ATTACHMENT B

DEMONSTRATION OF THE CONFORMANCE TO THE 10CFR50.46 ACCEPTANCE CRITERIA FOR THE SMALL BREAK LOSS-OF-COOLANT ACCIDENT FOR THE H. B. ROBINSON UNIT 2 NUCLEAR POWER PLANT WITH EXXON NUCLEAR COMPANY FUEL

Westinghouse Electric Corporation

Nuclear Technology Division

Nuclear Safety Department

Safeguards Engineering and Development

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ATTACHMENT

SAFETY EVALUATION
SMALL BREAK LOCA CONSEQUENCES OF OPERATION OF H. B. ROBINSON UNIT 2 WITH A MODIFICATION TO THE THIRD LINE SEGMENT OF THE OPERATING ENVELOPE

I. INTRODUCTION

A safety evaluation was performed to justify the operation of the H. B. Robinson Unit 2 nuclear power plant with a modification to the third line segment of the operating envelope. An analysis was performed to determine the results of a postulated small break loss-of-coolant accident (LOCA) assuming that the third line segment of the operating envelope for H. B. Robinson Unit 2 was raised from a 0.835 intercept to a 1.5 intercept.

Carolina Power & Light Company has observed shifts in the axial power profile of the H. B. Robinson Unit 2 nuclear power plant over a period of time, which upon projection indicates that the power distribution would eventually violate the plant technical specifications for the third line segment of the operating envelope of plant peaking factor.

The third line segment of the operating envelope of plant peaking factor for H. B. Robinson Unit 2 has an intercept value of 0.835 at the 12 foot elevation. A change to the plant technical specifications has been proposed which would raise the third line segment of the operating envelop from an intercept of 0.835 at the 12 foot elevation from the bottom of the core to a value of 1.5 at the 12 foot elevation which would allow increased operating flexibility with power shapes peaked toward the top of the core. In order to justify continued operating with a modification of the third line segment, a safety evaluation was prepared to assess and demonstrate compliance of the ECCS with the requirements of 10CFR50.46 and Appendix K assuming that the third line segment intercept value is raised from 0.835 to 1.5 at the 12 foot elevation.

II. METHOD OF EVALUATION

This evaluation considered the transient thermal-hydraulic response of the reactor coolant system and the transient thermal performance of the hot rod in the hottest assembly of H. B. Robinson Unit 2 during a postulated small break LOCA. As a technical basis to support the safety evaluation, a small break LOCA analysis was performed to demonstrate compliance of the emergency core cooling system (ECCS) with the requirements of 10CFR50.46 and Appendix K. The acceptance criteria for the loss-of-coolant accident is described in 10CFR50.46 as follows:

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

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

3. The amount of hydrogen generated by fuel element cladding that reacts chemically with water or steam does not exceed an amount corresponding to interaction of 1% of the total amount of Zircaloy in the reactor.

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, as required by the long-lived radioactivity remaining in the core.

These criteria were established to provide significant margin in ECCS performance following a LOCA. For H. B. Robinson Unit 2, small breaks (less than 1.0 ft^2) yield results with more margin to the acceptance criteria then large breaks.

Description of Small Break LOCA Transient

A minor pipe break (small break), as considered in this section, 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 a Condition III event, an infrequent fault.

Ruptures of small cross-section will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps. These pumps would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant which would be released to the containment contains the fission products existing at equilibrium.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the reactor Coolant System through the postulated break against the charging pump makeup flow at normal Reactor Coolant System pressure; i.e., 2250 psia. A makeup flow rate from one positive displacement charging pump is typically adequate to sustain pressurizer level at 2250 psia for a break through a 0.295-inch diameter hole. This break results in a loss of approximately 10.6 lbm/sec.

Should a larger break occur, depressurization of the Reactor Coolant System results. Unless the break is in the pressurizer, the break causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the low pressurizer pressure trip setpoints is reached. During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The Safety Injection System is actuated when the appropriate setpoint is reached. The consequences of the accident are limited in two ways:

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

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

Before the break occurs, the plant is in an equilibrium condition; i.e., the heat generated in the core is being removed via the secondary system. During the depressurization, heat from decay, hot internals, and the vessel continues to be transferred to the Reactor Coolant System. 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, system pressure increases and steam dump may occur.

Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates auxiliary feedwater flow by starting auxiliary feedwater pumps. The secondary flow aids in the reduction of Reactor Coolant System pressures.

When the RCS depressurizes to 600 psig, the cold leg accumulators begin to inject water into the reactor coolant loops. Due to the loss of offsite power assumption, the reactor coolant pumps are assumed to be tripped at the time of reactor trip during the accident and the effects of pump coastdown are included in the depressurization analysis.

Small Break LOCA Evaluation Model

The requirements of an acceptable ECCS Evaluation Model are presented in Appendix K of 10CFR50. The requirements of Appendix K regarding specific model features were met by selecting models which 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 that the LOCA occurs and include such items as the core peaking factors, axial power shape, and the performance of the ECCS system. Decay heat generated throughout the transient is also conservatively calculated as required by Appendix K of 10CFR50.46.

The WFLASH computer code was used in the analysis of the loss-of-coolant accident due to small breaks in the reactor coolant system. The WFLASH computer code is an extension of the FLASH-4 code developed at Westinghouse Bettis Atomic Power Laboratory. The WFLASH program permits a detailed spatial representation of the RCS.

The RCS is nodalized into volumes interconnected by flowpaths. The loop containing the break is modeled explicitly with the remaining non-faulted 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 WFLASH is given in Reference 1.

The use of WFLASH in the analysis involves, among other things, the representation of the reactor core as a heated control volume 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.

Cladding thermal analyses are performed with the LOCTA-IV (Reference 2) code which uses the RCS pressure, fuel rod power history, steam flow past the uncovered portion of the core, and mixture height history from the WFLASH hydraulic calculations, as input.

Small Break Input Parameters and Initial Conditions

As a technical basis for the safety evaluation, an analysis of a postulated small break LOCA was performed for H. B. Robinson Unit 2 using the WFLASH small break LOCA ECCS evaluation model. A WFLASH model for the Turkey Point Unit 3 (FPL) nuclear power plant was used as the basis for the H. B. Robinson WFLASH small break LOCA analysis. The H. B. Robinson and Turkey Point Unit 3 design parameters which influence the small break LOCA ECCS evaluation model analysis results are compared in Table 1. This comparison indicates that the H. B. Robinson plant design is similar to the Turkey Point plant design, except for the core power, safety injection flow rate, auxiliary feedwater flow rates, and fuel designs.

The WFLASH model for the Turkey Point Unit 3 (FPL) nuclear plant analysis represented the reactor coolant system as two loops. The RCS loop containing the break is represented as a single broken loop, while the two unfaulted loops are lumped together to form the intact loop. This method of representing the RCS and procedures for calculating the input were in conformance with the approved methods and procedures in place at the time the analysis was performed.

Safety injection flow rate to the RCS as a function of the system pressure is used as part of the input. The safety injection flow rate specific to the H. B. Robinson Unit 2 nuclear power plant was used in the analysis. The Safety Injection (SI) system was assumed to be delivering to the RCS 25 seconds after the generation of a safety injection signal.

The 25 second delay includes times required for diesel startup and loading of the safety injection pumps onto the emergency buses. The effect of flow from the RHR pumps is not considered here since their shutoff head is lower than the RCS pressure during the time portion of the transient considered here. Also, minimum safeguards Emergency Core Cooling System capability and operability has been assumed in the analysis.

The hydraulic analysis was performed with the WFLASH code using 102% of the licensed NSSS core power. The core hot rod thermal transient analysis was performed with the LOCTA-IV code using 102% of licensed NSSS core power. The basis of the analysis calculations are summarized in Table 2.

The worst small break for the Turkey Point Unit 3 (FPL) nuclear power plant was the 3 inch cold leg break, in a spectrum of cold leg breaks of equivalent diameters of 2 inches, 3 inches, 4 inches, and 6 inches. The limiting 3 inch break for FPL calculated a peak cladding temperature of 1605°F, while the 4 inch cold leg break resulted in 1506°F, and the 6 inch cold leg break resulted in 1233°F.

Higher power has the effect of not only increasing the peak cladding temperature (PCT), but tends to result in larger break sizes being more limiting. On the other hand, reducing the safety injection flow also tends to result in increasing PTs, but also has the effect of making the small break sizes more limiting. Based upon the limiting break size observed for the Turkey Point Unit 3 plant and the tradeoff in break size effects due to higher power and lower safety injection flow, the 3 inch equivalent diameter cold leg break was analyzed as the technical basis for this safety evaluation.

