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FRAP-T5 Uncertainty Study of Five Reactor Transient and Accident Events

Prepared by E. T. Laats, B. L. Hansen

EG&G Idaho, Inc.

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ABSTRACT

The FRAP-T5 fuel rod behavior code, with its recently developed automated uncertainty analysis option, was used at EG&G Idaho, Inc., to calculate rod behavior for five reactor transient and accident events. Included were locked rotor, rod ejection, steam line break, loss-of-flow, and turbine trip without bypass events. The intent was to identify the limiting Nuclear Regulatory Commission (NRC) fuel rod damage criteria based on best estimate calculations with associated uncertainties, rather than the traditional calculations that utilize conservative evaluation models. Conclusions are reported regarding the likelihood and subsequent consequences of exceeding the fuel rod damage limits.

SUMMARY

The FRAP-T5 fuel rod behavior code, with its recently developed automated uncertainty analysis option, was used to calculate rod behavior for five reactor transient and accident events. Included were the locked rotor, rod ejection, steam line break, loss of flow, and turbine trip without bypass events. The intent was to identify the limiting Nuclear Regulatory Commission (NRC) fuel rod damage criteria, based on best estimate calculations with associated uncertainties, rather than more traditional calculations that utilize conservative evaluation models.

From the results of the FRAP-T5 calculations, three overall conclusions were drawn.

1. Most events analyzed did not exceed any of the NRC damage criteria.

2. In cases where a criterion was exceeded, the criterion was thermal-hydraulic in nature [departure from nucleate boiling ratio (DNBR)], not mechanical or thermal.

3. Exceeding the departure from nucleate boiling ratio limit never led to a loss of cladding integrity.

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FRAP-T5 UNCERTAINTY STUDY OF FIVE REACTOR TRANSIENT AND ACCIDENT EVENTS

INTRODUCTION

During the past four years, the Core Performance Branch of the U.S. Nuclear Regulatory Commission (NRC) has sponsored the Fuel Performance Code Applications Program at the Idaho National Engineering Laboratory (INEL). The goals of the program are two fold: (a) to assess light water reactor (LWR) licensing criteria regarding fuel duty during various postulated transients and accidents, and (b) to develop capabilities for the NRC to audit vendor computer codes and calculations of both steady state and transient fuel rod behavior.

Toward the first goal, EG&G Idaho, Inc., examined the possibility of using the best estimate transient fuel behavior code FRAP-T4 for assessing current reactor licensing criteria.[1] Since the results were favorable, a sensitivity study was performed to determine the influence of varying four key fuel behavior code input parameters when modeling five postulated reactor transient and accident events.[2]

Efforts toward attaining the second goal have included the development of a new steady state fuel behavior audit code FRAPCON-1.[3,4,5] This development effort has been sponsored primarily by the Fuel Behavior Research Branch of the NRC, with some assistance from the Core Performance Branch. Also, initial groundwork has started toward the development of a transient fuel behavior audit code based on the FRAP-T5 code.[6,a]

In support of both goals of the Fuel Performance Code Applications Program at INEL, an uncertainty study was conducted. The FRAP-T5 code, with its automated uncertainty analysis option was used to perform this task. The option employs response surface methodology to estimate calculation uncertainty. In this study, the uncertainty of best estimate calculations

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a. EG&G Idaho, Inc., Code Configuration Control Number H002583B.

simulating five selected LWR transient and accident events was estimated. Included were locked rotor, rod ejection, steam line break, loss of flow, and turbine trip without bypass events. Code input was supplied by the four U.S. reactor vendors. Fifteen fuel rod design, model, and operation parameters were perturbed, and the uncertainties of fuel rod thermal, mechanical, and hydraulic performance were examined.

A dicussion of the transient and accident events analyzed are given, along with a description of the FRAP-T5 code and a presentation for each case. Modeling conventions used when assembling the input decks are presented, with results and conclusions.

EVENTS

Five LWR transient and accident events were selected for study. The selected events were the control element assembly (CEA), ejection or rod ejection, loss of coolant flow, locked pump rotor, steam line break, and turbine trip without bypass. Westinghouse Electric Corporation provided input for the rod ejection, locked rotor, and loss-of-flow events. Babcock and Wilcox (B&W) supplied input for rod ejection, locked rotor, and steam line break events. Combustion Engineering (CE) submitted input for rod ejection and loss-of-flow events, and General Electric Company (GE) provided input for turbine trip without bypass event.

To aid in the assimilation and interpretation of the results, a brief description of each event is presented below. Assumptions concerning power and coolant histories for each event were different among the vendors, although each vendor was depicting the same event.

Locked Rotor

Following seisure and stoppage of the reactor coolant pump shaft, the core flow rate rapidly decreases. An increase results in the average coolant temperature in the core; thereby, leading to a reactor scram by plant safety systems. Following scram, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to increase in temperature and expand. At the same time, heat transfer to the secondary side of the steam generator is reduced, because the reduced flow results in a decreased surface heat transfer coefficient and because the reactor coolant temperature in the tubes decreases, while the coolant temperature in the secondary side increases (turbine steam flow is reduced to zero upon trip). The rapid expansion of coolant in the reactor core, combined with the reduced heat transfer in the steam generator, causes an insurge of coolant into the pressurizer and a pressure increase throughout the reactor coolant system.

Rod Ejection

The ejection of a CEA results from physical failure of a pressure barrier component in the control rod drive assembly. The pressure differential acting on the control rod assembly rapidly ejects the assembly from the core region. The power excursion due to rapid increase in reactivity is limited by the negative Doppler reactivity feedback effect and terminated by reactor protection system trips.

Steam Line Break

Rupture of a main steam system pipe or valve results in an uncontrolled steam release from a steam generator. As a result, the rate of heat extraction by the steam generator increases, and causes a cooldown of the reactor coolant. With a negative moderator coefficient of reactivity, the cooldown produces a positive reactivity addition and subsequent scram due to an overpower condition. Assuming the most reactive control rod is stuck in its fully withdrawn position, there is a possibility that the core will again become critical and return to near operating power, even with the remaining control rods inserted.

Loss of Flow

A loss-of-coolant flow event results from a mechanical or electrical failure in one or more reactor coolant pumps, or from a fault in the power supply to these pumps. The reactor is operating at full power at the time of the incident. The immediate effect of a loss-of-coolant flow is the rapid increase in coolant temperature. This increase results in departure from nucleate boiling (DNB). Subsequent fuel damage is possible if the reactor is not tripped promptly.

Turbine Trip Without Bypass

During this event, the fast closure of the turbine stop valves, together with a failure of the bypass valves to open, produces a rapid increase in reactor system pressure. This pressure increase causes a compression of voids in the primary coolant system, inducing a rapid increase in neutron flux, with termination of the power increase by reactor scram. Opening of the relief valves limits the extent of the pressure rise, also limits the magnitude of the neutron flux peak, and the resulting peak heat flux.

CODE DESCRIPTION

The FRAP-T5 transient fuel rod behavior code calculates fuel rod transient temperature and deformation responses, resulting from changes in rod power level or cladding surface boundary conditions experienced during such events as a loss-of-coolant accident (LOCA), a power-cooling-mismatch, or a reactivity initiated accident. The transient heat conduction equation is solved at input or internally specified time intervals. Changes in material properties, pellet, gap, cladding surface heat transfer conditions, rod internal pressure distribution, mechanical interaction state, and rod deformation are taken into account. The mechanical response model computes cladding deformation resulting from thermal expansion, hydrostatic pressure differences, gap closure, and high temperature cladding rupture. Fuel deformation occurs by thermal expansion only. During each time step, output from the mechanical response model interacts with material properties and transient thermal models until individual node displacement, temperature, and rod internal pressure satisfy convergence criteria.

Rod geometry and design parameters, equivalent channel dimensions, inlet fluid conditions, power history, nodalization, convergence criteria, time step size, and various option flags are the minimum user input requirements. If necessary, thermal-hydraulic boundary conditions, time and location of critical heat flux, and heat transfer correlation multipliers can be user-supplied, based on results of experiments or supporting analyses. The code is dimensioned to handle rod arrays of limited size, but currently no feedback is provided to account for rod-to-rod interactions occurring as a result of flow redistribution, cladding deformation, or fuel rod failure.

An important feature in FRAP-T5 is the automated uncertainty analysis option. This option calculates the uncertainties of calculated fuel rod behavior variables due to uncertainties in fuel rod fabrication variables, materials properties, power, and cooling. The procedure to use the uncertainty analysis option is straightforward. The user selects a best estimate problem and makes the choice of input variables to be perturbed and output responses to be analyzed. Then, the code will follow a set procedure.

1. The statistical experimental design will be chosen. This design is simply a pattern for perturbing the specified variables of the problem. The problem is run as many times as the design dictates, each time varying the input variable perturbations according to the pattern.

2. A multiple regression routine is used to generate response surface equations using the information derived from Step 1. These equations are intended to replicate the responses of the code, with much lower costs.

3. The response surface equations are used to generate uncertainty distributions for the response parameters. Second order error propagation analysis is used to estimate the means and variances of the responses.

4. Finally, estimates of the fractional contributions to the response variances are made to indicate the relative importance of individual input variables.

FRAP-T5 is documented in three separate volumes. Reference 6 documents FRAP-T5 models, input format, and other running instructions. Reference 7 describes the fuel rod material property package, MATPRO. Reference 8 presents the results of independent FRAP-T5 model assessment studies.

CODE INPUT

Five reactor transient and accident events were analyzed in this study. Each vendor supplied input for up to three of the five events, for a total of nine cases. These nine cases are identified on Table 1 that lists the case identification number, the event being considered, the vendor who supplied the input, the rod bundle configuration from which the selected rod was chosen, and the event characteristics and duration. The input values for the fuel design parameters are consistent with those reported in reference safety analysis reports for each vendor. Initial conditions prior to each event correspond to beginning-of-life operation of the lead rod. These input are summarized on Table 2.

An uncertainty value was applied to fifteen parameters, which include fuel rod power, system pressure, coolant mass flux, fuel thermal conductivity, fuel specific heat, fuel thermal expansion, cladding diametral thermal expansion, internal gas thermal conductivity, pellet-cladding gap heat transfer, fuel pellet diameter, fuel pellet as-built density, cladding inside radius, cladding surface roughness, cladding oxidation, and rod surface critical heat flux. These parameters were selected because of their known influence on fuel rod behavior noted from previous experience, or because engineering judgment of each case identified these parameters as potentially being influential. The uncertainty value applied to each parameter is listed on Table 3. The uncertainties for the operation parameters were supplied by the respective vendors. The materials properties uncertainties were taken from the MATPRO-11 subcode. Rod geometry uncertainties were typical of those reported in test rod fabrication reports, and model uncertainties were usually the code default values based primarily on engineering judgment.

All input, both steady state and transient, were intended to be best estimate. However, residual conservatisms probably still exist for the operation parameters, namely, rod power, system pressure, and coolant mass flux. The histories for these parameters were based on other vendor calculations and assumptions that may have residual conservatisms themselves. This problem of residual conservatisms was more easily avoided for the rod geometry and materials properties parameters because the best estimate values and their associated uncertainties were based on fits of best estimate measurements.

TABLE 1. IDENTIFICATION OF EVENTS ANALYZED

Case Number	Event	Vendor	Rod Bundle Configuration	Event Characteristics[a]	Event Duration (s)
1	Locked rotor	Babcock and Wilcox	17 x 17	A,B	6.095
2	Locked rotor	Westinghouse	17 x 17	A	9.95
3	Rod ejection	Babcock and Wilcox	17 x 17	B,C	5.0
4	Rod ejection	Combustion Engineering	16 x 16	B	4.8
5	Rod ejection	Westinghouse	17 x 17	B	9.9
6	Steam line break	Babcock and Wilcox	15 x 15	A,B,C	10.0
7	Loss of flow	Combustion Engineering	16 x 16	A,B	6.0
8	Loss of flow	Westinghouse	17 x 17	A,B,C	30.0
9	Turbine trip without bypass	General Electric	8 x 8	A,B,C	10.0

a. A = flow decrease, B = power increase, C = pressure increase or decrease.

TABLE 2. INITIAL SYSTEM CONDITIONS PRIOR TO EVENT

Case Number	Rod Power (kW/m)	Coolant Mass Flux $(kg/s \cdot m^2)$	Coolant Pressure (MPa)
1	30.3	3637.0	15.5
2	27.8	3518.0	15.7
3	30.3	3532.0	15.5
4	29.1	3532.0	15.5
5	27.6	3518.0	15.7
6	37.4	3601.0	15.2
7	22.0	3532.0	15.5
8	27.6	3518.0	15.7
9	31.4	1440.0	7.4

TABLE 3. PARAMETER UNCERTAINTY VALUES

	Uncertainty								
Parameter	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6	Case 7	Case 8	Case 9
Fuel rod power	±8.4%	±2%	±6.7%	±0.62%	±2%	±8.4%	±4.6%	±2%	±11.8%
System pressure	±2%	±1.3%	±2%	±7.5%	±1.3%	±3%	±7.5%	±1.3%	±4.3%
Coolant mass flux	±15%	±5%	±15%	±15%	±5%	±15%	±17%	±5%	±23.6%
Fuel thermal conductivity	±0.4 W/m•K	±0.4 W/m•K	±0.4 W/m•K	±0.4 W/m•K	±0.4 W/m•K	±0.4 W/m•K	±0.4 W/m•K	±0.4 W/m•K	±0.4 W/m•K
Fuel specific heat	±2% when temperature <500 K, ± 6% when temperature >3000 K, linearly increasing value from ±2% to ±6% for temperature range 500 to 3000 K.								
Fuel thermal expansion	±(0.0000025 x temperature)m/m when temperature <500 K, ±0.00125 m/m when temperature >500 K.								
Cladding diametral thermal expansion	±10% when temperature <1073 K, ±50% when temperature >1073 K.								
Gas thermal conductivity	±(-0.0068 + 0.0000161 x temperature) W/m•K								
Pellet-cladding gap heat transfer	±25%	±25%	±25%	±25%	±25%	±25%	±25%	±25%	±25%
Fuel pellet diameter	±0.1%	±0.1%	±0.1%	±0.1%	±0.1%	±0.1%	±0.1%	±0.1%	±0.1%
Fuel pellet density	±0.67%	±0.67%	±0.67%	±0.67%	±0.67%	±0.67%	±0.67%	±0.67%	±0.67%
Cladding inside radius	±0.1%	±0.1%	±0.1%	±0.1%	±0.1%	±0.1%	±0.1%	±0.1%	±0.1%
Cladding roughness	±10%	±10%	±10%	±10%	±10%	±10%	±10%	±10%	±10%
Cladding oxidation	±17.5% when temperature <1523 K, ±6.5% when temperature >1523 K.								
Critical heat flux	±8%	±8%	±8%	±8%	±8%	±8%	±8%	±8%	±8%

MODELING TECHNIQUES

Consistent modeling was established and applied to all cases. Some modeling conventions were functional in nature and adopted from the standpoint of practicality and input limitations. Radial nodalization consisted of 10 fuel intervals, one gap interval, and two cladding intervals. Axially, the rods were divided into 15 equal intervals. The equivalent closed channel fluid model of FRAP-T5 calculated the steady state enthalpy rise at each time step based on inlet conditions. Initial condition axial power distributions were assumed constant throughout the events. A convergence criteria of 0.05 was specified for the temperature and pressure iterations. Azimuthal temperature distributions were not considered. The transient axial internal gas flow model was used to assure the most realistic cladding ballooning behavior, and no decay heat was added to the vendor supplied power histories since the vendor histories already accounted for this heat source.

Other modeling conventions were based on the need to generate best estimate results. Independent assessment results[8,9] supported the use of the fuel relocation, effective pellet conductivity, and Ross-and-Stoute gap conductance models in the thermal calculations. Cladding oxidation was calculated by the Cathcart-Pawel model. The CE-1 critical heat flux (CHF) correlation was used for Cases 1 through 8, and the GE correlation for Case 9. The nonuniform axial power factors were applied to the CHF correlations, but not the cold wall factors. The Groeneveld 5.9 film boiling correlation was used to model post-CHF surface heat transfer, based on relatively good agreement with Power Burst Facility cladding temperature measurements.[10] The FRACAS-I mechanical deformation model was used instead of FRACAS-II because of its more realistic relocation coupling[8,11] with thermal conductivity calculations and stress-dependent failure probability models in FRAIL.[12]

Input to the uncertainty analysis option was also consistent for all cases. To insure a high degree of resolution while maintaining a minimum number of runs, a one-fourth fractional factorial foldover design was used, which resulted in a total of 32 computer runs being required for each of the nine cases.

FUEL ROD DAMAGE PARAMETERS

Safety analysis reports[13-16] and NRC positions[17-19] were reviewed to define the criteria that specify the onset of fuel rod damage. Table 4 summarizes results of this survey in terms of parameters calculated by FRAP-T5. The various parameters and their respective limits most often applied for safety evaluation purposes have been listed. The prevent condition refers to undesirable consequences that are assumed to be avoided if the corresponding fuel rod performance limit is not exceeded during reactor operation. This report addresses those limits exceeded during the sensitivity study.

TABLE 4. CURRENTLY USED[a] BEGINNING-OF-LIFE FUEL ROD PERFORMANCE LIMITS

Parameter	Condition Prevented by Limit	Limit	Reference[b]
Surface heat flux	PWR rod temperature increase	1.16 MDNBR (CE-1)	17
		1.30 MDNBR (W-3)	17
		1.32 MDNBR (B&W-2)	17
	BWR rod temperature increase	1.06 MCPR	17
Cladding temperature	α - β phase transition	1500-1900°F	Physical limit
	Oxidation threshold	1800°F	17
	Prohibitive oxidation/loss of mechanical strength	2200°F	19
Fuel temperature	Incipient fuel melting	5080°F	17
Fuel enthalpy	BWR cladding failure threshold	170 avg cal/g	17
	PWR cladding failure threshold	200 avg cal/g	13,14,15
	Cladding fragmentation/fuel dispersal/pressure pulses	280 avg cal/g	17
Cladding oxidation	Excessive embrittlement effects	17% thickness	17
Cladding stress	Exceeding ultimate strength	$f(T_{clad})$	19
Cladding strain	Hard pellet-cladding interaction	1%	19
Internal pressure	Cladding tensile stress and ballooning	System pressure	17

a. As either accepted by NRC or submitted by vendors.

b. MBNBR = minimum departure from nucleate boiling ratio; MCPR = minimum critical power ratio. $f(T_{clad})$ = limit is function of cladding temperature.

RESULTS

The results of the FRAP-T5 uncertainty study are presented below for each of the five transient and accident events. Prime emphasis is placed on examining calculated fuel rod performance by comparing key fuel rod behavior parameters and their associated uncertainties, against the NRC rod damage limits discussed in the Section, Fuel Rod Damage Parameters. The level of detail discussed varies. For those parameters whose corresponding damage limit was exceeded or close to being exceeded, the discussion is more detailed. When a limit was not in jeopardy of being exceeded, results are briefly summarized.

