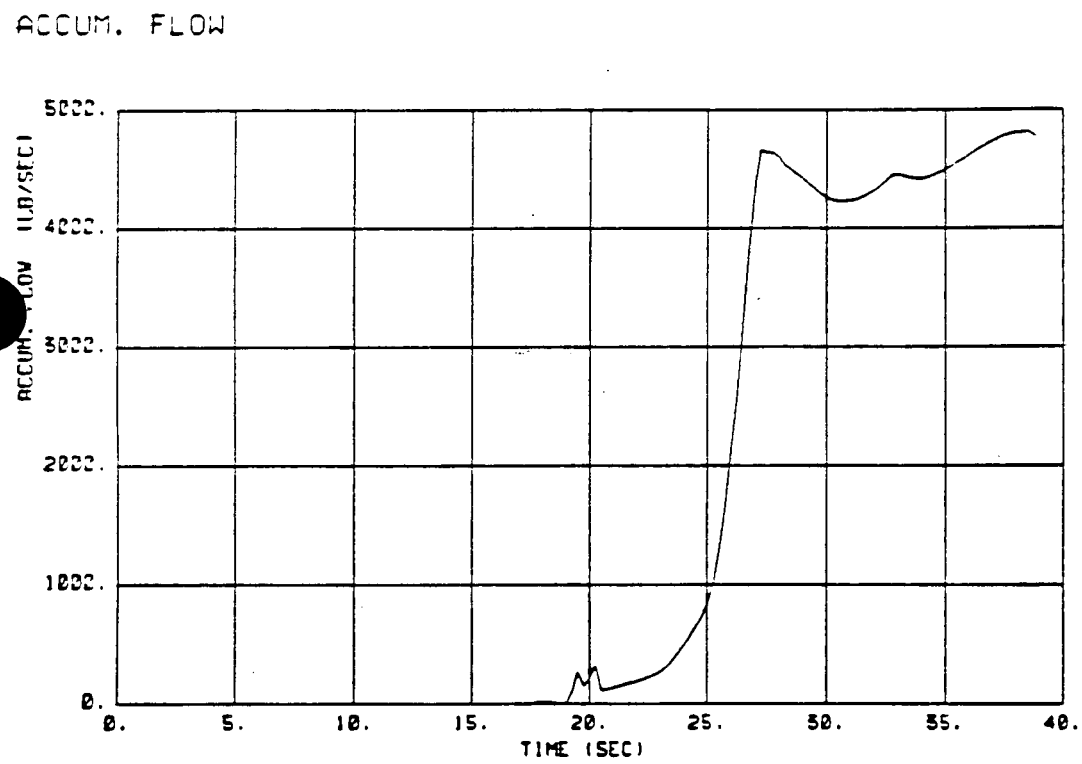
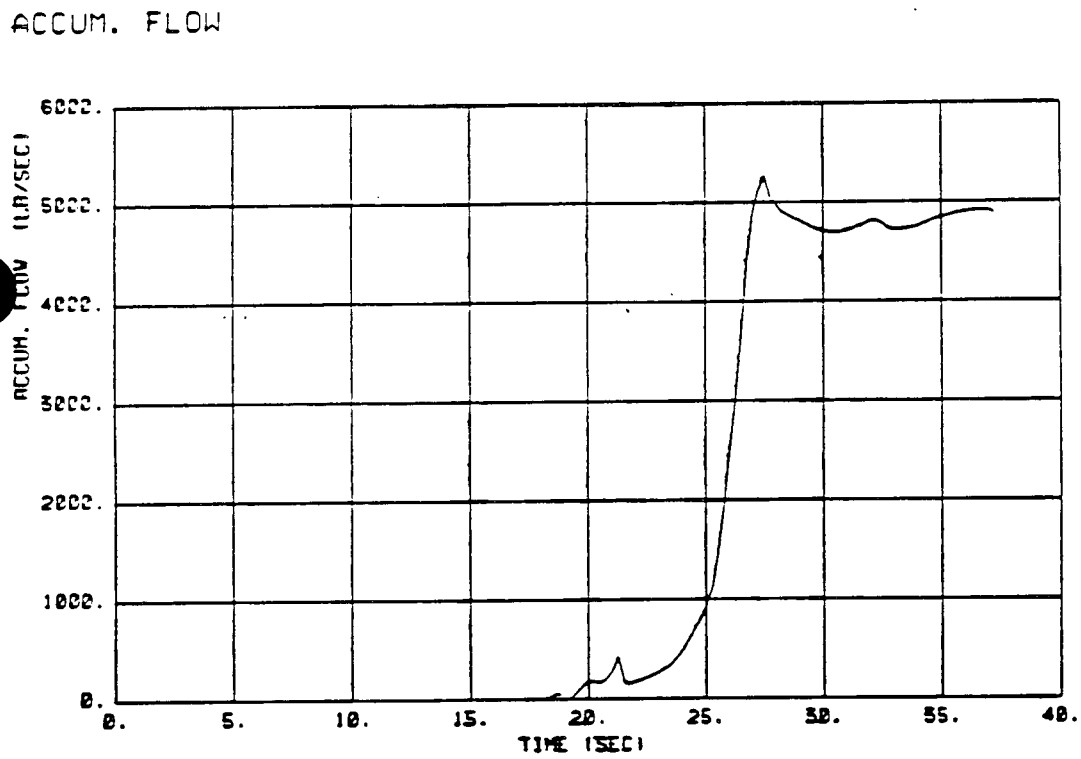
INDIAN POINT UNIT 2

Figure 14.3-39a
Accumulator Flow (Blowdown)-
DECLG (C_D=0.4)
(High T_{ave}) MINSI



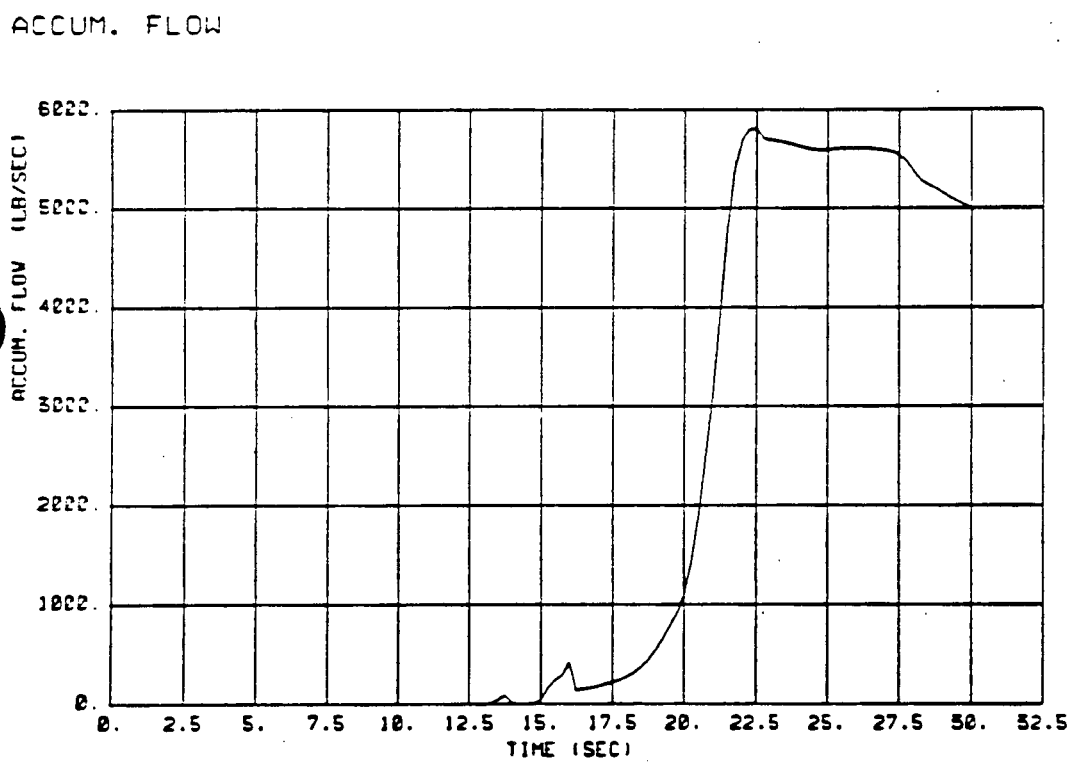
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-39b
Accumulator Flow (Blowdown)-
DECLG (C_D=0.4)
(Low T_{ave}) MINSI



CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-39c
Accumulator Flow (Blowdown)-
DECLG (C_D=0.4)
(High T_{ave}) MAXSI



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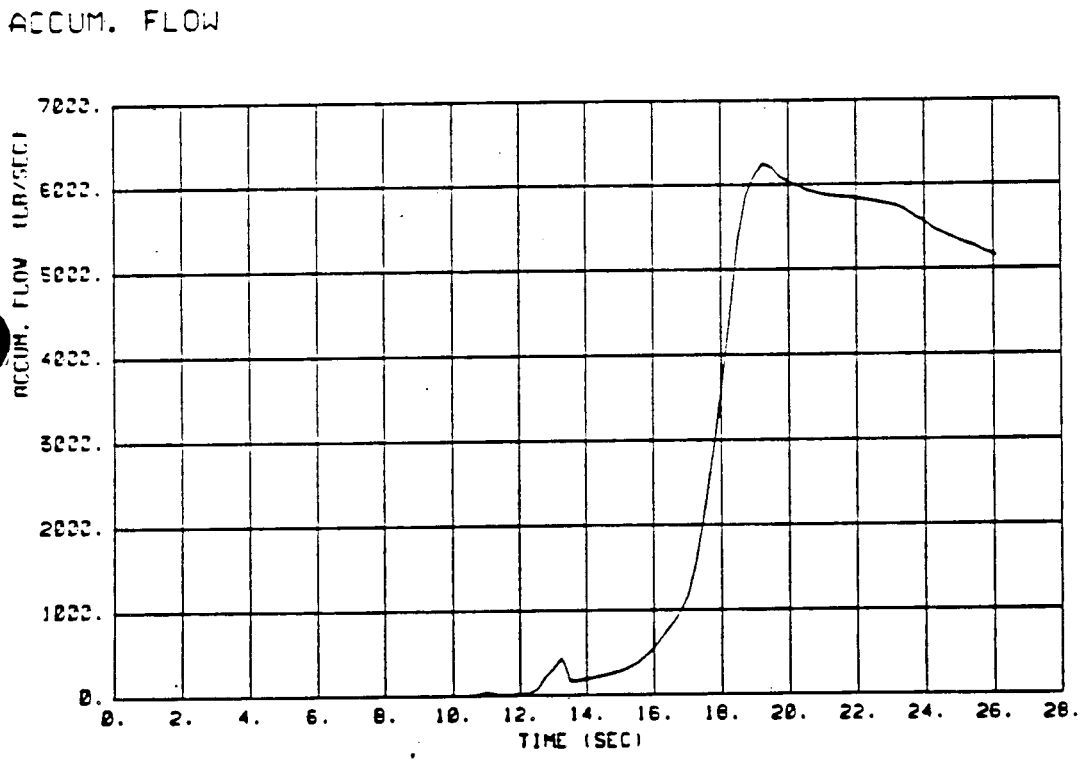
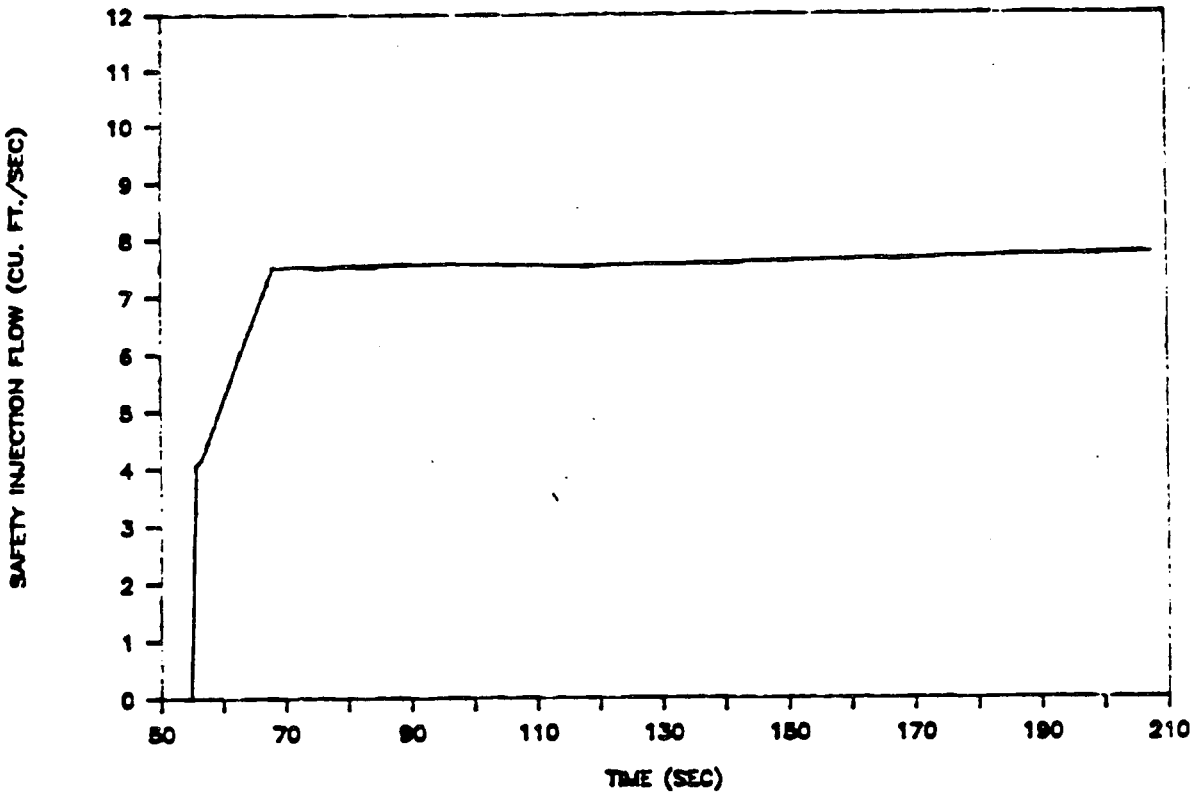
INDIAN POINT UNIT 2

Figure 14.3-40
Accumulator Flow (Blowdown)-
DECLG (C_D=0.6)
(High T_{ave}) MINSI



CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-41
Accumulator Flow (Blowdown)-
DECLG (C_D=0.8)
(High T_{ave}) MINSI



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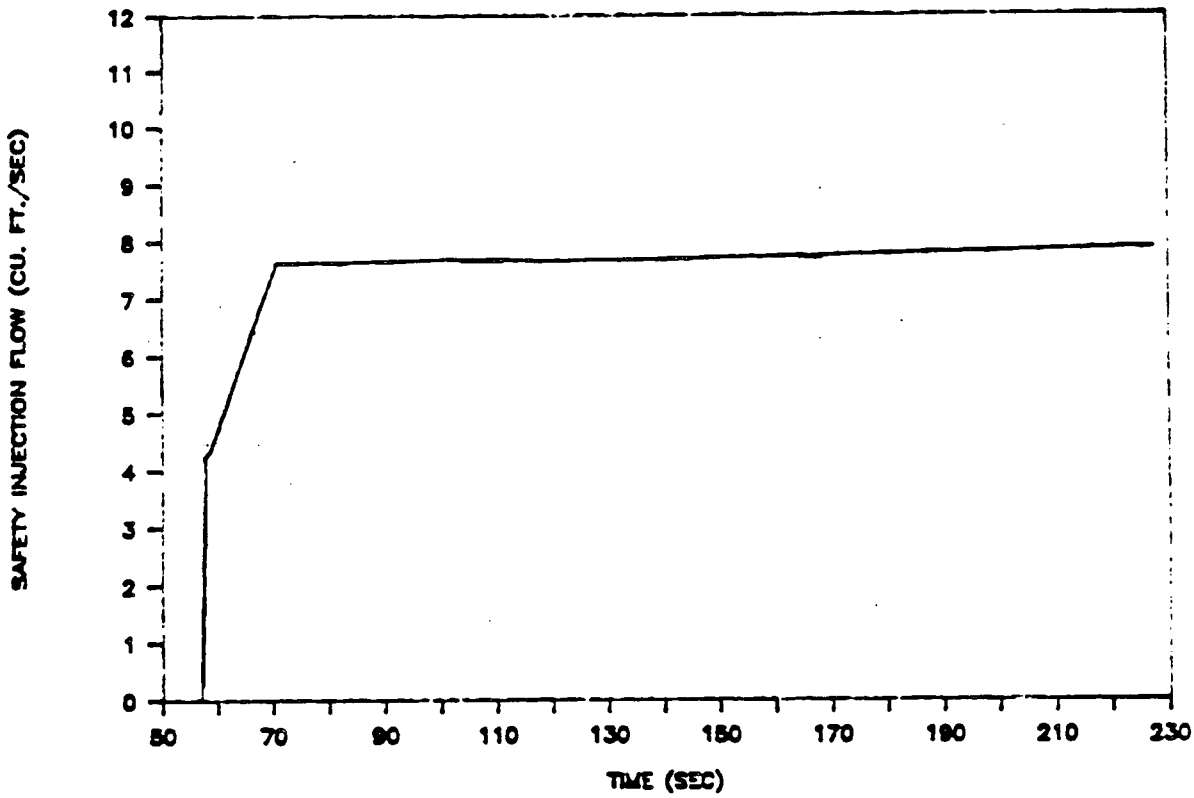
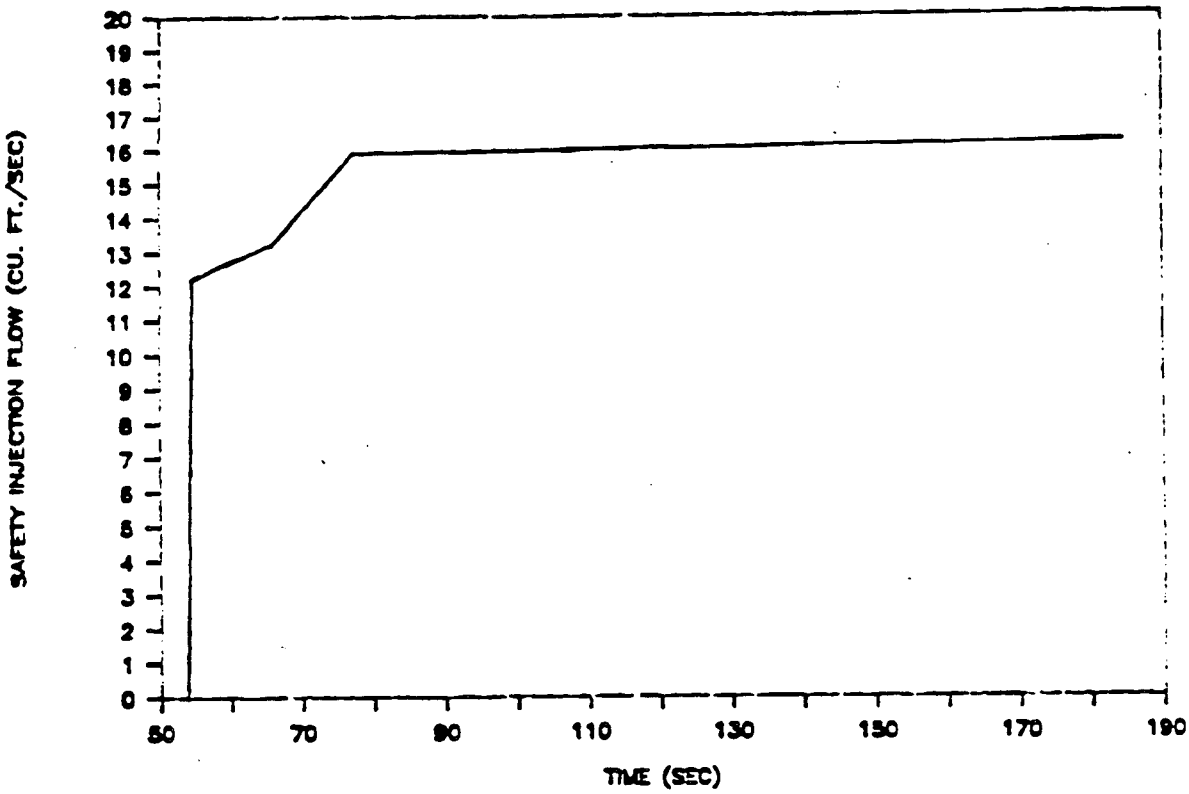
INDIAN POINT UNIT 2