The Turkey Point Unit 3 WFLASH and LOCTA-IV models were modified to represent the higher core power, plant specific safety injection flow and to take into account the difference in auxiliary feedwater flow design. Except for modification due to the power level differences, the Westinghouse 15 x 15 OFA fuel parameters were used to represent

the average fuel rod in the WFLASH analysis and the hot rod in the LOCTA-IV analysis. The WFLASH and LOCTA analyses incorporated the Westinghouse generic power shape with a third line segment intercept of 1.5 at the 12 foot elevation.

The Exxon 15 x 15 fuel design is similar to the Westinghouse 15 x 15 OFA fuel design. The Exxon 15 x 15 fuel rod has an outside diameter of 0.03533 ft., a cladding inside diameter of 0.03033 ft., a fuel pellet outside diameter of 0.02971 ft., and a fuel rod pitch of 0.04692 ft. The Exxon fuel rod maximum densification occurs at 1500 MWD/MTU at the theoretical density of 95.3%. The Westinghouse 15 x 15 OFA fuel rod has an outside diameter of 0.03517 ft., a cladding inside diameter of 0.03112 ft., a fuel pellet outside diameter of 0.03049 ft., and a fuel rod pitch of 0.04692 ft. During the small break LOCA transient the initial stored energy is removed to the RCS. Clearly the fuel rods are similar in design and the design differences will have only a small effect on the small break LOCA analysis results during the uncovery heatup.

III. RESULTS

The analysis results for H. B. Robinson Unit 2 are summarized in Table 3. Table 4 summarizes the analysis sequence of events. Figure 1 presents the reactor coolant system pressure response as a function of time after the break. Figure 2 presents the core mixture level transient response, and Figure 3 presents the hot spot clad temperature. Figure 4 presents the core steam flow rate, and Figure 5 shows the hot spot fluid temperature.

The maximum calculated peak cladding temperature for the 3 inch equivalent diameter small break in the cold leg of the H. B. Robinson Unit 2 nuclear power plant with a modified third line segment of the operating envelope is 1801°F. This results is well below all the acceptance criteria limits of 10CFR50.46 and is not limiting when compared to the results presented for large break LOCAs.

IV. CONCLUSIONS

The results of this safety evaluation indicate that the H. B. Robinson Unit 2 nuclear power plant with Exxon fuel and a raised third line segment to the operating envelop will be in compliance with the acceptance criteria as presented in 10CFR50.46 and Appendix K when analyzed with the Westinghouse WFLASH small break LOCA ECCS Evaluation Model.

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REFERENCES:

1. Esposito, V. J., Kesavan, K., and Maul, B. A., "WFLASH - A FORTRAN-IV Computer Program for Simulation of Transients in a Multi-Loop PWR," WCAP-8200, Revision 2 (Proprietary) and WCAP-8261, Revision 1 (Non-Proprietary), July 1974.

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

3. Skwarek, R., Johnson, W. J., and Meyer, P. E., "Westinghouse Emergency Core Cooling System Small Break October 75 Model," WCAP-8970 (Proprietary) and WCAP-8971 (Non-Proprietary), January 1979.

TABLE 1

COMPARISON OF H. B. ROBINSON UNIT 2 AND TURKEY POINT UNIT 3
DESIGN PARAMETERS WHICH INFLUENCE RESPONSE TO A
SMALL BREAK IN THE PRIMARY REACTOR COOLANT SYSTEM

Parameter	CP&L	FP&L
Core Power (MW_{th})	2300	2200
Fuel Type	Exxon 15 x 15	W 15 x 15 OFA
Barrel Baffle Design	Downflow	Downflow
Steam Generator Type	Model 44F	Model 44F
S.G. Safety Valve Setpoint (psia)	1100	1100
Safety Injection Flow ($lbm/sec/MW_{th}$) (* at 1000 psia)	0.01440*	0.01832*
Motor Driven Aux. Feed Pumps (Capacity GPM)	2 300	0
Steam Turbine Driven Aux. Feed Pumps (Capacity GPM)	1 600	3 600
Upper Head Temperature	THOT	THOT
Upper Support Plate Design	Flat	Flat
Thermal Design Flow (GPM)	88,600	89,500