Discussion of the locked rotor event is presented first, followed by the rod ejection, steam line break, loss of flow, and turbine trip without bypass events, respectively.

Locked Rotor

Two locked rotor cases were examined in this study. One case was submitted by B&W, and the other by Westinghouse. The general behavior characteristics of the event were a rapid reduction of core flow during the first second of the event, with either the rod power increasing or remaining essentially constant until shutdown at 1.7 s. The B&W case assumed a power increase of about 100%, where Westinghouse assumed a very steady, slightly decreasing power history prior to shutdown. The relative histories of key parameters are shown in Figures 1 and 2.

The calculations modeling the event submitted by B&W are discussed first, followed by the event submitted by Westinghouse.

Babcock and Wilcox Case

For the B&W case, the average fuel centerline temperature history of the hot node is shown in Figure 3. The maximum average temperature was about 2400 K with a 2σ uncertainty of 370 K. This temperature corresponded to 1.7 s into the transient. The fuel enthalpy at this temperature

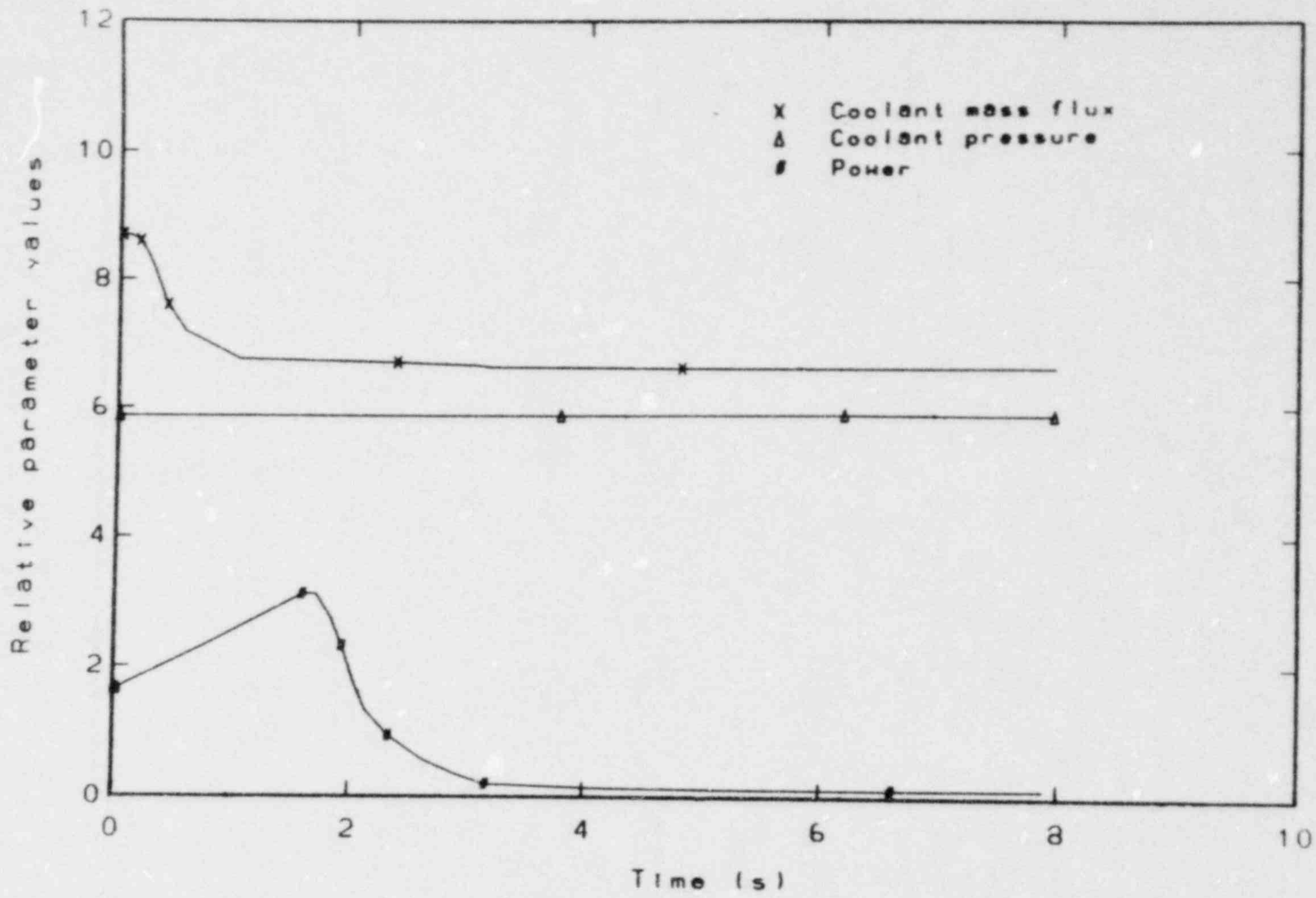

Figure 1. Operating history for B&W locked rotor event.

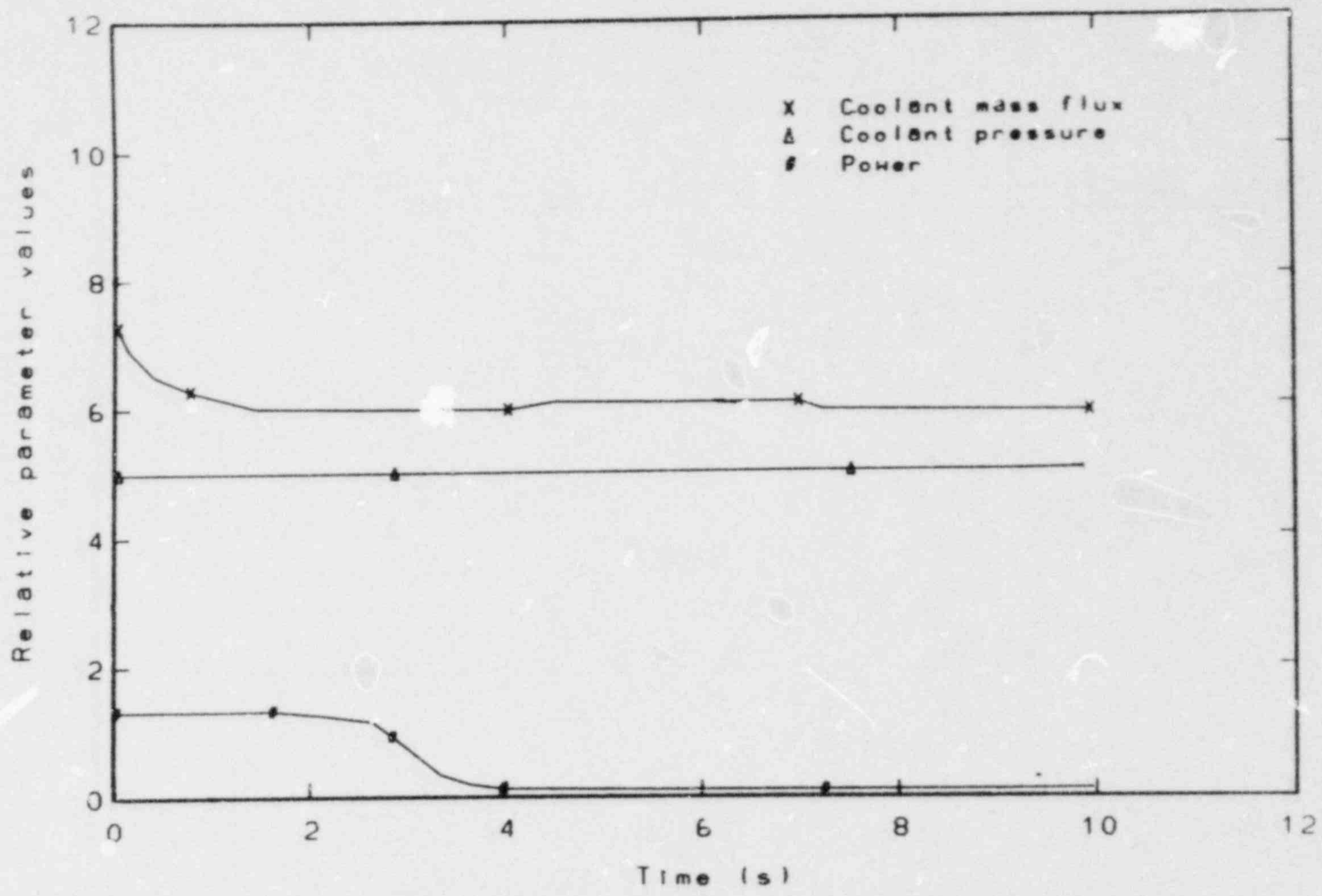

Figure 2. Operating history for Westinghouse locked rotor event.

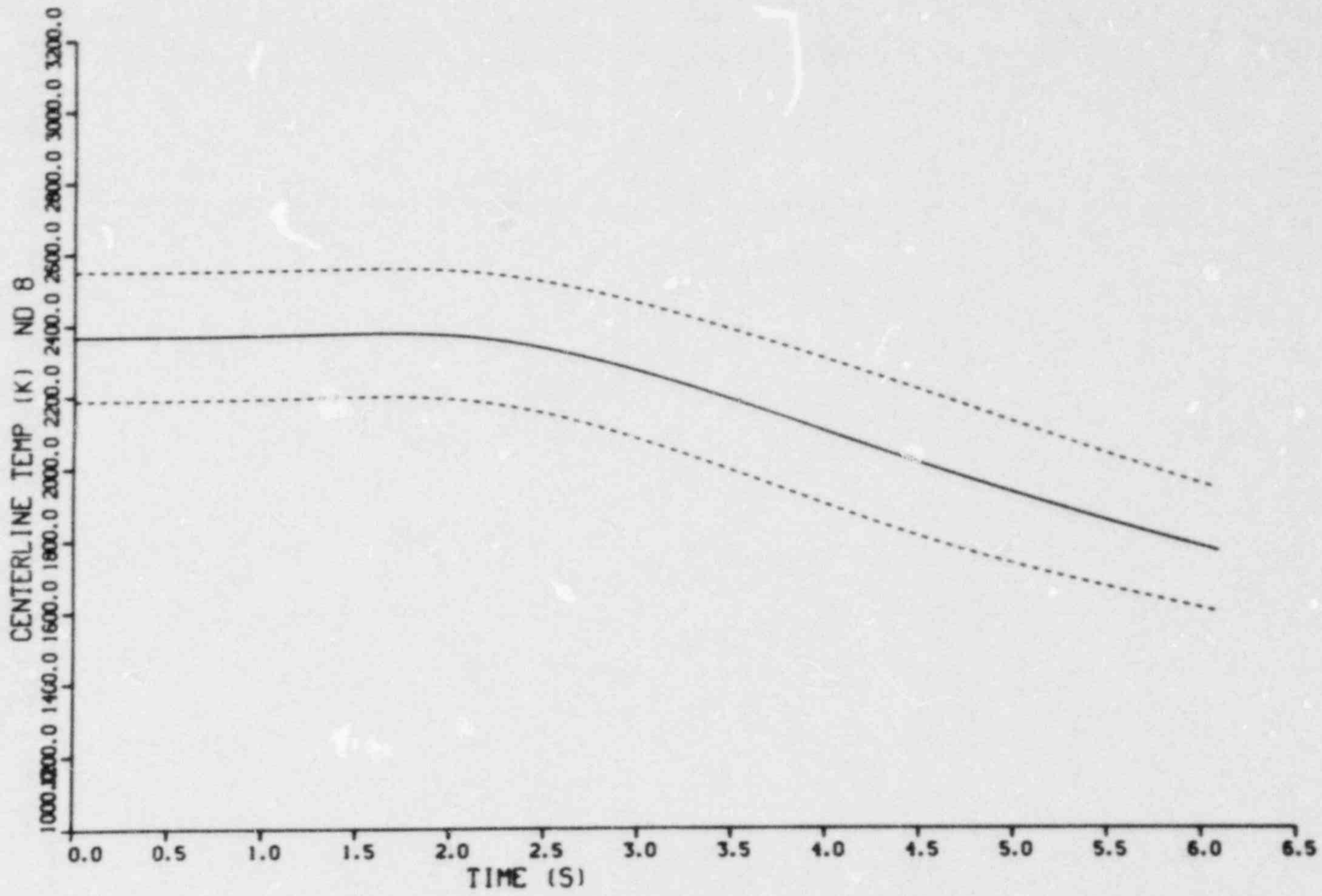

Figure 3. Average centerline temperature history and associated uncertainty for B&W locked rotor event.

was 89 cal/g, with a 2σ uncertainty of ±17 cal/g. About 90% of the statistical uncertainty in fuel centerline temperature was attributed to the uncertainty of the fuel thermal conductivity.

Comparing the centerline temperature calculations to the NRC damage limits, the best estimate calculations and the range of the 2σ bands about the best estimate calculations did not exceed the NRC limits. The same trend was noted for fuel enthalpy.

During this event, no cladding oxidation or hard pellet-cladding contact was noted, as illustrated by the cladding hoop stress history of the peak cladding temperature node shown in Figure 4. Thus, the cladding oxidation, stress, and strain limits were never in jeopardy of being exceeded.

Regarding the DNBR and cladding temperature limits, FRAP-T5 calculated that the fuel rod exceeded the physical DNBR limit of 1.0 in 4 of 32 computer runs used in this uncertainty study. The only uncertainty parameter that was used for all four runs was the rod power history. No other parameter was common to all four runs. Figure 5 illustrates the cladding surface temperature histories of each computer run. Four runs exhibited high temperatures when compared with the remaining 28 runs, due to the critical heat flux being exceeded. The best estimate calculated value of DNBR was 1.13, which already violated the NRC limit of 1.32.

Since FRAP-T5 is currently not programmed to calculate DNBR uncertainty, the probability of exceeding the DNBR of 1.0 was inferred from the cladding temperature uncertainty. At 1.4 s, the individual case either experienced DNB or never would after that time, and if the cladding temperature was greater than 645 K, the rod was in the DNB heat transfer mode. The probability density function (PDF) for cladding temperature at 1.4 s was used to estimate DNBR uncertainty. This procedure for inferring DNBR uncertainty is reasonable but very different from the more traditional vendor approach of using the CHF correlation uncertainty.

From this PDF shown in Figure 6, the probability of exceeding 640 K and experiencing DNB was about 20%. Since further information was not known about the characteristics of the DNBR uncertainty, no further description

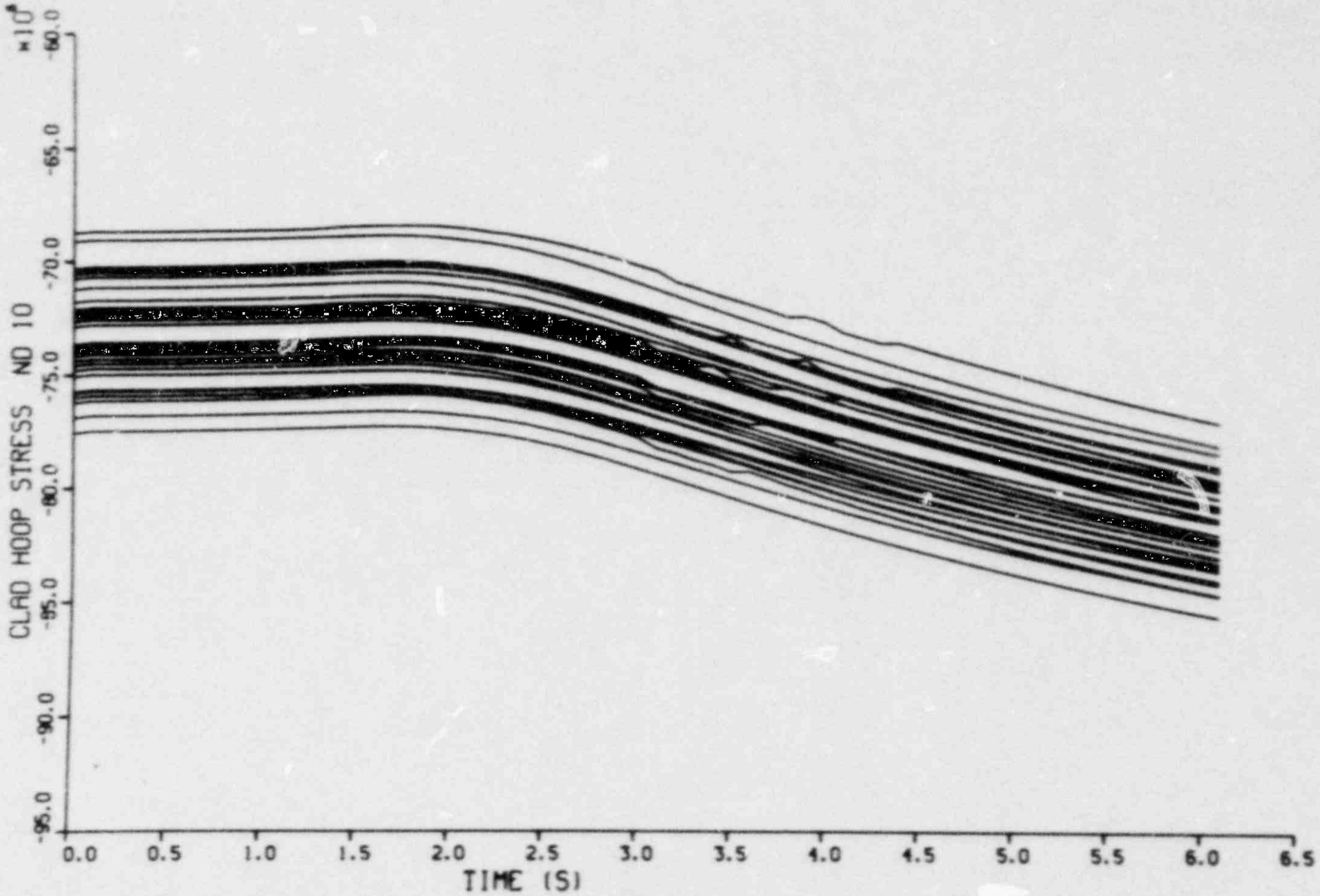

Figure 4. Individual cladding hoop stress histories for each B&W locked rotor calculation.

