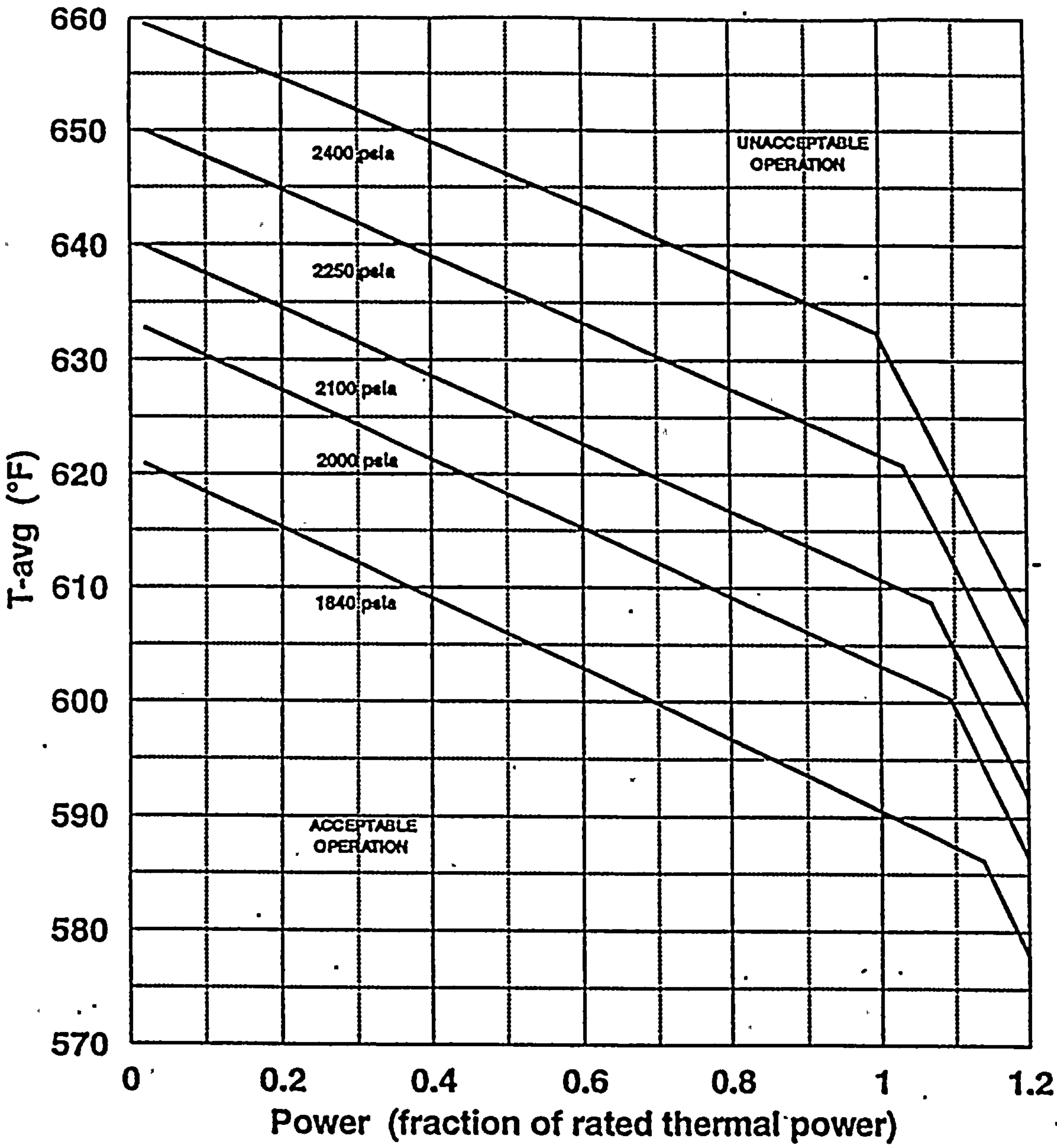

PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T-AVG IN °F)		
1840	(0.02, 620.86),	(1.136, 586.17),	(1.2, 577.94)
2000	(0.02, 632.79),	(1.094, 600.31),	(1.2, 586.52)
2100	(0.02, 639.85),	(1.068, 608.72),	(1.2, 591.77)
2250	(0.02, 649.96),	(1.031, 620.83),	(1.2, 599.40)
2400	(0.02, 659.52),	(0.996, 632.42),	(1.2, 606.63)

Figure 2.1-1 Reactor Core Safety Limits

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Range, Neutron Flux	Low Setpoint - less than or equal to 25% of RATED THERMAL POWER High Setpoint - less than or equal to 109% of RATED THERMAL POWER	Low Setpoint - less than or equal to 26% of RATED THERMAL POWER High Setpoint - less than or equal to 110% of RATED THERMAL POWER
3.	Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4.	Power Range, Neutron Flux, High Negative Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
5.	Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6.	Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3×10^5 counts per second
7.	Overtemperature Delta T	See Note 1	See Note 3
8.	Overpower Delta T	See Note 2	See Note 4
9.	Pressurizer Pressure -- Low	Greater than or equal to 1875 psig	Greater than or equal to 1865 psig
10.	Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11.	Pressurizer Water Level -- High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12.	Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

*Design flow is 1/4 Reactor Coolant System total flow rate from Table 3.2-1.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature $\Delta T \leq \Delta T_o \ [K_1 - K_2 \left[\frac{1+\tau_1 s}{1+\tau_2 s}\right] (T-T') + K_3 (P-P') - f_1 (\Delta I)]$

where:	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T'	=	Indicated T_{avg} at RATED THERMAL POWER ($\leq$ 576.3°F)
	P	=	Pressurizer pressure, psig
	P'	=	Indicated RCS nominal operating pressure (2235 psig or 2085 psig)
	$\frac{1+\tau_1 s}{1+\tau_2 s}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 22 secs. τ_2 = 4 secs.
	S	=	Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Operation with 4 Loops

$K_1 = 1.17$
$K_2 = 0.0230$
$K_3 = 0.00110$

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -37 percent and +3 percent, $f_1(\Delta I)=0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) For each percent that the magnitude of $(q_t - q_b)$ exceeds -37 percent, the ΔT trip setpoint shall be automatically reduced by 0.33 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +3 percent, the ΔT trip setpoint shall be automatically reduced by 2.34 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 (T - T'') - f_2 (\Delta I)\right]$

where:	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T''	=	Indicated T_{avg} at RATED THERMAL POWER ($\leq$ 563.0°F)
	K_4	=	1.083
	K_5	=	0.0177/°F for increasing average temperature and 0 for decreasing average temperature
	K_6	=	0.0015 for $T > T''$; $K_6=0$ for $T \leq T''$
	$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
	τ_3	=	Time constant utilized in the rate lag controller for T_{avg} $\tau_3 = 10$ secs.
	S	=	Laplace transform operator
	$f_2 (\Delta I)$	=	0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.4 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.5 percent ΔT span.

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - TAVG GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Delta k/k:

a. Within one hour after detection of an inoperable control rods(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5.

*See Special Test Exception 3.10.1.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3

a. One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

b. One charging flowpath associated with support of Unit 2 shutdown functions shall be available.*

APPLICABILITY: Specification 3.1.2.3.a. - MODES 5 and 6
Specification 3.1.2.3.b. - At all times when Unit 2 is in MODES 1, 2, 3, or 4.

ACTION:

a. With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.**

b. With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 152°F, unless the reactor vessel head is removed, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within one hour.

c. The provisions of Specification 3.0.3 are not applicable.

d. In addition to the above, when Specification 3.1.2.3.b is applicable and the required flow path is not available, return the required flow path to available status within 7 days, or provide equivalent shutdown capability in Unit 2 and return the required flow path to available status within the next 60 days, or have Unit 2 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours.

e. The requirements of Specification 3.0.4 are not applicable when Specification 3.1.2.3.b applies.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a differential pressure of greater than or equal to 2290 psid when tested pursuant to Specification 4.0.5.

*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F.

**For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration on the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying that on recirculation flow, each pump develops a differential pressure of greater than or equal to 2290 psid when tested pursuant to Specification 4.0.5.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum usable borated water volume of 4300 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum usable borated water volume of 90,000 gallons,
2. A minimum boron concentration of 2400 ppm, and
3. A minimum solution temperature of 70°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the water level volume of the tank, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water.

*For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

BORATED WATER SOURCES - OPERATIONS

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum usable borated water volume of 5650 gallons,

2. Between 20,000 and 22,500 ppm of boron, and

3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 350,000 gallons of water,

2. Between 2400 and 2600 ppm of boron, and

3. A minimum solution temperature of 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

TABLE 3.2-1
DNB PARAMETERS

PARAMETER	LIMITS 4 Loops in Operation at RATED THERMAL POWER
Reactor Coolant System Tavg	≤ 579.3°F*
Pressurizer Pressure	≥ 2050 psig**
Reactor Coolant System Total Flow Rate	≥ 341,100 gpm***

*Indicated average of at least three OPERABLE instrument loops.

**Limit not applicable during either a THERMAL POWER ramp increase in excess of 5 percent RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10 percent RATED THERMAL POWER.

***Indicated value.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line Pressure-Low					
Four Loops Operating	1 pressure/loop	2 pressures any loops	1 pressure any 3 loops	1,2,3##	
Three Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops	3##	14* 15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
COINCIDENT WITH					
T_{avg} -- Low-Low					
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops	1, 2, 3##	14*
Three Loops Operating	1 T_{avg}/operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops	3##	15
e. Steam Line Pressure-Low					
Four Loops Operating	1 pressure/loop	2 pressures any loops	1 pressure any 3 loops	1, 2, 3##	14*
Three Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops	3##	15
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1,2,3	14*

ENGINEERED SAFETY FEATURES INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-11	With 2 of 3 pressurizer pressure channels greater than or equal to 1915 psig.	P-11 prevents or defeats manual block of safety injection actuation on low pressurizer pressure.
P-12	With 2 of 4 T_{avg} channels less than or equal to Setpoint. Setpoint greater than or equal to 541°F	P-12 allows the manual block of safety injection actuation on low steam line pressure causes steam line isolation on high steam flow. Affects steam dump blocks. With 3 of 4 T_{avg} channels above the reset point, prevents or defeats the manual block of safety injection actuation on low steam line pressure.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN FEEDWATER PUMPS		
a. Manual Initiation	------------------------ See Functional Unit 9 ------------------------	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	Less than or equal to 1.1 psig	Less than or equal to 1.2 psig
d. Pressurizer Pressure--Low	Greater than or equal to 1815 psig	Greater than or equal to 1805 psig
e. Differential Pressure Between Steam Lines--High	Less than or equal to 100 psi	Less than or equal to 112 psi
f. Steam Line Pressure--Low	Greater than or equal to 500 psig steam line pressure	Greater than or equal to 480 psig steam line pressure

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Containment Radio-activity--High Train A (VRS-1101, ERS-1301, ERS-1305)	See Table 3.3-6	Not Applicable
3. Containment Radio-activity--High Train B (VRS-1201, ERS-1401, ERS-1405)	See Table 3.3-6	Not Applicable
4. STEAM LINE ISOLATION		
a. Manual	---------- See Functional Unit 9 ----------	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	Less than or equal to 2.9 psig	Less than or equal to 3 psig
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low	Less than or equal to 1.42 x 10^6 lbs/hr from 0% load to 20% load. Linear from 1.42 x 10^6 lbs/hr at 20% load to 3.88 x 10^6 lbs/hr at 100% load.	Less than or equal to 1.56 x 10^6 lbs/hr from 0% load to 20% load. Linear from 1.56 x 10^6 lbs/hr at 20% load to 3.93 10^6 lbs/hr at 100% load.
	T_{avg} greater than or equal to 541°F	T_{avg} greater than or equal to 539°F
e. Steam Line Pressure--Low	Greater than or equal to 500 psig steam line pressure	Greater than or equal to 480 psig steam line pressure
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level--High-High	Less than or equal to 67% of narrow-range instrument span each steam generator	Less than or equal to 68% of narrow-range instrument span each steam generator

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Manual Initiation	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure--High	S	R	M(3)	N.A.	1, 2, 3
d. Pressurizer Pressure--Low	S	R	M	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	N.A.	1, 2, 3
f. Steam Line Pressure--Low	S	R	M	N.A.	1, 2, 3
2. CONTAINMENT SPRAY					
a. Manual Initiation	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M(3)	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION					
a. Manual	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3,
c. Containment Pressure--High-High	S	R	M(3)	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low	S	R	M	N.A.	1, 2, 3
e. Steam Line Pressure-Low	S	R	M	N.A.	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	S	R	M	N.A.	1, 2, 3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level--Low-Low	S	R	M	N.A.	1, 2, 3
b. 4 kv Bus Loss of Voltage	S	R	M	N.A.	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	N.A.	1, 2, 3
d. Loss of Main Feed Pumps	N.A.	N.A.	R	N.A.	1, 2

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± 3%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE:

a. Immediately suspend all operations involving positive reactivity changes** and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. Immediately render all Safety Injection pumps and all but one charging pump inoperable by removing the applicable motor circuit breakers from the electric power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Requirement 4.4.3.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 3%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional surveillance requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection signal:

a) Centrifugal charging pump

b) Safety injection pump

c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5.

1. Centrifugal charging pump greater than or equal to 2290 psid

2. Safety injection pump greater than or equal to 1326 psid

3. Residual heat removal pump greater than or equal to 150 psid

g. By verifying the correct position of each mechanical stop for the following Emergency Core Cooling System throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 350,000 gallons of borated water.

b. Between 2400 and 2600 ppm of boron, and

c. A minimum water temperature of 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE when tested pursuant to Specification 4.0.5 by:

a. Verifying that each motor driven pump develops an equivalent discharge pressure of greater than or equal to 1240 psig at 60°F in recirculation flow.

b. Verifying that the steam turbine driven pump develops an equivalent discharge pressure of greater than or equal to 1180 psig at 60°F and at a flow of greater than or equal to 700 gpm when the secondary steam supply pressure is greater than 310 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

c. Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position.

d. Verifying that each automatic valve in the flow path is in the fully open position whenever the auxiliary feedwater system is placed in automatic control or when above 10% RATED THERMAL POWER. This requirement is not applicable for those portions of the auxiliary feedwater system being used intermittently to maintain steam generator water level.

e. Verifying at least once per 18 months during shutdown that each automatic valve in the flow path actuates to its correct position upon receipt of the appropriate engineered safety features actuation test signal required by Specification 3/4.3.2.

f. Verifying at least once per 18 months during shutdown that each auxiliary feedwater pump starts as designed automatically upon receipt of the appropriate engineered safety features actuation test signal required by Specification 3/4.3.2.

g. Verifying at least once per 18 months during shutdown that the unit cross-tie valves can cycle full travel. Following cycling, the valves will be verified to be in their closed positions.