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TABLE 2

H. B. ROBINSON UNIT 2
SMALL BREAK LOCA ECCS ANALYSIS BASIS
3-INCH COLD LEG BREAK

Calculational Basis	
Core Power, MW_{th}	2346*
Total Peaking Factor, F_q^T	2.32
Peak Linear Power Used in Analysis, kW/ft	10.99
Elevation of Peak Linear Power, ft	10.0
Operating Envelope Assumed in this Analysis	Figure 6
Hot Channel Enthalpy Rise Factor	1.65

* Including Factor of 1.02 for Calorimetric Uncertainty

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TABLE 3

H. B. ROBINSON UNIT 2
SMALL BREAK LOCA ECCS ANALYSIS RESULTS
3-INCH COLD LEG BREAK

Analysis Results	
Peak Clad Temperature (PCT), °F	1800.6
Elevation of Peak Clad Temperature, ft	12.0
Local Zr/H_2O Reaction (Max.), %	2.25
Local Zr/H_2O Elevation, ft. from Bottom	12.0
Total Zr/H_2O Reaction, %	less than 0.3
Hot Rod Burst Elevation	N/A*

* Burst Not Calculated to Occur

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TABLE 4

H. B. ROBINSON UNIT 2
SMALL BREAK LOCA ECCS ANALYSIS SEQUENCE OF EVENTS
3-INCH COLD LEG BREAK

Event	(Sec)
Break Initiation	0.0
Reactor Trip Signal Generated	20.5
S-Signal Generated	24.9
Top of Core Uncovered	507.8
Accumulator Injection Time	1184.0
Peak Clad Temperature Occurs	1201.1
Hot Rod Burst Time	N/A*
Top of Core Covered	2010.0