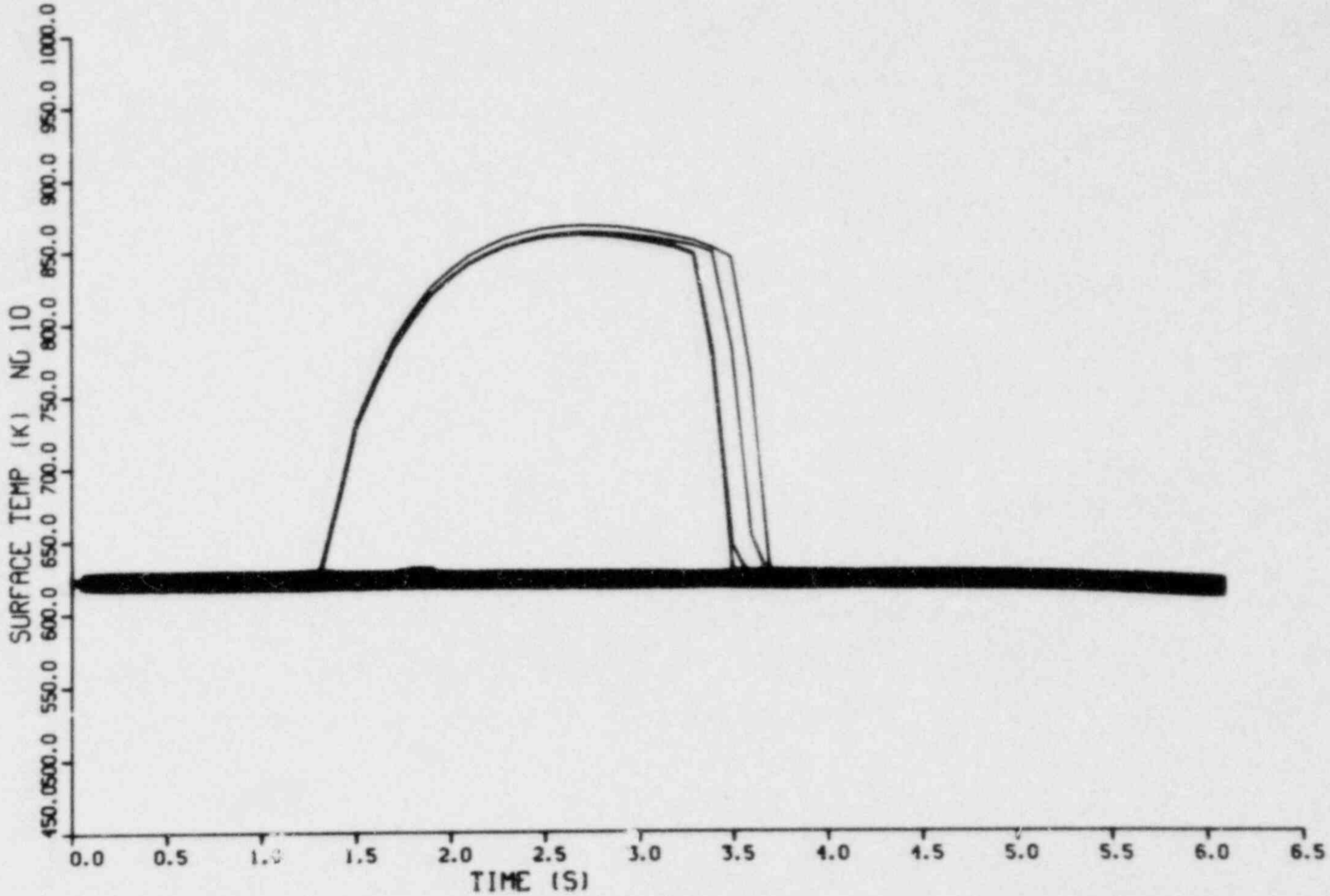

Figure 5. Individual surface temperature stress histories for each B&W locked rotor calculation.

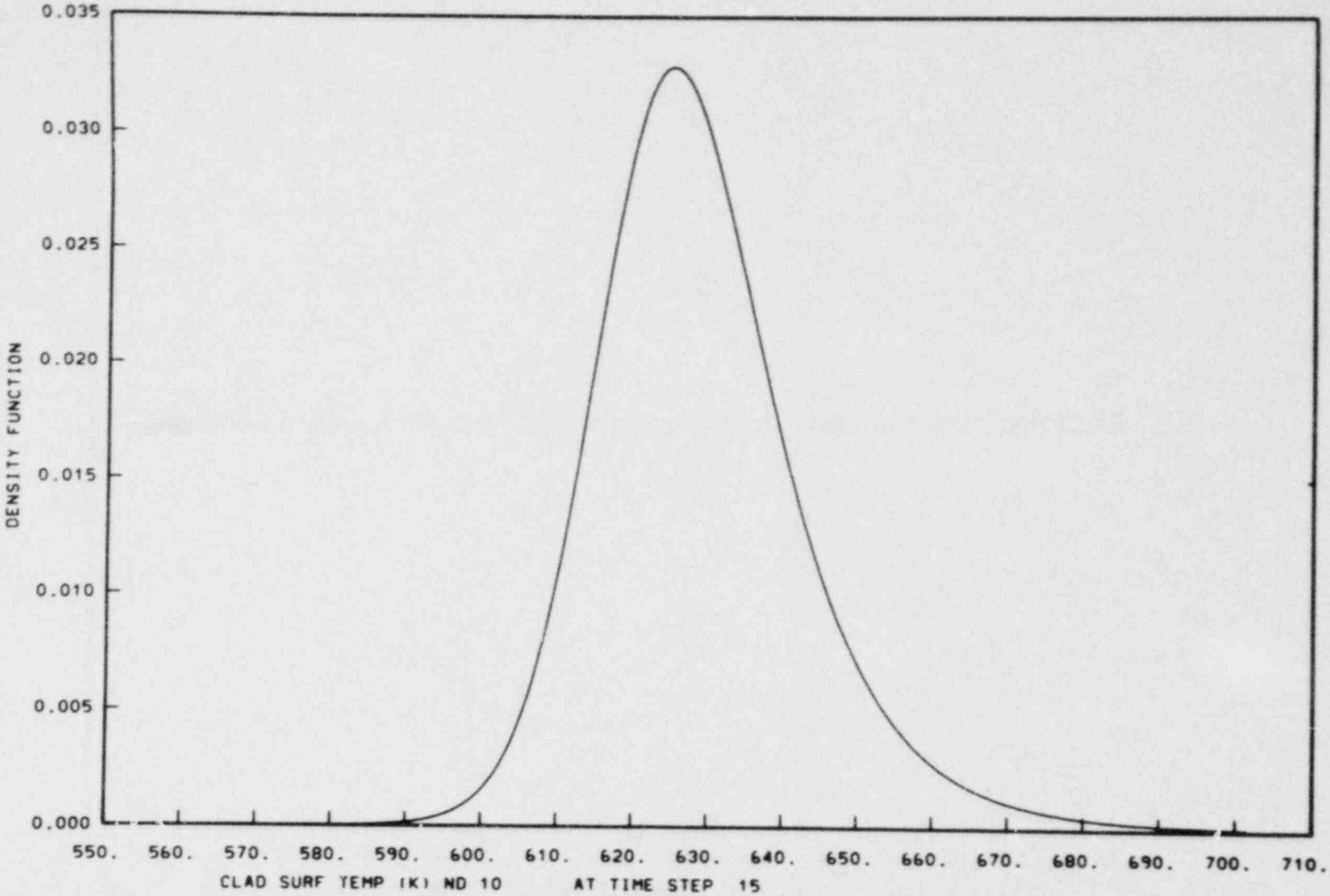

Figure 6. Probability density function for cladding surface temperature at 1.4 s, B&W locked rotor event.

could be made. Even though four runs indicated DNB to occur, the resulting calculated cladding surface temperatures remained well below any of the NRC cladding temperature limits. Also, no limits were exceeded even when comparing the 2σ uncertainty bound with the NRC limit.

Westinghouse Case

The best estimate calculation of the Westinghouse locked rotor event exceeded no NRC fuel damage limits, as compared with the B&W case that violated the DNBR limit of NRC in some cases. This result was not surprising since the Westinghouse case was less severe than the B&W case.

The history of the average fuel centerline temperature is shown in Figure 7, with the bounds of the 1σ standard deviations. Once again, about 90% of calculated fuel temperature uncertainty resulted from fuel conductivity uncertainty. At the peak centerline temperature of 2130 K, the 2σ uncertainty was 320 K. The corresponding average fuel enthalpy was 77 ± 13 cal/g. For both temperature and enthalpy, no NRC limit was exceeded by the 2σ bound.

For rod performance areas of cladding oxidation, stress, and strain, no limits were exceeded. Cladding temperatures always stayed far below the oxidation temperature threshold, producing no corrosion. Also, the hoop strain was always negative, thereby never producing positive permanent strains or high stress levels.

The best estimate calculation of the minimum DNBR was 1.35, versus the NRC limit of 1.30 for Westinghouse reactors. Since FRAP-T5 never calculated a minimum DNBR below 1.0, no accurate description of DNBR uncertainty can be stated, except that the uncertainty was small enough not to allow the critical heat flux to be exceeded. And even though the best estimate calculations did not exceed the NRC limit, the possibility of exceeding the NRC limit was within the bounds of the code uncertainty. One uncertainty study case had a minimum DNBR of 1.22, which exceeds the limit.

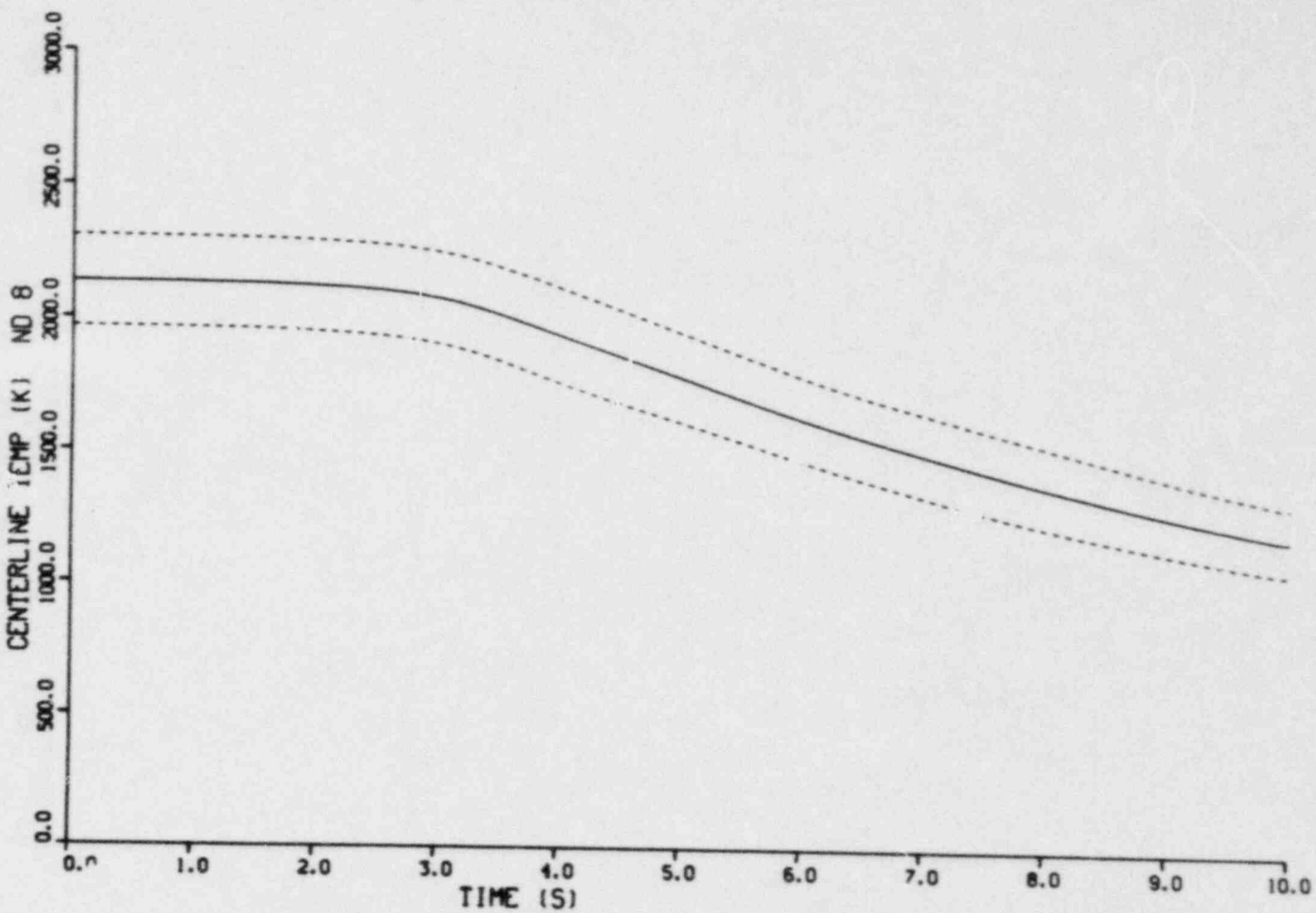

Figure 7. Average centerline temperature history and associated uncertainty for Westinghouse locked rotor event.

Rod Ejection

A rod ejection event was simulated for each of the PWR vendors; B&W, CE, and Westinghouse. The general performance characteristics of the event were a power excursion of 2 to 3 s duration due to an ejection of a control rod assembly. Other system parameters such as pressure and coolant mass flux remained essentially constant. The power history assumptions varied among vendors, as shown in Figures 8, 9, and 10, which illustrate histories of rod and system performance parameters.

Results of the B&W rod ejection event will be discussed first, followed by the CE and Westinghouse events, respectively.

Babcock and Wilcox Case

The duration of the B&W rod ejection event was about 3 s. The peak power was generated within the first quarter of a second, and peak fuel centerline temperature was calculated to occur at about 1.5 s. The average centerline temperature history of the uncertainty cases is shown in Figure 11 for the peak power elevation (Node 8) along with the one standard deviation uncertainty range. A noteworthy result is the 2σ upper bound of the calculation never exceeded the NRC fuel melt limit of 3078 K. The same trend was observed for fuel enthalpy, which was 106 cal/g at 1.5 s with a ±16 cal/g value for two standard deviations. The applicable NRC limit is 200 cal/g.

At the peak power elevation (Node 8), where the maximum centerline temperature occurred, the cladding surface heat flux never exceeded the critical heat flux. At a higher elevation, the centerline temperature was lower but the critical heat flux was also lower. For the best estimate calculation at this elevation, the minimum DNBR was 1.17, which violates the NRC limit of 1.32, but not the physical limit of 1.0. However, examining the cladding surface temperature histories for the uncertainty study cases (Figure 12), nearly half of the cases exceeded the physical DNBR limit of 1.0. Thus, when the best estimate DNBR value is 1.17, the possibility of exceeding a DNBR of 1.0 is highly probable.

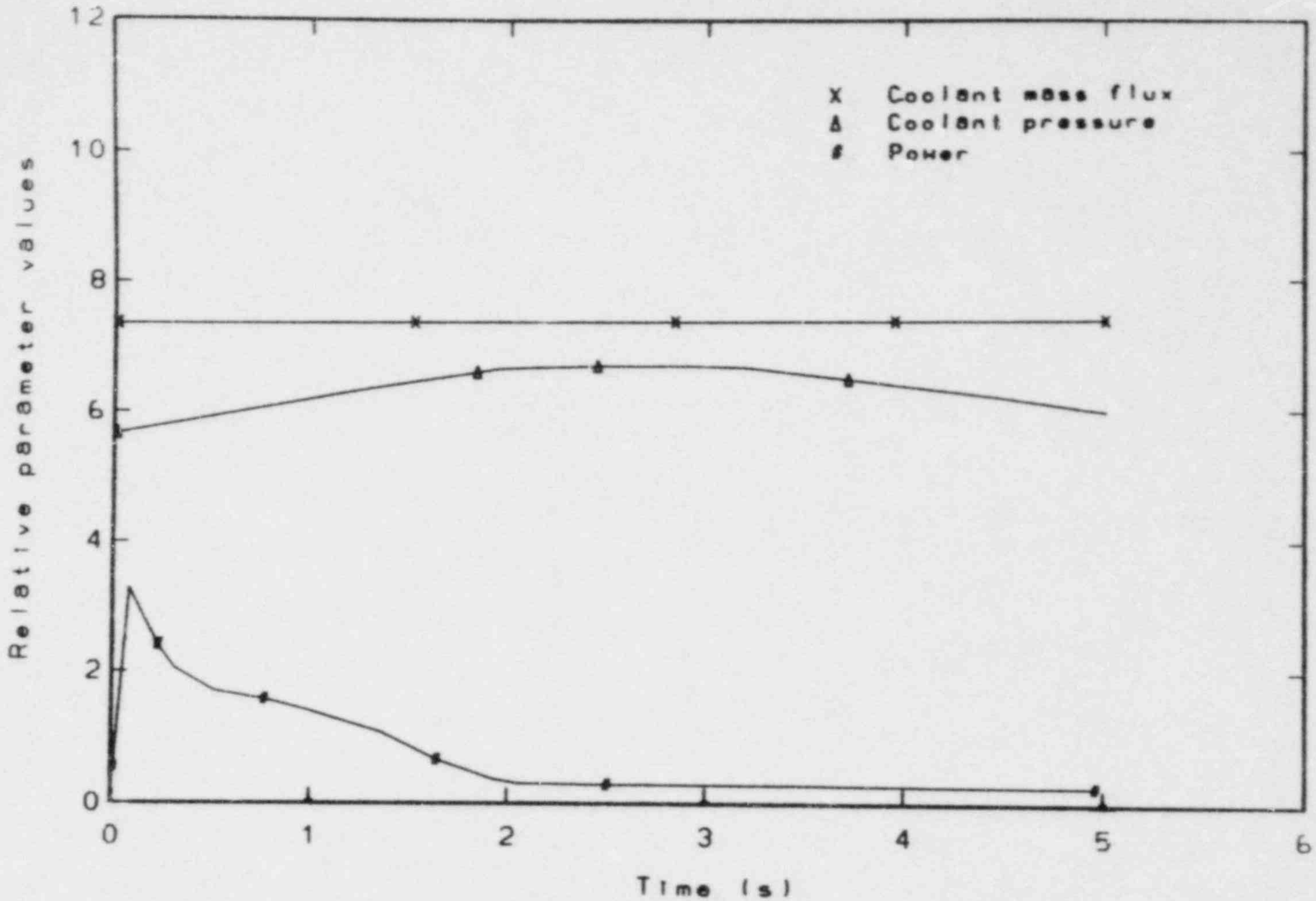

Figure 8. Operating history for B&W rod ejection event.