5.4 REACTOR COOLANT SYSTEM (Continued)

a. In accordance with the code requirements specified in Section 4.1.6 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total contained volume of the reactor coolant system is approximately 12,466 cubic feet at 0% steam generator tube plugging and 11,551 cubic feet at 30% steam generator tube plugging at a nominal T_{avg} of 70°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements, with one exception. This exception is the CVCS boron makeup system and the BIT.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water.

b. A nominal 8.97 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. 1. The fuel assemblies will be classified as acceptable for Region 1, Region 2, or Region 3 storage based upon their assembly average burnup versus initial nominal enrichment. Cells acceptable for Region 1, Region 2, and Region 3 assembly storage are indicated in Figures 5.6-1 and 5.6-2. Assemblies that are acceptable for storage in Region 1, Region 2, and Region 3 must meet the design criteria that define the regions as follows:

2.1 SAFETY LIMITS

BASES

4 Loop Operation

Westinghouse Fuel
(15x15 OFA)

(WRB-1 Correlation)

	Typical Cell*	Thimble Cell**
Correlation Limit	1.17	1.17
Design Limit DNBR	1.23	1.22
Safety Analysis Limit DNBR	1.40	1.42

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the applicable design DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

*represents typical fuel rod

**represents fuel rods near guide tube

SAFETY LIMITS

BASES

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBRs will be greater than the applicable design limit DNBR value for each fuel type.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The source Range Channels will initiate a reactor trip at about 10^{+5} counts per second, unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Delta T

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Overpower Delta T

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The overpower delta T reactor trip provides protection or back-up protection for at power steamline break events. Credit was taken for operation of this trip in the steamline break mass/energy releases outside containment analysis. In addition, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the reactor protection system.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. The pressurizer high water level trip precludes water relief for the Uncontrolled RCCA Withdrawal at Power event.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% Delta k/k is initially required to control the reactivity transient and automatic ESF is assumed to be available. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% Delta k/k SHUTDOWN MARGIN provides adequate protection for this event.

The SHUTDOWN MARGIN requirements are based upon the limiting conditions described above and are consistent with FSAR safety analysis assumptions.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 2000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 plus or minus 100 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC at the beginning, and near the end of each fuel cycle is adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above the safety analysis limit during all normal operations and anticipated transients. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (31 percent of RATED THERMAL POWER).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE. Three loops are required to be OPERABLE and to operate if the control rods are capable of withdrawal and the reactor trip breakers are closed. The requirement assures adequate DNBR margin in the event of an uncontrolled rod withdrawal in this mode.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 152°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a loss of coolant accident or a steam line rupture. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following a LOCA assuming mixing of the RWST, RCS, ECCS water, and other sources of water that may eventually reside in the sump, with all control rods assumed to be out. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 70°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

The maximum peak pressure resulting from a LOCA event is calculated to be 11.49 psig, which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 60°F will limit the peak pressure to 11.49 psig which is less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

PROPOSED CHANGES TO THE
DONALD C. COOK NUCLEAR PLANT UNIT NO. 2
TECHNICAL SPECIFICATIONS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{AVG} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

*See Special Test Exception 3.10.1.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3

a. One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

b. One charging flow path associated with support of Unit 1 shutdown functions shall be available.*

APPLICABILITY: Specification 3.1.2.3.a. - MODES 5 and 6
Specification 3.1.2.3.b. - At all times when Unit 1 is in MODES 1, 2, 3, or 4.

ACTION:

a. With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.**

b. With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 152°F, unless the reactor vessel head is removed, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within one hour.

c. The provisions of Specification 3.0.3 are not applicable.

d. In addition to the above, when Specification 3.1.2.3.b is applicable and the required flow path is not available, return the required flow path to available status within 7 days, or provide equivalent shutdown capability in Unit 1 and return the required flow path to available status within the next 60 days, or have Unit 1 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours.

e. The requirements of Specification 3.0.4 are not applicable when Specification 3.1.2.3.b applies.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above-required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a differential pressure of greater than or equal to 2290 psid when tested pursuant to Specification 4.0.5.

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*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F.

**For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a differential pressure of $\geq$ 2290 psid when tested pursuant to Specification 4.0.5.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum usable borated water volume of 4300 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum usable borated water volume of 90,000 gallons,
2. A minimum boron concentration of 2400 ppm, and
3. A minimum solution temperature of 70°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the contained borated water volume, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water.

*For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by Specification 3.1.2.7.b.2.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 5650 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 350,000 gallons of water,
2. Between 2400 and 2600 ppm of boron, and
3. A minimum solution temperature of 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion. †

e. At least once per 18 months, during shutdown, by:†

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection signal:

a) Centrifugal charging pump

b) Safety injection pump

c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1.	Centrifugal charging pump	Greater than or equal to 2290 psid
2.	Safety Injection pump	Greater than or equal to 1385 psid
3.	Residual heat removal pump	Greater than or equal to 160 psid

g. By verifying the correct position of each mechanical stop for the following Emergency Core Cooling System throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

†The provisions of Technical Specification 4.0.8 are applicable.

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 350,000 gallons of borated water,

b. Between 2400 and 2600 ppm of boron, and

c. A minimum water temperature of 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% Delta k/k is initially required to control the reactivity transient and automatic ESF is assumed to be available.

With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% Delta k/k SHUTDOWN MARGIN provides adequate protection for this event.

The SHUTDOWN MARGIN requirements are based upon the limiting conditions described above and are consistent with FSAR safety analysis assumptions.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 2000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a LOCA or steam line rupture. The limits of RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following a LOCA assuming mixing of the RWST, RCS, ECCS water, and other sources of water that may eventually reside in the sump, with all control rods assumed to be out. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 80°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

The maximum peak pressure resulting from a LOCA event is calculated to be 11.49 psig, which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 60°F will limit the peak pressure to 11.49 psig which is less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

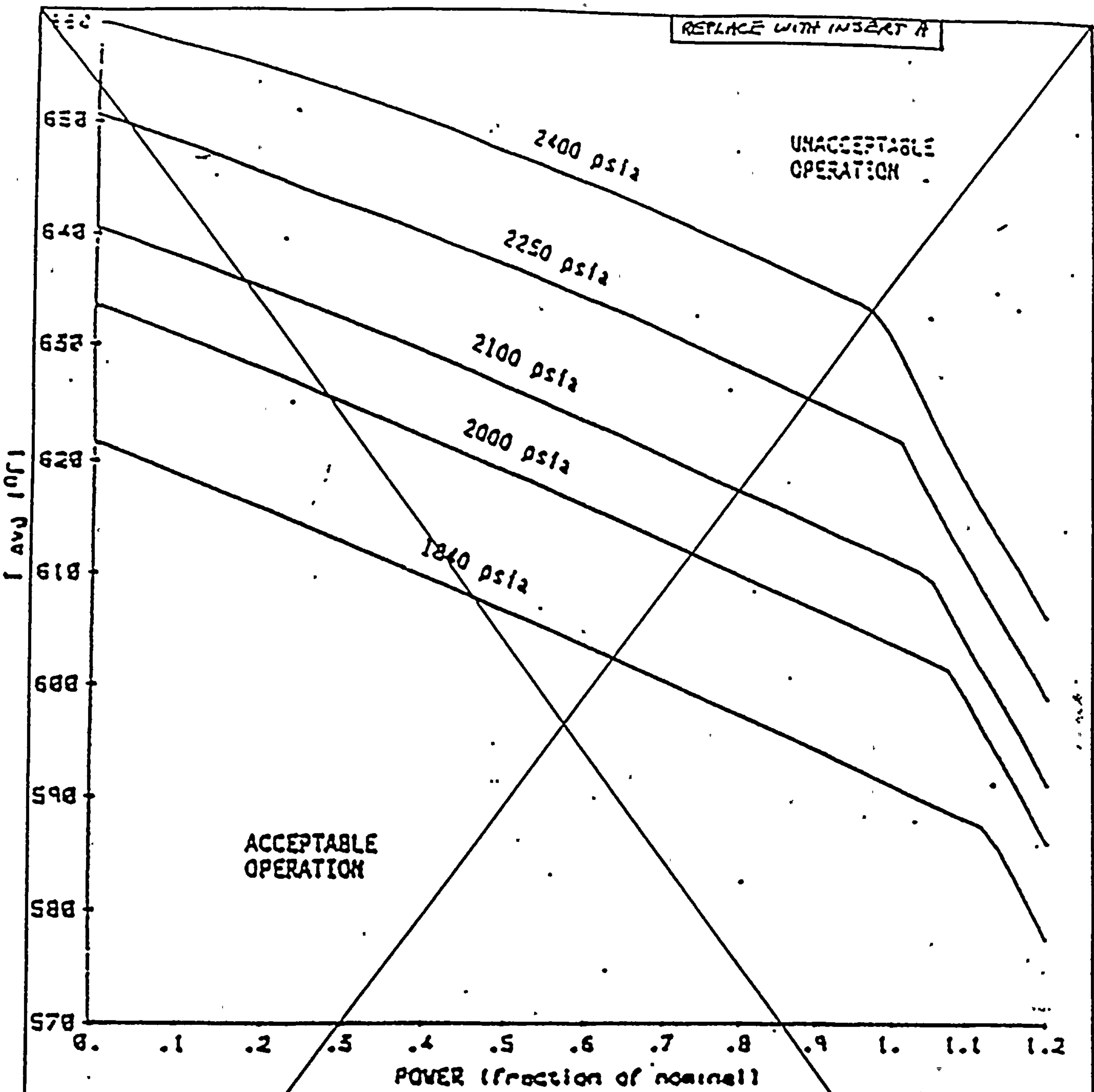
3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

ATTACHMENT 3 TO AEP:NRC:1207

CURRENT PAGES MARKED-UP TO REFLECT PROPOSED CHANGES TO THE
DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2
TECHNICAL SPECIFICATIONS

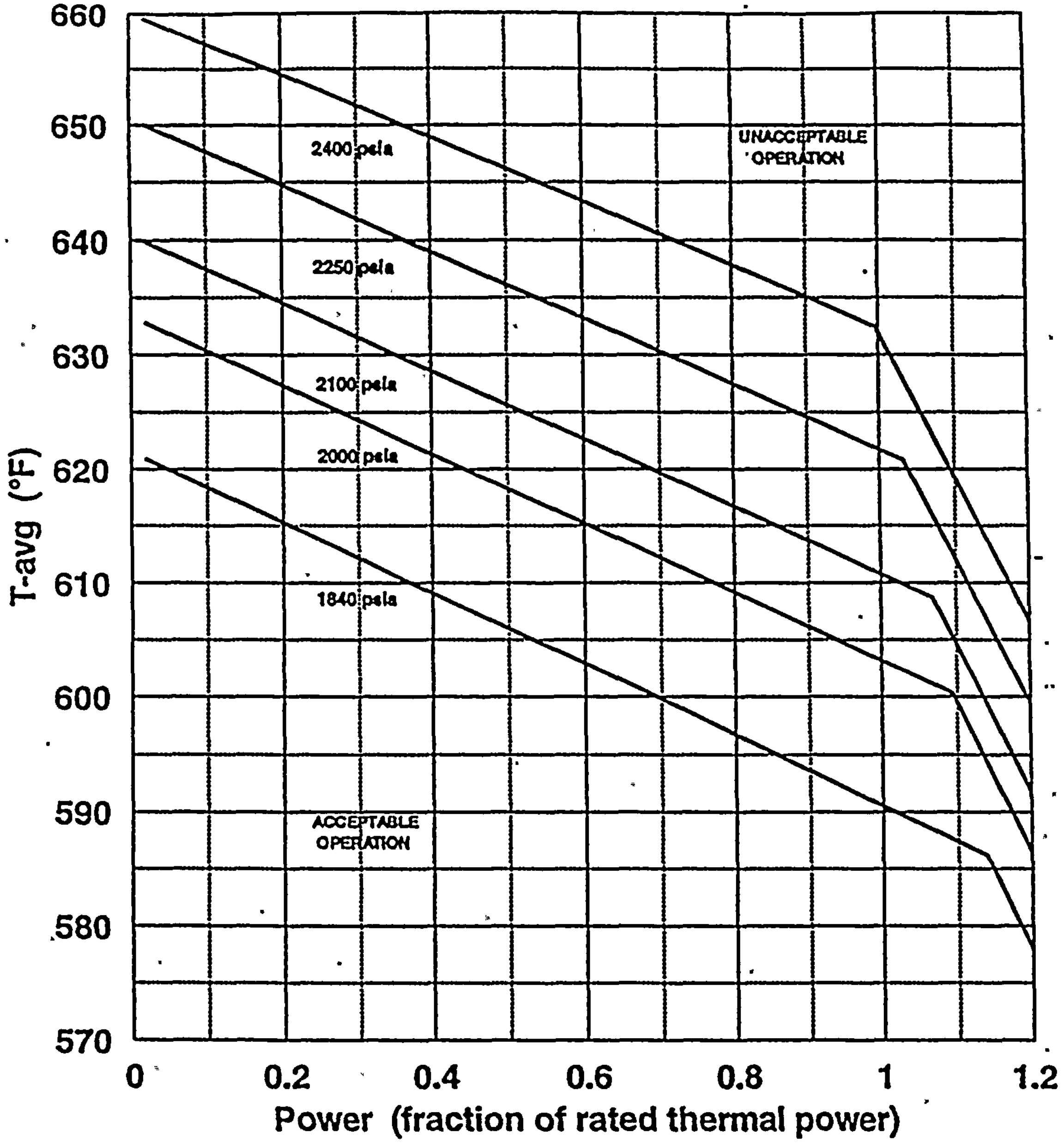
CURRENT PAGES MARKED-UP TO REFLECT PROPOSED CHANGES TO THE
DONALD C. COOK NUCLEAR PLANT UNIT NO. 1
TECHNICAL SPECIFICATIONS



PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T-AVG IN DEGREES F)
1840	(0.0, 622.1), (1.13, 587.3), (1.20, 577.5)
2000	(0.0, 633.8), (1.08, 601.4), (1.20, 586.0)
2100	(0.0, 640.8), (1.06, 609.8), (1.20, 591.3)
2250	(0.0, 650.7), (1.02, 621.9), (1.20, 598.9)
2400	(0.0, 660.1), (0.98, 633.7), (1.20, 606.2)

FIGURE 2.1-1 REACTOR CORE SAFETY LIMITS

INSERT A



PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T-AVG IN °F)		
1840	(0.02, 620.86),	(1.136, 586.17),	(1.2, 577.94)
2000	(0.02, 632.79),	(1.094, 600.31),	(1.2, 586.52)
2100	(0.02, 639.85),	(1.068, 608.72),	(1.2, 591.77)
2250	(0.02, 649.96),	(1.031, 620.83),	(1.2, 599.40)
2400	(0.02, 659.52),	(0.996, 632.42),	(1.2, 606.63)

FIGURE 2.1-1 REACTOR CORE SAFETY LIMITS

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - less than or equal to 25% of RATED THERMAL POWER High Setpoint - less than or equal to 109% of RATED THERMAL POWER	Low Setpoint - less than or equal to 26% of RATED THERMAL POWER High Setpoint - less than or equal to 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
5. Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3×10^5 counts per second
7. Overtemperature Delta T	See Note 1	See Note 3
8. Overpower Delta T	See Note 2	See Note 4
9. Pressurizer Pressure -- Low	Greater than or equal to 1875 psig	Greater than or equal to 1865 psig
10. Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11. Pressurizer Water Level -- High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12. Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

*Design flow is ~~90,400 gpm per loop.~~ 1/4 Reactor Coolant System total flow rate from Table 3.2-1.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left[\frac{1+\tau_1 S}{1+\tau_2 S}\right](T - T') + K_3 (P - P') - f_1(\Delta I)\right]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER (≤ ~~567.8~~ 576.3 °F)

P = Pressurizer pressure, psig

P' = Indicated RCS nominal operating pressure (2235 psig or 2085 psig)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1, τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 22 secs. τ_2 = 4 secs.

S = Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Operation with 4 Loops

$K_1 = \cancel{1.12}$ 1.17

$K_2 = 0.0230$

$K_3 = 0.00110$

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -37 percent and +~~2~~ 3 percent, $f_1(\Delta I)=0$ (where q_t and q_b are percent RATED TH[illegible]WER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) For each percent that the magnitude of $(q_t - q_b)$ exceeds -37 percent, the ΔT trip setpoint shall be automatically reduced by 0.33 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +~~2~~ 3 percent, the ΔT trip setpoint shall be automatically reduced by ~~2.17~~ 2.34 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 (T-T'') - f_2(\Delta I)\right]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, ^oF

T'' = Indicated T_{avg} at RATED THERMAL POWER ~~($\leq$ 567.8 ^oF)~~ 563.0

K_4 = 1.083

K_5 = $0.0177/^oF$ for increasing average temperature and 0 for decreasing average temperature

K_6 = 0.0015 for $T > T''$; $K_6 = 0$ for $T \leq T''$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator

$f_2(\Delta I)$ = 0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than ~~3.2~~ 3.4 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.1 2.5 percent ΔT span.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - TAVG GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3 ~~1.6%~~ Delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than 1.3 ~~1.6%~~ Delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

1.3 4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~1.6%~~ Delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5.

*See Special Test Exception 3.10.1.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3

a. One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

b. One charging flowpath associated with support of Unit 2 shutdown functions shall be available.*

APPLICABILITY: Specification 3.1.2.3.a. - MODES 5 and 6
Specification 3.1.2.3.b. - At all times when Unit 2 is in MODES 1, 2, 3, or 4.

ACTION:

a. With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.**

b. With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 152°F, unless the reactor vessel head is removed, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within one hour.

c. The provisions of Specification 3.0.3 are not applicable.

d. In addition to the above, when Specification 3.1.2.3.b is applicable and the required flow path is not available, return the required flow path to available status within 7 days, or provide equivalent shutdown capability in Unit 2 and return the required flow path to available status within the next 60 days, or have Unit 2 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours.

e. The requirements of Specification 3.0.4 are not applicable when Specification 3.1.2.3.b applies.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a ~~discharge~~ DIFFERENTIAL pressure of greater than or equal to ~~2390~~ 2290 ~~psig~~ psid when tested pursuant to Specification 4.0.5.

*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F.

**For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

DIFFERENTIAL

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a ~~discharge~~ pressure of greater than or equal to ~~2405~~ 2290 ~~psig~~ psid when tested pursuant to Specification 4.0.5.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum usable borated water volume of 4300 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum usable borated water volume of 90,000 gallons,
2. A minimum boron concentration of 2400 ppm, and
3. A minimum solution temperature of ~~80~~ 70°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the water level volume of the tank, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water.

* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

BORATED WATER SOURCES - OPERATIONS

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum usable borated water volume of 5650 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 350,000 gallons of water,
2. Between 2400 and 2600 ppm of boron, and
3. A minimum solution temperature of ~~80~~ 70 °F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

TABLE 3.2-1
DNB PARAMETERS

PARAMETER	LIMITS 4 Loops in Operation at RATED THERMAL POWER
Reactor Coolant System Tavg	≤ ~~570.9°F*~~ 579.3°F
Pressurizer Pressure	≥ 2050 psig**
Reactor Coolant System Total Flow Rate	≥ ~~361,600~~ 341,100 gpm***

---

* Indicated average of at least three OPERABLE instrument loops.

** Limit not applicable during either a THERMAL POWER ramp increase in excess of 5 percent RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10 percent RATED THERMAL POWER.

*** Indicated value.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

DELETE

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Flow in Two Steam Lines-High					
Four Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line	1, 2, 3##	14*
Three Loops Operating	2/operating steam line	1###/any operating steam line	1/operating steam line	3##	15
COINCIDENT WITH EITHER					
T_{avg}--Low-Low					
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops	1, 2, 3##	14*
Three Loops Operating	1 T_{avg}/operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops	3##	15
OR, COINCIDENT WITH					
f. Steam Line Pressure-Low					
Four Loops Operating	1 pressure/loop	2 pressures any loops	1 pressure any 3 loops	1, 2, 3##	14*
Three Loops Operating	1 pressure/operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops	3##	15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
COINCIDENT WITH ~~EITHER~~					
T_{avg} -- Low-Low					
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops	1, 2, 3##	14*
Three Loops Operating	1 T_{avg}/ operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops	3##	15
~~OR, COINCIDENT WITH~~					
e. Steam Line Pressure-Low					
Four Loops Operating	1 pressure/ loop	2 pressures any loops	1 pressure any 3 loops	1, 2, 3##	14*
Three Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops	3##	15
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1,2,3	14*

ENGINEERED SAFETY FEATURES INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-11	With 2 of 3 pressurizer pressure channels greater than or equal to 1915 psig.	P-11 prevents or defeats manual block of safety injection actuation on low pressurizer pressure.
P-12	With 2 of 4 T_{avg} channels less than or equal to Setpoint. Setpoint greater than or equal to 541°F	P-12 allows the manual block of safety injection ACTUATION ON ~~from high steam flow coincident with either~~ low steam line pressure. ~~or low-low T_{avg}. P-12 in coincidence with high steam flow will result in a steam line isolation.~~ CAUSES STEAM LINE ISOLATION ON HIGH STEAM FLOW. ~~P-12~~ Affects steam dump blocks. With 3 of 4 T_{avg} channels above the reset ~~value, the manual block of safety injection from high steam flow coincident with either low steam line pressure or low-low T_{avg} is prevented or defeated.~~ POINT, PREVENTS OR DEFEATS THE MANUAL BLOCK OF SAFETY INJECTION ACTUATION ON LOW STEAM LINE PRESSURE.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN FEEDWATER PUMPS		
a. Manual Initiation	------------- See Functional Unit 9 -------------	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	Less than or equal to 1.1 psig	Less than or equal to 1.2 psig
d. Pressurizer Pressure--Low	Greater than or equal to 1815 psig	Greater than or equal to 1805 psig
e. Differential Pressure Between Steam Lines--High	Less than or equal to 100 psi	Less than or equal to 112 psi
f. ~~Steam Flow in Two Steam Lines--High Coincident with Tavg --Low-Low or~~ Steam Line Pressure--Low	~~Less than or equal to 1.42×10^6 lbs/hr from 0% load to 20% load. Linear from 1.42×10^6 lbs/hr at 20% load to 3.88×10^6 lbs/hr at 100% load~~	~~Less than or equal to 1.56×10^6 lbs/hr from 0% load to 20% load. Linear from 1.56×10^6 lbs/hr at 20% load to 3.93×10^6 lbs/hr at 100% load~~
	~~T_{avg} greater than or equal to 541°F~~	~~T_{avg} greater than or equal to 539°F~~
	G ~~g~~reater than or equal to 500 psig steam line pressure	G ~~g~~reater than or equal to 480 psig steam line pressure

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Containment Radioactivity--High Train A (VRS-1101, ERS-1301, ERS-1305)	See Table 3.3-6	Not Applicable
3. Containment Radioactivity--High Train B (VRS-1201, ERS-1401, ERS-1405)	See Table 3.3-6	Not Applicable
4. STEAM LINE ISOLATION		
a. Manual	------------- See Functional Unit 9 -------------	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	Less than or equal to 2.9 psig	Less than or equal to 3 psig
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low ~~or Steam Line Pressure--Low~~	Less than or equal to 1.42×10^6 lbs/hr from 0% load to 20% load. Linear from 1.42×10^6 lbs/hr at 20% load to 3.88×10^6 lbs/hr at 100% load.	Less than or equal to 1.56×10^6 lbs/hr from 0% load to 20% load. Linear from 1.56×10^6 lbs/hr at 20% load to $3.93\ 10^6$ lbs/hr at 100% load.
	T_{avg} greater than or equal to 541°F	T_{avg} greater than or equal to 539°F
e. STEAM LINE PRESSURE-LOW	Greater than or equal to 500 psig steam line pressure	Greater than or equal to 480 psig steam line pressure
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level--High-High	Less than or equal to 67% of narrow-range instrument span each steam generator	Less than or equal to 68% of narrow-range instrument span each steam generator

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Manual Initiation	------------------ See Functional Unit 9 ------------------				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure-High	S	R	M(3)	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	S	R	M	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	N.A.	1, 2, 3
f. ~~Steam Flow in Two Steam Lines--High Coincident with Tavg --Low or Steam Line~~ STEAM LINE Pressure--Low	S	R	M	N.A.	1, 2, 3
2. CONTAINMENT SPRAY					
a. Manual Initiation	------------------ See Functional Unit 9 ------------------				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure-High-High	S	R	M(3)	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURED ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION					
a. Manual	--------------------	See Functional Unit 9	----------------		
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3
c. Containment Pressure--High-High	S	R	M(3)	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with Tavg--Low-Low ~~Pressure--Low~~	S	R	M	N.A.	1, 2, 3
e. STEAM LINE PRESSURE-LOW	S	R	M	N.A.	1,2,3
5. TURBINE TRIP AND FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	S	R	M	N.A.	1, 2, 3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level--Low-Low	S	R	M	N.A.	1, 2, 3
b. 4 kv Bus Loss of Voltage	S	R	M	N.A.	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	N.A.	1, 2, 3
d. Loss of Main Feed Pumps	N.A.	N.A.	R	N.A.	1, 2

REACTOR COOLANT SYSTEM

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± ~~1~~ 3%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE:

a. Immediately suspend all operations involving positive reactivity changes** and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. Immediately render all Safety Injection pumps and all but one charging pump inoperable by removing the applicable motor circuit breakers from the electric power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Requirement 4.4.3.

---

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 3 ~~1%~~.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional surveillance requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

   1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.

   2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal.

   2. Verifying that each of the following pumps start automatically upon receipt of a safety injection signal:

      a) Centrifugal charging pump

      b) Safety injection pump

      c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated ~~discharge~~ DIFFERENTIAL pressure on recirculation flow when tested pursuant to Specification 4.0.5.

   1. Centrifugal charging pump greater than or equal to ~~2405~~ 2290 ~~psig~~ psid

   2. Safety injection pump greater than or equal to ~~1409~~ 1326 ~~psig~~ psid

   3. Residual heat removal pump greater than or equal to ~~190~~ 150 ~~psig~~ psid

g. By verifying the correct position of each mechanical stop for the following Emergency Core Cooling System throttle valves:

   1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 350,000 gallons of borated water.

b. Between 2400 and 2600 ppm of boron, and

c. A minimum water temperature of ~~80~~ 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE when tested pursuant to Specification 4.0.5 by:

a. Verifying that each motor driven pump develops an equivalent discharge pressure of greater than or equal to ~~1375~~ 1240 psig at 60°F in recirculation flow.

b. Verifying that the steam turbine driven pump develops an equivalent discharge pressure of greater than or equal to ~~1285~~ 1180 psig at 60°F and at a flow of greater than or equal to 700 gpm when the secondary steam supply pressure is greater than 310 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

c. Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position.

d. Verifying that each automatic valve in the flow path is in the fully open position whenever the auxiliary feedwater system is placed in automatic control or when above 10% RATED THERMAL POWER. This requirement is not applicable for those portions of the auxiliary feedwater system being used intermittently to maintain steam generator water level.

e. Verifying at least once per 18 months during shutdown that each automatic valve in the flow path actuates to its correct position upon receipt of the appropriate engineered safety features actuation test signal required by Specification 3/4.3.2.

f. Verifying at least once per 18 months during shutdown that each auxiliary feedwater pump starts as designed automatically upon receipt of the appropriate engineered safety features actuation test signal required by Specification 3/4.3.2.

g. Verifying at least once per 18 months during shutdown that the unit cross-tie valves can cycle full travel. Following cycling, the valves will be verified to be in their closed positions.