Figure 14.3-42a
Pumped ECCS Flow (Reflood)-
DECLG (C_D=0.4)
(High T_{ave}) MINSI



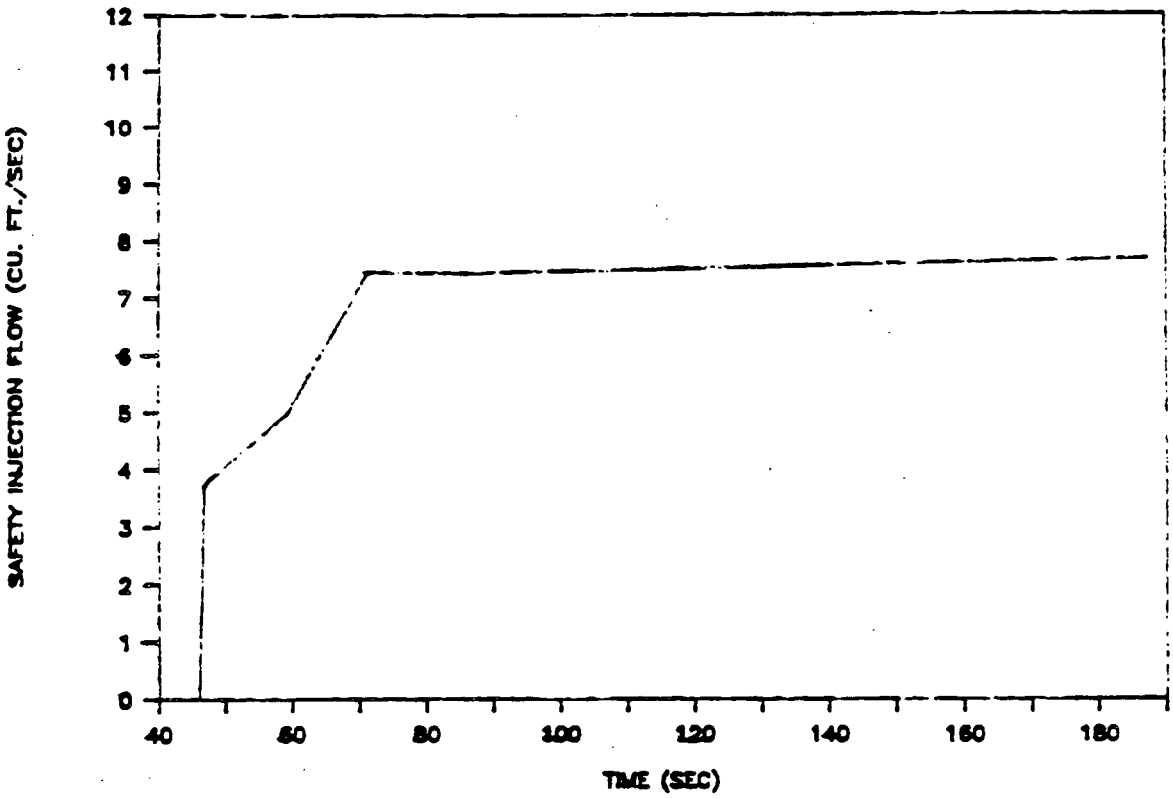
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-42b
Pumped ECCS Flow (Reflood)-
DECLG (C_D=0.4)
(Low T_{ave}) MINSI



CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-42c
Pumped ECCS Flow (Reflood)-
DECLG (C_D=0.4)
(High T_{ave}) MAXSI



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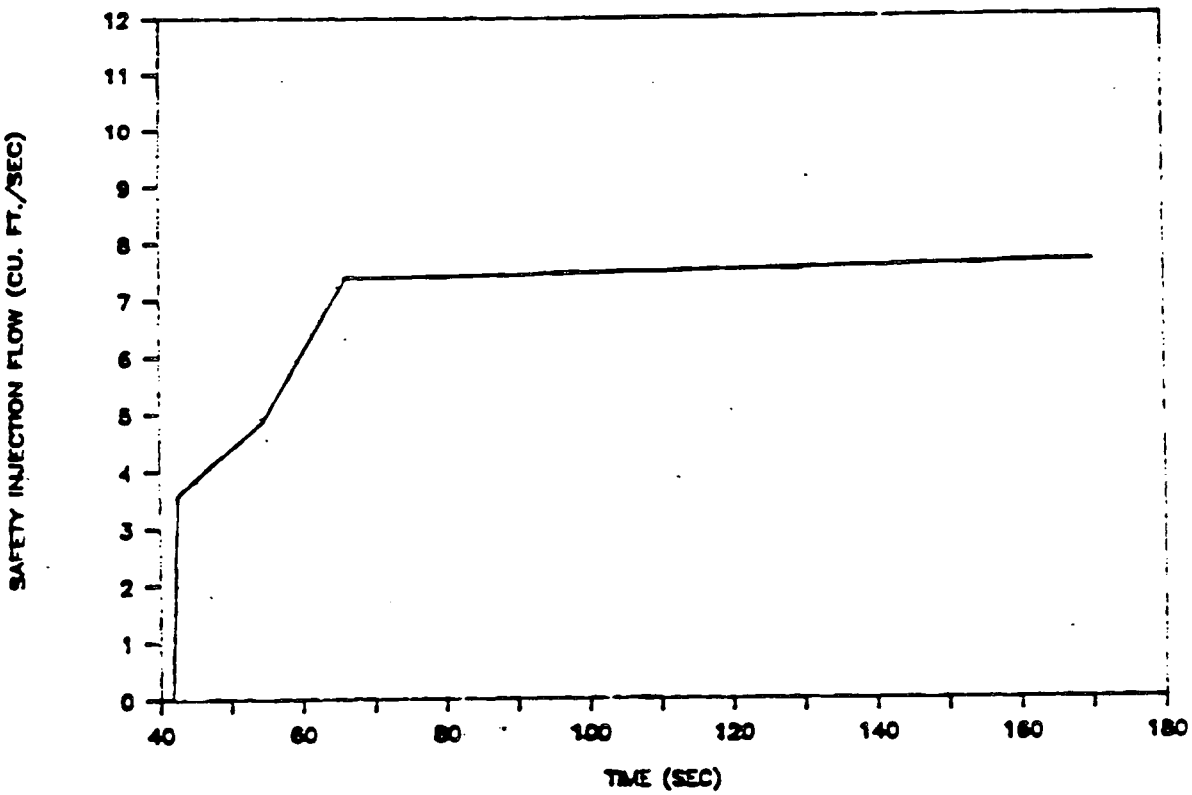
INDIAN POINT UNIT 2

Figure 14.3-43
Pumped ECCS Flow (Reflood)-
DECLG (C_D=0.6)
(High T_{ave}) MINSI



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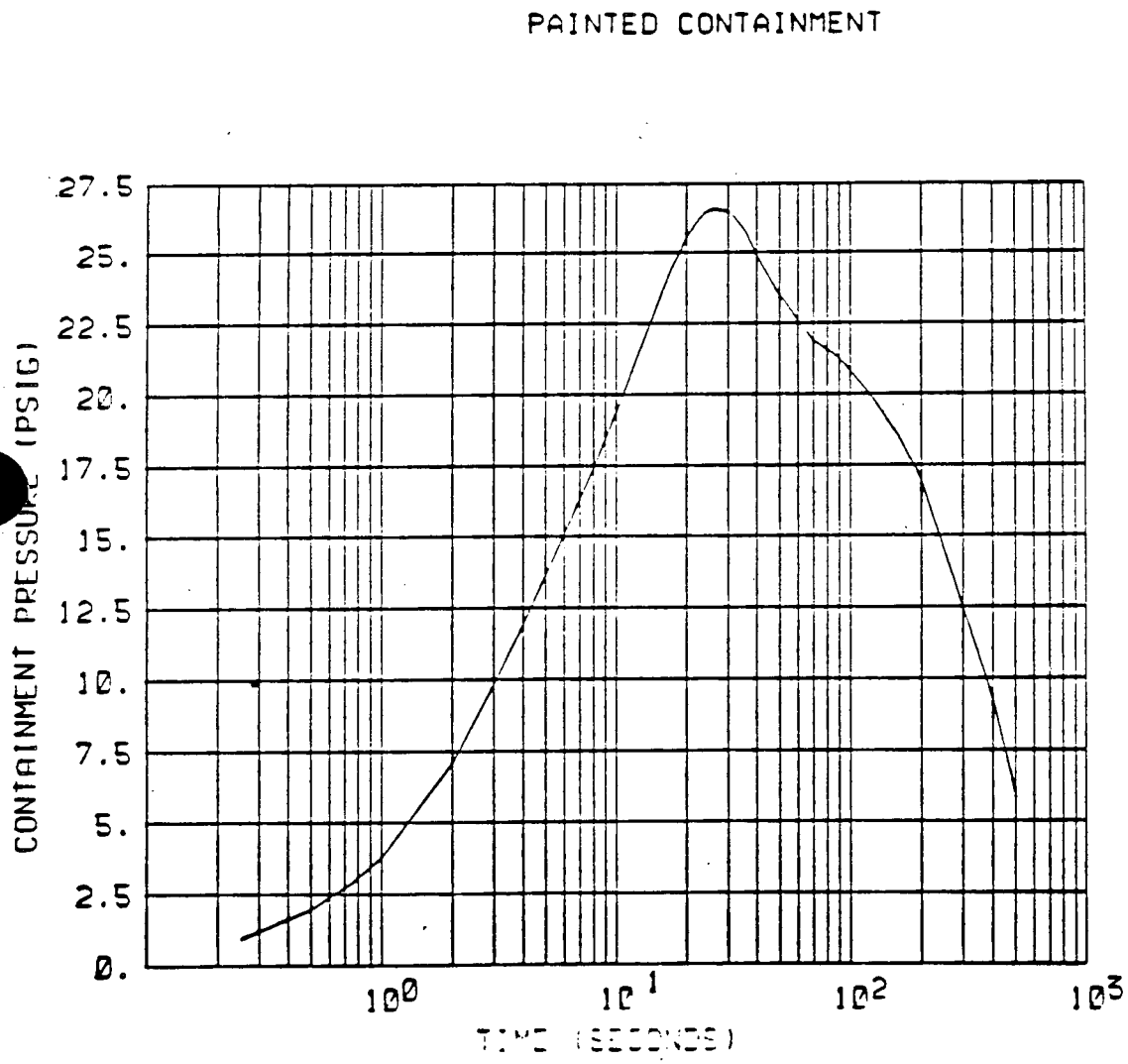
INDIAN POINT UNIT 2

Figure 14.3-44
Pumped ECCS Flow (Reflood)-
DECLG (C_D=0.8)
(High T_{ave}) MINSI



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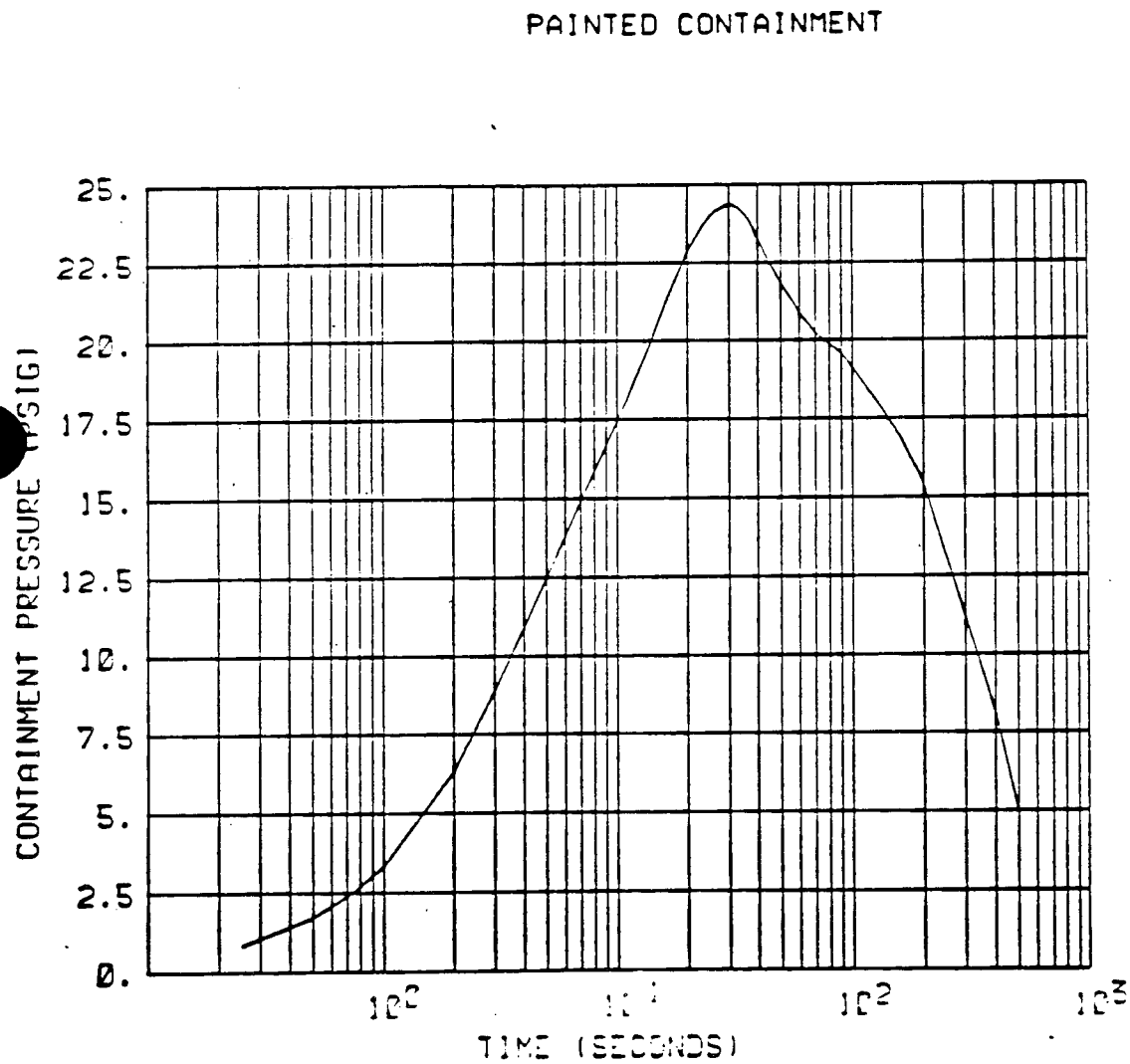
INDIAN POINT UNIT 2