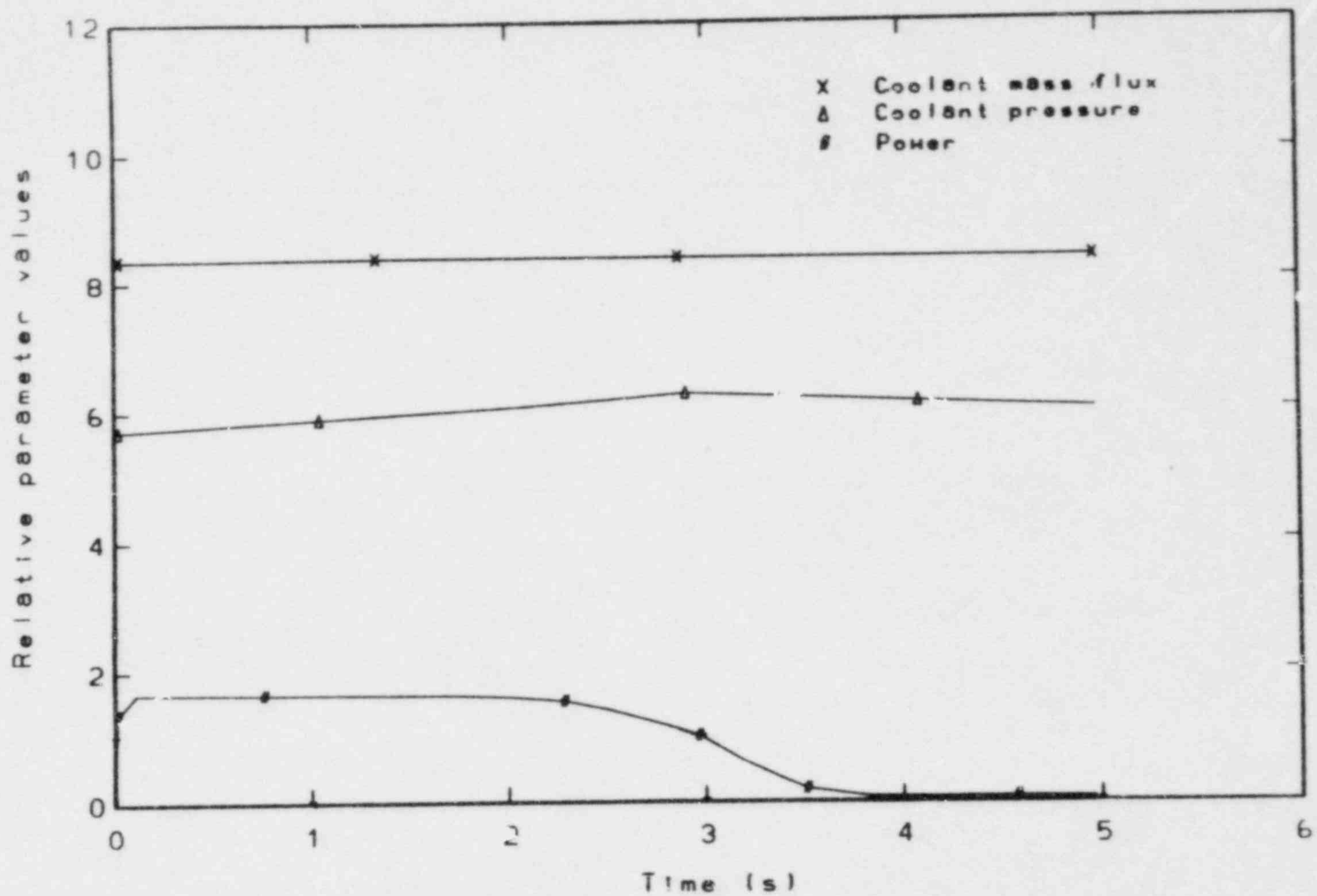

Figure 9. Operating history for CE rod ejection event.

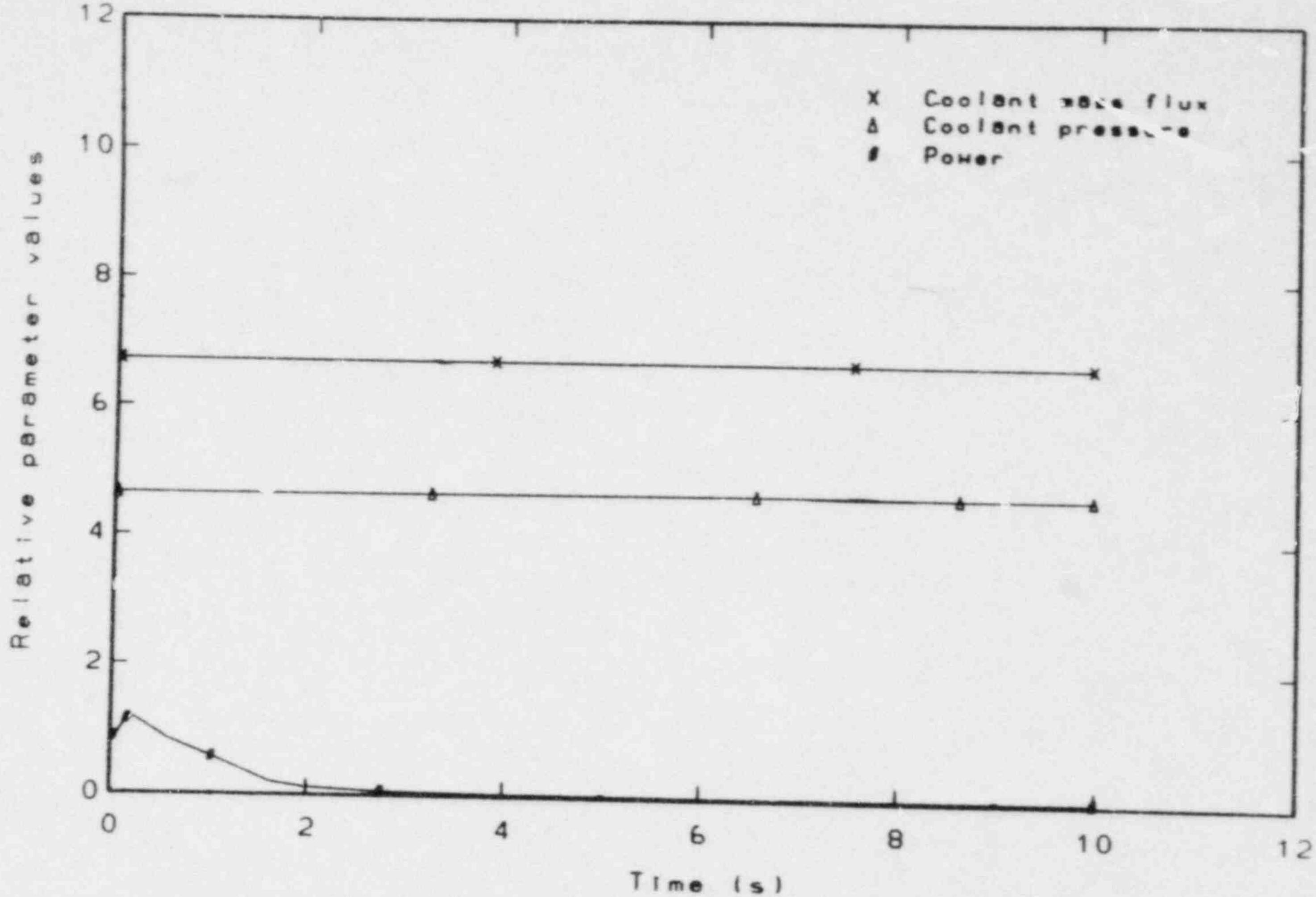

Figure 10. Operating history for Westinghouse rod ejection event.

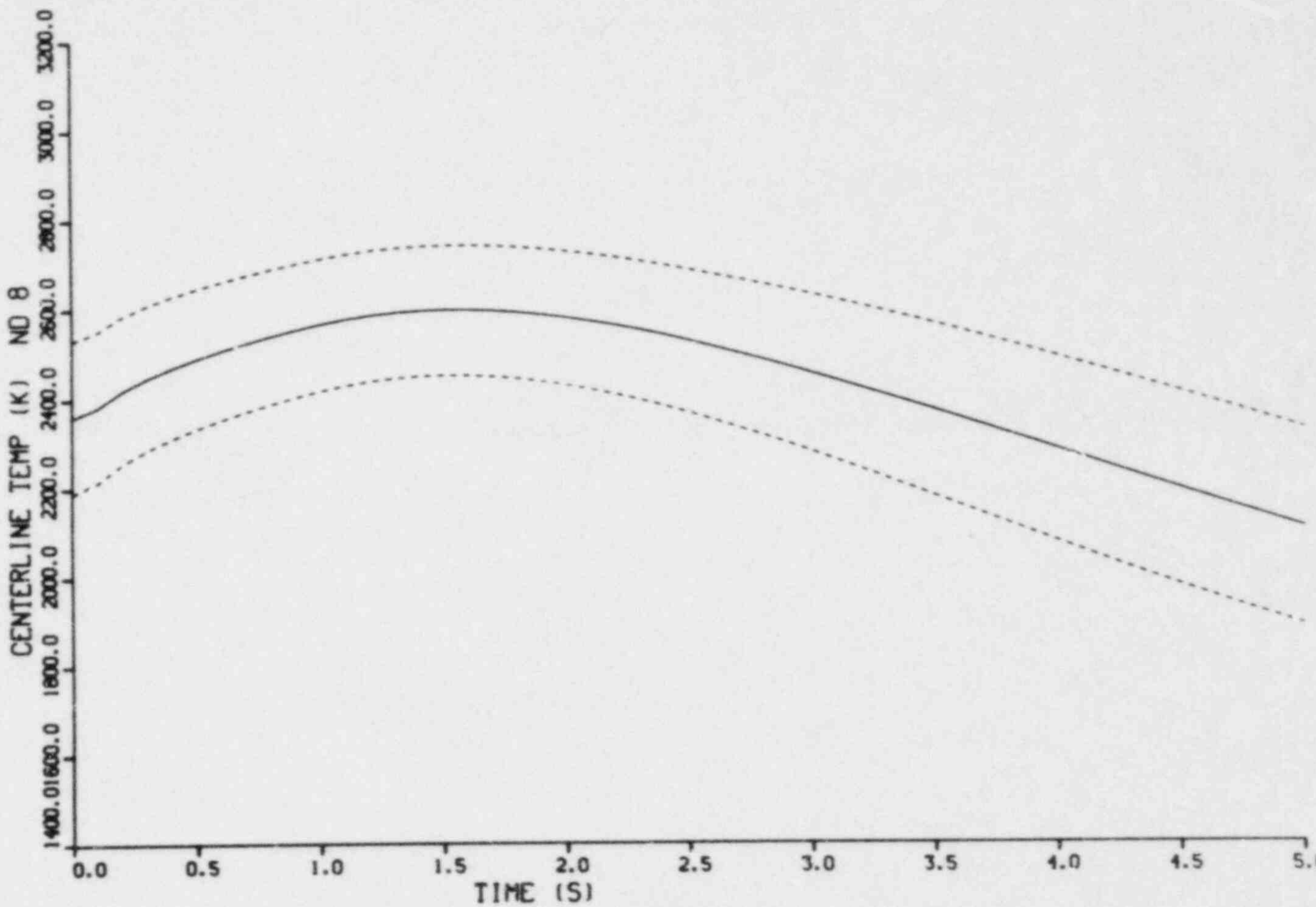

Figure 11. Average fuel centerline temperature history and associated uncertainty for B&W rod ejection event.

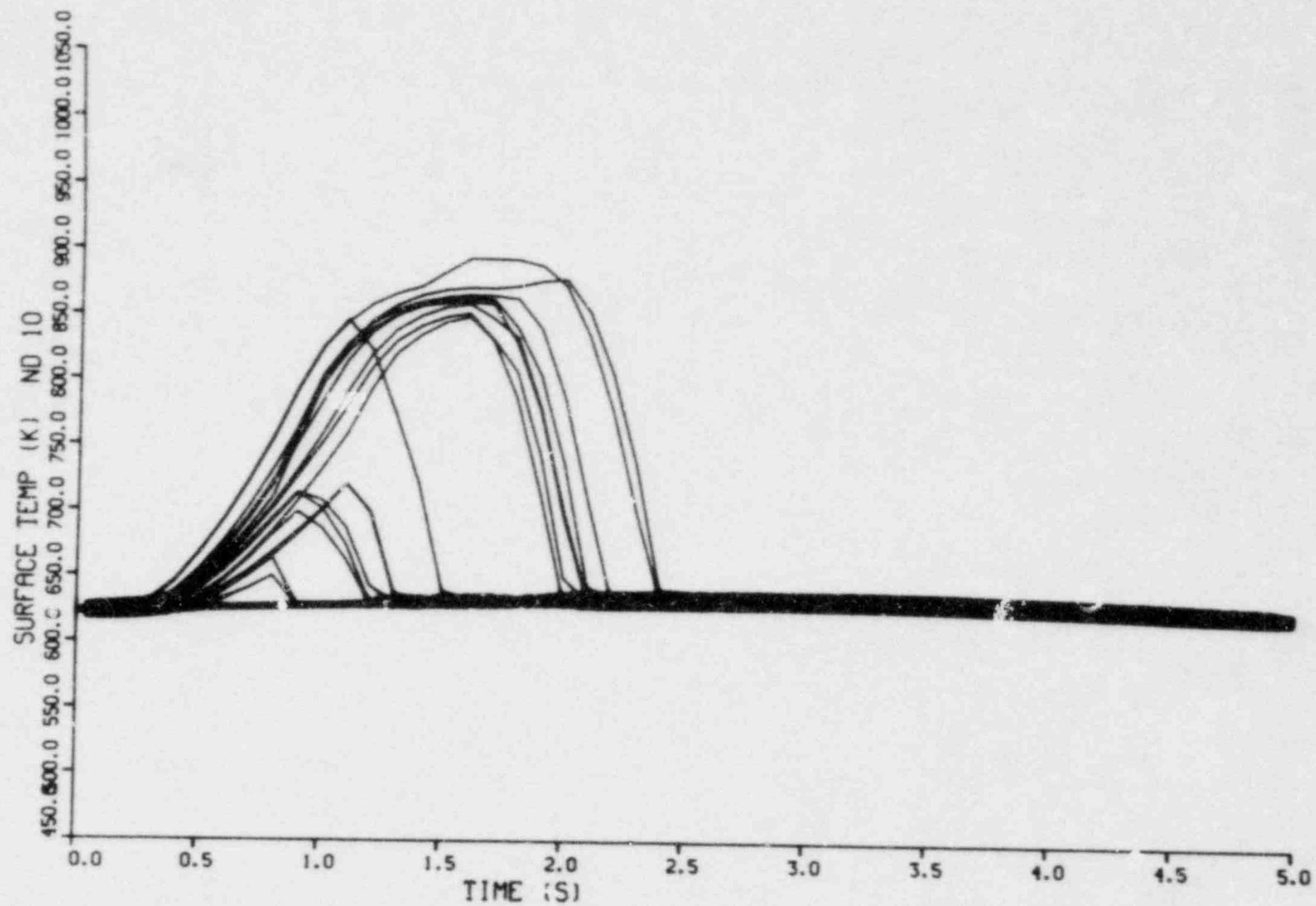

Figure 12. Individual cladding surface temperature histories for each B&W rod ejection calculation.

Even though a large number of cases predicted the rod to enter the DNB mode, the peak cladding surface temperature attained by the worst case was 900 K, which is far below the α - β transition temperature of 1189 K. Also, no cladding oxidation was calculated to occur during the short period of deficient cooling (less than 2 s). However, the permanent hoop strain histories in Figure 13 indicate that two of the 32 uncertainty study cases did predict cladding collapse. Cladding stress histories (Figure 14) show that stress levels for the collapsed rods did increase significantly but always remained low, peaking at 70 MPa, which are well below the NRC limit. Both of these runs utilized rod power, fuel conductivity, gas conductivity, and pellet diameter uncertainty parameters. No other parameters were used commonly by both runs. Lastly, no cladding failure was predicted to occur for any of the uncertainty study cases.

Combustion Engineering Case

The power history submitted by CE for the rod ejection event was less peaked than the B&W power history, but the CE event occurred over a longer period of time. The maximum fuel centerline temperature occurred at 2.6 s, as shown in Figure 15 for the peak power node. The fuel rod was never in jeopardy of exceeding the NRC incipient fuel melt limit within the two standard deviation range. The corresponding fuel enthalpy at 2.6 s was 91 ± 16 cal/g.

The fuel rod was never calculated to experience DNB. The best estimate calculation produced a minimum DNBR of 1.8. As a result, the cladding temperature remained low (Figure 16), as well as cladding stresses and strains.

No NRC limits were exceeded within the two standard deviation bounds of the calculation.

Westinghouse Case

The transient power history of the Westinghouse rod ejection event was a combination of the B&W and CE histories. As a result, rod behavior of the Westinghouse case was very uneventful. Namely, the power history peaked

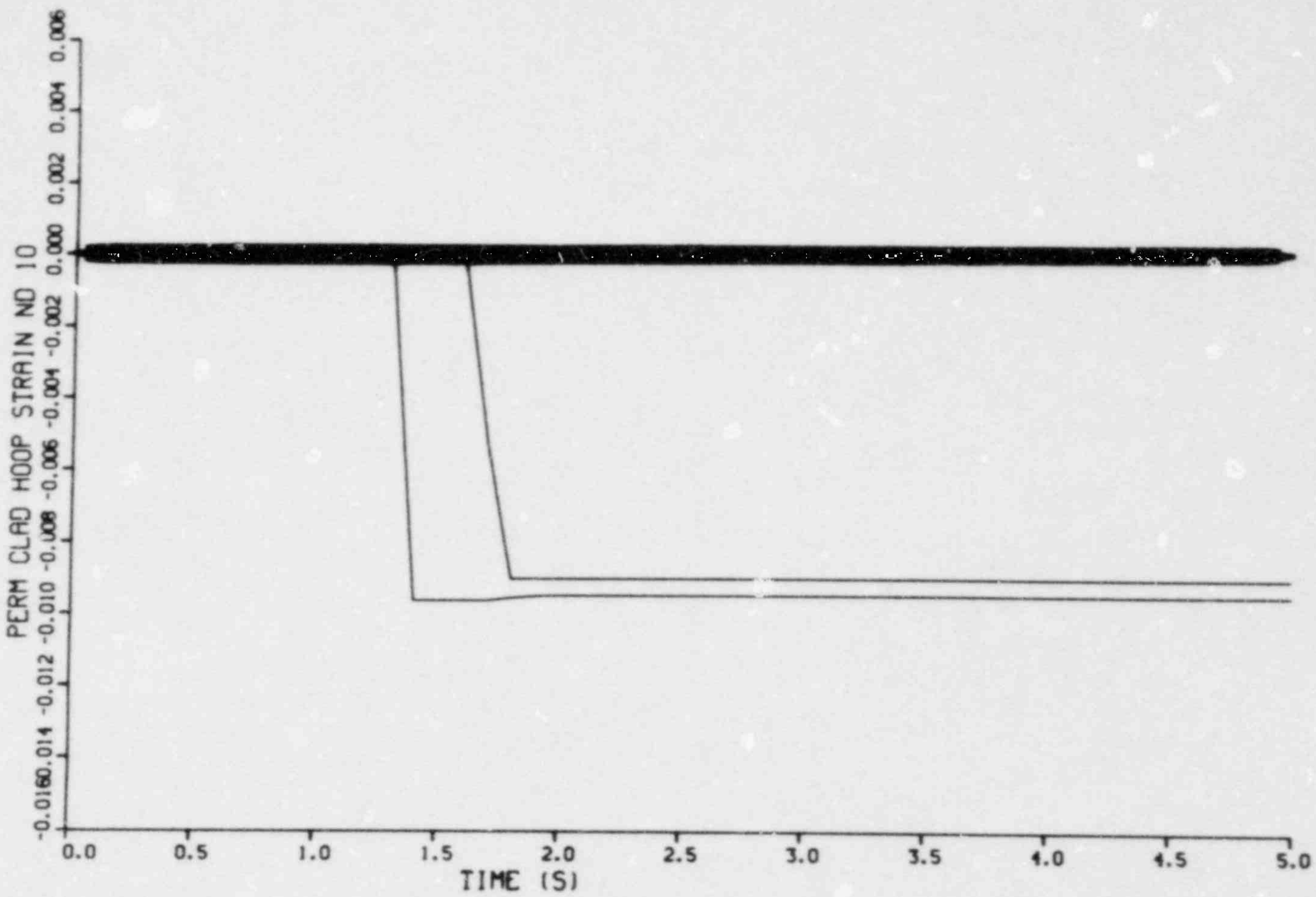

Figure 13. Individual permanent cladding hoop strain histories for each B&W rod ejection calculation.