DESIGN FEATURES

a. In accordance with the code requirements specified in Section 4.1.6 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total contained volume of the reactor coolant system is approximately 12,466 cubic feet at 0% steam generator tube plugging and 11,551 cubic feet at 30% steam generator tube plugging ~~12,612 ± 100 cubic feet~~ at a nominal T_{avg} of 70°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements, with one exception. This exception is the CVCS boron makeup system and the BIT.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water.

b. A nominal 8.97 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. 1. The fuel assemblies will be classified as acceptable for Region 1, Region 2, or Region 3 storage based upon their assembly average burnup versus initial nominal enrichment. Cells acceptable for Region 1, Region 2, and Region 3 assembly storage are indicated in Figures 5.6-1 and 5.6-2. Assemblies that are acceptable for storage in Region 1, Region 2, and Region 3 must meet the design criteria that define the regions as follows:

2.1 SAFETY LIMITS

BASES

4 Loop Operation

Westinghouse Fuel
(15x15 OFA)

(WRB-1 Correlation)

	Typical Cell*	Thimble Cell**
Correlation Limit	1.17	1.17
Design Limit DNBR	~~1.33~~ 1.23	~~1.32~~ 1.22
Safety Analysis Limit DNBR	~~1.45~~ 1.40	~~1.45~~ 1.42

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the applicable design DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

----

*represents typical fuel rod
**represents fuel rods near guide tube

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBR's will be greater than the applicable design limit DNBR value for each fuel type.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second, unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature delta T

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. ~~The reference average temperature (T') and the reference operating pressure (P') are set equal to the full power indicated Tavg and the nominal RCS operating pressure, respectively, to ensure protection of the core limits and to preserve the actuation time of the Overtemperature delta T trip for the range of full power average temperatures assumed in the safety analyses.~~ With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

DELETE

LIMITING SAFETY SYSTEM SETTINGS

BASES

Overpower Delta T

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. ~~The reference average temperature (T') is set equal to the full power indicated Tavg to ensure fuel integrity during overpower conditions for the range of full power average temperatures assumed in the safety analysis.~~ DELETE The overpower delta T reactor trip provides protection or back-up protection for at power steamline break events. Credit was taken for operation of this trip in the steam line break mass/energy releases outside containment analysis. In addition, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the reactor protection system.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. The pressurizer high water level trip precludes water relief for the Uncontrolled RCCA Withdrawal at Power event.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of ~~1.6%~~ 1.3 Delta k/k is initially required to control the reactivity transient and automatic ESF is assumed to be available. With Tavg less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% Delta k/k SHUTDOWN MARGIN provides adequate protection for this event.

The SHUTDOWN MARGIN requirements are based upon the limiting conditions described above and are consistent with FSAR safety analysis assumptions.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 2000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 plus or minus 100 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC at the beginning, and near the end of each fuel cycle is adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron

BASES

3/4.4.1 REACTOR COOLANT LOOPS

THE SAFETY ANALYSIS LIMIT

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above ~~1.69~~ during all normal operations and anticipated transients. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (31 percent of RATED THERMAL POWER).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE. Three loops are required to be OPERABLE and to operate if the control rods are capable of withdrawal and the reactor trip breakers are closed. The requirement assures adequate DNBR margin in the event of an uncontrolled rod withdrawal in this mode.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 152°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCP's to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a loss of coolant accident or a steam line rupture. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following ~~mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly~~. These assumptions are consistent with the LOCA analyses. INSERT A

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 70°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50. ~~The value of the minimum RWST temperature in Technical Specification 3.5.5 has been conservatively changed to 80°F to increase the consistency between Units 1 and 2. The lower RWST temperature results in lower containment pressure from containment spray and safeguards flow assumed to exit the break. Lower containment pressure results in increased flow resistance of steam exiting the core thereby slowing reflood and increasing PCT.~~

INSERT A

a LOCA assuming mixing of the RWST, RCS ECCS water, and other sources of water that may eventually reside in the sump, with all control rods assumed to be out.

CONTAINMENT SYSTEMS

BASES

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

The 11.49 maximum peak pressure resulting from a LOCA event is calculated to be ~~11.89~~ psig, which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 60°F will limit the peak pressure to 11.49 ~~11.39~~ psig which is less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

CURRENT PAGES MARKED-UP TO REFLECT PROPOSED CHANGES TO THE
DONALD C. COOK NUCLEAR PLANT UNIT NO. 2
TECHNICAL SPECIFICATIONS

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{AVG} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

- 1.3%

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to ~~1.6%~~ Delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

1.3%

With the SHUTDOWN MARGIN less than ~~1.6%~~ Delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~1.6%~~ Delta k/k:

1.3%

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

*See Special Test Exception 3.10.1

REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3

a. One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

b. One charging flow path associated with support of Unit 1 shutdown functions shall be available.*

APPLICABILITY: Specification 3.1.2.3.a. - MODES 5 and 6
Specification 3.1.2.3.b. - At all times when Unit 1 is in MODES 1, 2, 3, or 4.

ACTION:

a. With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.**

b. With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 152°F, unless the reactor vessel head is removed, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within one hour.

c. The provisions of Specification 3.0.3 are not applicable.

d. In addition to the above, when Specification 3.1.2.3.b is applicable and the required flow path is not available, return the required flow path to available status within 7 days, or provide equivalent shutdown capability in Unit 1 and return the required flow path to available status within the next 60 days, or have Unit 1 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours.

e. The requirements of Specification 3.0.4 are not applicable when Specification 3.1.2.3.b applies.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above-required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a ~~discharge~~ differential pressure of greater than or equal to ~~2390 psig~~ 2290 psid when tested pursuant to Specification 4.0.5.

*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F.

*For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a ~~discharge pressure~~ of ≥ ~~2405~~ psig when tested pursuant to Specification 4.0.5.

differential

2290 psid

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracting with:

1. A minimum usable borated water volume of 4300 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum usable borated water volume of 90,000 gallons,
2. A minimum boron concentration of 2400 ppm, and
3. A minimum solution temperature of ~~80°F.~~ 70°F

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the contained borated water volume, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water.

*For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by Specification 3.1.2.7.b.2.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 5650 gallons,

2. Between 20,000 and 22,500 ppm of boron, and

3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 350,000 gallons of water,

2. Between 2400 and 2600 ppm of boron, and

3. A minimum solution temperature of ~~80°F~~. 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.†

e. At least once per 18 months, during shutdown, by:†

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

a) Centrifugal charging pump

b) Safety injection pump

c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated ~~discharge~~ differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1. Centrifugal charging pump Greater than or equal to ~~2405 psig~~ 2290 psid

2. Safety Injection pump Greater than or equal to ~~1409 psig~~ 1385 psid

3. Residual heat removal pump Greater than or equal to ~~190 psig~~ 160 psid

g. By verifying the correct position of each mechanical stop for the following Emergency Core Cooling System throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

† The provisions of Technical Specification 4.0.8 are applicable.

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 350,000 gallons of borated water,

b. Between 2400 and 2600 ppm of boron, and

c. A minimum water temperature of ~~80°F~~. 90°F

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of ~~1.6%~~ 1.3% Delta k/k is initially required to control the reactivity transient and automatic ESF is assumed to be available.

With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% Delta k/k SHUTDOWN MARGIN provides adequate protection for this event.

The SHUTDOWN MARGIN requirements are based upon the limiting conditions described above and are consistent with FSAR safety analysis assumptions.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 2000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a LOCA or steam line rupture. The limits of RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following ~~mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly~~. These assumptions are consistent with the LOCA analyses. INSERT A

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of ~~80°F~~ 70°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

INSERT A

a LOCA assuming mixing of the RWST, RCS ECCS water, and other sources of water that may eventually reside in the sump, with all control rods assumed to be out.

BASES

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

11.49 The maximum peak pressure ~~expected to be obtained~~ resulting from a LOCA event is ~~9.4~~ psig, ~~The limit of 0.3 psig for initial positive containment pressure will limit the total pressure to 9.7 psig which is less than the design pressure and is consistent with the accident analyses.~~ which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 60°F will limit the peak pressure to ~~9.4~~ 11.49 psig which is less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

ATTACHMENT 4 TO AEP:NRC:1207

SUMMARY DESCRIPTION OF PROPOSED
INCREASED STEAM GENERATOR TUBE PLUGGING
TECHNICAL SPECIFICATIONS

Key for Summary Table

Page	Technical Specification Page	
Section	Technical Specification	
Group	Related Groups Discussed in Attachment 1, Description of Proposed Changes and 10 CFR 50.92 Significant Hazards Consideration Analysis	
	SGTP	Group 1, Changes Directly Related to Increased Steam Generator Tube Plugging
	Margin	Group 2, Changes Proposed to Increase unit 1 Operating Margin
	Both	Group 3, Changes Proposed to Increase the Operating Margin of Both units.
	Admin	Group 4, Administrative Change
Description	A Brief Description of Each Proposed Change	
Remarks	Brief Comments with a Cross Reference to the Analyses	

Page	Section	Group	Description	Remarks
2-2	Figure 2.1-1	Margin	Revise Reactor Core Safety Limits	The new thermal design is discussed in Section 3.3.2.1 of Attachment 6, WCAP 14285.
2-5	Table 2.2-1 Footnote	SGTP	Redefine design flow in footnote of Table 2.2-1 to be 1/4 MMF.	MMF for DNB is discussed in Section 3.3.2.1 of Attachment 6, WCAP 14285. T.S. MMF is 1.025 times thermal design flow (TDF). TDF is specified in Section 3.3.3.1 of WCAP 14285. The MMF employed in the DNB analysis is 1.019 times TDF. This was done to support a range of MMF's from 1.019 to 1.025 times TDF as indicated in Section 2.1 of WCAP 14285. Design flow in current technical specification Table 2.2-1 is MMF/4.
2-7	Table 2.2-1	Margin	The upper limit on T' increased to reflect analyses.	The OTDT trip is discussed in Section 3.3.2.1 of WCAP 14285. Details, including T', of the analyzed setpoint are in Table 3.3-3 of WCAP 14285.
2-8	Table 2.2-1	Margin	Decrease K1 from 1.32 to 1.17.	This change and the next change to f(ΔI) are being requested to optimize operating margin. Some load rejection capability is sacrificed for instrumentation margin, increased allowance for core burndown effects on hot leg streaming, and an increase in the positive ΔI break point for the f(ΔI) penalty. The OTDT trip is discussed in Section 3.3.2.1 of WCAP 14285. Details of the analyzed setpoint are in Table 3.3-3 of WCAP 14285.
2-8	Table 2.2-1	Margin	Change f(ΔI) to increase the region of positive ΔI which is without penalty.	See previous discussion of K1 decrease.
2-9	Table 2.2-1	Margin	Decrease the upper limit on T'' to reflect analyses.	Cook Nuclear Plant unit 1 is operated in a low temperature, low pressure mode to extend the life of the steam generators. Therefore, the analysis of the OPDT setpoint was analyzed with a low upper limit on T'' to convert unused margin to operating margin. The OPDT trip is discussed in Section 3.3.2.1 of WCAP 14285. Details, including T'', of the analyzed setpoint are in Table 3.3-3 of WCAP 14285.
2-9	Table 2.2-1	Margin	Change the allowable values in note 2 and 3.	The values indicated in the markups of Attachment 3 and in the proposed technical specifications of Attachment 2 were calculated by our organization.

Page	Section	Group	Description	Remarks
UNIT 1 3/4 1-1	UNIT 1 Section 3.1.1.1 4.1.1.1.1	Both	Reduce required shutdown margin.	The new value is supported by analyses. Unit 1: For core response steam break (CRSB), see Section 3.3.5.6 of WCAP 14285. For steamline mass and energy release (SM&E) inside containment, see Section 3.5.4.2 of WCAP 14285. For SM&E outside containment, see Section 3.3.4.7 of WCAP 14285.
UNIT 2 3/4 1-1	UNIT 2 Section 3.1.1.1 4.1.1.1			Unit 2: For CRSB, see Section B.3.11 of the Vantage 5 Reload Transition Safety Report for Cook Nuclear Plant unit 2 (RTSR). For SM&E inside containment, see Section 3.5.4.2 of WCAP 14285. For SM&E outside containment, see Section 3.3.4.7 of WCAP 14285. A copy of Section B.3.11 has been included in Attachment 7 to this submittal.
UNIT 1 3/4 1-11	UNIT 1 Section 4.1.2.3.1	Both	Change CCP surveillances to be consistent with 10% degradation. Change pump surveillance requirements from discharge pressure to differential pressure.	The new surveillance criterion is supported by analyses. The surveillance criteria are presented in Section 3.10.1.1 of WCAP 14285. The value given for CCP applies to both units Unit 1: For LOCA see Sections 3.1.1 and 3.1.2 of WCAP 14285. For core response steam break (CRSB), see Section 3.3.5.6 of WCAP 14285. For steamline mass and energy release (SM&E) inside containment, see Section 3.5.4.2 of WCAP 14285. For SM&E outside containment, see Section 3.3.4.7 of WCAP 14285.
UNIT 2 3/4 1-11	UNIT 2 Section 4.1.2.3.1			Unit 2: The technical specification changes supported by the RTSR are delineated in Table 6.1 of the RTSR. For LOCA see Sections C.3.1.2 and C.3.2 of the RTSR. For CRSB, see Section B.3.11 of the RTSR. For SM&E inside containment, see Section 3.5.4.2 of WCAP 14285. For SM&E outside containment, see Section 3.3.4.7 of WCAP 14285. Copies of Table 6.1 and Sections B.3.11, C.3.1.2, and C.3.2 have been included in Attachment 7 to this submittal.

Page	Section	Group	Description	Remarks
UNIT 1 3/4 1-12	UNIT 1 Section 4.1.2.4	Both	Change CCP surveillances to be consistent with 10% degradation. Change pump surveillance requirements from discharge pressure to differential pressure.	See previous discussion of CCP degradation increase for page 3/4 1-11.
UNIT 2 3/4 1-12	UNIT 2 Section 4.1.2.4			
UNIT 1 3/4 1-15	UNIT 1 Section 3.1.2.7	Both	Reduce the Minimum RWST temperature to 70°F.	The mode 5 and 6 minimum RWST temperature is conservatively maintained at the same value as that required for modes 1, 2, 3, and 4.
UNIT 2 3/4 1-15	UNIT 2 Section 3.1.2.7			
UNIT 1 3/4 1-16	UNIT 1 Section 3.1.2.8	Both	Reduce the Minimum RWST temperature to 70°F.	The new value is supported by the core response LBLOCA. Unit 1: See Section 3.1.1, Table 3.1-2, of WCAP 14285.
UNIT 2 3/4 1-16	UNIT 2 Section 3.1.2.8			Unit 2: See Table C.3.1-2 of the RTSR. A copy of this table is included in Attachment 7.
3/4 2-14	Table 3.2-1	SGTP	Increase DNB temperature limit	The calculation of the new DNB temperature limit is described in Section 1.2 of WCAP 14285 under the heading "DNB Parameters, RCS Tavg and RCS Flow". The readability error is 2.1°F. The resulting DNB temperature limit is 579.3°F.
3/4 2-14	Table 3.2-1	SGTP	Reduce MMF limit	See the discussion for page 2-5.