Figure 14.3-45a
Containment Pressure-DECLG (C_D=0.4)
(High T_{ave}) MINSI



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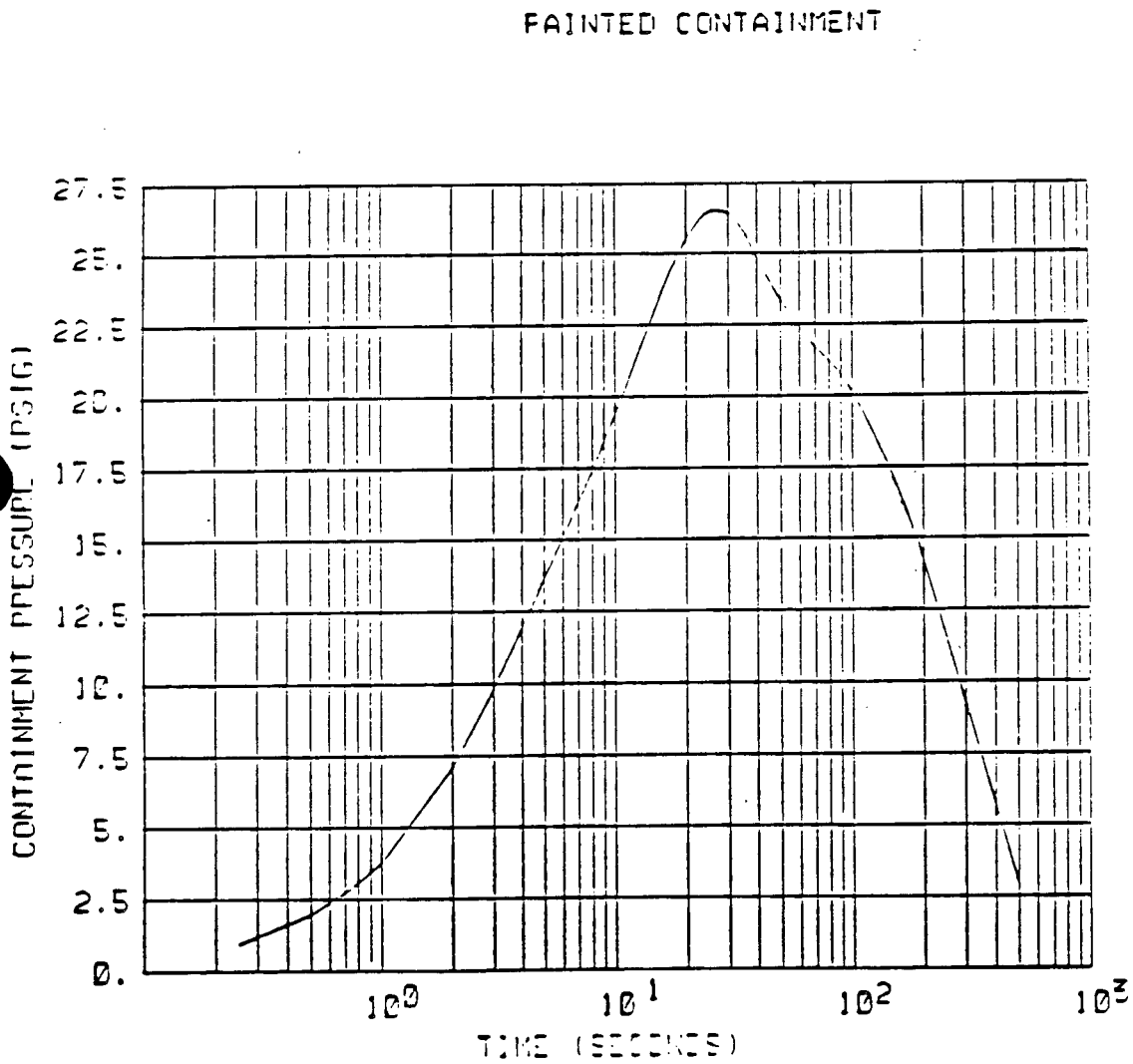
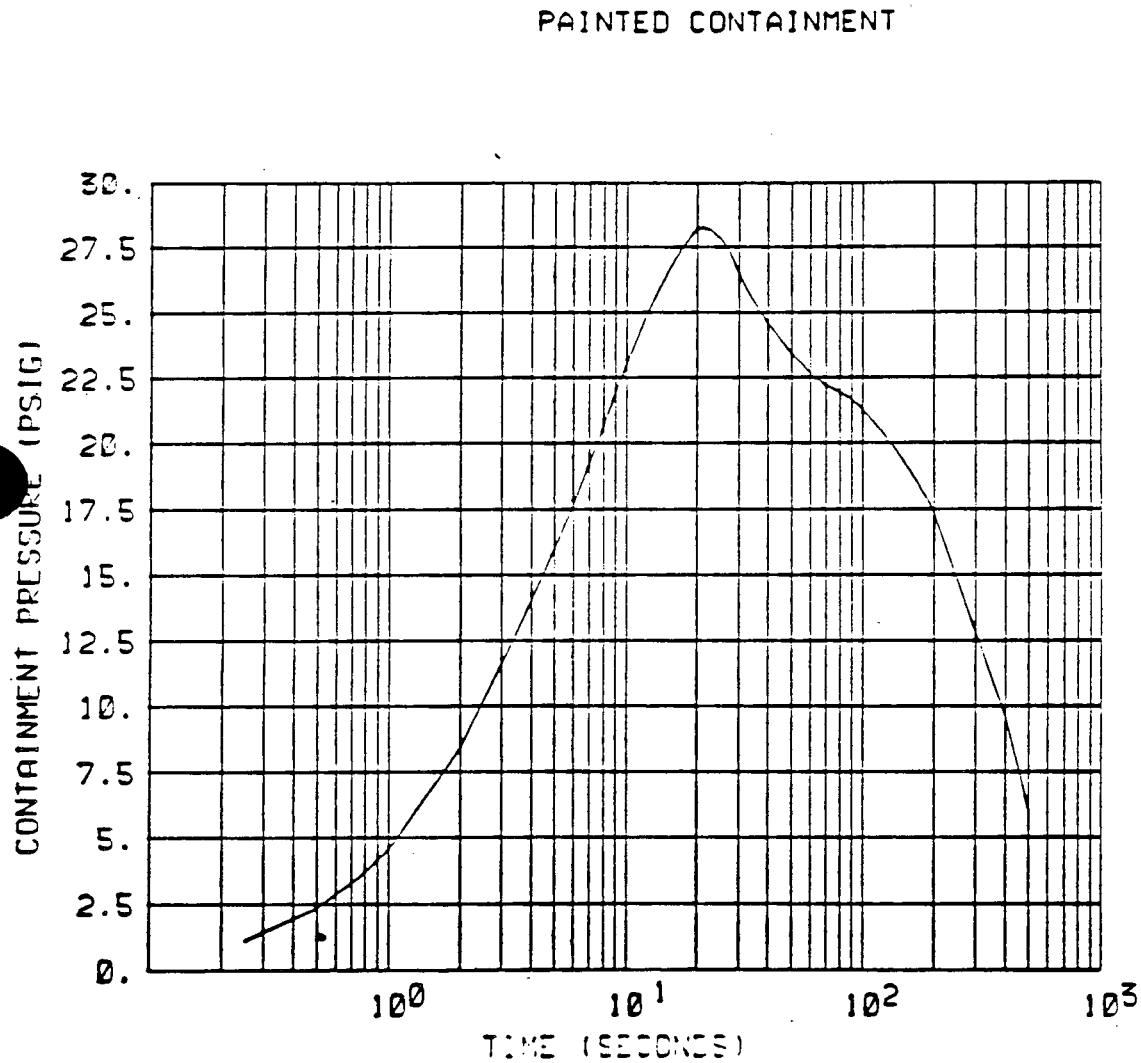
INDIAN POINT UNIT 2

Figure 14.3-45b
Containment Pressure-DECLG (C_D=0.4)
(Low T_{ave}) MINSI



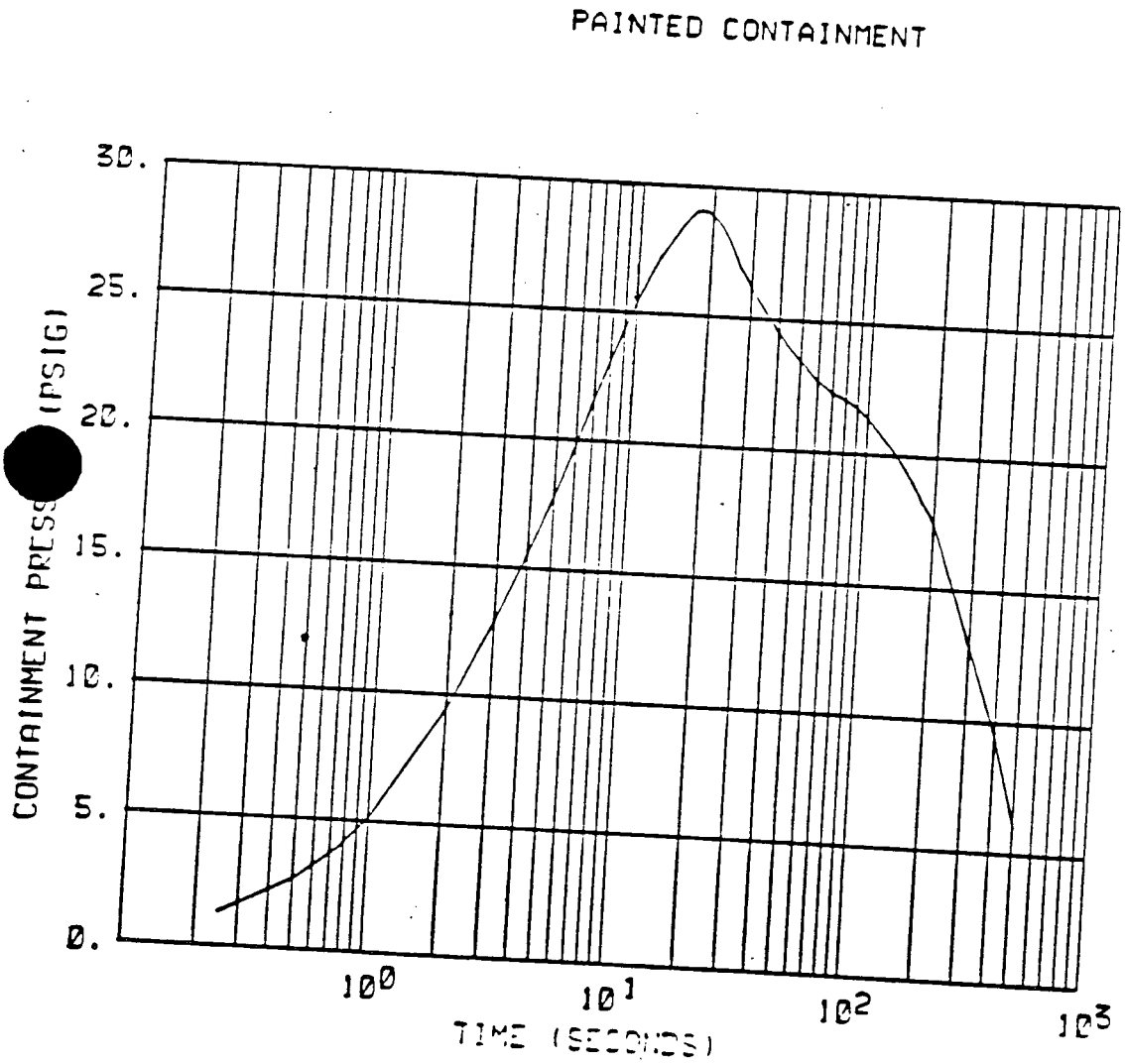
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-45c
Containment Pressure-DECLG (C_D=0.4)
(High T_{ave}) MAXSI



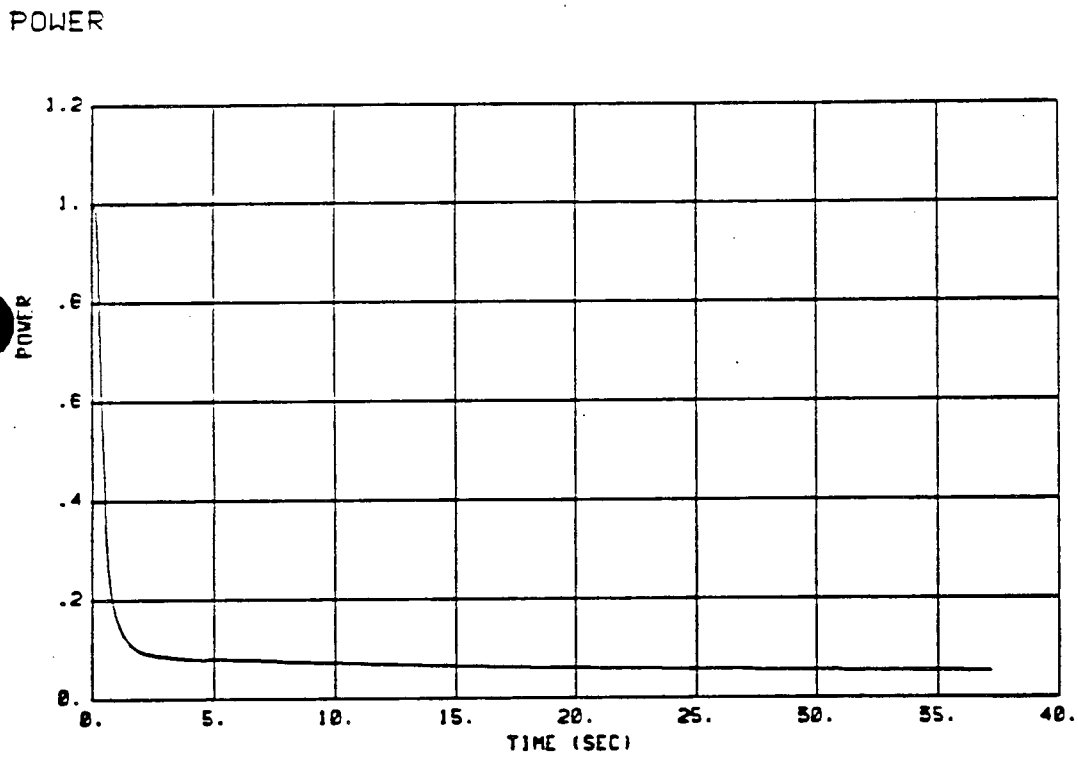
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-46
Containment Pressure-DECLG (C_D=0.6)
(High T_{ave}) MINSI



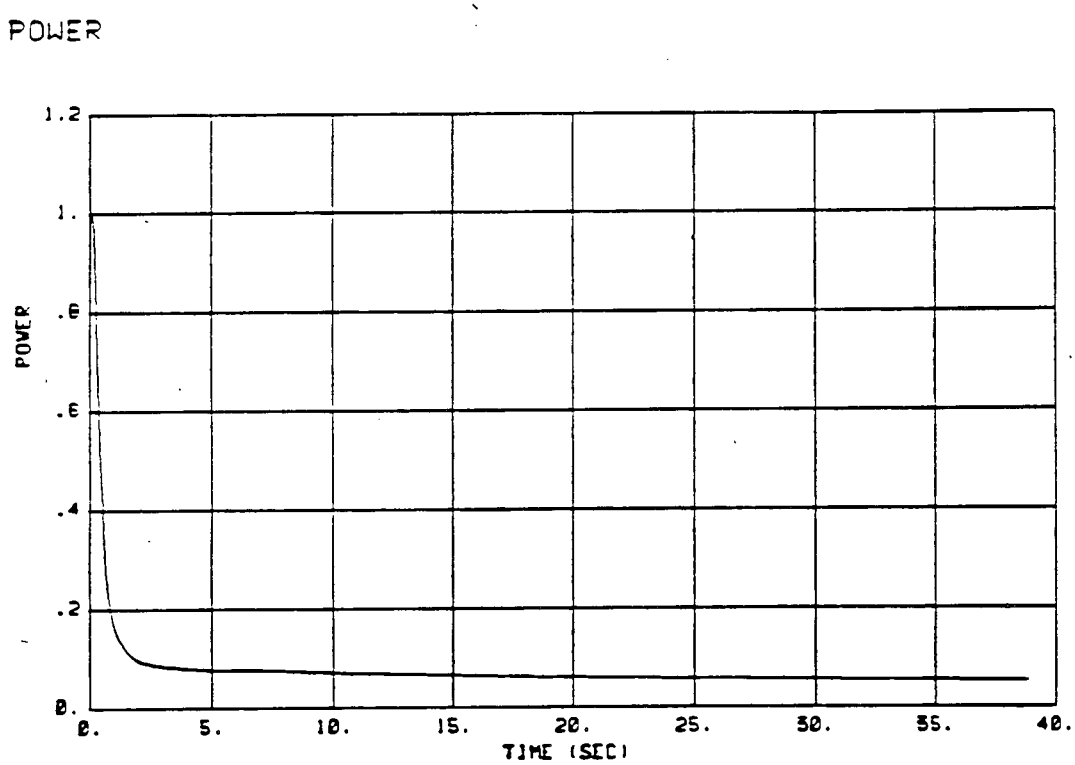
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-47
Containment Pressure-DECLG (C_D=0.8)
(High T_{ave}) MINSI



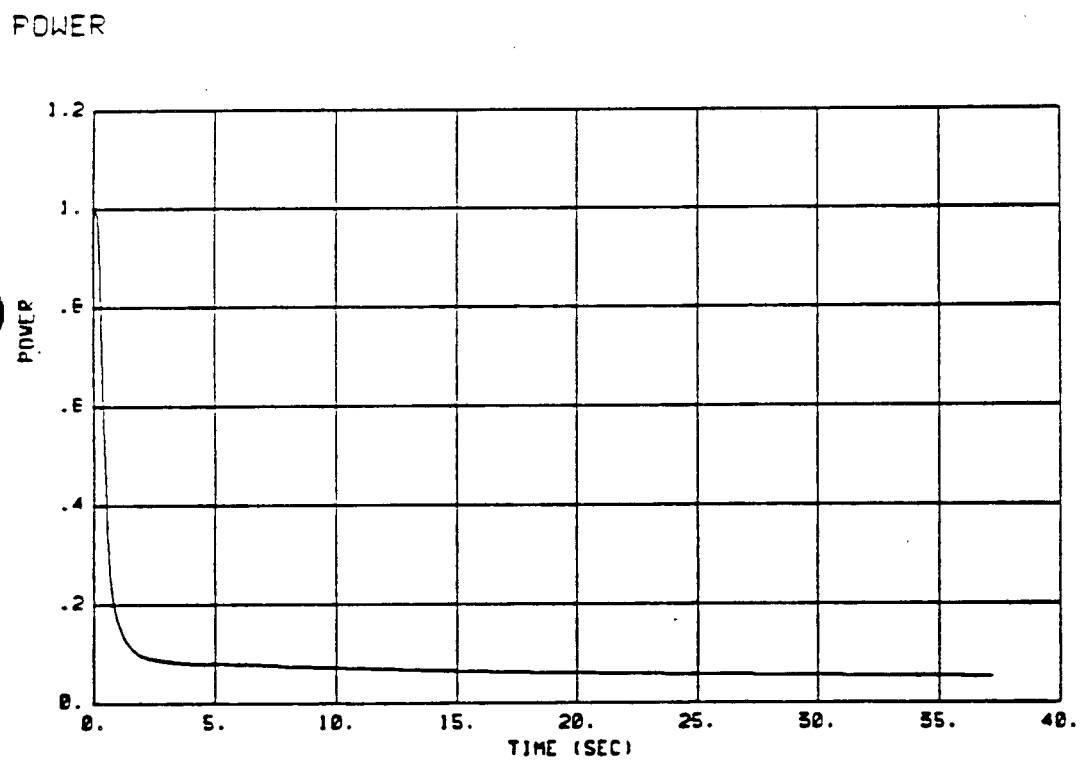
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-48a
Core Power Transient-
DECLG (C_D=0.4)
(High T_{ave}) MINSI



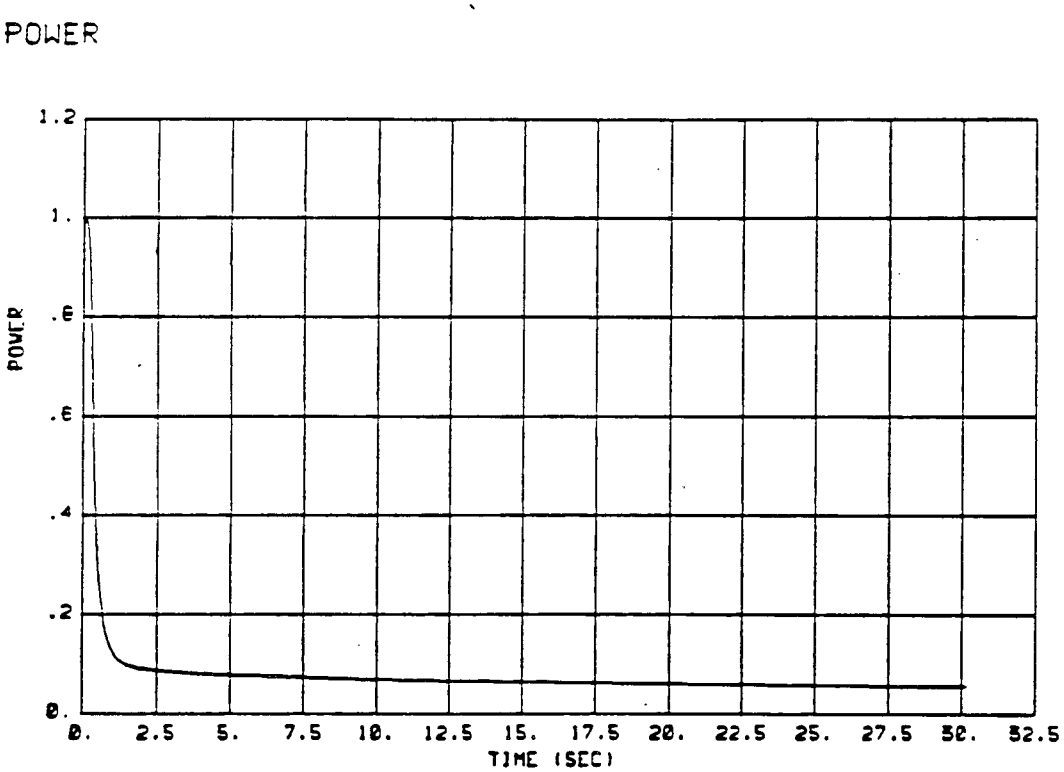
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-48b
Core Power Transient-
DECLG (C_D=0.4)
(Low T_{ave}) MINSI