* Burst Not Calculated to Occur

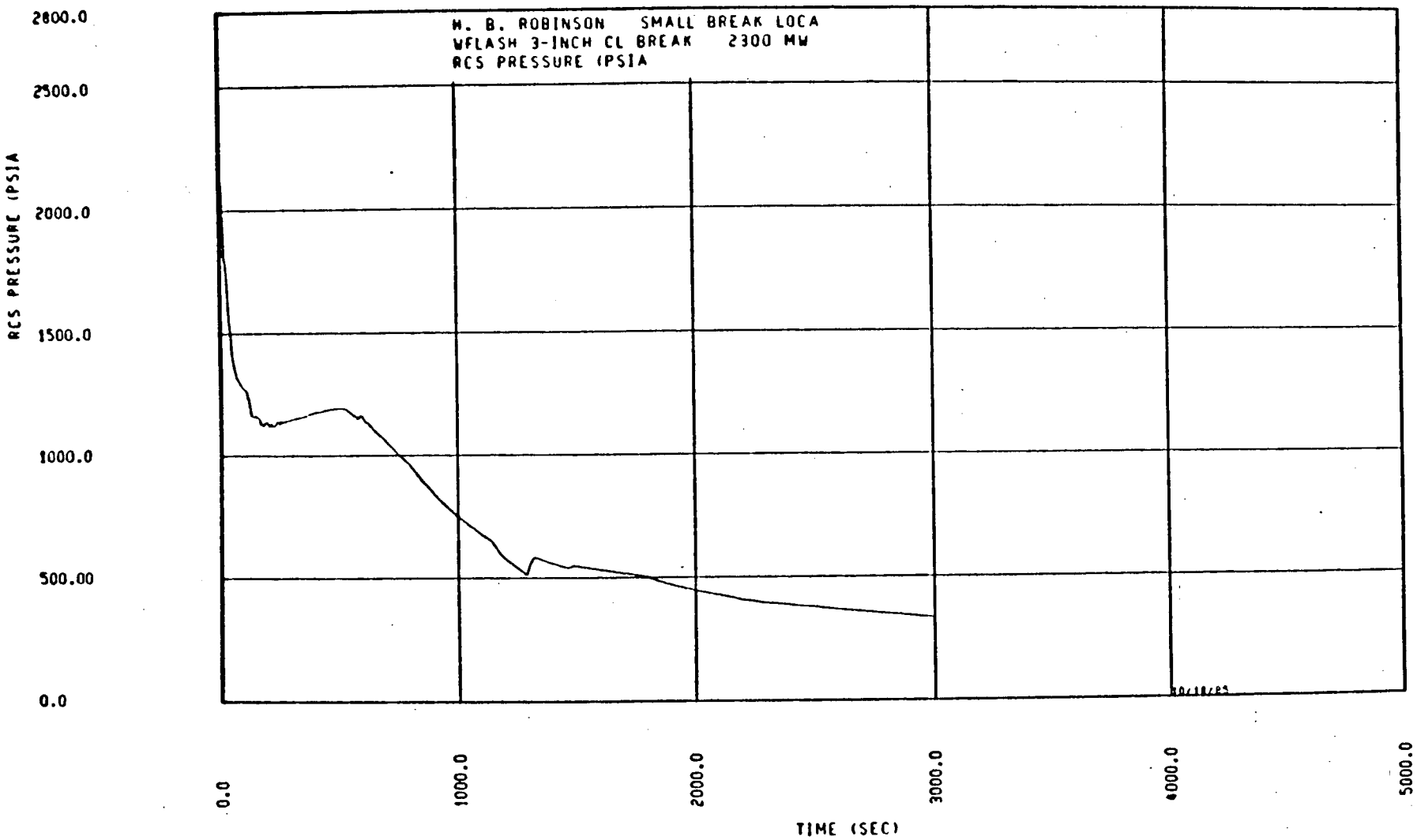

FIGURE 1

RCS PRESSURE VS TIME