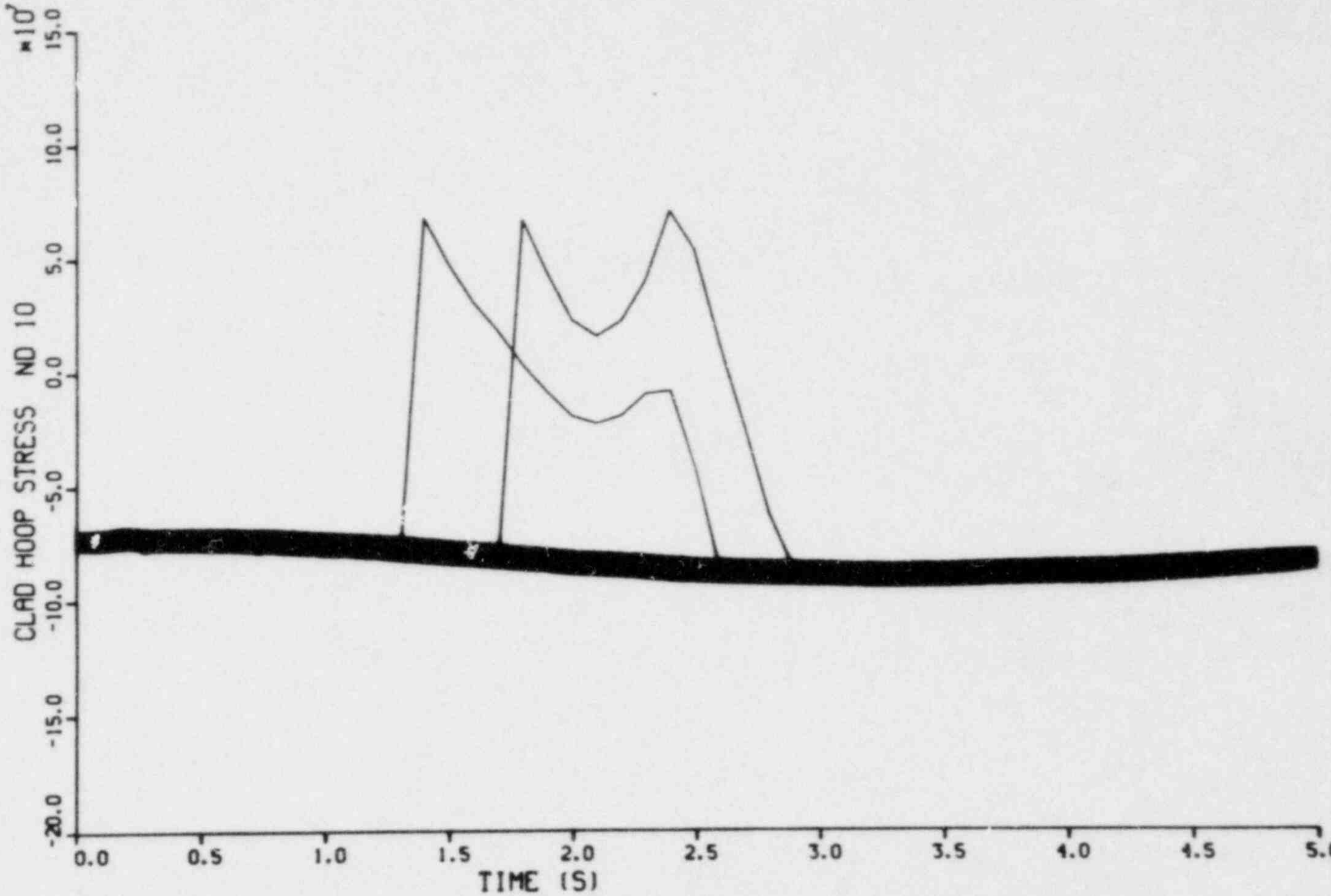

Figure 14. Individual permanent cladding hoop stress histories for each B&W rod ejection calculation.

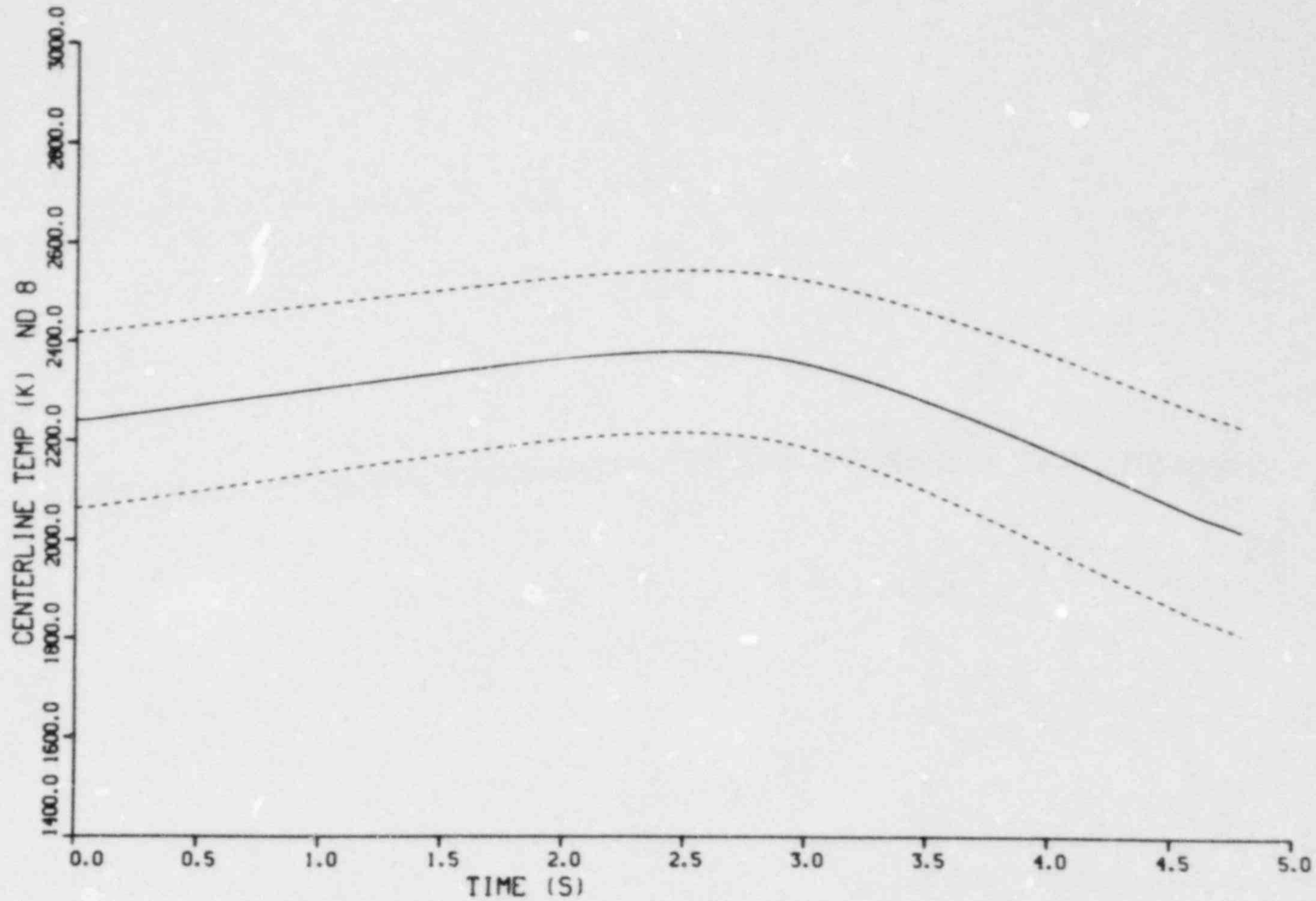

Figure 15. Average centerline temperature history and associated uncertainty for CE rod ejection event.

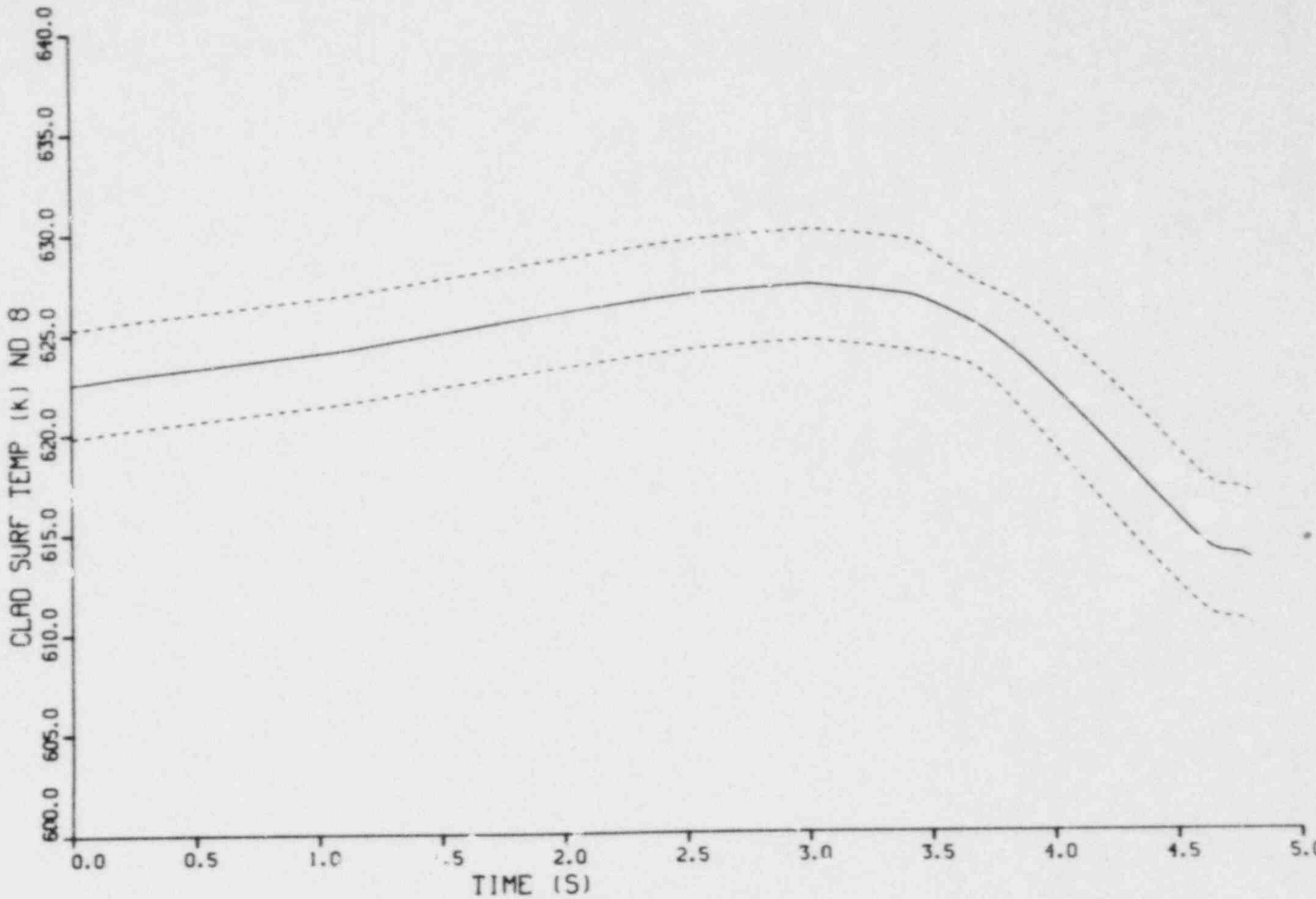

Figure 16. Average cladding surface temperature history and associated uncertainty for CE rod ejection event.

early, like the B&W event, but the absolute value of the peak was very low, like the CE event. The fuel centerline temperature history illustrated in Figure 17 shows a maximum value of 2130 K at about 1 s. Then, the temperature continually decreased and never approached the NRC limit, even within the two standard deviation range. The corresponding peak fuel enthalpy at 1 s was 79 ± 17 cal/g.

Other rod behavior indicators showed similar trends of acceptable performance. The cladding surface temperature history (Figure 18) showed no rise in temperature during the event. The minimum DNBR of the best estimate calculation was 2.5. Low cladding stresses and strains and no cladding oxidation were also observed.

Steam Line Break

The only steam line break event submitted for this study was provided by B&W. The general characteristics of the event were a decrease of coolant pressure and mass flux at the initiation of the event, followed by a decrease of rod power starting at 1 s. The relative histories of rod power, coolant conditions, and rod temperature are shown in Figure 19.

The maximum centerline temperature was calculated to occur at the peak power node (Node 8). The mean temperature history with one standard deviation bounds is illustrated in Figure 20. The NRC incipient fuel melt limit was not exceeded by the two sigma bounds of the FRAP-T5 calculation. The corresponding peak fuel enthalpy during the first second of the event was 97 cal/g with two standard deviations of 22 cal/g.

The minimum DNBR noted from the best estimate calculation for Node 8 was 1.77. None of the uncertainty study cases calculated minimum DNBR less than 1.0. However, near the top of the rod (Node 13), the best estimate calculated DNBR attained a minimum of 1.02, which exceeds the NRC limit of 1.32. Eleven of the uncertainty study cases went into DNB (DNBR $\leq$ 1.0) for up to 4.5 s, as shown in Figure 21, which illustrates Node 13 cladding surface temperature histories for the 32 computer runs. The highest cladding temperature attained was about 800 K. Slight disruptions of cladding

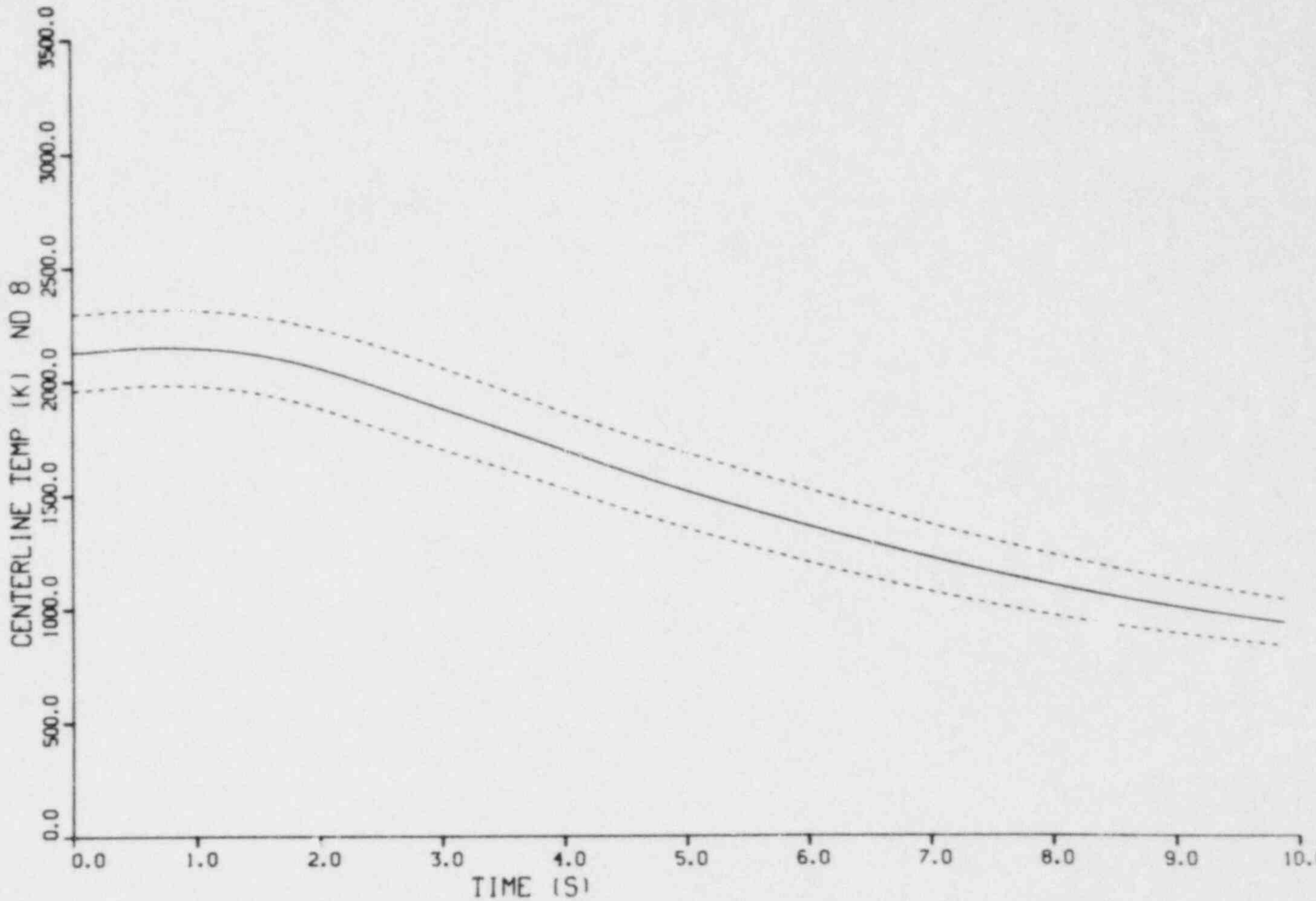

Figure 17. Average centerline temperature history and associated uncertainty for Westinghouse rod ejection event.