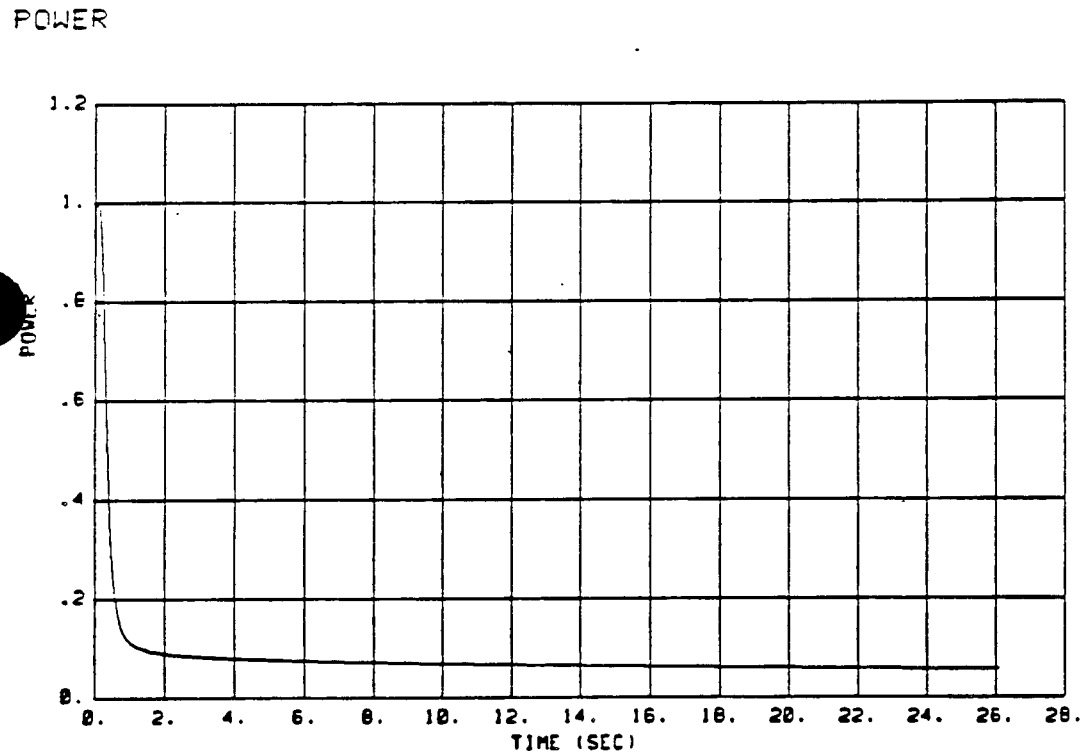
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-48c
Core Power Transient-
DECLG (C_D=0.4)
(High T_{ave}) MAXSI



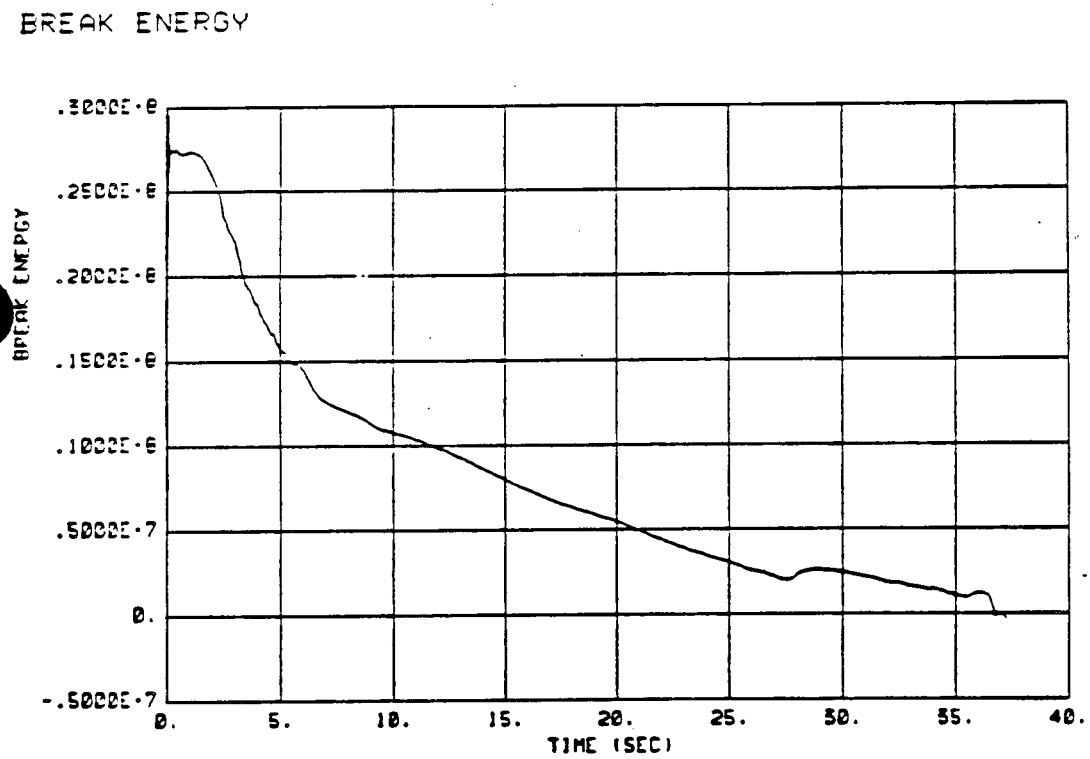
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-49
Core Power Transient-
DECLG (C_D=0.6)
(High T_{ave}) MINSI



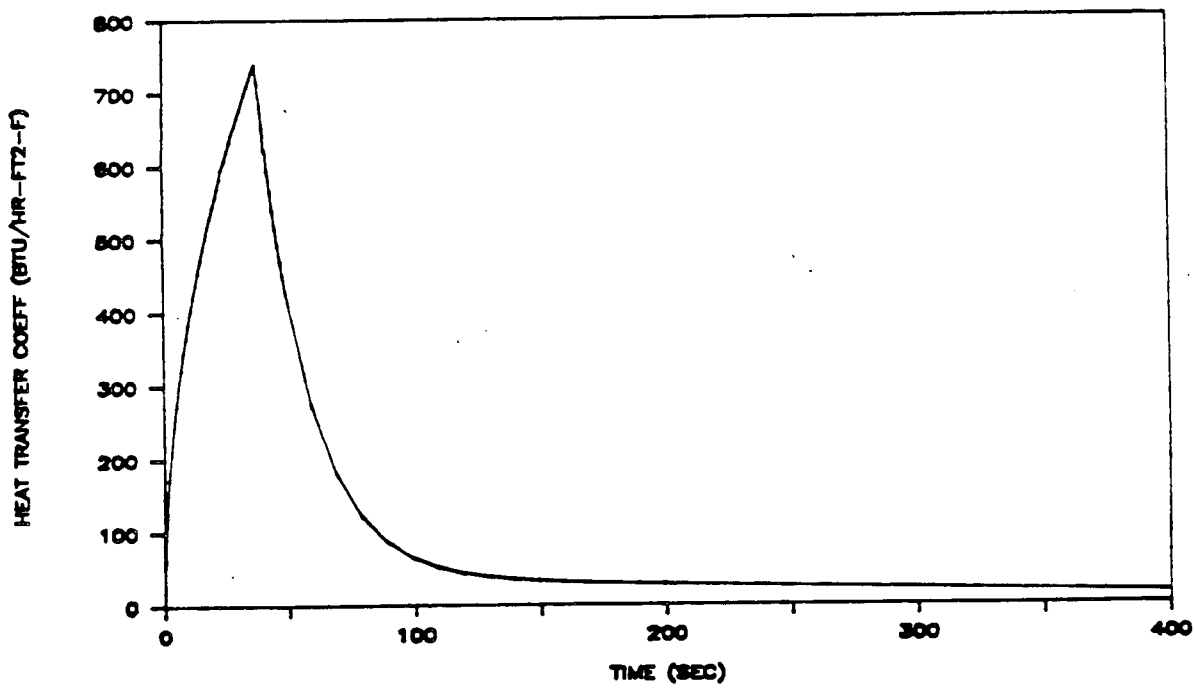
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-50
Core Power Transient-
DECLG (C_D=0.8)
(High T_{ave}) MINSI



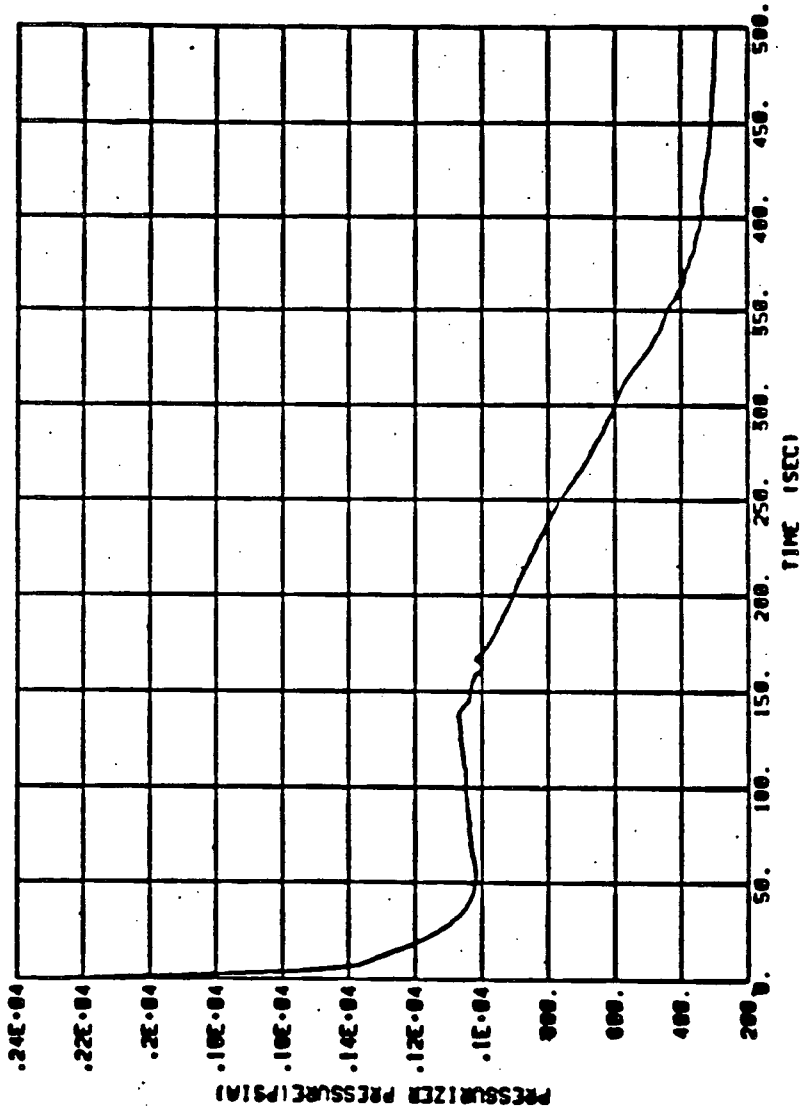
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-51
Break Energy Released to Containment-
DECLG (C_D=0.4)
(High T_{ave}) MINSI



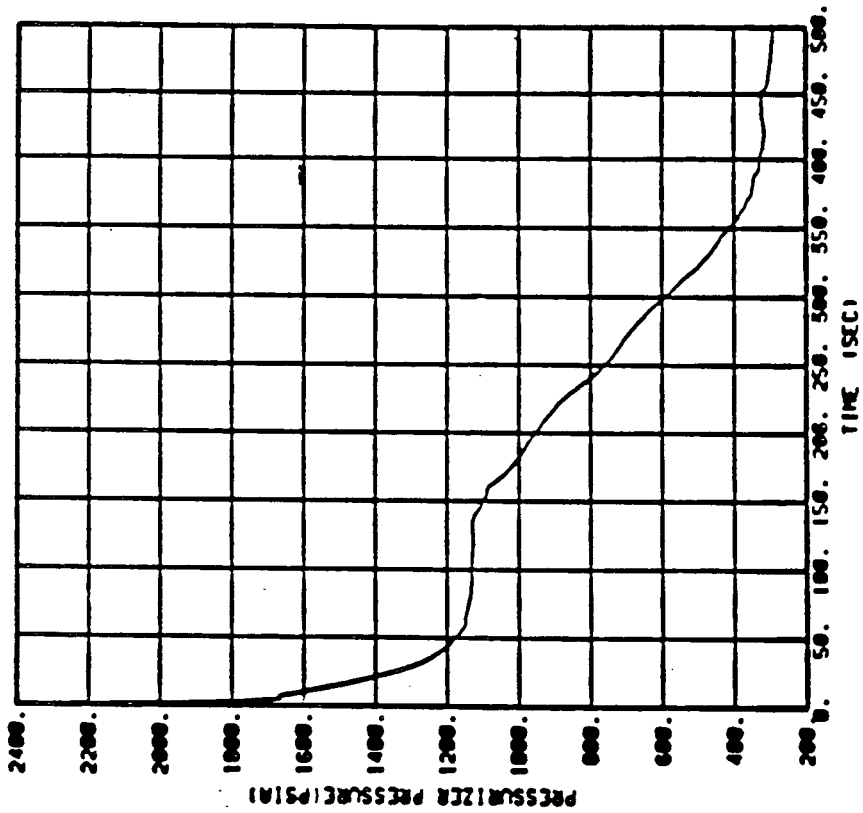
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-52
Containment Wall Heat Transfer
Coefficient-DECLG (C_D=0.4)
(High T_{ave}) MINSI



CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

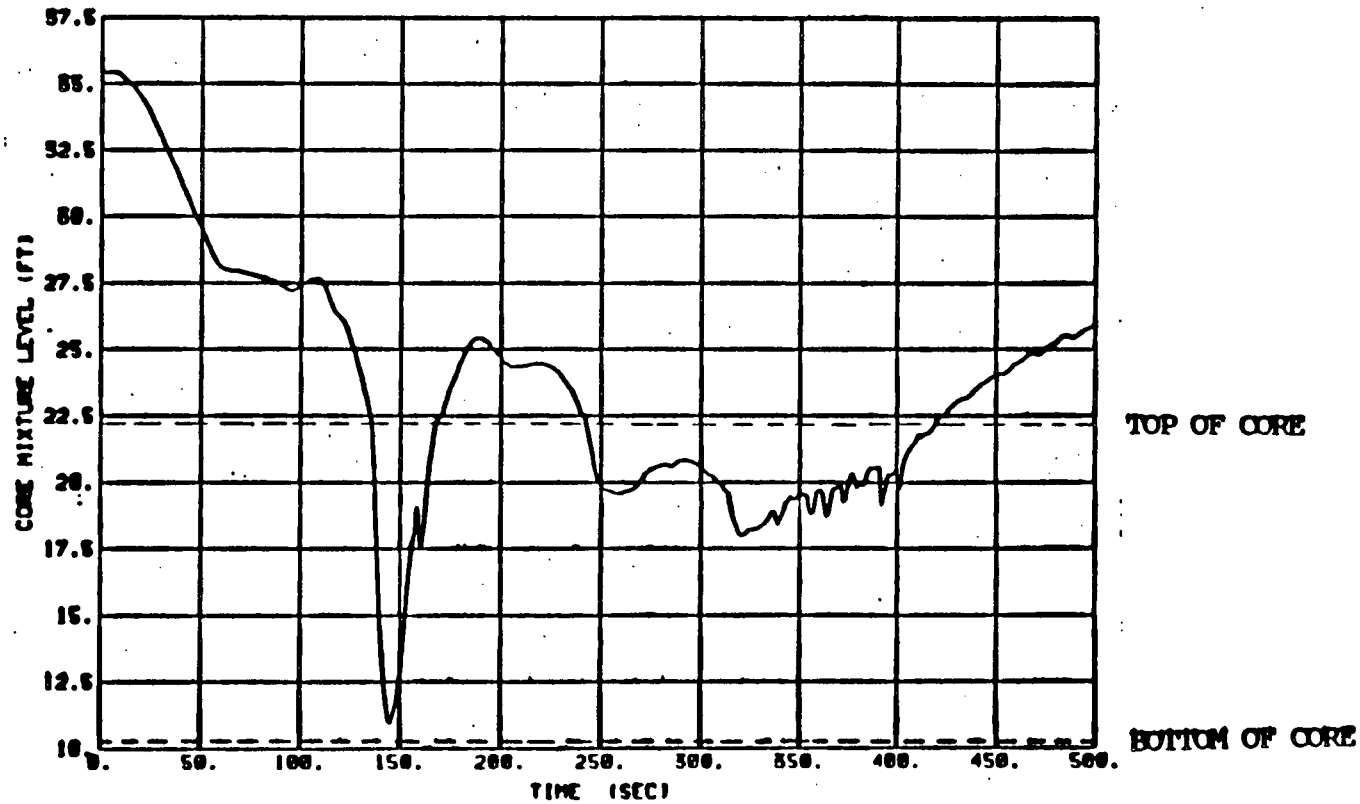
Figure 14.3-53a
(Low Tave) 6-Inch Cold Leg Break/RCS Pressure vs Time



CONSOLIDATED EDISON CO.