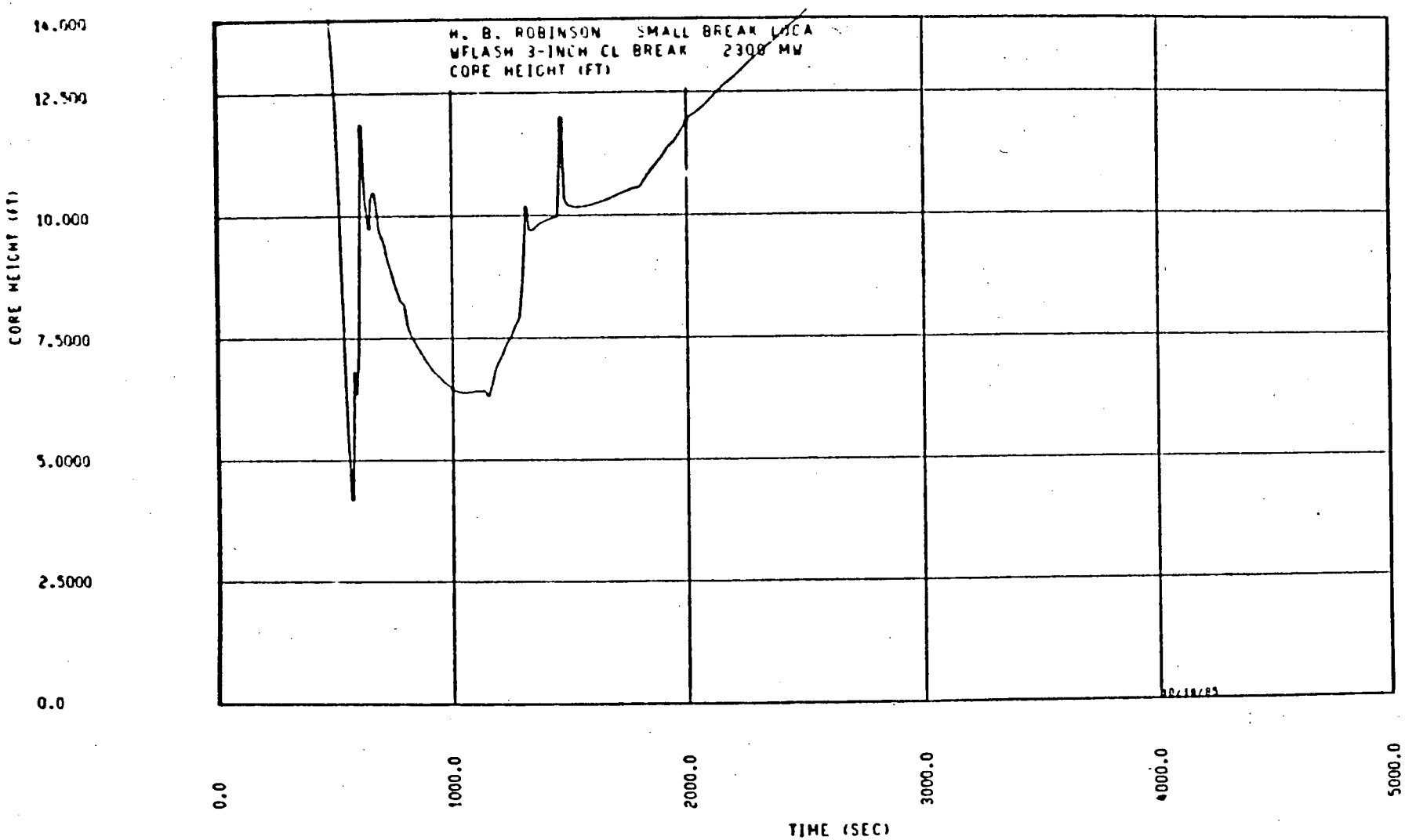

FIGURE 2
CORE MIXTURE LEVEL

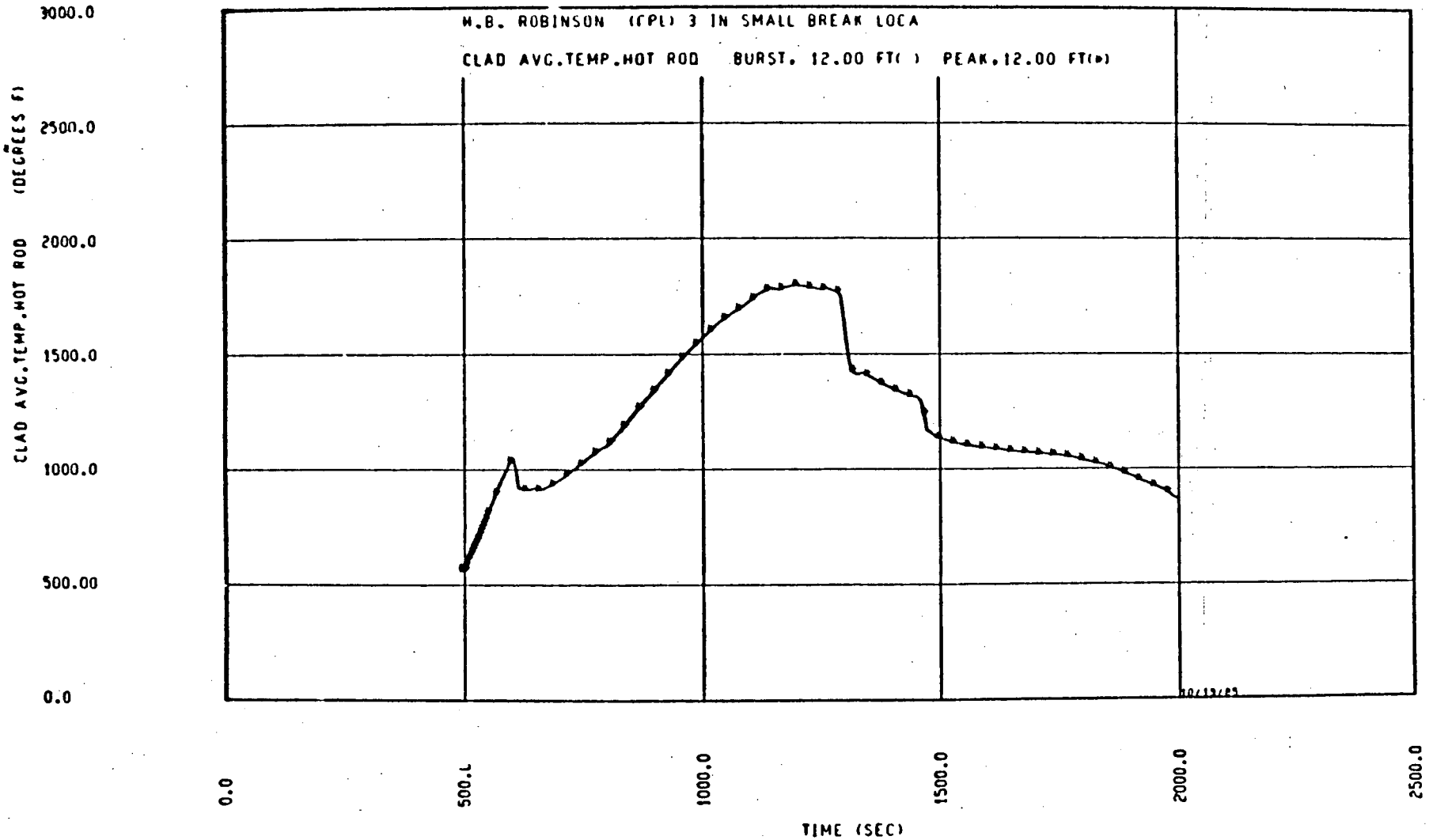