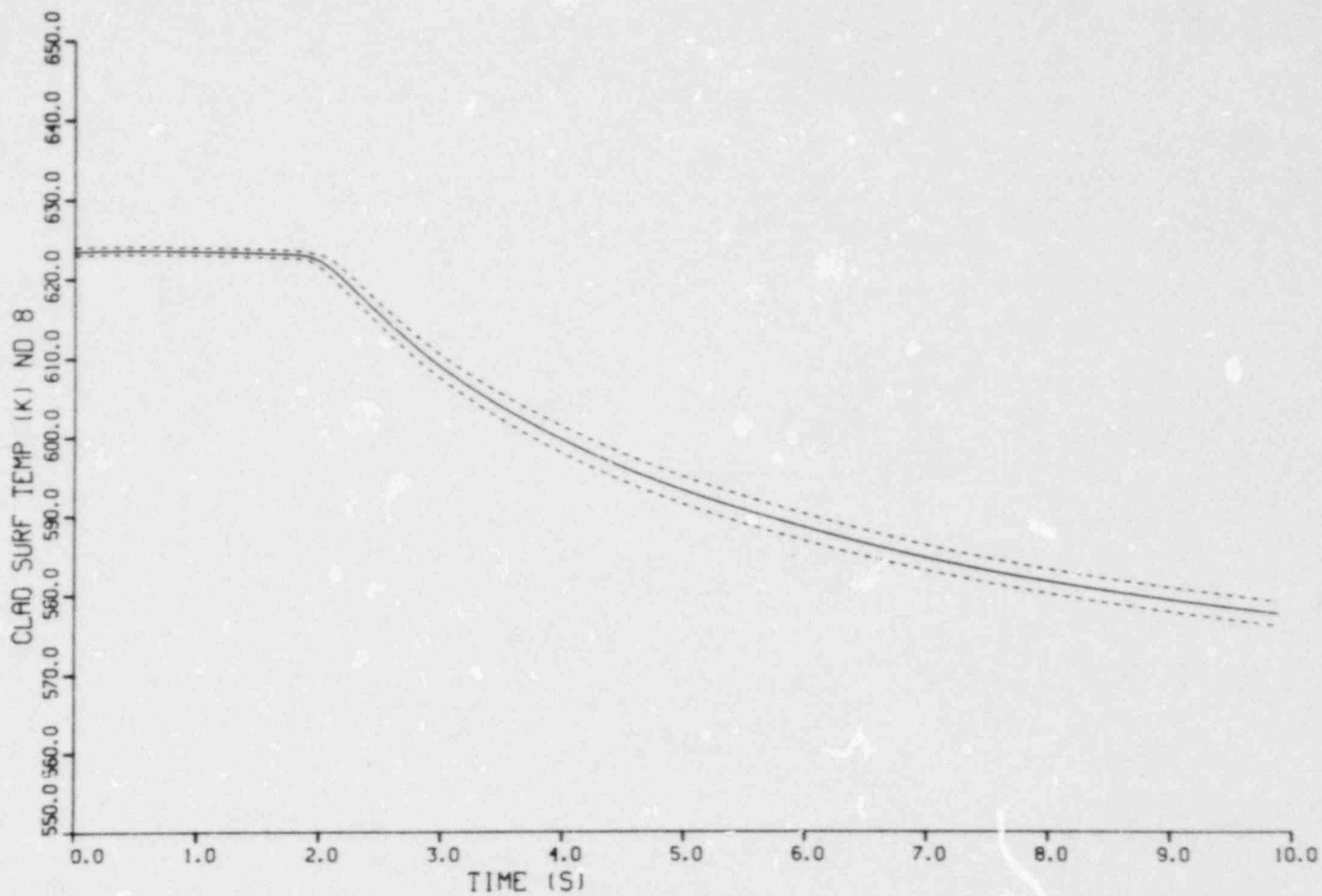

Figure 18. Average cladding surface temperature history and associated uncertainty for Westinghouse rod ejection event.

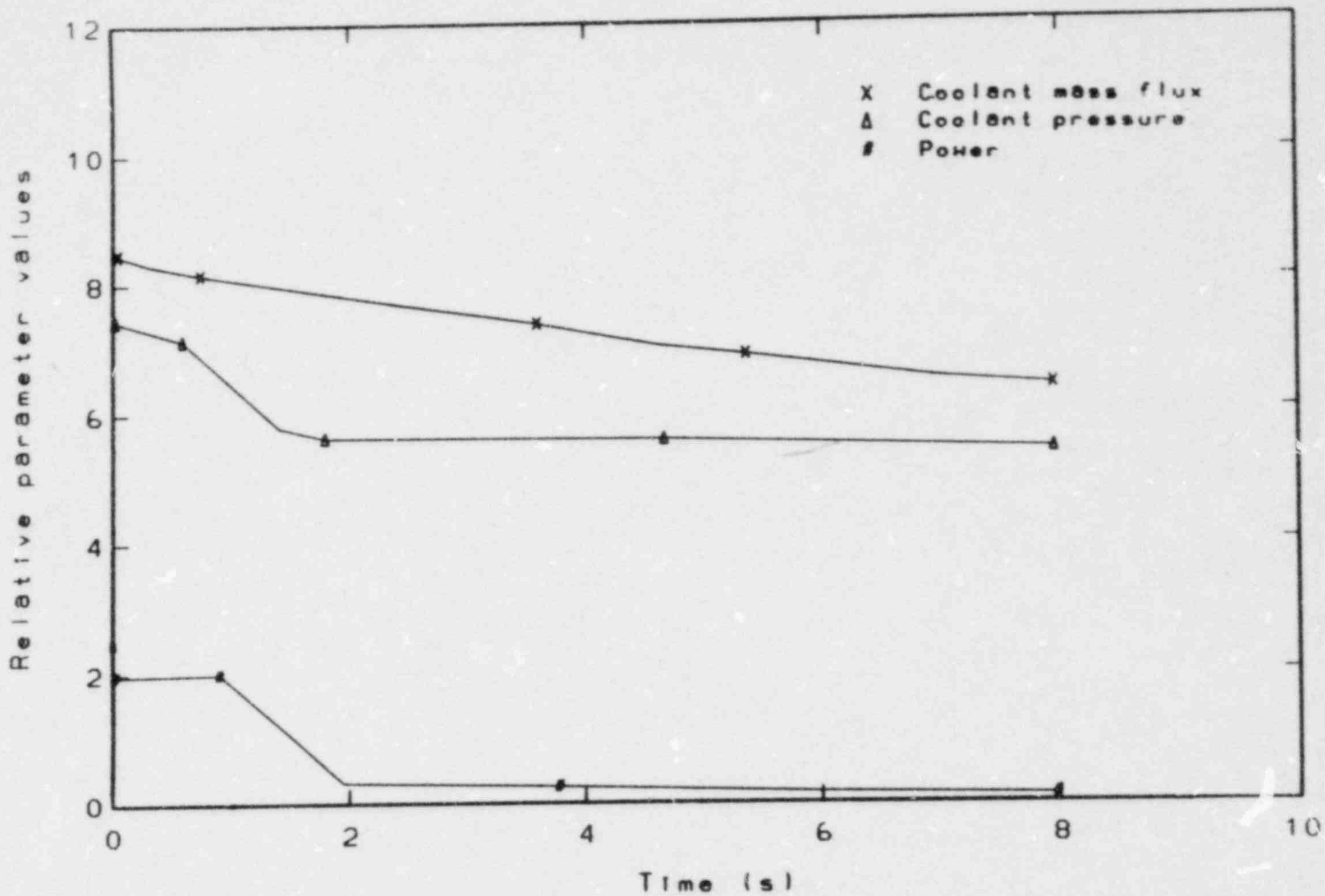

Figure 19. Operating history for B&W steam line break event.

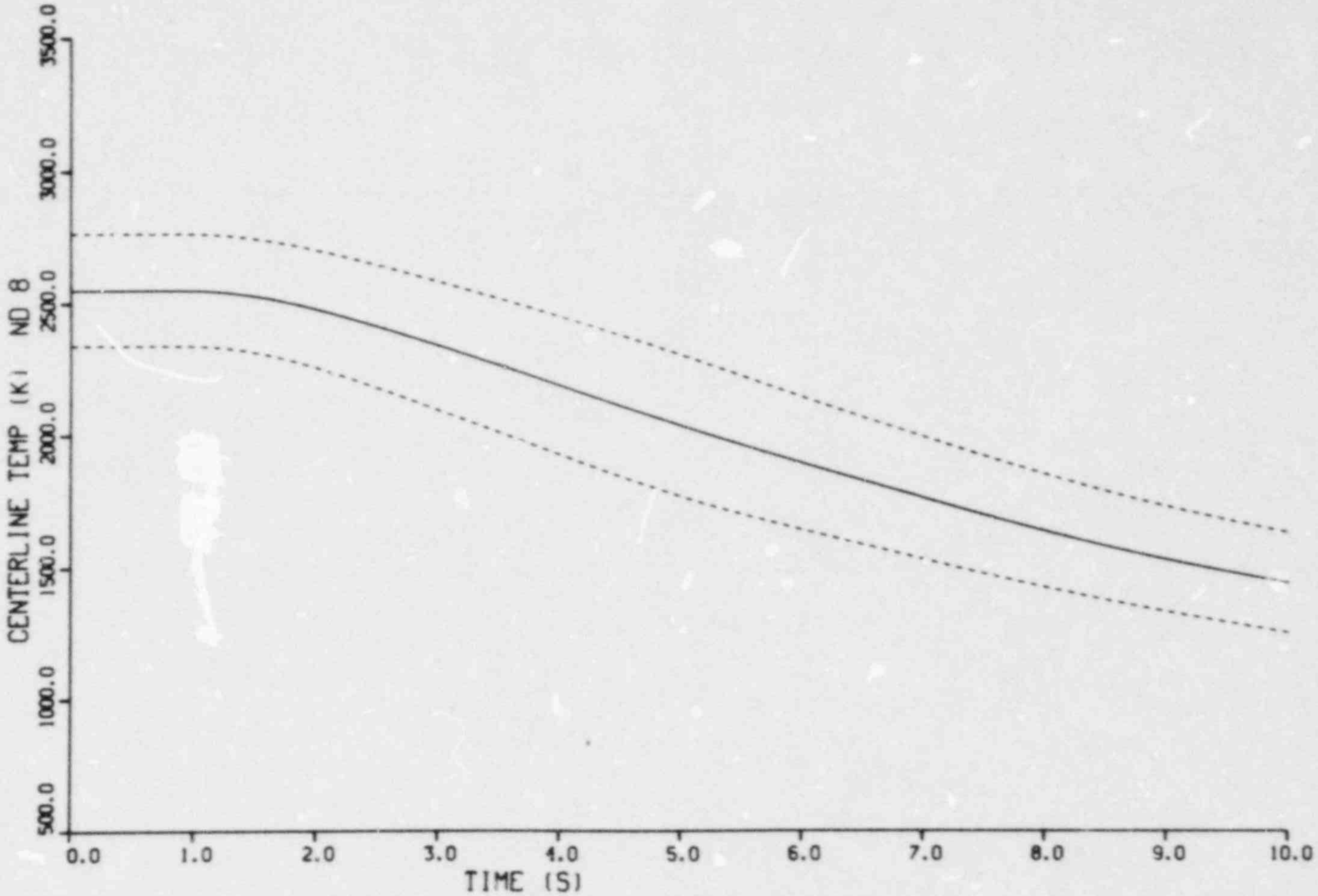

Figure 20. Average centerline temperature history and associated uncertainty for B&W steam line break event.

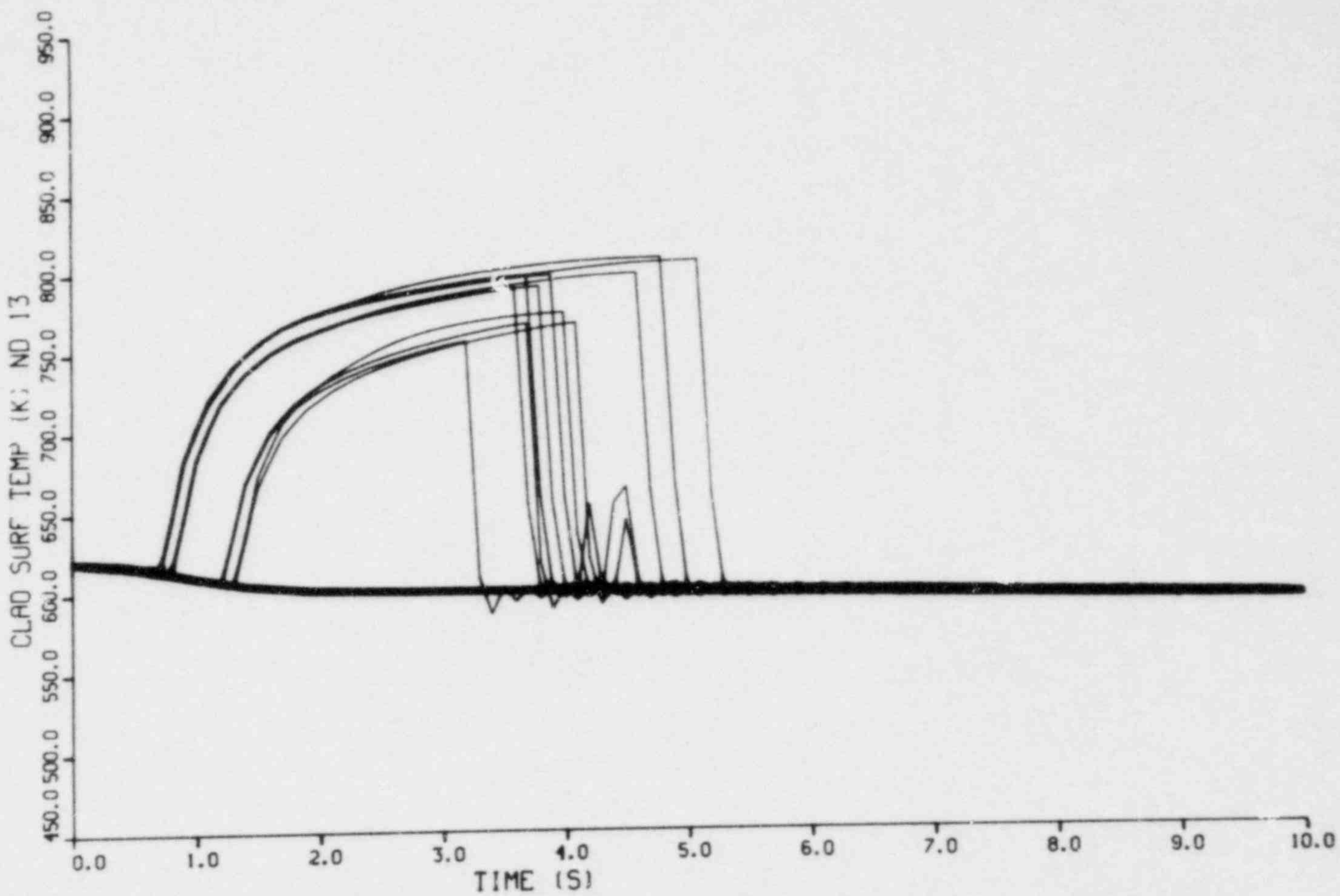

Figure 21. Individual cladding surface temperature histories for each B&W steam line break calculation.

hoop stress were noted during the event due to the elevated temperature, but hoop stresses always remained slightly negative. None of the uncertainty study cases predicted cladding failure or even cladding collapse. Because cladding surface temperatures remained below 800 K, even when experiencing film boiling, no cladding corrosion occurred.

Loss of Flow

Two vendors (CE and Westinghouse) submitted cases for the loss-of-flow event. Both submittals had the same basic event characteristics; a steady reduction of coolant flow, an essentially constant coolant pressure, and a reactor scram that occurred about 2 to 3 s after initiation of the event. The rod power, coolant flow, and pressure histories are illustrated in Figures 22 and 23.

Results of the CE case will be discussed first, followed by the Westinghouse case.

Combustion Engineering Case

Overall, results indicated that the CE loss-of-flow case was a very benign event. No NRC limits were exceeded or close to being exceeded. The fuel centerline temperature history (Figure 24) corresponds to the peak power rods. The 2σ bounds of the temperature calculation were well below the NRC limit for incipient fuel melting. The fuel enthalpy corresponding to the calculated peak centerline temperature was 84 cal/g, with a 2σ uncertainty of ±14 cal/g; again, well below the NRC limit.

Cladding surface temperature histories of the 32 computer runs used in this uncertainty study are shown in Figure 25. No case entered film boiling. This trend was not surprising since the minimum DNBR of the best estimate calculation was 2.4.

Because fuel and cladding always remained cool during the event, no high cladding stress conditions were encountered due to pellet-cladding interaction; thereby, never exceeding the corresponding NRC overstress/overstrain criteria. Also, cladding corrosion was never calculated to occur, thus avoiding the NRC oxidation limit.