Page	Section	Group	Description	Remarks
3/4 3-17	Table 3.3-3	Margin	Change ESF actuation logic to support 12% AFW Pump degradation	The revised part of Table 3.3-3 incorporates the safeguards logic used in Cook Nuclear Plant unit 2. This will allow for the use of 12% auxiliary feedwater head degradation (AFW) in unit 1. All analyses, other than an "information only" feedline break analyses, have been performed using the flow from an AFW pump with 12% head degradation. The safeguards logic itself will be modified via design change prior to implementation of these revised T/S pages (i.e. before unit 1, cycle 16). After this modification, the unit 2 feedline break analysis using 12% degraded flow will bound unit 1. The evaluation which shows that the unit 2 feedwater line break will bound unit 1 is discussed in Section 3.3.4.8 of WCAP 14285.
3/4 3-21	Table 3.3-3	Margin	Change ESF actuation logic to support 12% AFW Pump degradation	See discussion for page 3/4 3-17
3/4 3-23a	Table 3.3-3	Margin	Change ESF actuation logic to support 12% AFW Pump degradation	See discussion for page 3/4 3-17
3/4 3-24	Table 3.3-4	Margin	Change ESF actuation logic to support 12% AFW Pump degradation	See discussion for page 3/4 3-17
3/4 3-26	Table 3.3-4	Margin	Change ESF actuation logic to support 12% AFW Pump degradation	See discussion for page 3/4 3-17
3/4 3-31	Table 4.3-2	Margin	Change ESF actuation logic to support 12% AFW Pump degradation	See discussion for page 3/4 3-17

Page	Section	Group	Description	Remarks
3/4 3-33	Table 4.3-2	Margin	Change ESF actuation logic to support 12% AFW Pump degradation	See discussion for page 3/4 3-17
3/4 4-4	Section 3.4.2	Margin	Increase Pressurizer Valve Tolerance	The Non-LOCA accidents were reanalyzed or reevaluated based on a pressurizer valve setpoint tolerance of 3%. This is noted in section 1.1 and 3.3.2.3 of WCAP 14285.
3/4 4-5	Section 3.4.3	Margin	Increase Pressurizer Valve Tolerance	See discussion for page 3/4 4-5.
Unit 1 3/4 5-5	Section 4.5.2.f.2 4.5.2.f.3	Margin	Change RHR/SI pump surveillances to be consistent with 15% degradation. Change RHR/SI pump surveillance requirements from discharge pressure to differential pressure.	15% degradation of the SI and RHR pumps is discussed in Sections 1.1 and 1.2 of WCAP 14285. LBLOCA is discussed in Section 3.1.1; SBLOCA is discussed Section 3.1.2; and LOCA mass and energy release (M&E) is discussed in Section 3.5.2.1 of WCAP 14285. The new surveillance criteria are supported by analyses. The surveillance criteria are presented in Section 3.10.1 of WCAP 14285.
Unit 2 3/4 5-5	Section 4.5.2.f.2 4.5.2.f.3	Admin	Change RHR/SI pump surveillance requirements from discharge pressure to differential pressure.	This an administrative change. The discharge pressure criteria in the current technical specifications correspond to the same pump performance characteristics as the proposed differential pressure criteria. The change ensures that surveillance criteria use similar acceptance criteria. The surveillance acceptance criteria for 10% degraded pumps were provided in the technical specification mark-ups of the RTSR. A copy of the mark-up of page 3/4 5-5 is included in Attachment 7.
UNIT 1 3/4 5-5	UNIT 1 Section 4.5.2.f.1	Both	Change CCP surveillance to be consistent with 10% degradation. Change pump surveillance requirements from discharge pressure to differential pressure.	Refer to the discussion given for page 3/4 1-11 for information concerning the 10% CCP degradation.
UNIT 2 3/4 5-5	UNIT 2 Section 4.5.2.f.1			

Page	Section	Group	Description	Remarks
UNIT 1 3/4 5-11	UNIT 1 Section 3.5.5	Both	Reduce minimum RWST temperature to 70°F.	See discussion for page 3/4 1-16.
UNIT 2 3/4 5-11	UNIT 2 Section 3.5.5			
3/4 7-6	Sections 4.7.1.2.a and 4.7.1.2.b	Margin	Change AFW Pump surveillance to be consistent with 12% degradation	See discussion for page 3/4 3-17. The proposed surveillance criteria is identical to the criteria in the unit 2 technical specifications. These criteria correspond to the auxiliary feedwater flows used in all analyses for both units except the "information only" unit 1 feedwater line break. As noted in the discussion for page 3/4 3-17, after the changes to the unit 1 safeguards actuation logic, unit 1 will be bounded by the unit 2 feedwater line break.
5-5	Section 5.4.2	SGTP	Reduce system volume to account for plugged steam generator tubes.	A volume range corresponding to 0% to 30% plugging is specified. See section 1.2 of WCAP 14285.
B 2-1(a)	Bases Section 2.1.1	Margin	Change DNB Values for Fuel	The values for DNBR for typical and thimble cells are being revised. The revised values are noted in Sections 1.2 and 3.3.2.1 of WCAP 14285. This change is related to the new thermal design and the new OTDT and OPDT protection trip setpoints. Therefore, see also discussions for pages 2-2, 2-7, 2-8, and 2-9.
B 2-4	Bases Section 2.2.1	Margin	Remove detail from the discussion of the OTDT protection trip.	The discussion of the proper normalization of T′ and P′ is being removed. This information is documented in Section 3.3.2.1 of WCAP 14285 and will be controlled administratively.
B 2-5	Bases Section 2.2.1	Margin	Remove detail from the discussion of the OPDT protection trip.	The discussion of the proper normalization of T′′ is being removed. This information is documented in Section 3.3.2.1 of WCAP 14285 and will be controlled administratively.

Page	Section	Group	Description	Remarks
UNIT 1 B 3/4 1-1	UNIT 1 Bases Section 3/4.1.1.1 and 3/4.1.1.2	Both	Reduce required shut down margin	See discussion for page 3/4 1-1.
UNIT 2 B 3/4 1-1	UNIT 2 Bases Sections 3/4.1.1.1 and 3/4.1.1.2			
B 3/4 4-1	Bases Section 3/4.4.1	Margin	Change DNB Values for Fuel	Change "1.69" to "the safety analysis limit".
UNIT 1 B 3/4 5-3	UNIT 1 Bases Section 3/4.5.5	Both	Reduce the minimum RWST temperature to 70°F.	Clarify conditions under which the reactor will remain subcritical. Specifically, LBLOCA is called out as the initiating condition and the control rods are assumed to be out instead of being inserted except for the most reactive assembly. In addition, the explanation that a conservatively high value of the RWST temperature is included in the technical specifications for unit 1 is being removed because the proposed value of 70°F is based on the analyses. See the discussion for page 3/4 1-16.
UNIT 2 B 3/4 5-3	UNIT 2 Bases Section 3/4.5.5			
UNIT 1 B 3/4 6-2	UNIT 1 Bases Sections 3/4.6.1.4 3/4.6.1.5	Both	Change peak containment pressure to reflect analysis result	Discussion of maximum calculated containment pressure is given in Sections 1.2 and 3.5.3.4 of WCAP 14285.
UNIT 2 B 3/4 6-2	UNIT 2 Bases Sections 3/4.6.1.4 3/4.6.1.5			

ATTACHMENT 5 TO AEP:NRC:1207

DISCUSSION OF PREVIOUS RELATED SUBMISSIONS

Introduction

Attachment 6 to this submittal is WCAP 14285. It describes the analyses and evaluations performed by Westinghouse Electric Corporation in order to support a reduced thermal design flow and a reduced minimum measured flow which are expected to result from increased steam generator tube plugging to the level of 30% in the unit 1 steam generators. It also describes analyses and evaluations performed simultaneously to support certain increases in operating margin such as increased setpoint tolerance for the pressurizer safety valves.

As discussed in Section 2.0 of WCAP 14285, the new analyses replace analyses performed earlier to support the operation of Cook Nuclear Plant and the evaluations described in WCAP 14285 are based on those earlier analyses. The earlier analyses are described in WCAP 11902 and WCAP 11902 Supplement 1, references 3 and 10. They are referred to as the "Rerating Program" in WCAP 14285.

The purposes of this attachment are to:

1. indicate those aspects of earlier analyses which have been submitted for NRC review and approved,
2. indicate those portions of these analyses which have not previously been submitted for review.
3. describe the earlier analyses,
4. provide references for previous submittals for the convenience of the reviewer, and

This submittal includes some proposed technical specification changes for both units. Therefore, the discussion of this attachment describes submittals for both units.

The discussion of this attachment describes the applications made by us to implement the features supported by the earlier analyses and the approvals received. This is significant because it will assist in clarifying available margin and because there are increased operating margins supported by the earlier analyses which we have not previously implemented. In some cases, we have not submitted a request due to the desire to maintain the technical specifications for the two Cook units as nearly alike as possible.

The following lists summarizes the status of analysis features of earlier analyses:

Principal Features of the Earlier Analyses Which Have been Reviewed and Approved

1. Reduced temperature and pressure operation for unit 1.
2. Reduced temperature operation for unit 2.
3. 10% degradation for the RHR and HHSI pumps for both units.
4. Increased MSIV response time for both units.
5. BIT 0 ppm boric acid concentration for both units.
6. Reduced MMF for unit 1.

Principal Features of the Earlier Analyses Which Have Not Been Submitted for Review

1. unit 1 rerate to 3413 MWt. The available power margin is allocated in this submittal to allow for increased steam generator tube plugging.
2. unit 2 rerate to 3588 MWt. Additional analytic work remains to be completed.
3. 10% degradation for the centrifugal charging pumps for both units. Approval to implement this feature is requested in this submittal.
4. Minimum RWST temperature of 70°F. Approval to implement this feature is requested in this submittal.
5. SDM requirement of 1.3%. Approval to implement this feature is requested in this submittal.

Purpose of the Earlier Analyses (Rerating Program)

The earlier analyses were performed to accomplish a number of goals. The most urgent of these was to permit operation of unit 1 at reduced primary temperature and pressure. The benefit of operating in a reduced primary temperature and pressure mode was to slow the degradation of the unit 1 steam generators. In addition, since essentially all of the analytic basis of the Cook units had to be reviewed or revised, all the analyses were performed to position unit 1 for subsequent uprating to 3413 MWt core power and unit 2 to 3588 MWt core power. As of this time, We have not requested NRC review of uprating either unit. This submission proposes to use the margin between the unit 1 analyzed, uprated power and licensed, rated thermal power to accommodate the increased tube plugging. Finally, the earlier analyses supported increased operating margins in selected areas. Among these were increased allowable ECCS pump degradation, reduction of required shutdown margin (SDM), a reduction in the minimum temperature of the refueling water storage tanks (RWST), removal of the boron injection tanks (BIT), and slower response times for certain components and systems. This submittal requests approval for the implementation of an allowed 10% degradation for the ECCS centrifugal charging pumps, reduction of required SDM, and a reduction in the minimum temperature of the RWST's for both units, which is supported in part by the earlier analyses.

Description and Review History of Prior Submittals

The first of the earlier analyses is described in reference 1, WCAP-11908, Containment Integrity Analysis for Cook Nuclear Plant units 1 and 2. It was submitted for NRC review by reference 2. Reference 1 presented a long term containment analysis which bounded both units at a core power of 3413 MWt, operation at a reduced temperature and pressure, and operation of the ECCS with residual heat removal (RHR) crossties closed. Reference 2 requested approval for operation with RHR cross ties closed.

The next group of analyses is described in reference 3, WCAP-11902, Reduced Temperature and Pressure Operation for Cook Nuclear Plant unit 1 Licensing Report. Reference 3 presented the remainder of the analyses and evaluations necessary to support operation of unit 1 at reduced temperature and pressure. The

analyses presented in reference 3 were performed at a core power of 3413 MWt. However, the evaluations described in reference 3 supported operation at a core power of 3250 MWt. Reference 3 also supported 10% degradation of the unit 1 RHR and high head safety injection (HHSI) pumps and a minimum RWST temperature of 70°F. Reference (3) was submitted for NRC review by reference 4.

The letters of references 5, 6, 7, and 8 provided supplementary information to the staff related to the request for approval (references 2 and 4) to operate unit 1 at reduced temperature and pressure with 10% degraded RHR and HHSI pumps. The request to operate unit 1 in this manner was approved by reference 9.

Reference 10, WCAP 11902, Supplement 1, Rerated Power and Revised Temperature and Pressure Operation for Cook Nuclear Plant units 1 & 2 Licensing Report, describes the balance of the analyses which were performed by Westinghouse Electric Corporation to support the operation of unit 1 at 3413 MWt. In particular, an analysis of the steam mass and energy release (SM&E) to containment, the associated containment analysis, and the SM&E outside containment are included in this report. These two analyses were performed to bound both units at the unit 2 uprated core power of 3588 MWt. Together with reference 3, the analyses of reference 10 completed the Westinghouse scope of analyses to support an additional 3 seconds for the response time of the main steam isolation valves (MSIV), 0 ppm boric acid concentration in the BIT, and 10% degradation of the CCP's for unit 1. The two SM&E analyses were performed assuming a SDM of 1.3%. However, the core response steam break analysis reported in reference 3 assumed a SDM of 1.6%.

Reference 11, Vantage 5 Reload Transition Safety Report for Cook Nuclear Plant unit 2 (RTSR), together with reference 1, Containment Integrity Analysis, reference 3, WCAP 11902, Reduced Temperature and Pressure Operation, and reference 10, WCAP 11902, Supplement 1, Rerated Power and Revised Temperature and Pressure Operation, support reduced temperature and pressure operation for unit 2 at an uprated core power of 3588 MWt. However, reference 1 and the RHR and HHSI cross tie closed LOCA cases of reference 11 only support a unit 2 core power of 3413 MWt. The analyses reported in references 1, 10, and 11 support 10% degradation of the CCP's, HHSI pumps, and RHR pumps, an increase of 3 seconds in MSIV response time for unit 2, 0 ppm boric acid concentration in the BIT for unit 2, a minimum RWST temperature of 70°F for unit 2, and a SDM of 1.3% for unit 2.

The letter of reference 13 submitted reference 11, RTSR, and the portions of reference 10, WCAP 11902, Supplement 1, which addressed the SM&E to the containment.