FIGURE 3
HOT SPOT CLAD TEMPERATURE

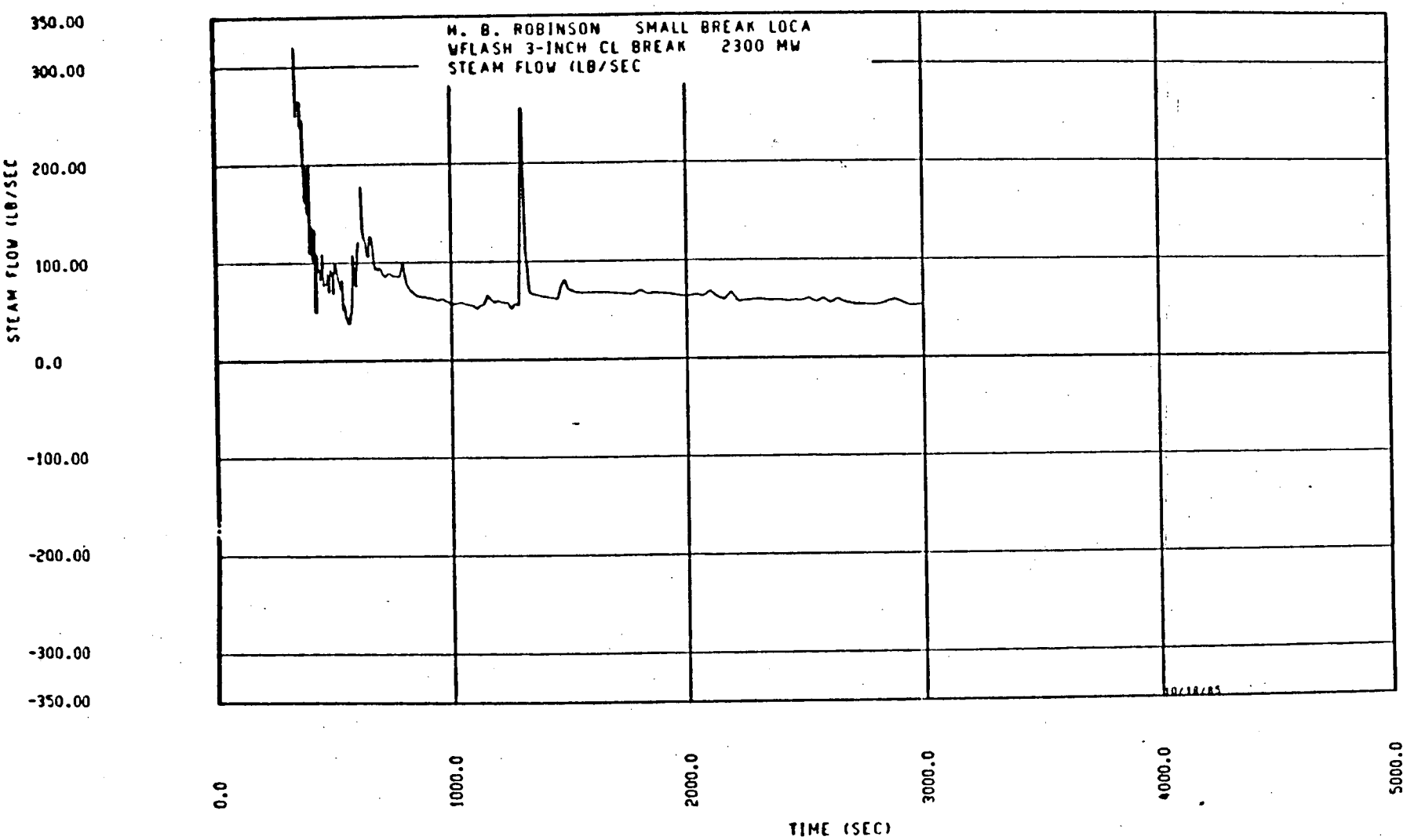

FIGURE 4

CORE STEAM FLOW RATE