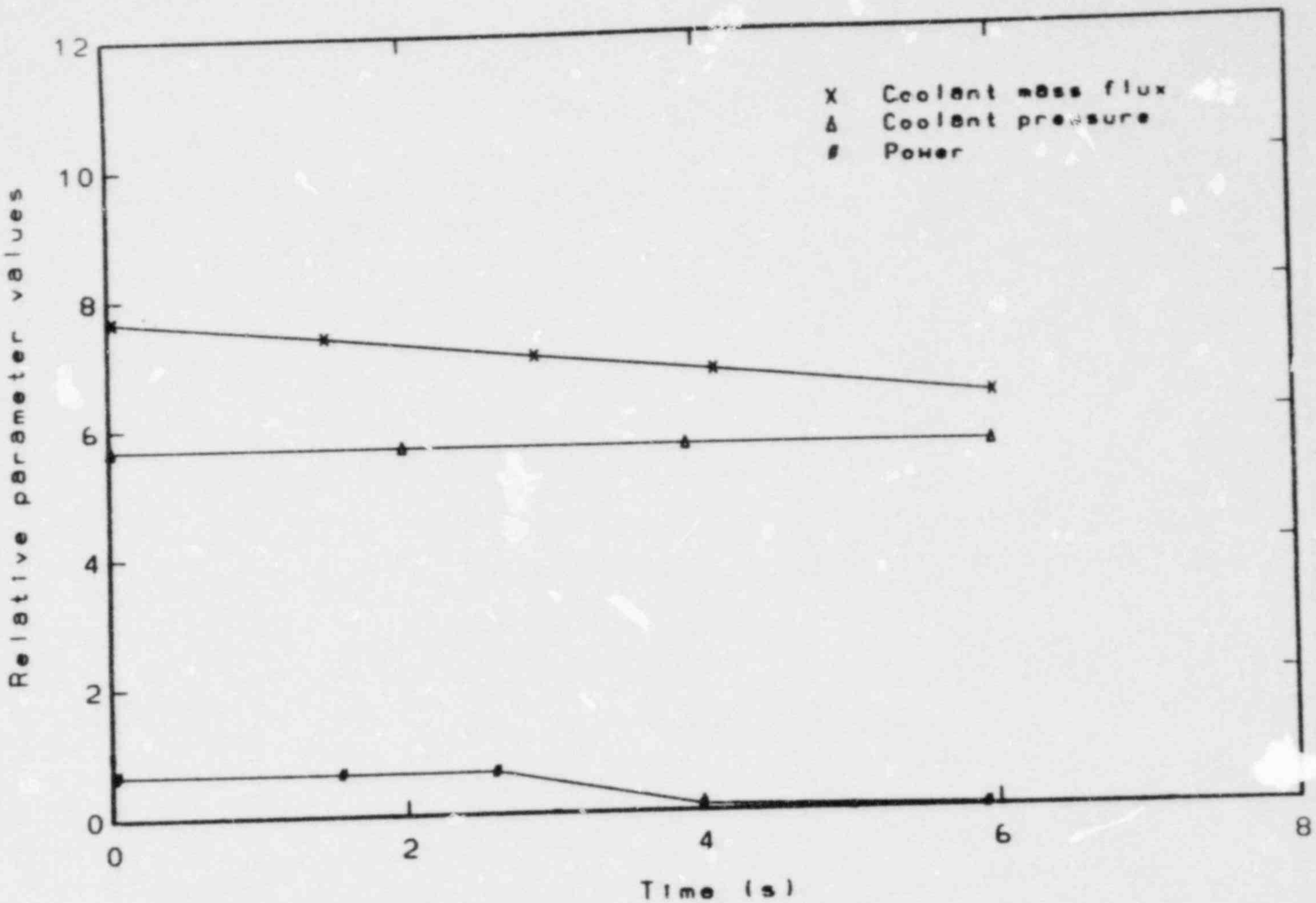

Figure 22. Operating history for CE loss-of-flow event.

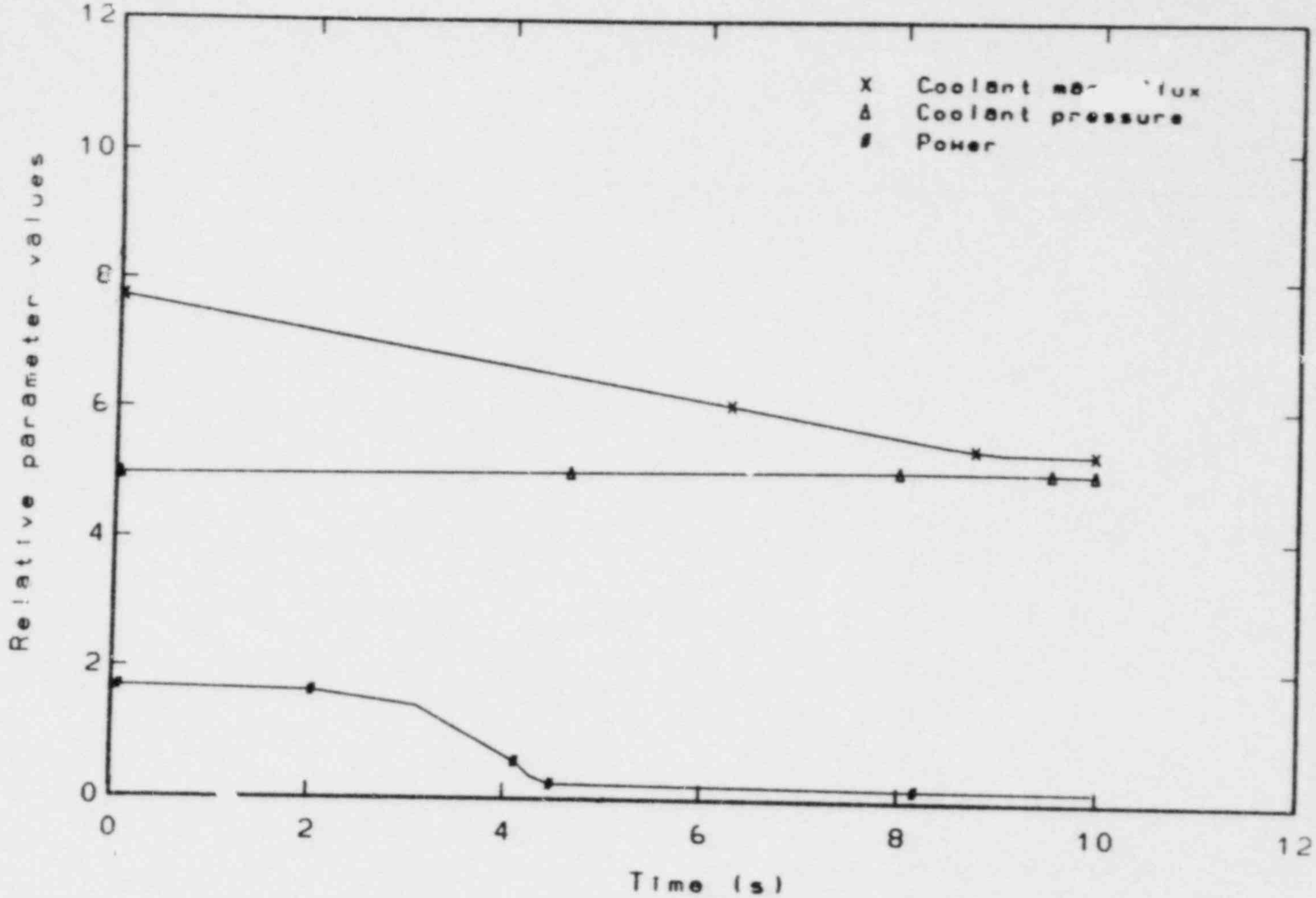

Figure 23. Operating history for Westinghouse loss-of-flow event.

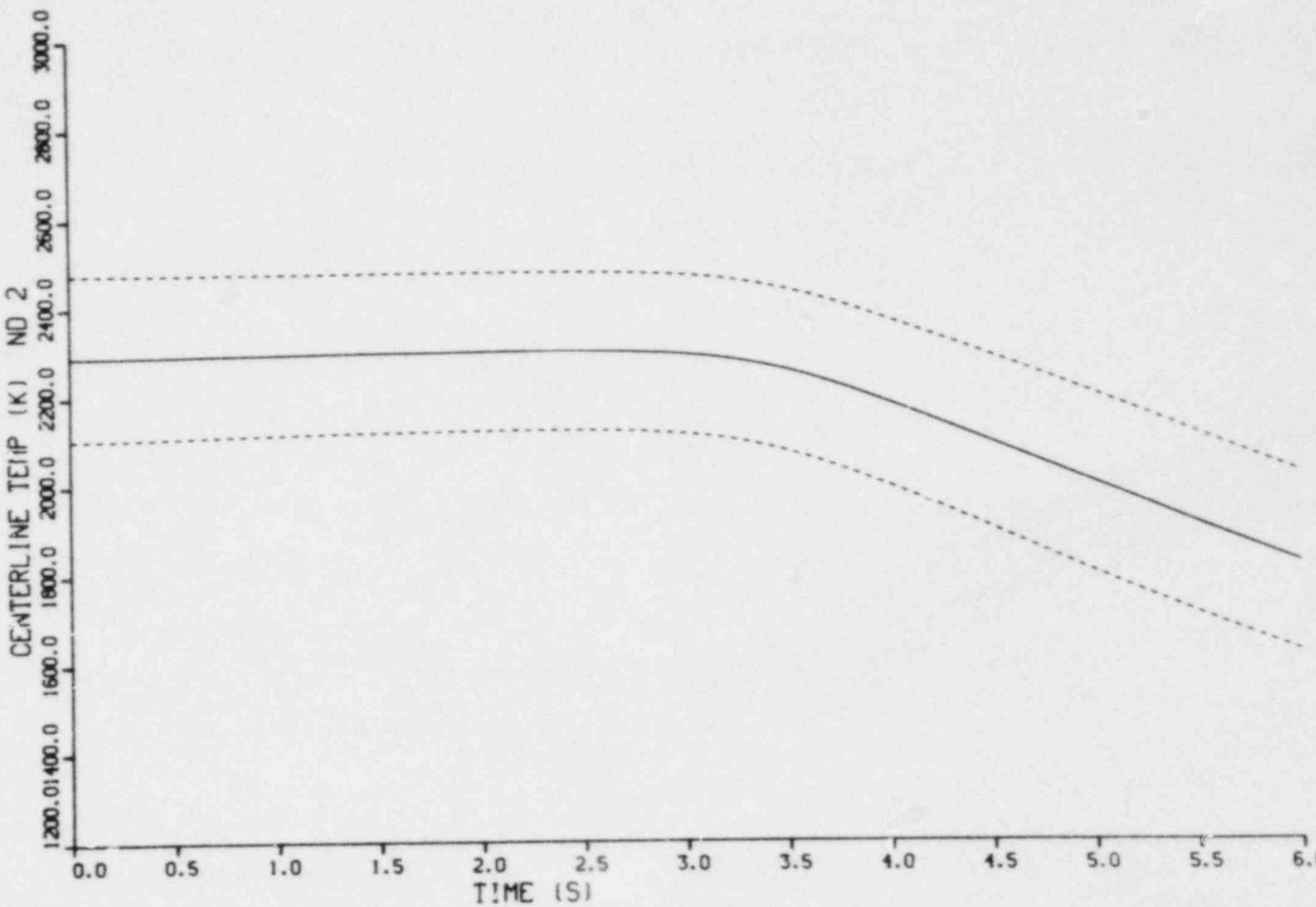

Figure 24. Average centerline temperature history and associated uncertainty for CE loss-of-flow event.

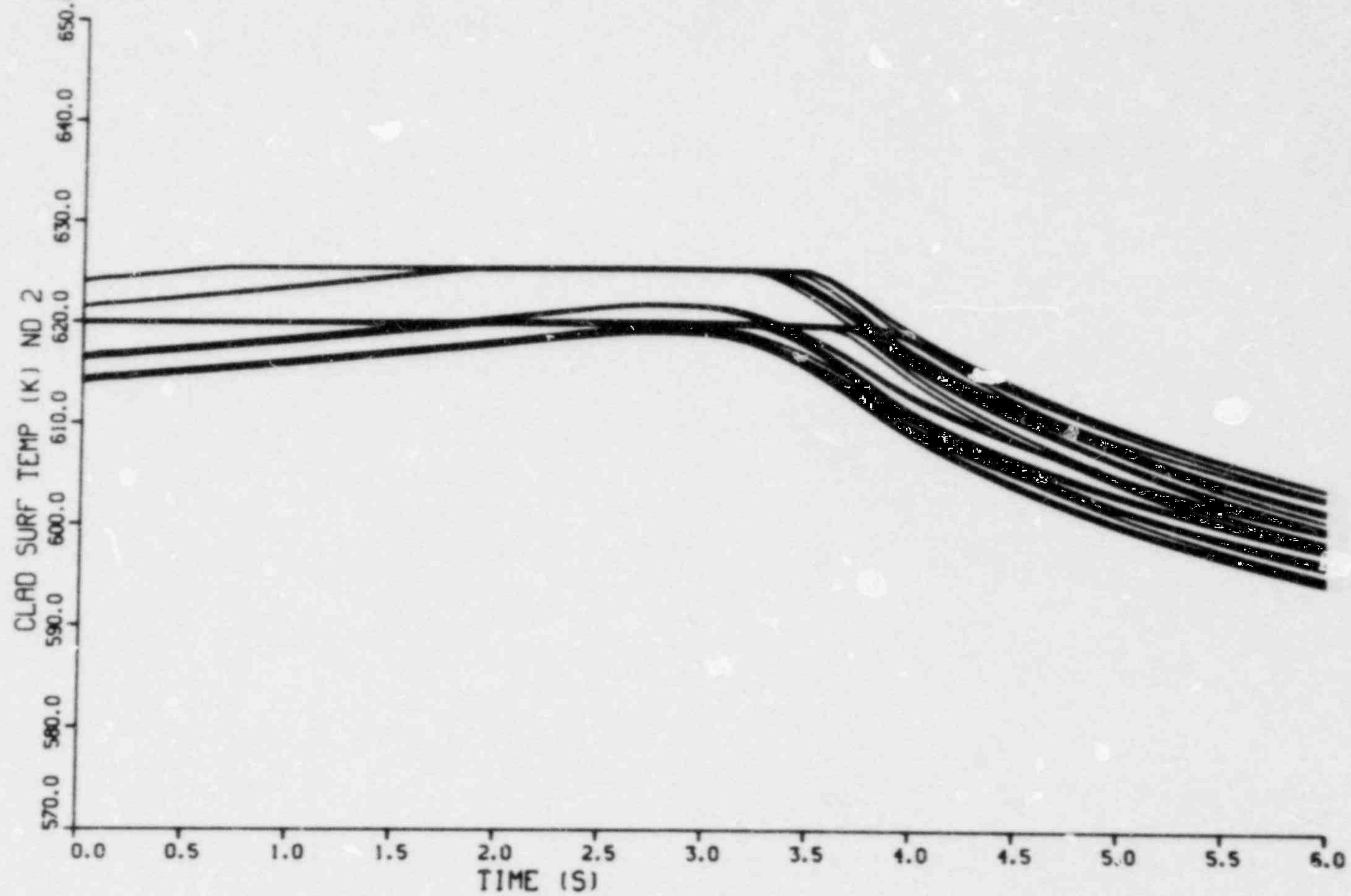

Figure 25. Individual cladding surface temperature histories for each CE loss-of-flow calculation.

Westinghouse Case

Since the operating histories of both the CE and Westinghouse events were very similar, the Westinghouse event was also very uneventful. The fuel centerline temperature history illustrated in Figure 26 shows essentially no increase in temperature during the event; only a period of initially constant temperature, then a decrease. The cladding surface temperature history showed the same trend. Film boiling was never entered by the best estimate of the 32 uncertainty study calculations. The minimum DNBR of the best estimate calculation was 2.0. The cool cladding and fuel temperatures resulted in calculated cladding hoop stresses always negative; thereby, never producing any positive permanent cladding strains. Also, no noteable cladding oxidation occurred.

Turbine Trip Without Bypass

The turbine trip without bypass event was submitted by GE. The event characteristics were a power increase during the first second, followed by reactor shutdown. Also, occurring simultaneously was a short period of increased mass flux and coolant pressure, followed by decreases of both variables. The operating history is shown schematically in Figure 27.

The fuel centerline temperature history of the peak power elevation is shown in Figure 28. At the time of peak temperature, the NRC incipient fuel melt limit was not exceeded by the best estimate temperature calculation or corresponding 2σ bound. The average fuel enthalpy and one standard deviation at time of peak temperature was 123 ± 14 cal/g, which again does not exceed any NRC fuel performance limits.

In all previous events, the uncertainty in fuel centerline temperature was dominated by the uncertainty of the fuel thermal conductivity. For the turbine trip without bypass event, fuel conductivity was again most important, but significant contributions to temperature uncertainty were also obtained from pellet-cladding gap conductance and rod power history. Other parameters provided insignificant contributions.

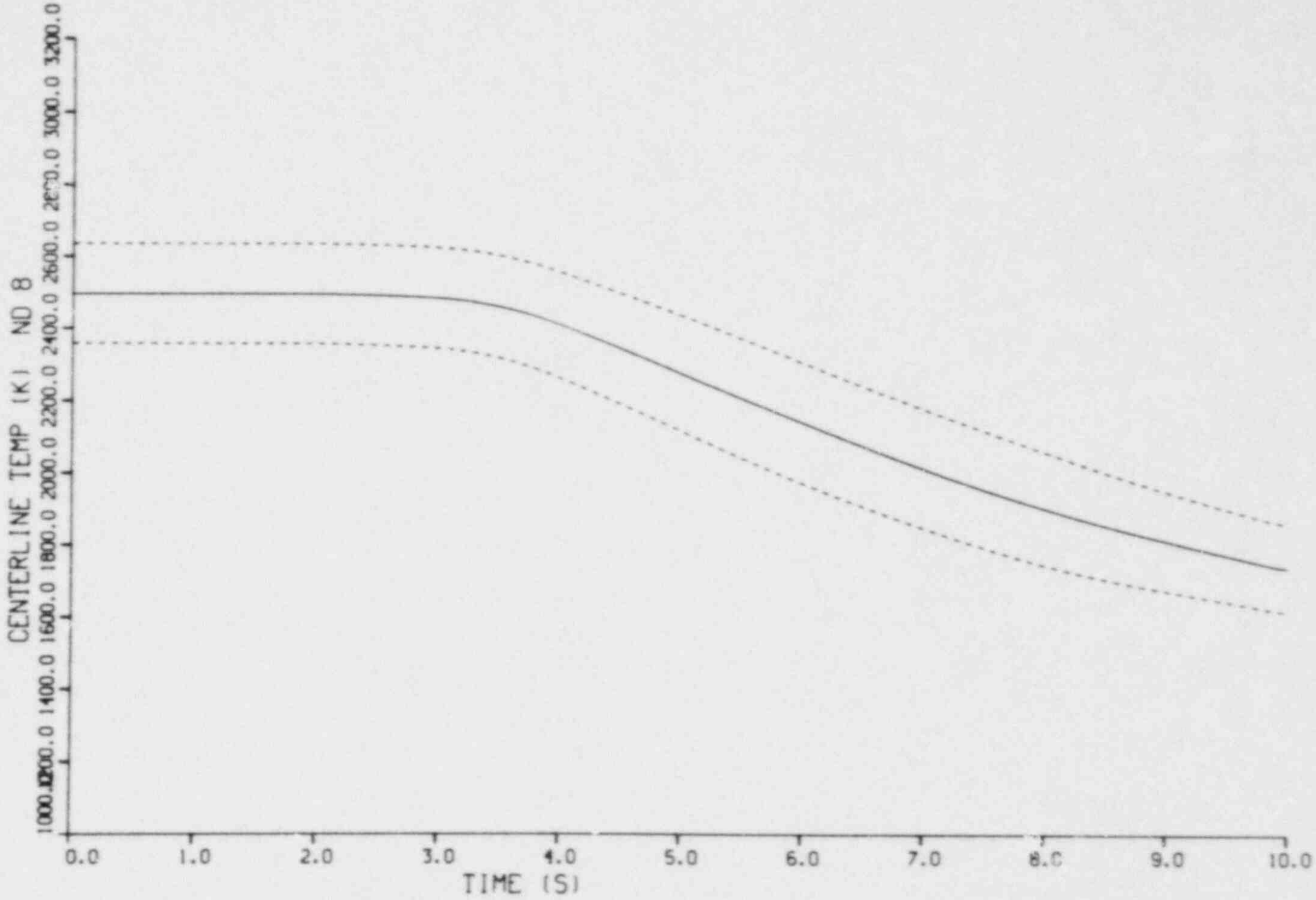

Figure 26. Average centerline temperature history and associated uncertainty for GE turbine trip without bypass event.