The letters of references 14, 15, and 16 provided supplementary information to the staff related to reference 13. Operation of unit 2 at reduced temperature with 10% degradation of the RHR and HHSI pumps was approved by reference 17. Some changes to both the unit 1 and unit 2 technical specifications which returned certain activities to administrative control were also made.

The letters of references 18 and 19 proposed technical specifications that implemented an increase of 3 seconds in the MSIV response times. These proposals were supported by reference 3, WCAP-11902, reference 10, WCAP-11902, Supplement 1, reference 11, RTSR, and evaluations performed by us. The letters in references 18 and 19 submitted the portions of reference 10, WCAP 11902, Supplement 1, which addressed the SM&E to the containment. The proposals to increase the MSIV response times by 3 seconds were approved by references 20 and 21.

The letter of reference 22 proposed to reduce the primary system minimum measured flow (MMF) for unit 1. An evaluation, performed by Westinghouse Electric Corporation, allocated available margin in MMF to the flow reduction. The evaluation was included in the submittal. This proposal was approved by reference 23.

The letter of reference 24 proposed to reduce the boron concentration in the BIT's of both units to 0 ppm. This proposal was supported by reference 3, WCAP-11902, reference 10, WCAP-11902, Supplement 1, reference 11, RTSR, and analyses performed by us. Reference 10, WCAP 11902, Supplement 1, was submitted in its entirety in support of this proposal. The proposal was approved by reference 25.

The letters of references 26 and 27 proposed to relax the tolerance of the main steam safety valve setpoints for both Cook units. The proposal was based on new analyses and on evaluations performed by Westinghouse Electric Corporation. The evaluations were based on the analyses described in reference 1 WCAP-11908, Containment Integrity Analysis, reference 3, WCAP-11902, reference 10, WCAP-11902, Supplement 1, and reference 11, RTSR. The descriptions of the new analyses and evaluations were included as attachments to these letters. This proposal was approved by reference 28.

References

1. WCAP-11908, Containment Integrity Analysis for Cook Nuclear Plant units 1 and 2, M. E. Wills, July 1988.

2. Letter AEP:NRC:1024D, Containment Long Term Pressure Analysis to Support RHR Cross Tie Closure, from M. P. Alexich to T. E. Murley, August 22, 1988.

3. WCAP-11902, Reduced Temperature and Pressure Operation for Cook Nuclear Plant unit 1 Licensing Report, D. L. Cecchett and D. B. Augustine, October 1988.

4. Letter AEP:NRC:1067, Reduced Temperature and Pressure Program Analyses and technical specification Changes, from M. P. Alexich to T. E. Murley, October 14, 1988.

5. Letter AEP:NRC:1067A, Supplemental technical specification Changes for Reduced Temperature and Pressure Program, from M. P. Alexich to T. E. Murley, December 30, 1988.

6. Letter AEP:NRC:1067B, Additional Information on Reduced Temperature and Pressure Submittal: Boron Dilution Accident, from M. P. Alexich to T. E. Murley, February 6, 1989.

7. Letter AEP:NRC:1067C, unit 1 RTP Program: Additional Information on Containment Structural Analysis, from M. P. Alexich to T. E. Murley, March 14, 1989.

8. Letter AEP:NRC:1067D, Modification of Reduced Temperature and Pressure Program technical specification Changes, from M. P. Alexich to T. E. Murley, June 5,1989.

9. Amendment No. 126 to Facility Operating License No. DPR-58.

10. WCAP 11902, Supplement 1, Rerated Power and Revised Temperature and Pressure Operation for Cook Nuclear Plant units 1 & 2 Licensing Report, September 1989.

11. Vantage 5 Reload Transition Safety Report for Cook Nuclear Plant unit 2, B. W. Gergos, Editor, January 1990.

12. No reference 12.

13. Letter AEP:NRC:1071E, unit 2 Cycle 8 Reload Licensing, Proposed technical specifications for unit 2 Cycle 8, and Related unit 1 Proposals, from M. P. Alexich to T. E. Murley, February 6, 1990.

14. Letter AEP:NRC:1071H, Modification to Our Previous Submittal AEP:NRC:1071E; Revised Figures for the Loss of Load Event, from M. P. Alexich to T. E. Murley, April 6, 1990.

15. Letter AEP:NRC:1071I, Information to Supplement Our Previous Submittals AEP:NRC:1071E and 1071H, from M. P. Alexich to T. E. Murley, May 29, 1990.

16. Letter AEP:NRC:1071K, Offsite Dose Calculation for the Reactor Coolant Pump Locked Rotor Event for unit 2 Cycle 8, from M. P. Alexich to T. E. Murley, July 23, 1990.

17. Amendment No. 148 to Facility Operating License No. DPR-58 and Amendment No. 134 to Facility Operating License No. DPR-74.

18. Letter AEP:NRC:1120, Expedited technical specification Change Request Steam Generator Stop Valves, from M. P. Alexich to T. E. Murley, January 31, 1990.

19. Letter AEP:NRC:1123, technical specification Change Request, Steam Generator Stop Valves, from M. P. Alexich to T. E. Murley, May 14, 1990.

20. Amendment No. 147 to Facility Operating License No. DPR-58.

21. Amendment No. 135 to Facility Operating License No. DPR-74.

22. Letter AEP:NRC:1130, technical specification Change for unit 1 Cycle 11, from M. P. Alexich to T. E. Murley, July 23,1990.

23. Amendment No. 152 to Facility Operating License No. DPR-58.

24 Letter AEP:NRC:1140, technical specification Change Request, BIT Boron Concentration Reduction, from M. P. Alexich to T. E. Murley, March 26, 1991.

25. Amendment No. 158 to Facility Operating License No. DPR-58 and Amendment No. 142 to Facility Operating License No. DPR-74.

26 Letter AEP:NRC:1169, technical specifications Change to Increase the Allowable Tolerance for Main Steam Safety Valves, from E. E. Fitzpatrick to T. E. Murley, November 11, 1992.

27. Letter AEP:NRC:1169A, Update for technical specification Change to Increase the Allowable Tolerances for Main Steam Safety Valves, from E. E. Fitzpatrick to T. E. Murley, December 17, 1993.

28. Amendment No. 182 to Facility Operating License No. DPR-58 and Amendment No. 167 to Facility Operating License No. DPR-74.

ATTACHMENT 6 TO AEP:NRC:1207

DESCRIPTION OF ANALYSES PERFORMED BY
WESTINGHOUSE ELECTRIC CORPORATION FOR
COOK NUCLEAR PLANT UNIT 1

WCAP 14285

ATTACHMENT 7 TO AEP:NRC:1207

DESCRIPTION OF ANALYSES PERFORMED BY
WESTINGHOUSE ELECTRIC CORPORATION FOR
COOK NUCLEAR PLANT UNIT 2

ATTACHMENT 6 TO AEP:NRC:1207

DESCRIPTION OF ANALYSES PERFORMED BY
WESTINGHOUSE ELECTRIC CORPORATION FOR
DONALD C. COOK NUCLEAR PLANT UNIT 1

WCAP 14285

WCAP-14285, Revision 1

DONALD C. COOK NUCLEAR PLANT UNIT 1
STEAM GENERATOR TUBE PLUGGING PROGRAM
LICENSING REPORT

MAY 1995

WESTINGHOUSE ELECTRIC CORPORATION
Energy Systems Business Unit
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LIST OF ACRONYMS AND ABBREVIATIONS

AFWPR	Auxiliary Feedwater Pump Runout
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
APC	Alternate Plugging Criteria
BIT	Boron Injection Tank
BOP	Balance of Plant
CCWS	Component Cooling Water System
CHG/SI	Charging/Safety Injection
COLR	Core Operating Limits Report
CRDM	Control Rod Drive Mechanism
CS	Condensate System
CVCS	Chemical and Volume Control System
DBE	Design Basis Earthquake
DECL	Double-Ended Cold Leg
DEHL	Double-Ended Hot Leg
DEPS	Double-Ended Pump Suction
DF	Decontamination Factor
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EAB	Exclusion Area Boundary
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
ECT	Eddy Current Testing
EDG	Emergency Diesel Generator
EFPM	Effective Full Power Months
EOP	Emergency Operating Procedure
ESF	Engineered Safety Features
ESFAS	Engineered Safety Feature Actuation System
ESW	Essential Service Water
$F\Delta H$	Hot Channel Enthalpy Rise Factor
F_Q	Total Peaking Factor
FHA	Fuel Handling Accident
FSAR	Final Safety Analysis Report
GPM	Gallons per Minute
HELB	High Energy Line Break
HFP	Hot Full Power
HZP	Hot Zero Power
IFBA	Integral Fuel Burnable Absorbers
IFM	Intermediate Flow Mixing
ITDP	Improved Thermal Design Procedure
LB	Large Break

LIST OF ACRONYMS AND ABBREVIATIONS (continued)

LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
LOL/TT	Loss of Load/Turbine Trip
LOOP	Loss of All AC Power to the Station Auxiliaries
LPZ	Low Population Zone
M/E or M&E	Mass and Energy
MMF	Minimum Measured Flow
MSLB	Main Steam Line Break
MWt	Megawatt Thermal
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OPΔT	Overpower Delta T
OTΔT	Overtemperature Delta T
PCT	Peak Clad Temperature
PLOF	Partial Loss of Reactor Coolant Flow
PORV	Power Operated Relief Valve
PTS	Pressurized Thermal Shock
PSSM	Power Shape Sensitivity Model
PWR	Pressurized Water Reactor
RC	Reactor Coolant
RCCA	Rod Cluster Control Assembly
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RHRS	Residual Heat Removal System
RPS	Reactor Protection System
RSE	Reload Safety Evaluation
RSR	Relative Stability Ratio
RTDP	Revised Thermal Design Procedure
RTP	Rated Thermal Power
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
RWFS	RCCA Bank Withdrawal from a Subcritical Condition
SAL	Safety Analysis Limit
SDM	Shutdown Margin
SER	Safety Evaluation Report
SB	Small Break
SFPCS	Spent Fuel Pool Cooling System
SI	Safety Injection

LIST OF ACRONYMS AND ABBREVIATIONS (continued)

SIS	Safety Injection System
SG	Steam Generator
SGTP	Steam Generator Tube Plugging
SGTR	Steam Generator Tube Rupture
SLB	Steam Line Break
SLB-CR	Steam Line Break Core Response
SR	Surveillance Requirement
TA	Total Allowance
T_{AVG}	RCS Average Temperature
T_{HOT}	Vessel Outlet Temperature
T_{COLD}	Vessel Inlet Temperature
TDF	Thermal Design Flow

DEFINITIONS

Rerating Program:

WCAP-11902 documented the analyses and evaluations performed to support reduced temperature and pressure operation of Donald C. Cook Nuclear Plant Units 1 and 2. Subsequently, a supplement to WCAP-11902 was issued to summarize the additional efforts performed to support a rerating of Cook Nuclear Plant Unit 1 and to provide part of the support for a Unit 2 rerating. These analyses and evaluations are described in Sections 2.0 and are documented in References 1 and 2 of Section 2.0. Throughout this report, the analyses and evaluations documented in WCAP-11902 and Supplement are referred to as the Rerating Program.

Steam Generator Tube Plugging Program:

Analyses and evaluations to support operation of Cook Nuclear Plant Unit 1 with up to a level of 30% steam generator tube plugging. In addition to the increased tube plugging level (and corresponding reduced thermal design flow), several increased operating margins were also addressed in the Steam Generator Tube Plugging Program. These operating margins are described in Section 1.0.

SUMMARY AND CONCLUSIONS

PROGRAM SUMMARY

The purpose of this document is to provide the safety analysis and evaluation results to support operation of Donald C. Cook Nuclear Plant Unit 1 with up to a level of 30% steam generator tube plugging (SGTP). In addition to the increased level of steam generator tube plugging, the analyses and evaluations support a corresponding reduction in Thermal Design Flow (TDF) and a 5% loop flow asymmetry. The analyses and evaluations were performed over a range of primary temperatures (553°F and 576.3°F) and for two values of primary pressure (2100 psia and 2250 psia).

The evaluations in this report are based on analyses performed for the Rerating Program. This program is discussed in more detail in Section 2.0.

In addition to addressing an increased SGTP level of 30%, the following increased operating margins were also addressed:

(1) Reduction of SI and RHR discharge pressure on recirculation - The RHR and SI minimum safeguards pump head curves were reduced by 15%, an additional 5% reduction from the current analysis degradation of 10%. The charging pump head curve degradation is maintained at the current value of 10%.

(2) The emergency diesel generator (EDG) start time was increased from 10 seconds to 30 seconds

(3) To support increased ΔT drift, the margin between the safety analysis limits (SAL) and the nominal values of the K_1 and K_4 gains of the Donald C. Cook Nuclear Plant Unit 1 OTΔT and OPΔT setpoint equations were adjusted.

(4) An increase in the pressurizer code safety valve (PSV) setpoint tolerance from +/-1% to +/-3%

(5) Decreased shutdown margin for T_{avg} greater than 200°F.

The analyses and evaluations in this document support all of these changes. Discussions of specific analyses address issues most relevant to those analyses.

The operating parameters for the increase in steam generator tube plugging level and the additional operating margins listed above will be referred to throughout this report as the "Steam Generator Tube Plugging (SGTP) Program". This report provides the necessary documentation to support the Technical Specification changes associated with the Steam Generator Tube Plugging Program. The topics addressed in this report are as follows:

- Description of License Amendment
- Summary of Technical Specification Changes
- Basis for Evaluations/Analyses Performed
- Loss of Coolant Accident Analyses
- Post-LOCA Hydrogen Production
- Post-LOCA Hot Leg Recirculation Time
- LOCA Hydraulic Forces
- Non-LOCA Analyses
- Containment Analyses
- Steam Generator Tube Rupture Analyses
- Reactor Cavity Pressure Evaluation
- Radiological Analysis
- Primary Components Evaluations
- Fluid and Auxiliary Systems Evaluations
- Fuel Structural Evaluation

Also provided in the Appendix to this report are the proposed Technical Specification changes. A brief summary of the results of each analysis and evaluation is provided below.

ACCIDENT ANALYSIS CONCLUSIONS

The results of the accident analyses and evaluations performed for the SGTP Program demonstrate that safe operation of the Donald C. Cook Nuclear Plant Unit 1 is maintained. The bases for the evaluations and analyses performed are provided in Section 2.1 A summary of the conclusions of each of the accident analyses is provided below.