INDIAN POINT UNIT 2

Figure 14.3-53b
(High Tave) 6-Inch Cold Leg
Break/RCS Pressure vs Time

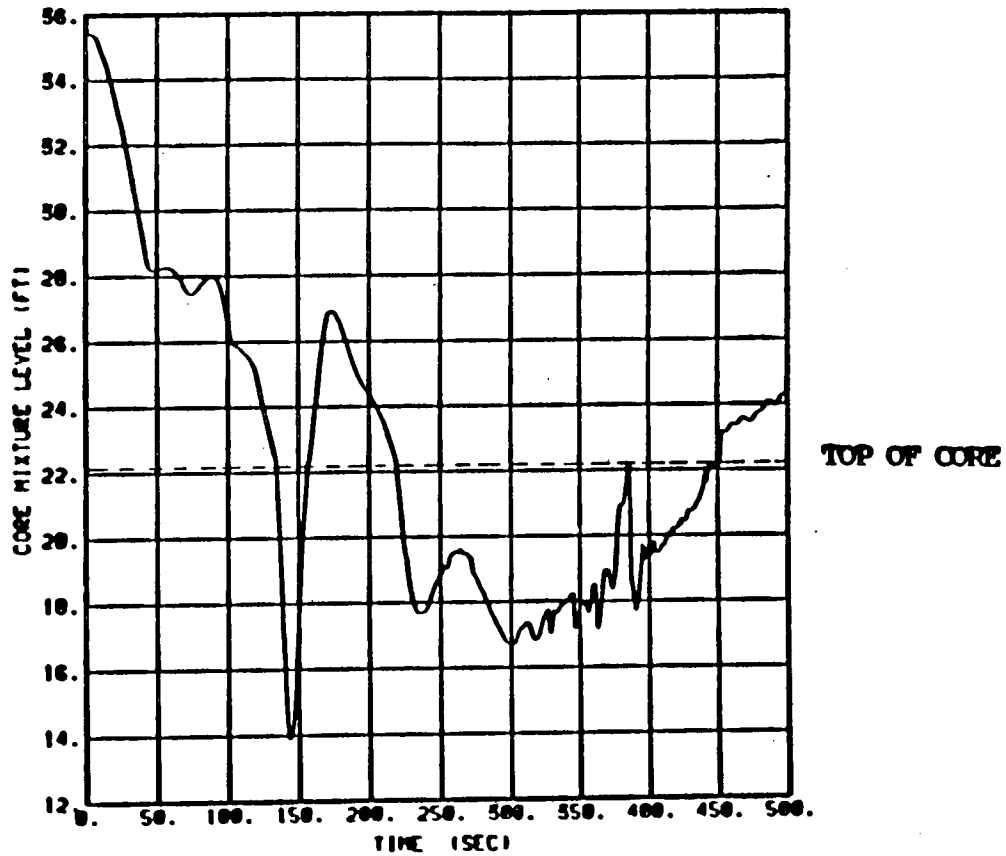


CONSOLIDATED EDISON CO.

INDIAN POINT UNIT 2

Figure 14.3-54 a

(Low Tave) 6-Inch Cold Leg Break/Core Mix Height vs Time

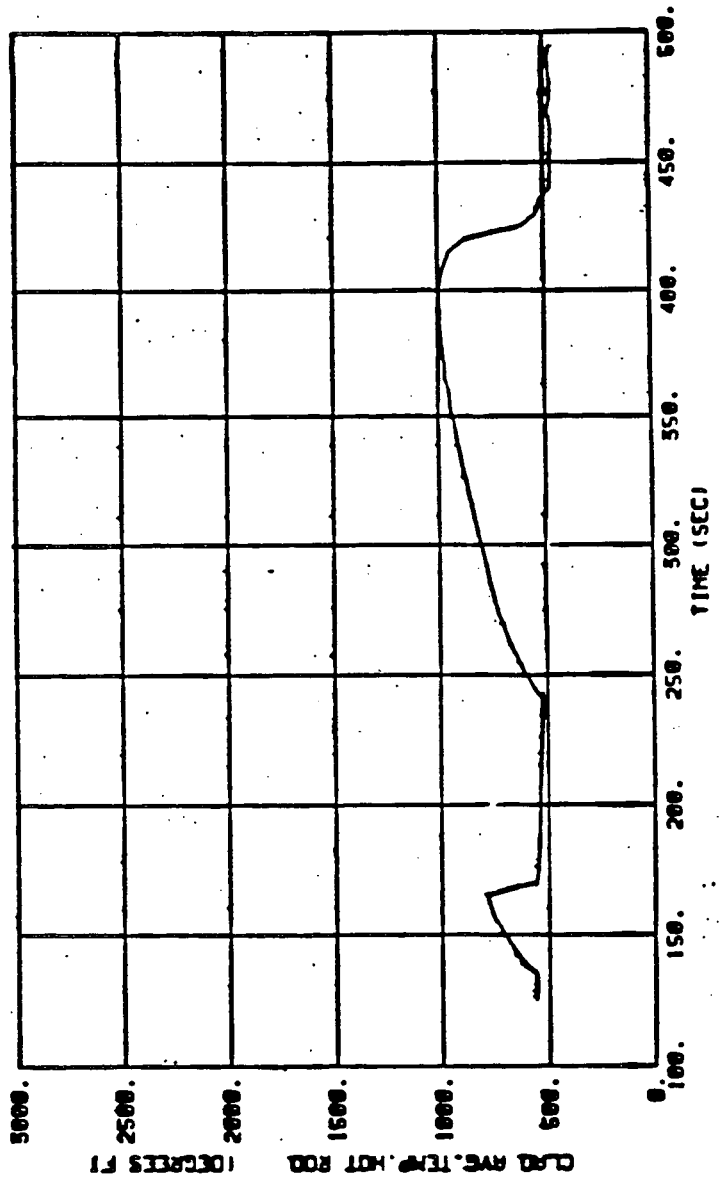


CONSOLIDATED EDISON CO.

INDIAN POINT UNIT 2

Figure 14.3-54b

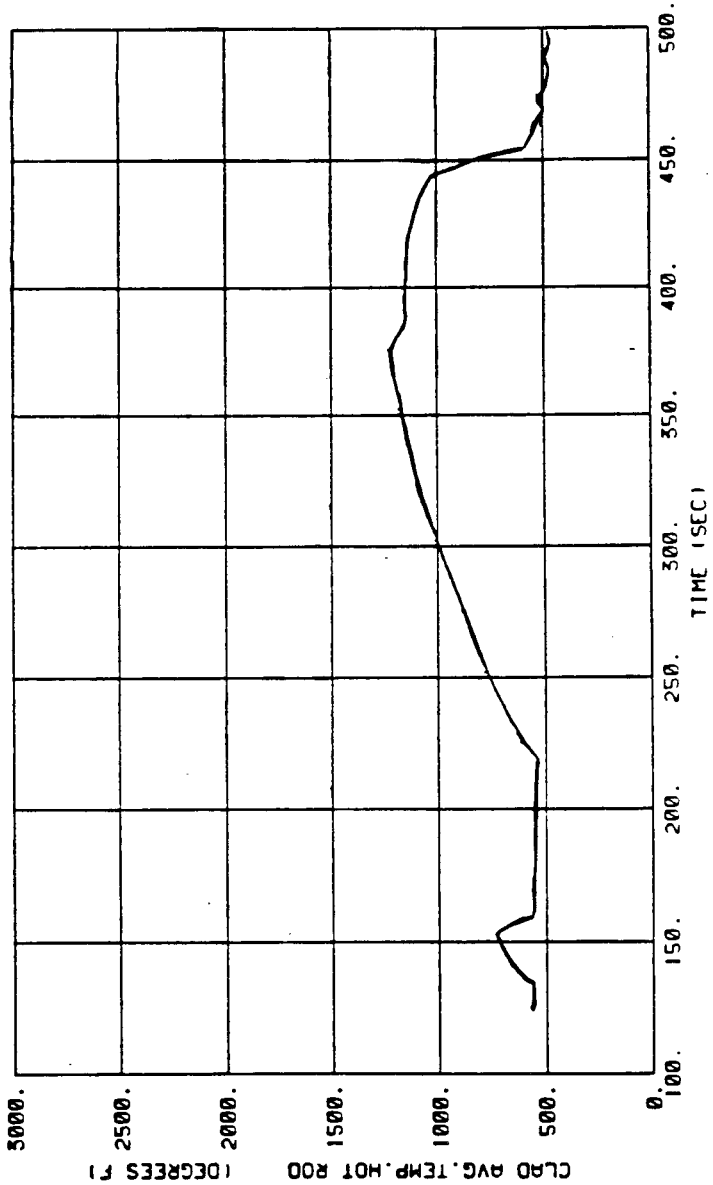
(High Tave) 6-Inch Cold Leg Break/Core Mix Height vs Time



CONSOLIDATED EDISON CO.

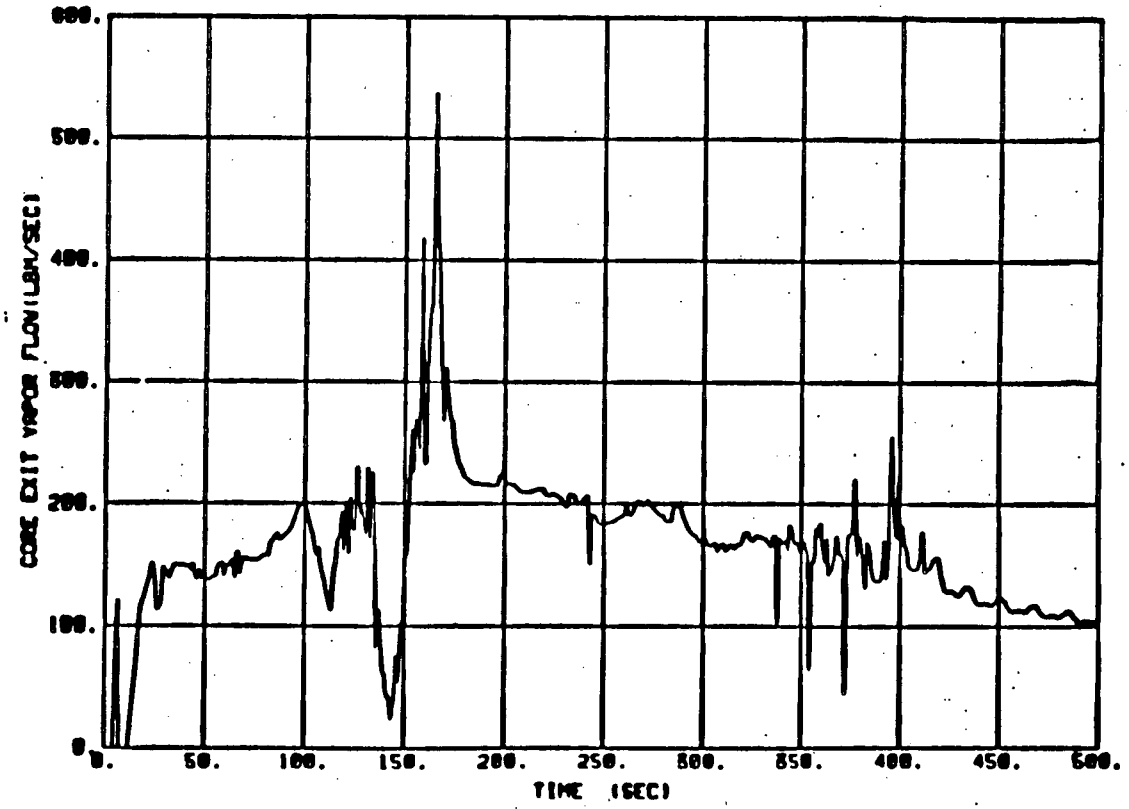
INDIAN POINT UNIT 2

Figure 14.3-55a
(Low Tave) 6-Inch Cold Leg Break/Clad Average Temperature Hot Rod vs Time



CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-55b
(High Tave) 6-Inch Cold Leg Break/Clad Average Temperature Hot Rod vs Time

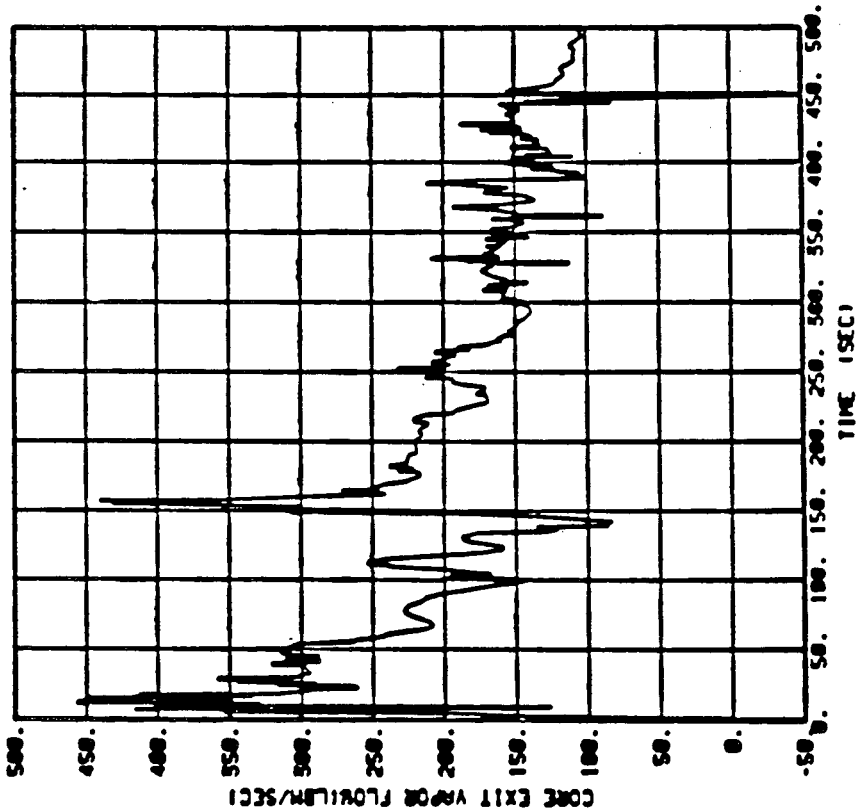


CONSOLIDATED EDISON CO.

INDIAN POINT UNIT 2

Figure 14.3-56a

(Low Tave) 6-Inch Cold Leg Break/Steam Flow vs Time

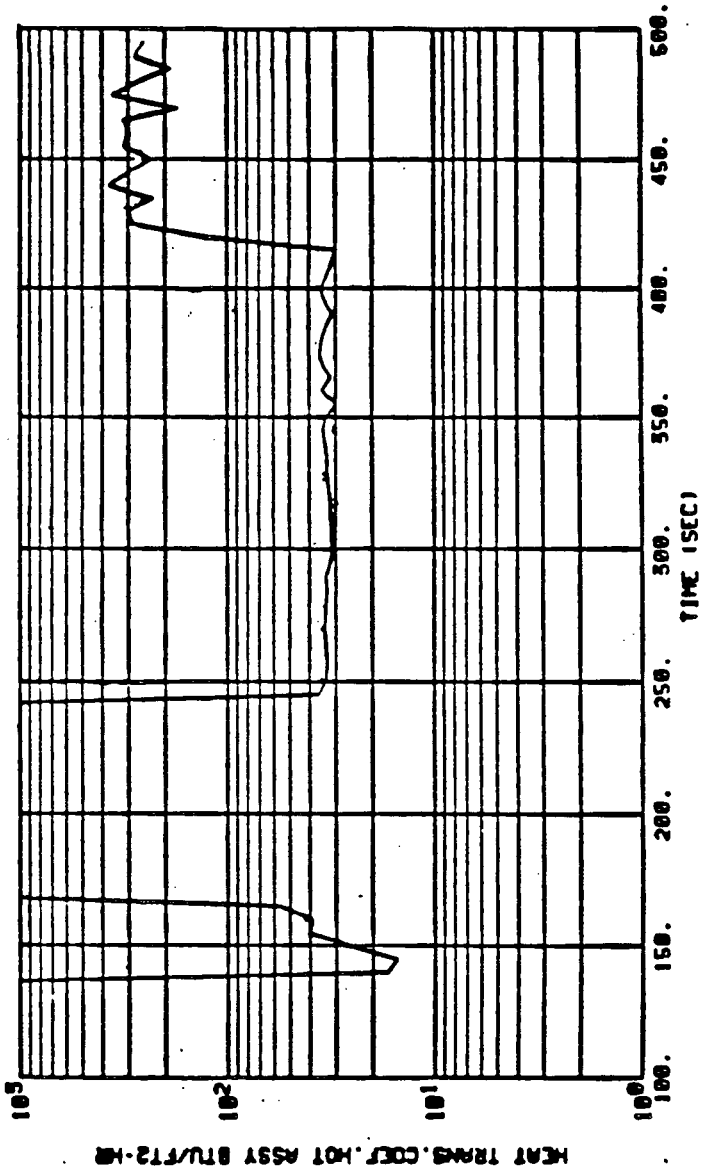


CONSOLIDATED EDISON CO.

INDIAN POINT UNIT 2

Figure 14.3-56b

(High Tave) 6-Inch Cold Leg Break/Steam Flow vs Time

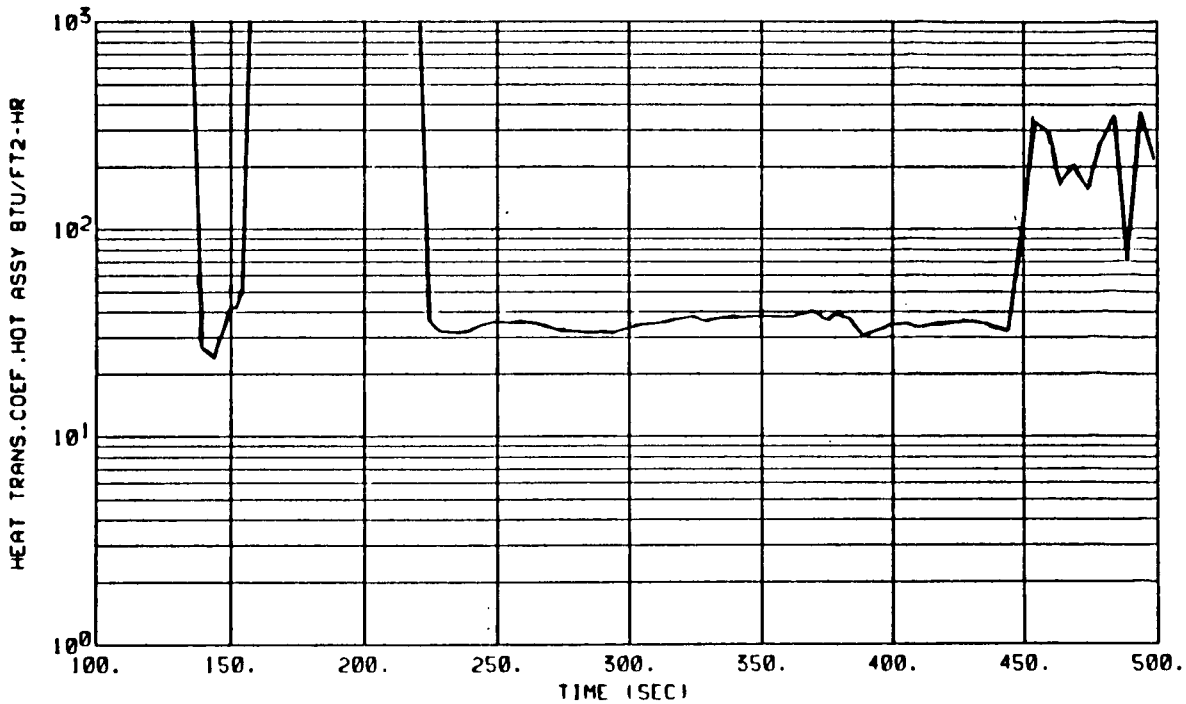


CONSOLIDATED EDISON CO.