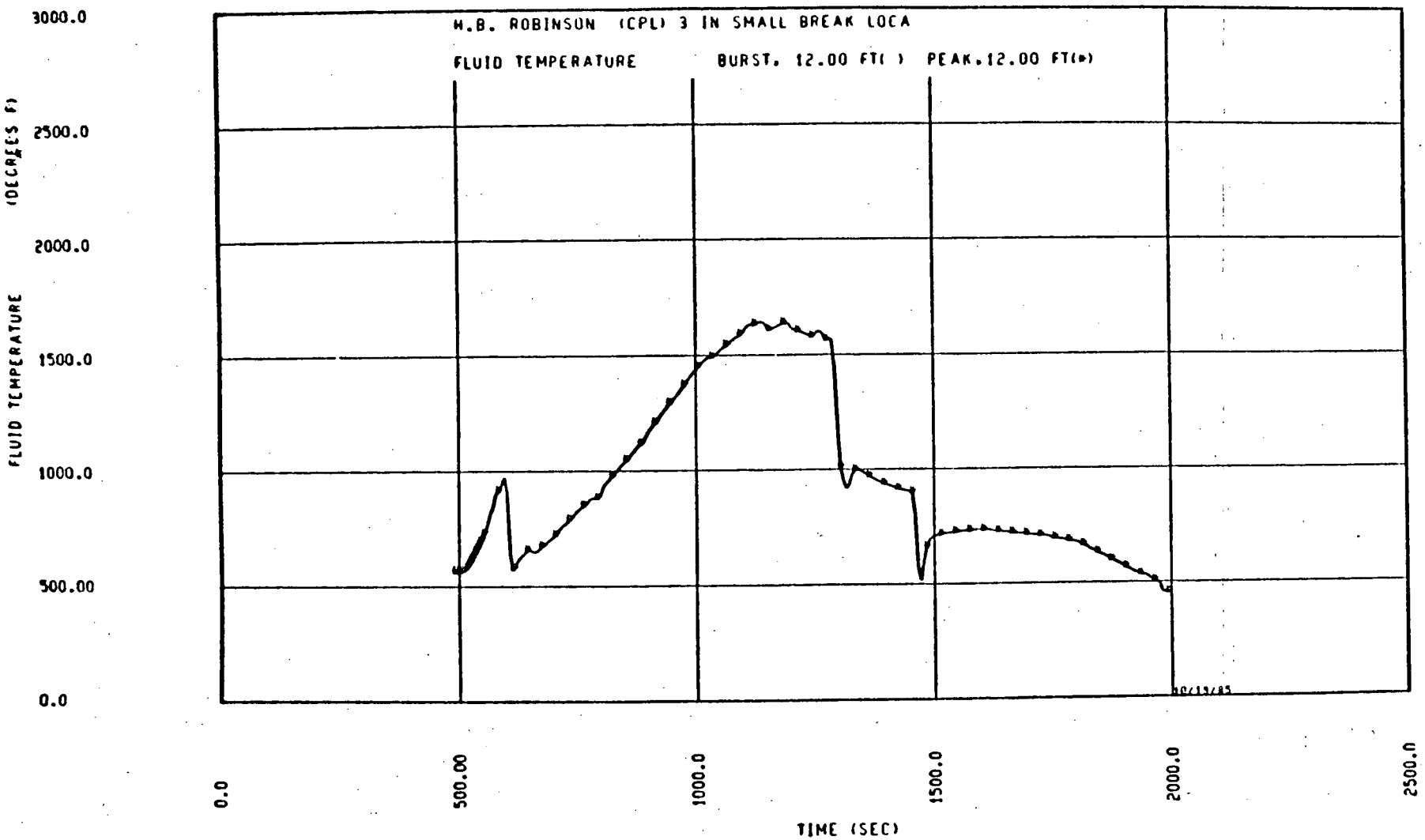

FIGURE 5
HOT SPOT FLUID TEMPERATURE

FIGURE 6
SMALL BREAK LOCA POWER SHAPE