Large Break LOCA (Section 3.1.1)

The large break LOCA analysis was reanalyzed for the impact of the increased tube plugging level, reduced TDF, loop flow asymmetry, revised ECCS flows, and the increased EDG start time. The large break LOCA analysis was not impacted by the pressurizer code safety valve tolerance increase, the revised K1/K4 values, or the decreased shutdown margin. The large break LOCA analysis was performed with the 1981 version of the Westinghouse ECCS Evaluation Model using the BASH computer code. Analysis assumptions included ECCS flow with the RHR cross-tie valves closed, a total peaking factor of 2.15, a hot channel enthalpy rise peaking factor of 1.55, and an accumulator temperature of 100°F. A full spectrum break analysis was performed at the nominal RCS conditions (initial RCS pressure of 2250 psia and initial hot leg temperature of 609.1°F) from which the limiting break discharge coefficient was determined. The limiting break was then reanalyzed at the reduced hot leg temperature and nominal RCS pressure of 2250 psia, and also at nominal hot leg temperature and an initial RCS pressure of 2100 psia. The above cases were all analyzed with minimum safety injection flow, which was determined to be limiting. The limiting break was determined to be C_D= 0.4 at the nominal hot leg temperature (T_{HOT}= 609.1°F) and a pressure of 2100 psia with

minimum safety injection flow. The peak cladding temperature was calculated to be 2164°F, which is less than the 2200°F limit in 10CFR50.46.

Small Break LOCA (Section 3.1.2)

The small break LOCA analysis was reanalyzed for the impact of the increased tube plugging level, reduced TDF, loop flow asymmetry, revised ECCS flows, and the increased EDG start time. The small break LOCA analysis was not impacted by the pressurizer code safety valve tolerance increase, the revised K1/K4 values, or the decreased shutdown margin. The small break LOCA analysis was performed with the Westinghouse small break LOCA ECCS Evaluation Model using the NOTRUMP code (including the recent model changes submitted in WCAP-10054-P, Addendum 2 and WCAP-10081-NP, Addendum 2). The key analysis input assumptions included ECCS flows with the HHSI cross-tie discharge valves closed, a total peaking factor of 2.32 and hot channel enthalpy rise peaking factor of 1.55. Other analysis input assumptions incorporated in the small break LOCA analysis are reduced hot assembly average power (P_{HA}) and a power shape based on a reduced axial offset of +20%. A single break size analysis was performed at the previously-limiting break size of three inches. The calculation used the reduced temperature, reduced pressure operating condition. An evaluation of the break spectrum and the range of operating conditions concluded that the analyzed case would remain bounding with respect to peak clad temperature. The calculation was performed with minimum safety injection flow, which was limiting. The peak cladding temperature was calculated to be 1443°F, which is less than the 2200°F limit in 10CFR50.46.

LOCA Hydraulic Forcing Functions (Section 3.2)

LOCA hydraulic forces are relatively insensitive to specific SGTP levels. The Donald C. Cook Nuclear Plant LOCA hydraulic forces were most recently analyzed for the Rerating Program. The RCS parameters used in the existing analysis-of-record conservatively bound the conditions at 30% tube plugging. Therefore, the existing LOCA forces analyses remain conservative relative to the SGTP Program.

Non-LOCA Analyses (Section 3.3)

The non-LOCA events were addressed by a combination of evaluations and analyses for the impact of the increased tube plugging level, reduced TDF, loop flow asymmetry, revised ECCS flows, pressurizer code safety valve tolerance increase, increased EDG start time, revised K1/K4 values, and decreased shutdown margin. The computer codes and methods used for the non-LOCA analyses have been previously approved by the NRC. The non-LOCA safety analyses were reviewed on the basis of both DNB and non-DNB acceptance criteria. All DNB event reanalyses were found to yield a minimum DNBR which remains above the limit value. The analyses demonstrate that all licensing basis criteria continue to be met and the conclusions presented in the UFSAR remain valid.

Post-LOCA Hydrogen Generation (Section 3.4)

The post-LOCA hydrogen generation rates that were reviewed as part of the Rerating Program were determined to remain applicable to the SGTP Program.

Containment Integrity (Section 3.5)

The containment integrity analyses were addressed for the impact of the increased level of tube plugging, reduced thermal design flow, loop flow asymmetry, revised ECCS flows, and the increased EDG start time. The containment analyses were not impacted by the pressurizer code safety valve tolerance increase, the revised K1/K4 values, or the increased shutdown margin. The increase in the containment pressure and temperature following a LOCA was analyzed. The mass and energy release rates calculated as part of the SGTP Program formed the basis to evaluate the structural integrity of the containment following a postulated accident to satisfy the acceptance criteria, General Design Criterion 38. Even though Cook Nuclear Plant is licensed to GDC's in Appendix H of the original FSAR, more conservative acceptance criteria were used. The containment integrity analysis for the most limiting case (i.e., RHR cross tie valve closed) resulted in a maximum calculated containment pressure of 11.49 psig, for the double-ended pump suction minimum safeguards break case. Since the calculated pressure is below the design pressure of 12.0 psig, the results of the LOCA containment integrity analysis are acceptable.

The Mainsteam Line Break (MSLB) mass and energy releases were used as input into the containment integrity analysis to demonstrate that the peak containment temperature resulting from a design basis MSLB will not exceed the equipment qualification criterion for the plant. The containment pressure response determined for the LOCA containment integrity analysis is calculated to be more severe than for the MSLB, and therefore, bounds the MSLB analysis. For the large break case, the limiting case among the double-ended ruptures is the 1.4 ft^2 double-ended rupture, 102% power, MSIV failure case. This case yielded a calculated peak temperature of 322.7°F. For the small break case, the most limiting case in terms of peak calculated temperature is the 0.942 ft^2 split break, 30% power with an MSIV failure. This case resulted in a calculated peak temperature of 326°F. Both cases are within the Environmental Acceptance Criteria. Therefore, the analysis demonstrates that the containment heat removal systems function to rapidly reduce the containment pressure and temperature in the event of a MSLB. General Design Criterion 50 and Appendix K are satisfied.

Short Term Containment Analysis (Section 3.5.1)

The short term containment analysis that was performed for the Rerating Program was reviewed and it was determined that the conclusions provided for the Rerating Program remain valid for the SGTP Program. That is, the resulting peak pressures remain below the allowable design peak pressures for the pressurizer enclosure, the fan accumulator room and the steam generator enclosure.

Steam Generator Tube Rupture (Section 3.6)

The SGTR event was analyzed for the impact of the increased tube plugging level and associated reduced TDF and loop flow asymmetry. The SGTR analysis was not impacted by any of the SGTP Program increased operating margins. The thyroid and whole body doses estimated for Cook Nuclear Plant Unit 1, based on the 30% SGTP evaluation, remain within a "small fraction" (10%) of the 10CFR100 exposure limit guidelines. Small fraction is the smallest of the exposure guidelines defined in NUREG-0800. Therefore, the conclusions of the UFSAR remain valid.

Post-LOCA Hot Leg Recirculation Time (Section 3.7)

The hot leg switchover to preclude boron precipitation and post-LOCA long term cooling are not adversely affected by the 30% SGTP Program. The proposed changes do not significantly affect the normal plant operating parameters, the safeguards systems actuations, the accident mitigation capabilities important to these events, or the assumptions used in the analysis of these events. The proposed changes do not create conditions more limiting than those assumed in the LOCA-related analyses.

Reactor Cavity Pressure Analysis (Section 3.8)

The Reactor Cavity Pressure Analysis that was performed for the Rerating Program was reviewed and it was determined that the conclusions provided for the Rerating Program remain valid for the SGTP Program. The SGTP Program parameters affect the Reactor Cavity Pressure Analysis through the mass and energy releases provided as input into the analysis. There is no direct impact of SGTP level on short-term mass and energy release rate calculations and containment subcompartment response analysis. The mass and energy releases used as input for the Reactor Cavity Pressure Analysis reflected limiting conditions and therefore, the NSSS performance parameters for the SGTP Program did not impact the results.

Radiological Doses (Section 3.9)

A reanalysis of the offsite doses following a large break LOCA was performed for the increase in emergency diesel generator start time to 30 seconds. While there was a slight increase in the offsite thyroid doses, the doses are within the applicable limits. The source terms for LOCA and the fuel handling accident are unaffected by the increase in SGTP level or any of the other SGTP Program adders.

FLUID AND AUXILIARY SYSTEMS EVALUATION CONCLUSIONS (Section 3.10)

The fluid systems proof of design calculations were reviewed for the SGTP conditions. This review demonstrated that the NSSS fluid systems will continue to function adequately as designed for all conditions of the SGTP Program. ECCS flowrates were revised as part of the SGTP Program and were used in the safety analyses and evaluations.

In the NSSS/BOP interface area, the proposed NSSS Performance Parameters for the SGTP Program were compared with those of the Rerating Program. The results of the evaluation show that a SGTP level of 30% will have no adverse effects on the Balance Of Plant (BOP) systems performance (Main Steam System,Condensate and Feedwater System, Auxiliary Feedwater System, Steam Generator and Blowdown System). They will continue to perform acceptably at the conditions associated with 30% SGTP.

The evaluations for the fluid and auxiliary systems are described in more detail in Section 3.10.

PRIMARY COMPONENTS EVALUATION CONCLUSIONS

Steam Generators (Section 3.11.1)

In the thermal-hydraulic areas: the modified moisture separator packages on Cook Nuclear Plant Unit 1 will permit operation at steam pressures down to 700 psia and below without exceeding 0.25% moisture carryover. Steam Generator operating characteristics will be acceptable down to the minimum steam pressure of 589 psia. The evaluation of thermal-hydraulic stability indicates satisfactory results for all SGTP cases.

The evaluation performed for the effects of the SGTP program on U-bend tube fatigue for Cook Unit 1 is documented in WCAP-13814. It was concluded in WCAP-13814 that four tubes were susceptible to high cycle fatigue at the 30% SGTP Program conditions and would require preventative action.

Structural analyses and evaluations performed for the Cook Unit 1 steam generators indicate that the steam generator components remain in compliance with the applicable ASME Code requirements under the SGTP conditions.

Reactor Vessel (Section 3.11.2)

The results of the structural evaluations performed for the reactor vessel demonstrate that operation of Cook Nuclear Plant Unit 1 within the parameters of the SGTP program does not result in stress intensities or fatigue usage factors which exceed the acceptance criteria of the applicable ASME Code versions. The SGTP Program does not result in an increase in the fast neutron fluence values calculated for the Rerating Program. Therefore, the reactor vessel

integrity analyses performed as part of the Rerating Program will remain applicable after 30% SGTP.

Reactor Internals (Section 3.11.3)

Results of the thermal-hydraulic analyses performed for the reactor internals indicate that the SGTP Program for Cook Nuclear Plant Unit 1 results in acceptable values of core bypass flow, pressure drops, component lift forces, and momentum flux values. It was also confirmed that the control rod drop time limit of 2.4 seconds remains applicable for the SGTP Program conditions.

From the component stress analysis and the flow induced vibration evaluations, it is concluded that the margins of safety are within acceptable limits per the original design basis.

Control Rod Drive Mechanisms (Section 3.11.4)

The conclusion of structural evaluations performed for the 30% SGTP conditions for the CRDMs demonstrate that the operability, service life, and structural integrity of the CRDM latch assembly, drive rod, and coil stack will not be adversely affected.

Reactor Coolant Pumps (Section 3.11.5)

The review performed of the reactor coolant pumps for the 30% SGTP conditions demonstrate that the conditions are acceptable for the 93A RCP, and no additional thermal or structural analyses are required to demonstrate compliance with the applicable codes and standards. The RCP motor evaluation revealed that the motors are acceptable for operation at the 30% SGTP conditions.

Pressurizer (Section 3.11.6)

A fatigue analysis performed for the Cook Unit 1 pressurizer, incorporating the most conservative conditions of the SGTP program, demonstrated that the pressurizer remains in compliance with the applicable ASME Code criteria.

Reactor Coolant Piping and Supports (Section 3.11.7)

An evaluation was performed to determine the effects of the 30% SGTP conditions on the primary loop piping, primary equipment supports, and the primary equipment nozzles. Operation at the SGTP Program conditions was found to be acceptable because these conditions are already enveloped by the Rerating Program. In addition, the rerating transients and plant parameters associated with the Rerating and SGTP Programs for Donald C. Cook Nuclear Plant Unit 1 have been reviewed. The impact on the design basis analysis for the

NRC Bulletin 88-08 evaluation of the auxiliary spray piping and the NRC Bulletin 88-11 evaluation of the pressurizer surge line piping is insignificant.

Auxiliary Components (Section 3.11.8)

Evaluations were performed for the auxiliary tanks, pumps, valves, and heat exchangers to determine the effects of the revised RCS parameters due to the SGTP Program. The results of these evaluations demonstrated that, due to conservatively specified parameters used in the procurement of the auxiliary equipment, the 30% SGTP parameters are not expected to adversely affect the function or structural integrity of this equipment.

Fuel Structural Evaluation (Section 3.12)

Evaluations were performed of the fuel for Cook Nuclear Plant Unit 1 for the SGTP Program conditions in the areas of fuel rod and fuel assembly structural integrity, core design, and thermal-hydraulic design.

The fuel assembly structural integrity is not affected by the SGTP Program, and the core coolable geometry is maintained for the 15x15 OFA fuel in the Unit 1 core. The evaluation of the fuel rod structural integrity indicates these conditions will be acceptable, although it is noted that cycle-specific verification during the normal reload will still be performed.

The results of the core design evaluation indicated that the SGTP Program conditions result in no impact to the core design except for the values of the statepoint for the Steamline Break Analysis, and the Dropped Rod Analysis.

Thermal-hydraulic analyses were made for the fuel for the limiting 30% SGTP parameters using RTDP methodology. The analysis showed that the DNBR design basis was met for the limiting DNB events. This analysis caused the available DNB margin to increase. This margin can be used for flexibility of design and to offset unanticipated DNBR penalties.

2.1 DESIGN POWER CAPABILITY PARAMETERS

This section describes the parameters which were used as the basis for the evaluations and analyses performed to support the SGTP Program for Cook Nuclear Plant Unit 1. The NSSS performance parameters feature the current licensed NSSS power of 3262 MWt, a T_{avg} temperature range from 553°F to 576.3°F, two primary pressure values of 2250 psia or 2100 psia, a maximum average and peak SGTP level of 30%, reduced TDF, and 5% loop flow asymmetry. The RCS temperature range is bounded by the Rerating Program.