INDIAN POINT UNIT 2

Figure 14.3-57a

(Low Tave) 6-Inch Cold Leg Break/Heat Transfer Coefficient Peak Location vs Time



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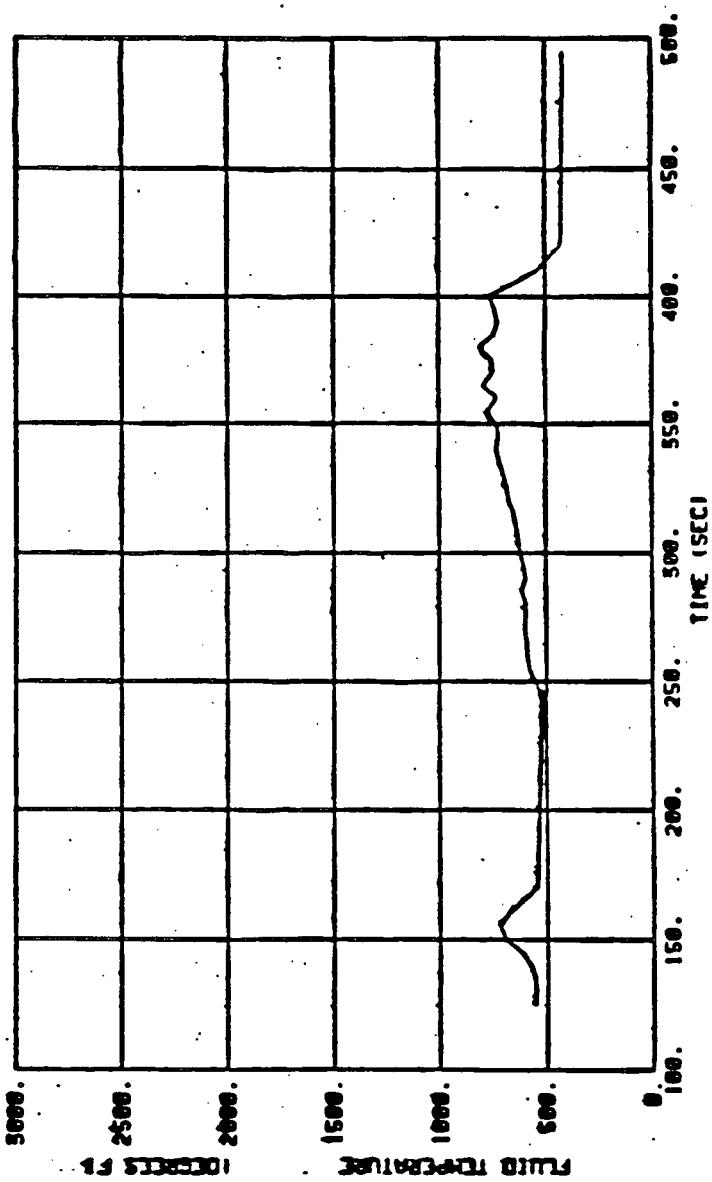
INDIAN POINT UNIT 2

Figure 14.3-57b

(High Tave) 6-Inch Cold Leg
Break/Heat Transfer Coefficient
Peak Location vs Time



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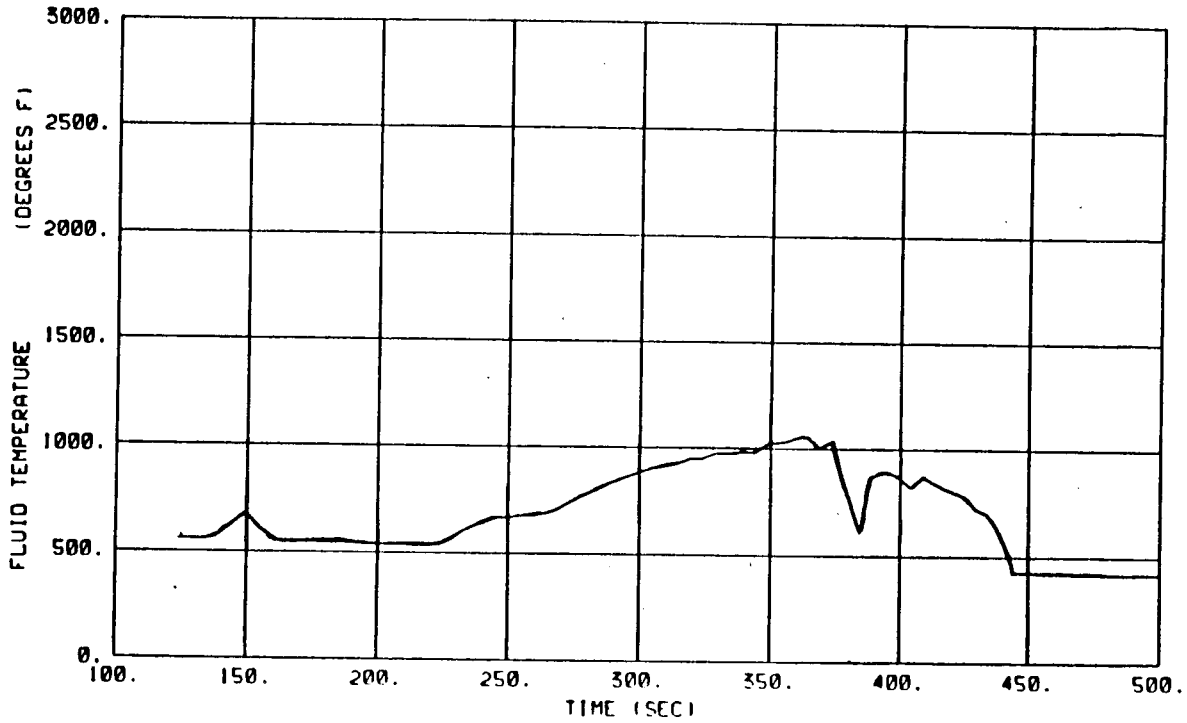
INDIAN POINT UNIT 2

Figure 14.3-58a

(Low Tave) 6-Inch Cold Leg Break/Fluid Temperature Peak Location vs Time



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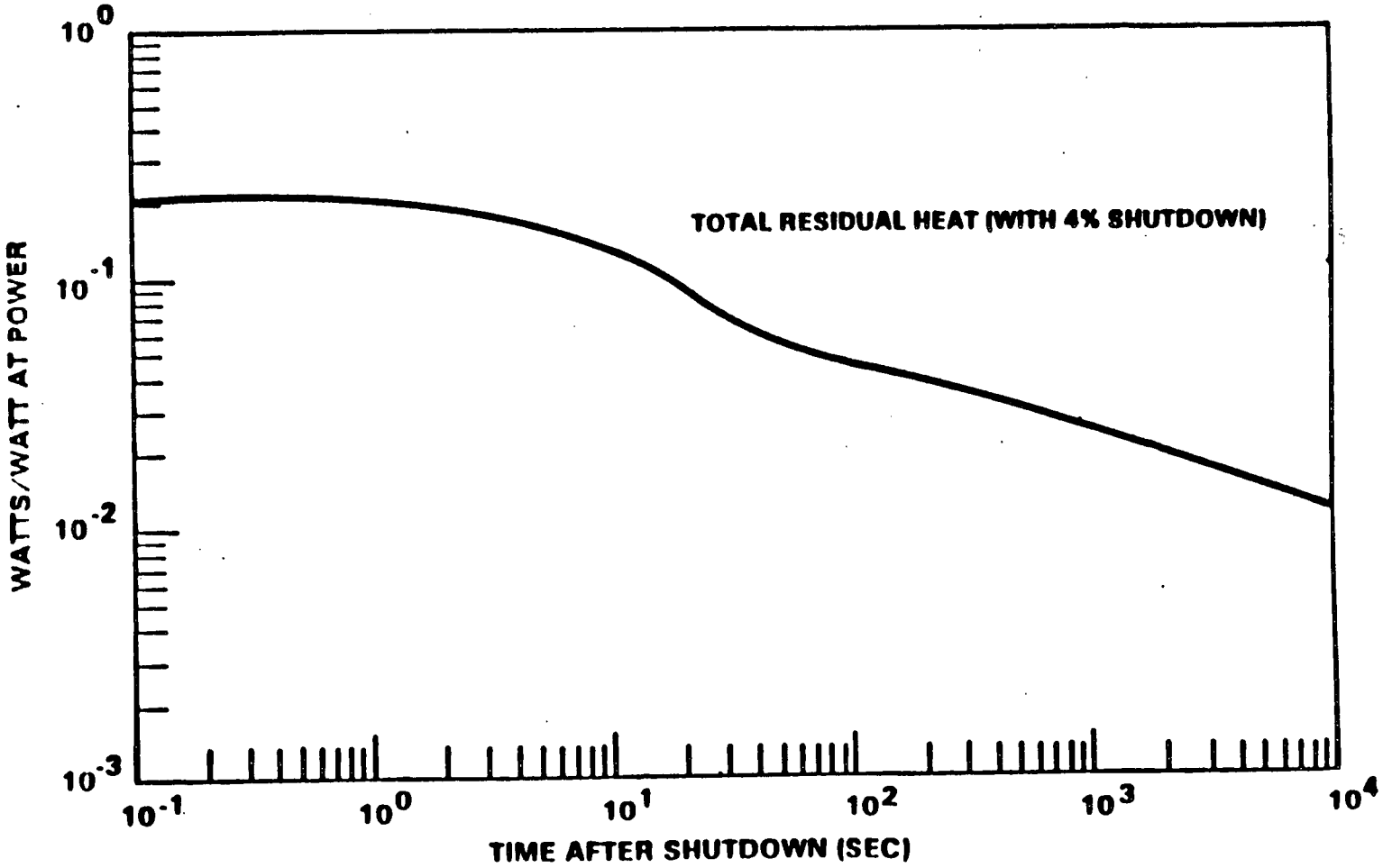
INDIAN POINT UNIT 2

Figure 14.3-58b

(High Tave) 6-Inch Cold Leg Break/Fluid Temperature Peak Location vs Time



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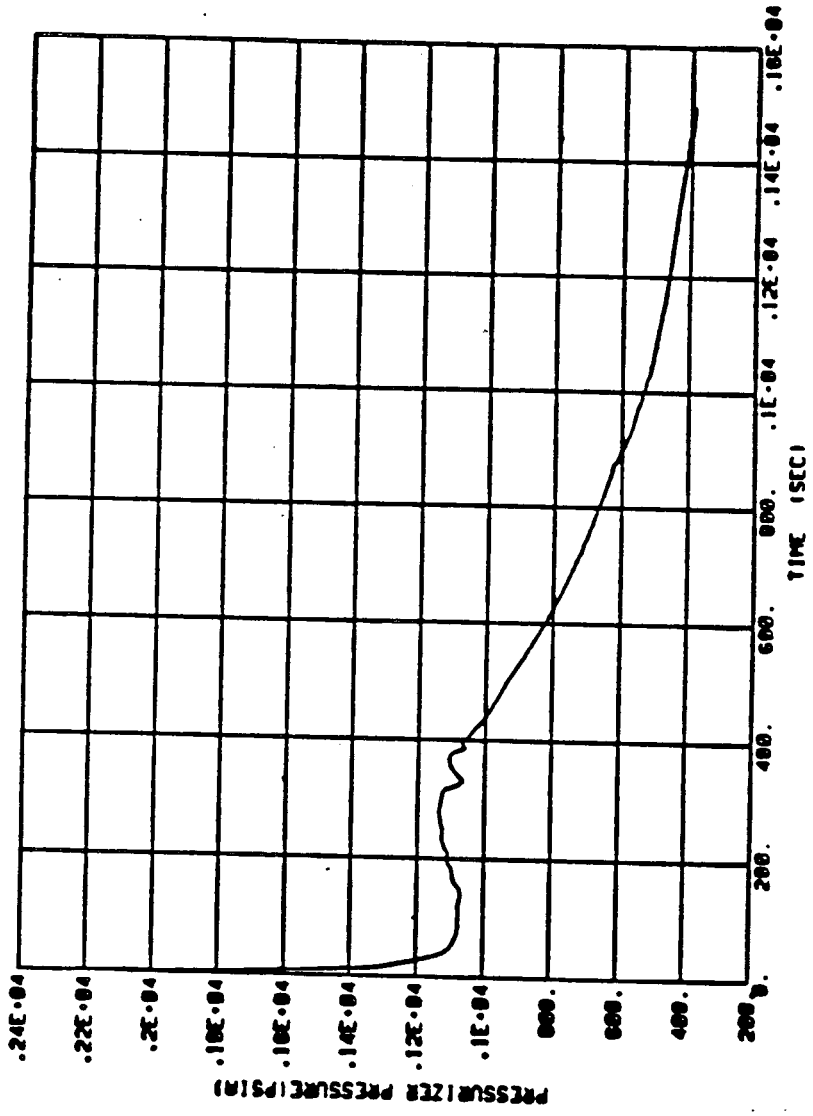
INDIAN POINT UNIT 2

Figure 14.3-59
Core Power Distribution - Total Residual Heat vs Time



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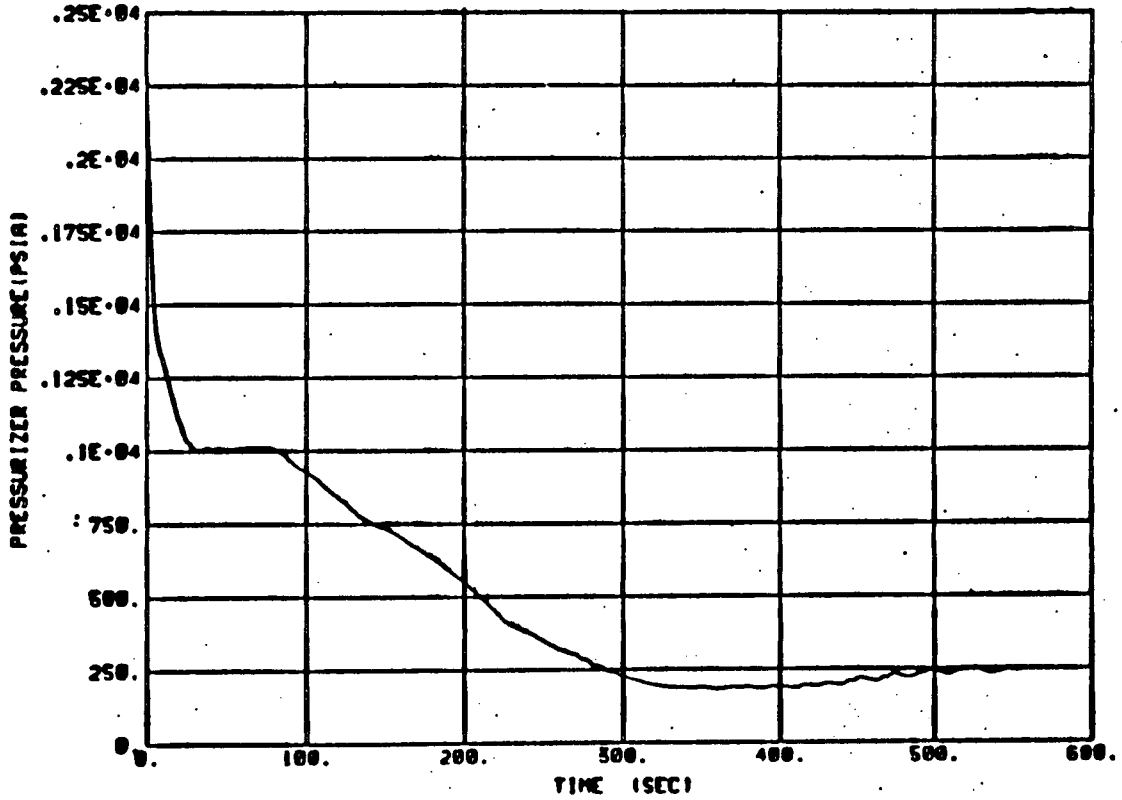
INDIAN POINT UNIT 2

Figure 14.3-60

4-Inch Cold Leg
Break/RCS Pressure vs Time



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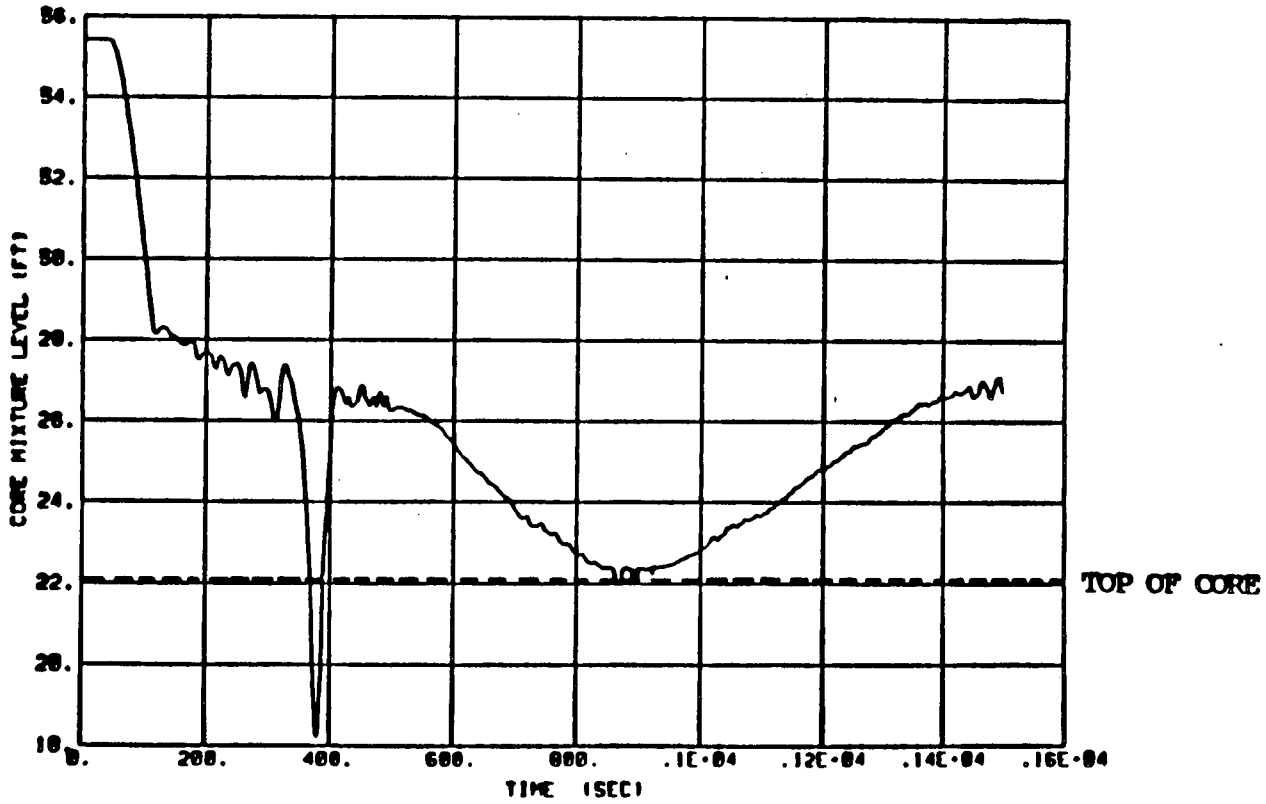
INDIAN POINT UNIT 2