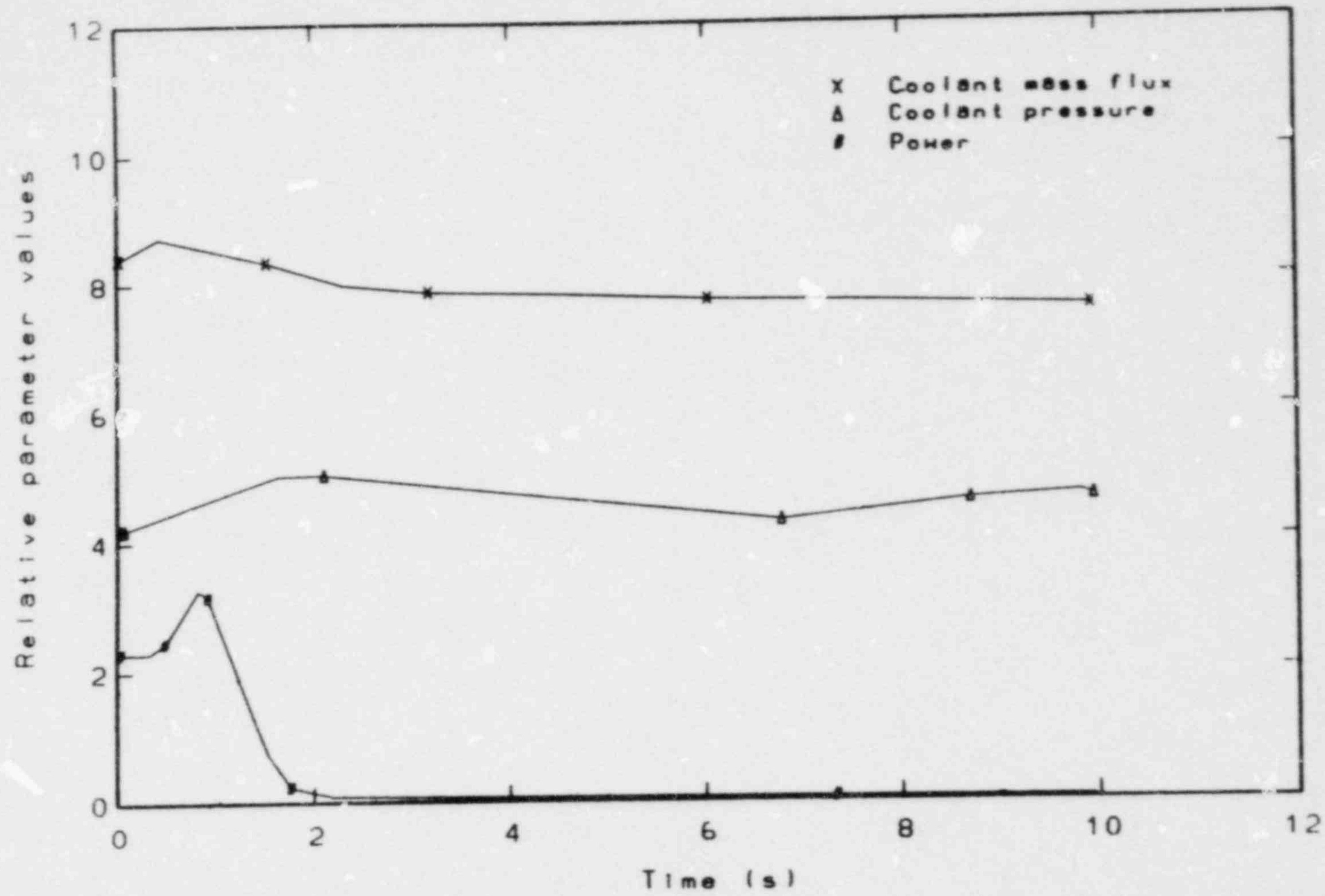

Figure 27. Operating history for GE turbine trip without bypass event.

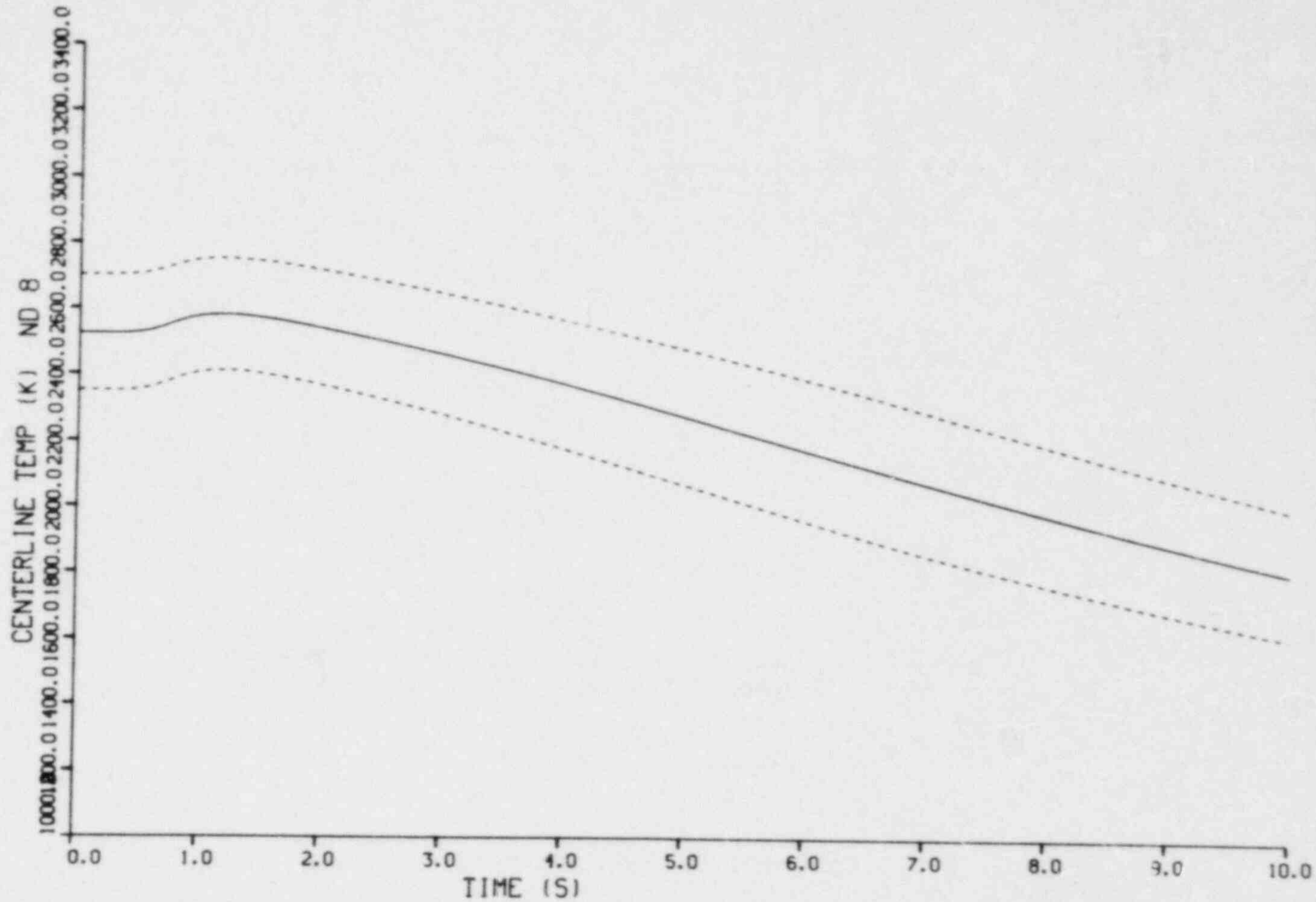

Figure 28. Average centerline temperature history and associated uncertainty for GE turbine trip without bypass event.

The cladding surface temperature at the peak power node is presented in Figure 29 for the 32 uncertainty study runs. Four of the 32 runs went into film boiling, although the best estimate minimum DNBR was only 1.53. The DNBR calculation for BWR coolant system conditions appears to be very sensitive to parameter perturbations, a sensitivity not observed for the PWR calculations.

For three of the four cases that entered film boiling, the rod returned to nucleate boiling within 1 s. Slight disruptions of cladding surface temperature and stress were noted, but no permanent change of rod geometry was predicted.

For the case that never returned to nucleate boiling, the cladding surface attained a peak temperature of about 1200 K, exceeding the α to $\alpha - \beta$ phase transition limit. However, the oxidation rate at this temperature is very low.

The cladding did collapse onto the fuel at about 2.3 s, as shown in Figure 30. Collapse was followed by a momentary spike of the cladding stress to 60 MPa. No rod failure ever occurred. Rather, the hot cladding continued to collapse as the fuel was cooled and contracted.

The above trends were noted for the elevation that corresponded to the axial peak power location. Further up the rod, the minimum DNBR calculated by the best estimate case was somewhat smaller, but no film boiling was calculated to occur by any of the 32 uncertainty study cases. This result shows the importance of considering rod performance at the elevation not only corresponding to the lowest DNBR, but also the highest stored energy.

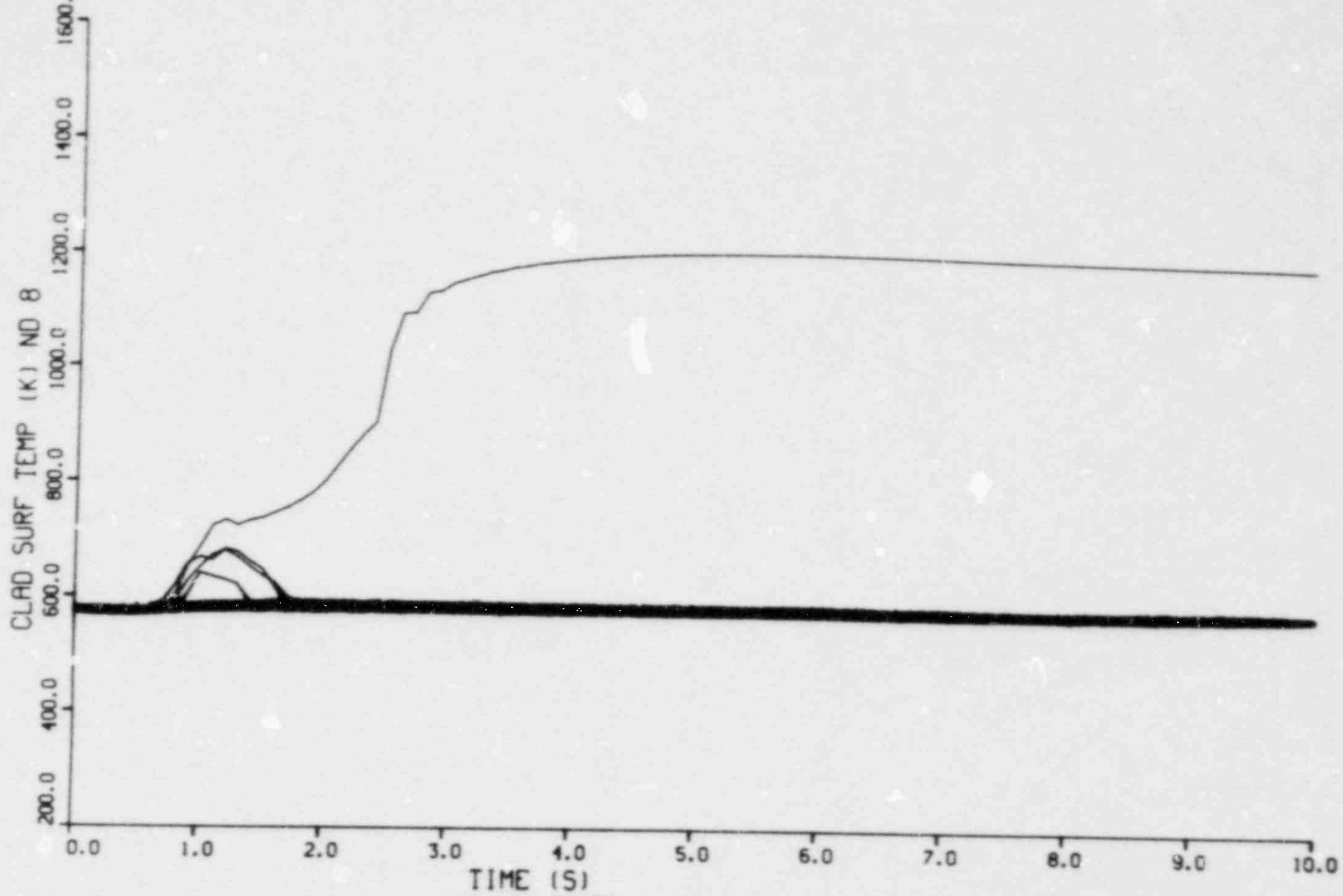

Figure 29. Individual cladding surface temperature histories for each GE turbine trip without bypass calculation.

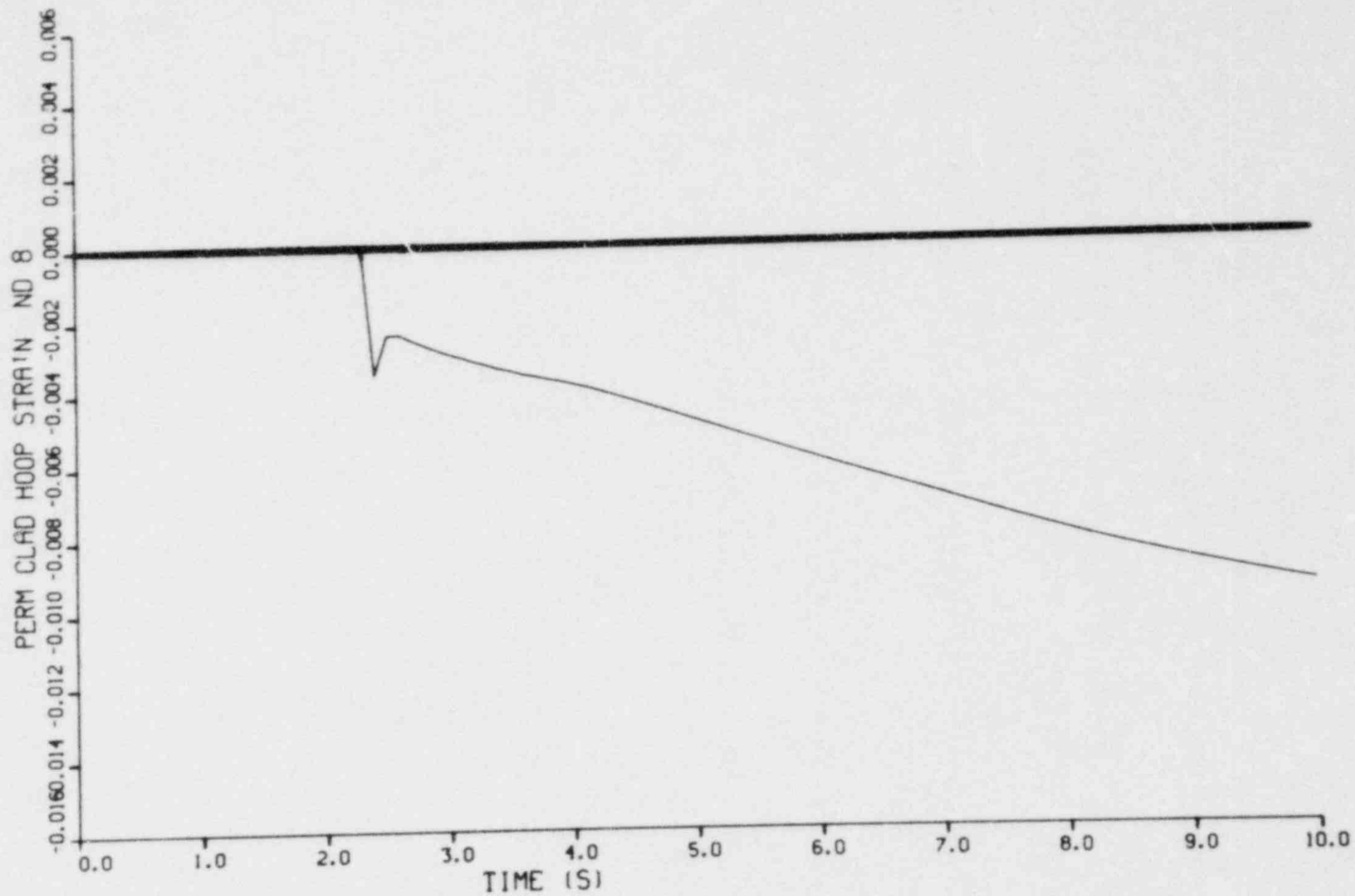

Figure 30. Individual permanent cladding hoop strain histories for each GE turbine trip without bypass calculation.

CONCLUSIONS

In this uncertainty study, five reactor operating events were examined; locked rotor, rod ejection, steam line break, loss of flow, and turbine trip without bypass. Input for these events was provided by four U.S. reactor vendors; Westinghouse Electric Company, Babcock and Wilcox, Combustion Engineering, and General Electric Company.

The results of the uncertainty studies of the five reactor operating events have led to three conclusions:

1. In general, the NRC fuel rod damage criteria were not exceeded by the events considered.

   Six of the nine base-case calculations never exceeded any NRC damage limits. The other three calculations were very close to the limit, and exceeded the limit only momentarily.

2. When an NRC rod damage limit was exceeded, the limit was always thermal-hydraulic in nature, rather than thermal, mechanical, or chemical.

   The only NRC damage limit ever exceeded was the departure from nucleate boiling ratio.

3. Exceeding the DNBR limit never led to a loss of cladding integrity.

   None of the base-case calculations predicted permanent rod deformation during these events. However, under extreme conditions, cladding collapse did occur during the Babcock and Wilcox rod ejection and General Electric Company turbine trip without bypass events. But even then, cladding failure never occurred.

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16. ABSTRACT (200 words or less)

The FRAP-T5 fuel rod behavior code, with its recently developed automated uncertainty analysis option, was used at EG&G Idaho, Inc., to calculate rod behavior for five reactor transient and accident events. Included were locked rotor, rod ejection, steam line break, loss-of-flow, and turbine trip without bypass events. The intent was to identify the limiting Nuclear Regulatory Commission (NRC) fuel rod damage criteria based on best estimate calculations with associated uncertainties, rather than the traditional calculations that utilize conservative evaluation models. Conclusions are reported regarding the likelihood and subsequent consequences of exceeding the fuel rod damage limits.

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