Figure 14.3-61
8-Inch Cold Leg
Break/RCS Pressure vs Time



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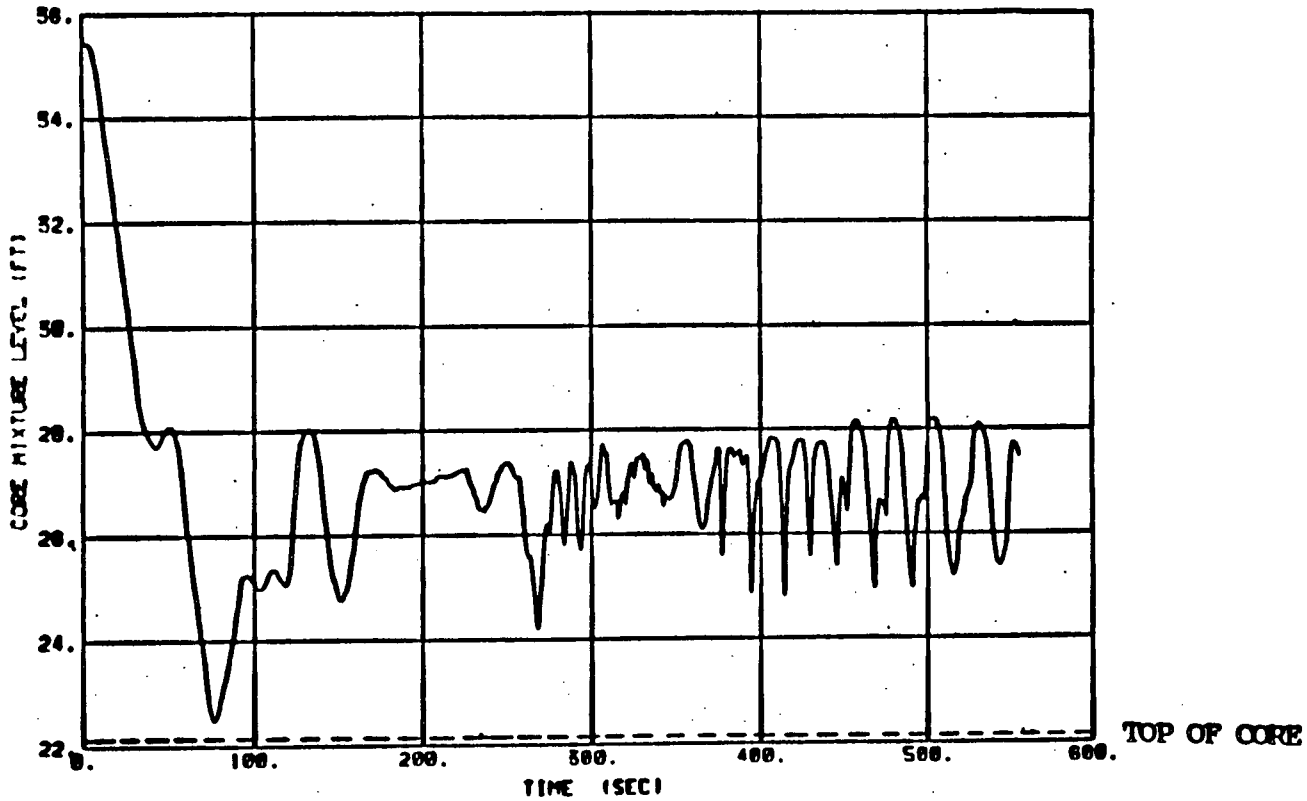
INDIAN POINT UNIT 2

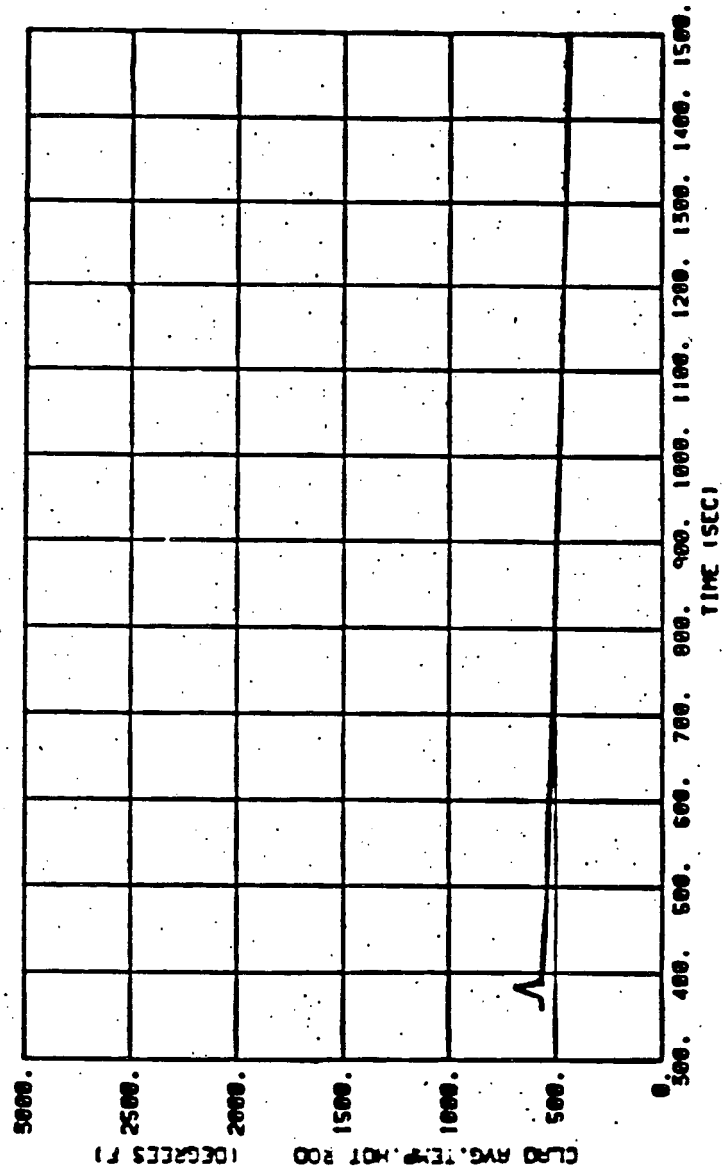
Figure 14.3-62

4-Inch Cold Leg
Break/Core Mix Height vs Time



CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-63
8-Inch Cold Leg
Break/Core Mix Height vs Time



CONSOLIDATED EDISON CO.
INDIAN POINT UNIT 2

Figure 14.3-64

4-Inch Cold Leg Break/Clad Average Temperature Hot Rod vs Time

Enclosure 3
to
Attachment C

No Significant Hazards Consideration Evaluation

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
September 30, 1988

SIGNIFICANT HAZARDS EVALUATION SUMMARY

1.0 Administrative Changes

2.0 ITDP and WRB-1 Correlation

3.0 Low Pressurizer Pressure Reactor Trip Setpoint

4.0 OTΔT and OPΔT Setpoints

5.0 Boric Acid Storage System Volume

6.0 Safety Injection Accumulators

7.0 Boron Concentration Shutdown Margin

8.0 Power Distribution - FΔH

9.0 Rod Drop Time

10.0 Hot Channel Factor $F_Q(z)$

11.0 Low Pressurizer Pressure Safety Injection Setpoint

1.0 ADMINISTRATIVE CHANGES

1.1 Description of Change

The proposed technical specification changes discussed below involve changes which are administrative in nature and are made to delete obsolete requirements, relocate requirements to other sections of the technical specifications, make typographical corrections and clarification. Requirements which refer to three loop operation are deleted since the plant is not licensed for this mode of operation and no current safety analyses exist for three loop operation. Restrictions which applied to a specific previous fuel cycle have also been deleted. RCS pressure, temperature and flow limits have been relocated from the safety limits section of the technical specifications to the limiting conditions of operation section of the specifications. In the design features section of the specifications the stated RCS water volume is not applicable without an allowance for steam generator tube plugging and the volume is identified as the "nominal" volume. Specific references to LOPAR fuel assemblies have been deleted because of the general application of the Technical Specification to all fuel assemblies.

Specifications or bases containing these administrative changes appear in the following pages:

Page vii
Page 2.1-1
Figure 2.1-2
Pages 2.3-2, 4, 5, 6, and 7
Page 3.1.G-1
Page 3.10-5
Figures 3.10-3 and 4
Page 3.11-1
Page 5.3-2

1.2 Safety Assessment

There are no current safety analyses performed for three loop operation, as the plant is not licensed for this mode of operation. The proposed removal of specifications for three loop operation is a purely administrative change which deletes incorrect and inappropriate portions of the technical specifications. The proposed change has no effect on the safety function of any systems, components, or operations.

An obsolete requirement to limit the Region 1 fuel residence time to 21,000 effective full power hours is deleted from Tech Spec. 2.1. This paragraph does not in any way refer to or apply to any other fuel, and is therefore meaningless because there is no longer any Region 1 fuel in the reactor.

The technical specification in Section 2.3.3 for automatic backup to manual tripping is specific to LOPAR fuel and will also be applicable to OFA fuel.

The proposed change to relocate the DNB parameters from Section 2.1 to a new Section 3.1.G, is purely an administrative change to improve the organization of the Tech Specs. This reorganization has no effect on the application of any specifications, and is therefore purely an administrative change.

Proposed changes to pages identified in Section 1.1 to correct typographical errors and clarify requirements are administrative in nature. The proposed change to the RCS volume listed in Specification 5.3.B.3 identifies the volume with 0% steam generator tube plugging and identifies the volume as "nominal". This change corrects errors introduced when plugging levels change and identifies the volume as a nominal rather than a limiting value. The specification being changed is descriptive in nature, and thus the change has no effect on the safety function of any components, systems or operations. No safety analyses are affected by this change, because the RCS volumes assumed in safety analyses vary and are conservative for each specific case.

1.3 Basis For No Significant Hazards Consideration Determination

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (51 FR 7751) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

The proposed change to delete all specifications related to three loop operation is to acknowledge that safety analyses have not been performed for this mode of operation, and Indian Point No. 2 is not licensed for power operation with only three loops operating. The proposed change to delete a paragraph in Tech Spec 2.1 which limits the Region 1 fuel residence time is an administrative change to delete an obsolete requirement.

The proposed change to reorganize the Technical Specifications by moving from Section 2.1 to a new Section 3.1.G, the DNB related parameters of RCS T_{avg} total flow rate, and pressurizer pressure, is purely an administrative change to improve the organization of the Technical Specifications.

The proposed change to the design features to identify the RCS water volume as the nominal volume with no plugged steam generator tubes is an administrative change for clarification.

Consistent with the Commission's criteria in 10CFR50.92, we have determined that the proposed changes described above do not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revisions do not affect plant operations. The proposed revisions delete obsolete specifications, relocate existing specifications and add corrections and clarifications.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed revisions delete obsolete specifications, relocate existing specifications and add corrections and clarifications. The proposed changes do not modify the plant's configuration or operation. Nothing would be added or removed that could conceivably introduce a new or different kind of accident mechanism or initiating circumstances than those previously evaluated.

(3) involve a significant reduction in a margin safety. With the proposed changes, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety. Because these changes are administrative in nature their implementation does not affect any margin of safety.

2.0 ITDP AND WRB-1 CORRELATION

2.1 Description of Change

The proposed revisions to Technical Specification Figure 2.1-1 would include a change to the thermal hydraulic design method employed in satisfying the DNB design bases for Indian Point Unit 2. The use of the Improved Thermal Design Procedure allows for increased design and operational flexibility. In addition, the use of the WRB-1 correlation for evaluation of the departure from nucleate boiling ratio for the OFA assemblies has been included.

2.2 Safety Assessment

The DNB thermal design bases applicable to Indian Point Unit 2 has been and continues to be that there must be at least a 95% probability that the minimum DNBR of the most limiting fuel rod is greater than or equal to the DNBR limit of the DNB correlation for any Condition I or II occurrence. With the previous design procedure used on Indian Point Unit 2, reactor parameters used in analyses were assigned fixed values such that the limiting value of each parameter would be met. With the use of the Improved Thermal Design Procedure, a limit DNBR is chosen with reactor parameters at their nominal values. Variations in the parameters are considered in choosing this limit DNBR such that the thermal design basis continues to be met on a more realistic basis. The use of the Improved Thermal Design Procedure continues to satisfy the Westinghouse DNB design basis and provides additional DNB margin for use in plant design and operational flexibility.

The WRB-1 correlation has been selected for use in DNB evaluation of the Optimized fuel assemblies. Analysis of the LOPAR fuel continues to be based on the W-3 L-Grid correlation. The WRB-1 correlation was developed by Westinghouse based on a much larger set of test results than the W-3

correlation, has a wider range of application in design and a better fit to rod bundle data. Based on over 1000 test data points, the correlation limit which satisfies the criteria for no DNB occurring on a 95%/95% confidence level for the WRB-1 is a 1.17. This same criteria applied to the W-3 L-Grid correlation resulted in a correlation limit of 1.24. Use of the WRB-1 correlation for the Optimized fuel assemblies satisfies the same design criteria as applied to Indian Point Unit 2 in the past and continues to apply to the LOPAR fuel. Use of this correlation does not result in any reduction of DNB safety margin.

2.3 Basis for No Significant Hazards Consideration Determination

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (51 FR 7751). Example (vi) of those involving no significant hazards consideration discusses a change which may reduce a safety margin but where the results are clearly within all acceptable criteria with respect to the system or component. The proposed change for the use of the accepted Improved Thermal Design Procedure (ITDP) and WRB-1 correlation for DNB evaluation of the Optimized fuel assemblies is discussed below.

Consistent with the Commission's criteria in 10CFR50.92, we have determined that the proposed change does not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The ITDP and WRB-1 represent changes to analyses methods only. The probability of an accident occurring is not impacted by the methods selected to evaluate the DNB design basis associated with that accident once it has been postulated to occur. The consequences of the accident must satisfy the same DNB design basis as previously evaluated. Use of ITDP and the WRB-1 do not decrease the available DNB margins when evaluating an accident.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The noted changes are to the methods used in evaluating the DNB design basis only and are involved in analyses only after an accident has been postulated to occur.

(3) involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety. The DNB design criteria continues to be satisfied with the use of ITDP and the WRB-1. As described in the safety assessment, use of this improved method and correlation do not decrease DNB margin over methods and correlations previously used in Indian Point Unit 2

3.0 LOW PRESSURIZER PRESSURE REACTOR TRIP SETPOINT

3.1 Description of Change

The proposed revision to Technical Specification 2.3.1.B(3) would increase the minimum allowable value for the low pressurizer pressure reactor trip setpoint.

3.2 Safety Assessment

The proposed revision to Technical Specification 2.3.1.B(3) would revise the lower limit at which a reactor trip signal would be produced as a result of low pressurizer pressure. Such a revision does not limit operation flexibility nor affect the safety function of the reactor trip signal on pressurizer low pressure, but merely revises the allowable setpoint limit to a value more consistent with plant operation. Under the proposed amendment, this Technical Specification would be revised to permit the reactor trip on low pressurizer pressure to be established at value greater than or equal to 1870 psig.

The safety function of the reactor trip signal on low pressurizer pressure is to initiate shutdown of the reactor core for severe depressurization events and to ensure that the reactor coolant system pressure does not exceed the applicable lower limit for the overtemperature and overpower ΔT protection.

Among the most severe of the rapid depressurization events are large and small break Loss-Of-Coolant Accidents (LOCA). Worst case large and small break LOCAs have been reanalyzed using currently approved computer techniques including the assumption of the increased setpoint for reactor trip on pressurizer low pressure. For the large break LOCA reactor core shutdown is provided initially by massive void formation in the core and later by through the injection of borated water via the Emergency Core Cooling System. For the small break LOCA, reactor trip is a result of depressurization of the reactor coolant system to the low pressurizer pressure reactor trip setpoint.

The results of each of these analyses have clearly demonstrated conformance with the applicable design and regulatory requirements assuming the increased lower limit for reactor trip on low pressurizer pressure.

As discussed in Section 4.0 that follows, revisions to the OTΔT and OPΔT setpoints given in Technical Specification 2.3.1.B(4) and (5), respectively, have been proposed to reflect the proposed revision to Technical Specification Figure 2.1-1, Reactor Core Safety Limit, discussed in Section 2.0. To ensure the applicability of the proposed Technical Specification revision given in revised Figure 2.1-1, the pressure in the reactor coolant system must be maintained at a value greater than or equal to 1775 psia. The proposed change in the reactor trip setpoint on low pressurizer pressure to a value greater than or equal to 1870 psig clearly demonstrates conformance with this requirement.

With the exception of those Non-LOCA accident analyses that model a reactor trip signal on OTΔT or OPΔT protection which implicitly takes credit for low pressurizer pressure trip as discussed above, no other non-LOCA safety analyses for Indian Point Unit 2 take credit for reactor trip on low pressurizer pressure. Therefore, there is no direct impact on the Non-LOCA safety analyses by the proposed change.

3.3 Basis for Non Significant Hazards Consideration Determination

Consistent with the Commission's criteria in 10 CFR 50.92, we have determined that the proposed change does not involve a significant hazards consideration because the operation of Indian Point Unit 2 in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision is supported by conservative analyses utilizing the latest approved computer codes and methodology. These analyses have demonstrated conformance to the applicable design and regulatory criteria.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change to the minimum allowable setpoint for reactor trip on low pressurizer pressure does not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require evaluation and resolution. Nothing would be added or removed that could conceivably introduce a new or different kind of accident mechanism or initiating circumstance than that previously evaluated.

In general, the proposed change does not adversely effect the ability of the pressurizer low pressure reactor trip signal to perform its safety function to initiate reactor core shutdown during a rapid depressurization event.

(3) involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety.

The safety function of reactor trip on low pressurizer pressure is to initiate reactor core shutdown during a severe depressurization event and to ensure that the reactor coolant system pressure does not exceed the applicable lower limit for the overtemperature and overpower ΔT protection. Worst case large and small break LOCA transients were reanalyzed using the latest approved computer codes and methodology as a basis for evaluating this proposed change. For the Non-LOCA accidents, analyses and evaluations demonstrate continued conformance to all applicable design and safety criteria.

4.0 OT ΔT and OP ΔT SETPOINTS

4.1 Description of Change

The proposed revision to Technical Specifications 2.3.1.B(4) and 2.3.1.B(5) consist of; 1) changes to the constant terms currently included in the Overtemperature ΔT and Overpower ΔT protection logic functions.

The proposed changes to the constant terms currently included in Technical Specifications 2.3.1.B(4) and 2.3.1.B(5) are necessary to reflect the revised reactor core safety limits given in the proposed revision to Technical Specification Figure 2.1-1. The proposed revision to Technical Specification Figure 2.1-1 results from the implementation of a change in the Westinghouse DNB methodology for Indian Point Unit 2 as discussed in Section 2.0 and the proposed change in the allowable value of $F_{\Delta H}$ as discussed in Section 8.0

4.2 Safety Assessment

The proposed revisions to Technical Specification 2.3.1.B(4) and 2.3.1.B(5) would revise the ΔT limit at which a reactor trip signal would be produced as a result of increases in ΔT from either overtemperature or overpower transient conditions. Such a revision does not limit operation flexibility nor affect the safety function of the reactor trip signals on the ΔT protection. These revisions only change the allowable ΔT limit functions to be consistent with plant protection required to bound the reactor core safety limits as specified for the proposed change to Technical Specification Figure 2.1-1 discussed in Section 2.0.

All of the licensing basis accidents described in FSAR Chapter 14 which take credit for an OT ΔT or OP ΔT reactor trip have been either analyzed or evaluated considering the proposed changes to the OT ΔT and OP ΔT setpoint functions as provided in the proposed changes to Technical Specification 2.3.1.B(4) and (5). The results of these analyses and evaluations have demonstrated conformance with the applicable design and regulatory requirements assuming the proposed change for reactor trip on the ΔT protection setpoint functions.

4.3 Basis for No Significant Hazards Consideration Determination

Consistent with the Commission's criteria in 10 CFR 50.92, we have determined that the proposed changes do not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision is supported by conservative evaluations and analyses utilizing the latest approved computer codes and methodology. These analyses have demonstrated conformance to the applicable design and regulatory criteria.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes to the OT ΔT and OP ΔT setpoint functions for reactor trip do not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require evaluation and resolution. Nothing would be added or removed that could conceivably introduce a new or different kind of accident mechanism or initiating circumstance than that previously evaluated.

In general, the proposed changes do not adversely affect the ability of OT ΔT and OP ΔT reactor trip signals to perform their safety function to initiate reactor core shutdown during an overtemperature ΔT or overpower ΔT transient condition, respectively.

(3) involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety.

The safety function of reactor trip on overtemperature ΔT and overpower ΔT is to initiate reactor core shutdown during ΔT transient events to ensure that the reactor core safety limits as defined in Technical Specification Figure 2.1-1 are not exceeded. Evaluations and/or analyses for all of the licensing basis accidents described in FAR Chapter 14 which take credit for an OT ΔT or OP ΔT reactor trip have been performed and the results of these analyses and evaluations have demonstrated conformance with the applicable design and regulatory requirements.

5.0 BORIC ACID STORAGE SYSTEM VOLUME

5.1 Description of Change

The proposed revision to Technical Specification 3.2 would maintain the performance capability and reliability of the boric acid storage system.

5.2 Safety Assessment

The proposed Technical Specification revision to the boric acid storage system is proposed to maintain plant cold shutdown boration capability of the boric acid storage system. The proposed increase also reflects methodology changes which have added conservatism to the calculation.

The safety function of the boric acid storage system is to provide a source of concentrated boric acid which can be used by the chemical and volume control system to borate the reactor coolant system to a cold shutdown condition.

The revised boric acid volume requirement continues to be met with a single Boric Acid Storage Tank (BAST) even though two tanks are provided. Hence, the proposed increase in the minimum boric acid storage system volume can be instituted without having adverse effects upon the health and safety of the public.

5.3 Basis For No Significant Hazards Consideration Determination

As required by 10CFR50.91(a)(1) this analysis is provided to demonstrate that a proposed license amendment to increase the boric acid storage system minimum volume requirements from 4400 to 6000 gallons does not represent a significant hazards consideration.

As stated in the Basis section of Technical Specification 3.2, a minimum volume of boric acid in storage is specified to allow for the reactor coolant to be borated to a cold shutdown condition. The proposed increase in boric acid volume is due to two key reasons: 1) increased core reactivity and 2)

change in the assumptions used to calculate cold shutdown boration requirements.

Consistent with the Commission's criteria in 10CFR50.92, we have determined that the proposed change does not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with this change would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The volume of boric acid required in the boric acid storage system is not considered in the mitigation of Chapter 14 events. The volume is required to ensure that a sufficient volume of boric acid solution is available to borate the reactor coolant system to a cold shutdown condition.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The larger volume requirement is well within the capacity of the boric acid storage system. The RWST provides an alternative source of boric acid to meet redundancy requirements.

(3) Involve a significant reduction in a margin of safety. The use of the more conservative shutdown margin assumptions have not decreased, but actually increased cold shutdown boration capability.

6.0 SAFETY INJECTION ACCUMULATORS

6.1 Description of Change

The proposed revision to Technical Specification 3.3.A.1.c would maintain the performance capability and reliability of the Emergency Core Cooling System accumulators.

6.2 Safety Assessment

The proposed revision to Technical Specification 3.3.A.1.c would revise the minimum cover gas pressure and required water volume limits for the Emergency Core Cooling System accumulators. Such a revision does not limit operational flexibility nor affect the safety function of the accumulators. Under the proposed amendment, this Technical Specification would be revised to maintain the accumulator cover gas pressure for each accumulator at a value greater than or equal to 615 psig and to maintain the water volume for each accumulator at a value between 787.5 and 802.5 cubic feet for each accumulator. These values are within the capability of existing systems and instrumentation.

The safety function of the Emergency Core Cooling System accumulators is to provide passive injection of borated water to the reactor coolant system for events resulting in massive depressurization of the reactor coolant system and loss of reactor coolant system inventory. Among the most severe of these events is the large break Loss-Of-Coolant Accident (LOCA). The worst cast large break LOCA was reanalyzed using the latest computer techniques including the assumption of the increased accumulator cover gas pressure and revised accumulator water volume. For the large break LOCA, the accumulators inject borated water into the reactor pressure vessel via the reactor coolant system cold legs when the system pressure has fallen below the pressure of the accumulator cover gas. This borated water aids in the refilling of the reactor vessel following a postulated large break LOCA. The results of this analysis have clearly demonstrated conformance with the applicable design and regulatory requirements assuming the increased accumulator cover gas pressure and revised accumulator water volume. For a small break LOCA,

depressurization is much slower than for a large break LOCA and refilling of the reactor vessel is accomplished primarily by the high head safety injection system. Accumulators are functional during the small break LOCA, however, evaluations have shown that the small break LOCA analytical results are insensitive to changes of this magnitude. Hence, the design and regulatory requirements associated with large and small break LOCA will continue to be satisfied with the proposed change to the accumulator cover gas pressure and water volume.

6.3 Basis for No Significant Hazards Consideration Determination

Consistent with the Commission's criteria in 10CFR50.92, we have determined that the proposed change does not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revisions are supported by conservative analysis utilizing the latest approved computer codes and methodology for large break LOCA and by evaluation of conformance to the applicable design and regulatory criteria in the unlikely event of a small or large break LOCA.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes to the accumulator cover gas pressure and water volume do not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require evaluation and resolution. Nothing would be added or removed that could conceivably introduce a new or different kind of accident mechanism or initiating circumstances than those previously evaluated.

The proposed changes are within the capabilities of the system and do not adversely effect the ability of the emergency core cooling system accumulators to perform their safety function to provide passive injection of borated water to the reactor coolant system.

(3) involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety.

The safety function of the emergency core cooling system accumulators is to provide passive injection of borated water to the reactor coolant system in the event of massive depressurization and loss of reactor coolant inventory. The worst case large break LOCA transient was reanalyzed using the latest approved computer codes and methodology as a basis for evaluating these proposed changes, and evaluations have determined that these changes will not adversely affect the results of small break LOCA analyses. These analyses/evaluations demonstrate continued conformance to all applicable design and safety criteria.

7.0 BORON CONCENTRATION SHUTDOWN MARGIN

7.1 Description of Change

The proposed revision to Technical Specification 3.8.A.8 and the associated bases would decrease the required shutdown margin during refueling from 10% $\Delta k/k$ to 5% $\Delta k/k$ while fixing the minimum refueling boron concentration at 2000 ppm. To maintain consistency, a proposed change to Technical Specification 3.6.A.1 and its associated bases is also required to reflect the proposed change to the required shutdown margin during refueling given in Technical Specification 3.8.A.8. The proposed change in the minimum required shutdown margin during refueling allows for increased flexibility in fuel management and provides consistency with the required shutdown margin of the spent fuel pit given in Technical Specification section 5.4.

7.2 Safety Assessment

Whenever the reactor vessel head is less than fully tensioned, the plant operation must adhere to the Technical Specification 3.8 requirements for shutdown margin. The purpose of this requirement is to ensure that the reactor core is sufficiently subcritical during refueling such that adequate operator response time exists to mitigate the possible occurrence of an uncontrolled boron dilution transient as described in FSAR Section 14.1.5.2.1. The proposed change to Technical Specification 3.8.A.8 would require that the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following is met:

(1) a shutdown margin greater than or equal to 5% $\Delta k/k$, or,

(2) a boron concentration of greater than or equal to 2000 ppm.

A safety analysis for the boron dilution during refueling event as described in FSAR Section 14.1.5.2.1 has been performed based on the proposed change to Technical Specification 3.8.A.8. The results of this analysis have demonstrated conformance with the acceptable design and regulatory requirements assuming the proposed changes to Technical Specification 3.8.A.8.

The purpose of Technical Specification 3.6.A.1 regarding whenever the reactor vessel head is less than fully tensioned and the refueling shutdown margin requirement is not met is to ensure containment integrity is maintained until acceptable shutdown margin requirements are met for refueling. Since the proposed revision to Technical Specification 3.8.A.8 changes the refueling shutdown margin requirements, a proposed revision to Technical Specification 3.6.A.1 is necessary to maintain consistency. This proposed revision does not change the purpose of Technical Specification 3.6.A.1. and, therefore, is considered to be an administrative change and does not affect any margin to safety.

7.3 Basis for No Significant Hazards Consideration Determination

Consistent with the Commission's criteria in 10 CFR 50.92, we have determined that the proposed changes do not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision is supported by conservative analyses utilizing approved methodology. These analyses have demonstrated conformance to the applicable design and regulatory criteria.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change to the refueling shutdown margin and minimum boron concentration does not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require

evaluation and resolution. Nothing would be added or removed that could conceivably introduce a new or different kind of accident mechanism or initiating circumstance than that previously evaluated.

In general, the proposed change does not adversely affect the ability to keep the reactor safely shutdown during refueling operations.

(3) involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety.

The safety function of refueling shutdown margin and minimum boron concentration is to keep the reactor core shutdown during refueling operations. Safety analyses for the licensing basis accident described in FSAR Chapter 14 which take credit for refueling boron concentration have been performed and the results of these analyses have demonstrated conformance with the applicable design and regulatory requirements.

8.0 POWER DISTRIBUTION FΔH

8.1 Description of Change

The proposed revisions to Technical Specification 3.10.2.1 and the associated bases would increase the allowable peak value of $F_{\Delta H}$ at 100% power from 1.55 to 1.62. The increase in 100% rated power $F_{\Delta H}$ limit from 1.55 to 1.62 allows for more flexibility in the fuel management.

8.2 Safety Assessment

The LOCA and non-LOCA safety evaluations performed in support of the transition to the Westinghouse Optimized Fuel Assembly explicitly considered a 1.62 limit $F_{\Delta H}$ at 100% power in the analyses. The potential for reductions in available DNB margins associated with an increased peaking factor have been offset by the use of improved analyses methods. Specifically, the use of the Improved Thermal Design Procedure and WRB-1 correlation in satisfying the DNB basis have compensated for the increase in $F_{\Delta H}$. Worst case large and small break LOCAs were reanalyzed using the latest approved computer techniques including the proposed increase in $F_{\Delta H}$. Non-LOCA events have been reanalyzed or evaluated for $F_{\Delta H}$ of 1.62 as applicable. Based on these analyses/evaluations, the design and criteria applicable to LOCA and non-LOCA continue to be satisfied for the increased peaking factor.

8.3 Basis for No Significant Hazards Consideration Determination

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (51 FR 7751). Example (vi) of those involving no significant hazards consideration discusses a change which may reduce a safety margin but where the results are clearly within all acceptable criteria with respect to the system or component. The proposed change is to increase allowable peak value of $F^N_{\Delta H}$ at 100% power from 1.55 to 1.62.

Consistent with the Commission's criteria in 10CFR50.92, we have determined that the proposed change does not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The peak $F^N_{\Delta H}$ value represents a design limit on peaking factors which must be satisfied for plant operation. This proposed change is supported by conservative analyses and evaluations based on approved codes and methodologies. All applicable design and safety criteria continue to be satisfied.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change in the design and operational limit value of $F_{\Delta H}$ does not modify the plant's configuration or operation, and therefore the previously postulated accidents are the only ones that require evaluation or resolution. Nothing would be added or removed that could conceivably introduce a new or different kind of accident mechanism or initiating circumstances than that previously evaluated.

(3) involve a significant reduction in a margin of safety. With the proposed changes, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety. Approved analysis codes and methodologies were employed as the basis for evaluating this proposed change.

All applicable LOCA and non-LOCA design and safety criteria continue to be satisfied including the impact of an increased $F_{\Delta H}$.

9.0 ROD DROP TIME

9.1 Description of Change

The proposed revision to Technical Specification 3.10.8, Rod Drop Time, consist of changing the control rod drop time interval of 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry to a control rod drop time interval of 2.4 seconds from gripper release to dashpot entry.

9.2 Safety Assessment

The proposed revision to Technical Specification 3.10.8 is necessary to reflect an increase in rod drop travel time resulting from the planned insertion of OFA fuel. Compared to the current 15x15 LOPAR fuel design, the 15x15 OFA design has a slight reduction in the ID and OD of the guide thimble tube above the dashpot. Although adequate nominal diametral clearance exists for the control rod/guide thimble tube interface, the slight reduction in annular clearance results in an increased resistance for control rod drop. This increased resistance results in a longer time interval for rod travel from release of the gripper to dashpot entry. The rod drop time interval of 2.4 seconds from gripper release to dashpot entry was determined from conservative analytical calculations and bounds all LOPAR to OFA transition fuel cycles up to and including a full OFA core.

Such a revision does not limit operational flexibility nor affect the safety function of the reactor protection system. The revision only changes the allowable control rod drop time to be consistent with the proposed plant operation with OFA fuel.

All of the licensing basis accidents described in FSAR Chapter 14 which take credit for a reactor trip have been either analyzed or evaluated considering the proposed change to the control rod time as provided in the proposed change to Technical Specification 3.10.8. The results of these analyses and evaluations have demonstrated conformance with the applicable design and regulatory requirements assuming the proposed change in the control rod drop time.

9.3 Basis for No Significant Hazards Consideration Determination

Consistent with the Commission's criteria in 10 CFR 50.92, we have determined that the proposed changes do not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision is supported by conservative evaluations and analyses utilizing the latest approved computer codes and methodology. These analyses have demonstrated conformance to the applicable design and regulatory criteria.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change to the control rod drop time for reactor trip does not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require evaluation and resolution. Nothing would be added or removed that could conceivably introduce a new or different kind of accident mechanism or initiating circumstance than that previously evaluated.

In general, the proposed change does not adversely affect the ability of control rods to perform their safety function of initiating core shutdown in response to a reactor trip signal.

(3) involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety.

The safety function of control rod drop in response to a reactor trip signal is to initiate reactor core shutdown. Safety evaluations and analysis for all of the licensing basis accidents described in FSAR Chapter 14 which take credit for a reactor trip have been performed and the results of these analyses and evaluations have demonstrated conformance with the applicable design and regulatory requirements.

10.0 HOT CHANNEL FACTOR $F_Q(Z)$

10.0 Description of Change

The proposed revision to Technical Specification Figure 3.10-2 would maintain limitations on the core axial power distribution to ensure safe operation of the plant.

10.2 Safety Assessment

The proposed revision to Technical Specification 3.10 (Figure 3.10-2) would revise the normalized total peaking factor as a function of core height. Such a revision does not increase the allowable linear heat generation rate at upper elevations of the reactor core. Under the proposed amendment, this Technical Specification would be revised to increase the allowable normalized total peaking factor at the upper elevations of the reactor core.

The purpose of the normalized total peaking factor as a function of core height (K(z)) is to ensure that the linear heat generation rate at all core axial elevations is such that the regulatory requirements of 10CFR50.46 will be satisfied in the unlikely event of a Loss-Of-Coolant Accident (LOCA). Worst case large and small break LOCAs were reanalyzed using the latest computer techniques including realistic axial power distributions. The results of each of these analyses clearly demonstrated conformance with the applicable design and regulatory requirements for axial power distributions limited by the proposed revision.

10.3 Basis For No Significant Hazards Consideration Determination

Consistent with the Commission's criteria in 10CFR50.92, we have determined that the proposed change does not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision is supported by conservative analyses utilizing the latest approved computer codes and methodology. These analyses have demonstrated conformance to the applicable design and regulatory criteria in the unlikely event of a small or large break LOCA.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change to the allowable core axial power distribution limits does not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require evaluation and resolution. Nothing would be added or removed that could conceivably introduce a new or different kind of accident mechanism or initiating circumstances than that previously evaluated.

(3) involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety.

Worst case large and small break LOCA transient were reanalyzed using the latest approved computer codes and methodology as a basis for evaluating this proposed change. These analyses demonstrate continued conformance to all applicable design and safety criteria.

11.0 LOW PRESSURIZER PRESSURE SAFETY INJECTION SETPOINT

11.1 Description of Change

The proposed revision to Technical Specification Table 3.5-1 increases the minimum allowable value for the low pressurizer pressure safety injection setpoint.

11.2 Safety Assessment

The proposed revision to Technical Specification Table 3.5-1 increases the minimum setpoint at which a safety injection signal would be produced as a result of low pressurizer pressure. The existing setpoint corresponds to the low pressure SI trip assumed in the safety analysis. This safety analysis assumption remains unchanged. However, the proposed setpoint limit differs from the existing limit by increasing the low pressure safety injection setpoint in the technical specifications by an amount that accounts for possible instrument error. Thus the proposed change assures that the safety analysis assumption relative to SI initiation would be met even with trip instrument uncertainty at its largest expected value.

Such a revision does not limit operational flexibility nor affect the safety function of the safety injection signal on pressurizer low pressure, but merely revises the allowable setpoint limit to a value more consistent with plant operation. Under the proposed amendment, this Technical Specification would be revised to permit the safety injection setpoint on low pressurizer pressure to be established at a value greater than or equal to 1829 psig, changed from 1700 psig.

11.3 Basis for No Significant Hazards Consideration Determination

Consistent with the Commission's criteria in 10 CFR 50.92, we have determined that the proposed change does not involve a significant hazards consideration because the operation of Indian Point Unit 2 in accordance with this change would not:

1. involve a significant increase in the probability or consequence of an accident previously evaluated. The proposed revision assures that assumptions are met for the existing safety analyses.

2. create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change to the minimum allowable setpoint for safety injection on low pressurizer pressure does not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require evaluation and resolution. Nothing would be added or removed that could conceivably introduce a new or different kind of accident mechanism or initiating circumstances than those previously evaluated.

3. involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety.

   The safety function of the safety injection on low pressurizer pressure is to initiate safety injection flow during a severe depressurization event. The proposed change will increase the allowable pressure setpoint and assure that safety injection flow will be delivered to the reactor core as assumed in the safety analyses.