Also incorporated into the SGTP Program was an RCS flow measurement uncertainty range of 1.9% to 2.5%. The Technical Specifications will be revised to incorporate the Minimum Measured Flow (MMF) corresponding to a flow measurement uncertainty of 2.5%. This increase will provide additional margin for instrumentation. If additional flow margin is needed in the future. the margin can be reallocated from instrumentation margin to RCS flow margin by revising the MMF Technical Specification because all of the accident analyses evaluated a MMF of 339,100 gpm. A MMF of 339,100 gpm total reflects a 1.9% flow measurement uncertainty. A MMF of 341,100 gpm total reflects a 2.5% flow measurement uncertainty. The RCS flow margin has been reviewed and sufficient margin exists to maintain the TDF of 83,200 gpm/loop with the 2.5% flow measurement uncertainty.

A brief description of each set of parameters is provided below:

Case 1: These are the original NSSS performance parameters for Unit 1 and are shown for comparison with the revised parameters. The NSSS power level of 3250 MWt does not account for reactor coolant pump heat; at the time that Unit 1 was designed, it was the custom to indicate only the core power level value.

Case 2: These parameters incorporate a core power level of 3250 MWt, an NSSS power level of 3262 MWt (which includes 12 MWt of reactor coolant pump heat), an average steam generator tube plugging level of 30%, primary pressures of either 2250 psia or 2100 psia, and a lower bound vessel average temperature of 553.0°F.

Case 3: These parameters incorporate the same features as case 2, except that the primary temperatures and resulting secondary parameters incorporate an upper bound vessel average temperature of 576.3°F. This case was used as the basis for selected analyses, where high primary temperatures were limiting.

Case 4: These parameters incorporate the same features as case 2, except that the TDF was reduced to 79,000 gpm/loop to bound 5% loop flow asymmetry.

Case 5: These parameters incorporate the same features as case 4, except that the primary temperatures and resulting secondary parameters incorporate an upper bound vessel average temperature of 576.3°F (the highest vessel average temperature considered for the SGTP

1.0 INTRODUCTION - DESCRIPTION OF LICENSE AMENDMENT REQUEST

1.1 PURPOSE FOR CHANGE

The Donald C. Cook Nuclear Plant Unit 1 has experienced tube corrosion problems in its steam generators and, as a result, an increasing number of tubes have been plugged during the last several outages. Steam generator tube plugging potentially decreases reactor coolant system flow due to increased flow resistances through the steam generators. As the number of plugged tubes increases, the RCS flow may be reduced to a value below that which is currently analyzed in the licensing basis.

Currently, the licensing basis analyses for the Donald C. Cook Nuclear Plant Unit 1 are documented in the Updated Final Safety Analysis Report (UFSAR). These analyses are bounding for a maximum average steam generator tube plugging (SGTP) level of up to 10% with a peak level of 15% in any one steam generator. This amendment request reflects the changes to the safety analysis assumptions and results due to the revised operating conditions resulting from an increased level of steam generator tube plugging. While the analyses and evaluations were being performed for the increased level of tube plugging, several operating margins were increased and incorporated into the analyses and evaluations in order to maximize the benefit of the reanalysis. Therefore, in addition to addressing an increased SGTP level of 30%, the following increased operating margins were also addressed:

(1) Reduction of SI and RHR discharge pressure on recirculation - The RHR and SI minimum safeguards pump head curves were reduced by 15%, an additional 5% reduction from the current analysis degradation of 10%. The charging pump head curve degradation is maintained at the current value of 10%.

(2) The emergency diesel generator startup time was increased from 10 seconds to 30 seconds

(3) To support increased ΔT drift, the margin between the safety analysis limits (SAL) and the nominal values of the K_1 and K_4 gains of the Donald C. Cook Nuclear Plant Unit 1 OTΔT and OPΔT setpoint equations were adjusted.

(4) An increase in the pressurizer code safety valve (PSV) setpoint tolerance from +/- 1% to +/-3%

(5) Decreased shutdown margin for T_{avg} greater than 200°F.

The revised parameters associated with the increase in tube plugging level to 30% SGTP, both directly and indirectly, are referred to throughout this report as the "Steam Generator

Tube Plugging Program". This program resulted in changes to the Donald C. Cook Nuclear Plant Unit 1 Technical Specifications, including:

- Core Safety Limits
- OPΔT/OTΔT Setpoints
- Shutdown Margin for Modes 1,2,3, and 4
- DNB Parameters
- RTS Response Times
- ESFAS Instrumentation Logic
- Pressurizer Code Safety Valve Lift Setting Tolerance
- ECCS Pump Discharge Pressure on Recirculation
- Minimum RWST Temperature
- Containment Internal Pressure
- RCS Volume
- Emergency Diesel Generator Start Time

been based on the new core safety limits and account for instrument uncertainties. The reference temperatures are now indicated values and the temperature range that was analyzed is specified.

Shutdown Margin for MODES 1, 2, 3, and 4

Shutdown margin (SDM) requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of the highest reactivity worth is fully withdrawn.

The Limiting Condition for Operation (LCO) 3.1.1.1 Shutdown Margin has been revised from 1.6% $\Delta k/k$ to 1.3% $\Delta k/k$. This change is also included in the associated Bases.

The acceptability of the decrease in the SDM is based on re-analysis of the most limiting accident, core response to a steam line break.

DNB Parameters, RCS Tavg and RCS Flow

LCO 3.2.5, DNB Parameters specifies RCS parameters assumed as initial conditions in the transient and accident analyses.

In Table 3.2-1, DNB Parameters, the RCS T_{avg} has been changed from ≤ 570.9°F to ≤ (576.3 + 5.1) - (readability error)°F. The 576.3 °F value has been verified in the re-analyses. The additional 5.1°F is included to account for temperature uncertainty factors such as cold leg streaming, as documented in WCAP-12568, Rev. 1. The readability error will be determined by AEPSC.

Additionally, in Table 3.2-1, the RCS Total Flow Rate has been reduced from ≥ 361,600 to ≥ 341,100 gpm. This reduction is based on the increase in the SGTP limit to 30% and includes a 2.5% instrument uncertainty.

RTS Response Times

Reactor Trip System (RTS) Instrumentation response times are assumed in accident analyses for the time interval from when the monitored parameter (level, pressure, temperature, etc.) exceeds its setpoint at the sensor until loss of stationary gripper coil voltage.

RTS Instrumentation response times specified in Table 3.3-2 for Functions 9, 10, 14, and 16 (Pressurizer Pressure--Low, Pressurizer Pressure--High, Steam Generator Water Level--Low-Low, and Undervoltage-Reactor Coolant Pumps) relaxed from 1.0 second to 2.0 seconds

(Functions 9, 10, and 14) or to 1.5 seconds (Function 16). The acceptability of these relaxations has been verified by accident analyses.

ESFAS Instrumentation Logic

The Engineered Safety Feature Actuation System (ESFAS) instrumentation initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits, protect the RCS pressure boundary, and to mitigate accidents.

ESFAS Instrumentation logic for Functional Units 1 and 4 and for the P-12 interlock have been revised. These revisions reflect the Unit 1 implementation of the "hybrid" steamline break protection logic that is presently used on Unit 2.

Function 1 - Safety Injection, Turbine Trip, Feedwater Isolation and Motor-driven Feedwater Pumps:

Actuation on Steam Flow in Two Steam Lines-High coincident with either T_{avg}-Low-Low or Steam Line Pressure-Low has been replaced by actuation on Steam Line Pressure-Low

Function 4 - Steam Line Isolation:

Actuation on Steam Flow in Two Steam Lines-High coincident with either T_{avg}-Low-Low or Steam Line Pressure-Low has been replaced by actuation on Steam Flow in Two Steam Lines-High coincident with T_{avg}-Low-Low and on Steam Line Pressure-Low

Pressurizer Code Safety Valve Lift Setting Tolerance

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. Accident and safety analyses which require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The limit protected by this specification is the reactor coolant pressure boundary Safety Limit of 110% of design pressure.

The LCO 3.4.2 (MODES 4 and 5) and LCO 3.4.3 (MODES 1, 2, and 3) pressurizer code safety valve lift setting tolerance has been increased from ± 1% to ± 3%.

The acceptability of the increased safety valve tolerance has been established by evaluation or analysis of applicable events including loss of load, turbine trip, locked rotor, loss of normal feedwater, feedwater line break, and loss of all power to station auxiliaries.

ECCS Pump Flows

LCO 3.5.2, ECCS Subsystems - $T_{avg} \geq 350°F$, specifies requirements applicable to pumps, heat exchangers, and flow paths credited with core cooling following an accident.

The Surveillance Requirement (SR) 4.5.2.f (centrifugal charging pump, safety injection pump and residual heat removal pump tests) requirements have been revised to specify measurement of differential pressure along with revised acceptance criteria pressures. The revised centrifugal charging pump differential pressure criteria represents a 10% pump head degradation. The differential pressure criteria specified for the safety injection pump and the residual heat removal pump reflects relaxations in the associated pump curves which represent 15% pump head degradation.

The relaxation for centrifugal charging pumps is also applicable to LCOs 3.1.2.3 and 3.1.2.4.

The acceptability of the relaxed pump curves has been verified for all applicable accidents.

Minimum RWST Temperature

The minimum Refueling Water Storage Tank (RWST) temperature has been changed from 80°F to 70°F in LCOs 3.1.2.7, 3.1.2.8, and 3.5.5 and Bases 3.5.5. The 70°F minimum temperature is acceptable based on the LOCA and non-LOCA analyses performed for the Cook Nuclear Plant Unit 1 licensing basis.

Containment Internal Pressure

The maximum calculated post-accident containment pressure must remain below the containment design pressure of 12.0 psig. The results of the containment integrity analyses performed for the SGTP program resulted in a maximum calculated containment pressure of 11.49 psig. Thus, the value in the Bases for LCO 3.6.1.4 (11.89 psig), Internal Pressure, is being revised to reflect the analysis results.

Emergency Diesel Generator (EDG) Start Time

LCO 3.8.1, AC Sources - Operating, specifies requirements for off-site and on-site (diesel generator) AC sources, including EDG testing requirements to demonstrate the capability to achieve the required voltage and frequency within the specified time. The EDG start time, plus the load sequencer loading times, plus the equipment actuation/start times, establishes the total time until the function of ESF equipment is assumed in accident analyses.

The SR 4.8.1.1.2.a.4, SR 4.8.1.1.2.e.4.b, SR 4.8.1.1.2.e.6.b, and SR 4.8.1.1.2.f.3 requirements have been revised to specify a relaxed EDG start time of 30 seconds. In the safety analysis, the 30 second time is the time at which the load sequencer is assumed to

start loading. Additionally, the SRs have been revised to specify that voltage and frequency shall be achieved rather than engine RPM, consistent with the safety analysis, Regulatory Guide 1.9, and NUREG-1431.

Relaxation of the EDG start time to 30 seconds has been shown to be acceptable based on re-analysis of limiting accidents.

Only the response times specified in Table 3.3-5, Engineered Safety Features Response Times which include the diesel generator start time are affected by this change. The longer response times assumed in LOCA, Non-LOCA, and containment analyses have been specified in Table 3.3-5.

RCS Volume

Design Features Section 5.4.2 specifies the total contained volume of the RCS. With the increase of the SGTP limit to 30%, a corresponding reduction in RCS volume must be specified. Since the actual level of tube plugging may change each outage, a range of RCS volume corresponding to the range of 0% to 30% tube plugging has been specified: approximately 12,466 ft^3 to 11,551 ft^3.

TABLE 1.2-1
SUMMARY OF TECHNICAL SPECIFICATION CHANGES

PAGE	SECTION	DESCRIPTION OF CHANGE	BASIS
2-2 B 2-1(a)	2.1 Bases 2.1	**Core Safety Limits** Replace Figure 2.1-1, Reactor Core Safety Limits The safety analysis limit DNBR specified in the Bases for Section 2.1 has been revised from 1.45 to 1.42.	Figure 2.1-1 has been replaced with a new figure based on the latest analyses, reflecting 30% SGTP, reduced rated thermal power, reduced RCS flow, etc. Section 3.3.2.1
2-7 2-8 2-9 B 2-4 B 2-5	2.2 B 2.2.1	**OTΔT & OPΔT Setpoints** Revise OTΔT and OPΔT Trip Setpoint and Allowable Value notes.	Proposed settings based on new core safety limits and account for instrument uncertainties. Section 3.3.2.1 & Table 3.3-3
3/4 1-1 B 3/4 1-1	3.1.1.1	**Shutdown Margin** Shutdown Margin limit relaxed from 1.6 to 1.3 % Δk/k	Relaxation based on re-analysis of limiting accident - core response to steam line break. Sections 3.3.4.7 & 3.3.5.6
3/4 2-14	3.2.5 Table 3.2-1	**DNB Parameters** Revise Table 3.2-1, DNB Parameters, RCS T_{avg} from 570.9 °F to (581.4 -readability error) °F Readability error is responsibility of AEPSC.	T_{avg} input assumption verified by reanalyses. Sections 2.1, 3.3.2.1 & 3.12.4

TABLE 1.2-1 (continued)
SUMMARY OF TECHNICAL SPECIFICATION CHANGES

PAGE	SECTION	DESCRIPTION OF CHANGE	BASIS
2-5 3/4 2-14	2.2 Table 2.2-1 3.2.5 Table 3.2-1	**DNB Parameters** Revise Table 3.2-1, DNB Parameters, RCS Total Flow from ≥ 361,600 to ≥ 341,100 gpm.	Change based on 30% SGTP limit. The value of 341,100 includes a 2.5% instrument uncertainty. Analysis used 339,100 which includes a 1.9% instrument uncertainty. Sections 2.1, 3.3.2.1 & 3.12.4
3/4 3-10 3/4 3-11	3.3.1 Table 3.3-2	**RTS Response Times** RTS Instrumentation response times for Functions 9, 10, 14, and 16 (Pressurizer Pressure--Low, Pressurizer Pressure--High, Steam Generator Water Level--Low-Low, and Undervoltage-Reactor Coolant Pumps) relaxed from 1.0 second to 2.0 seconds (Functions 9, 10, and 14) or to 1.5 seconds (Function 16).	The acceptability of these relaxations verified by accident analyses. Sections 3.1 & 3.3
3/4 3-17 3/4 3-21 3/4 3-23a 3/4 3-24 3/4 3-26 3/4 3-28 3/4 3-31 3/4 3-33	Table 3.3-3 Table 3.3-4 Table 3.3-5 Table 4.3-2	**ESFAS LOGIC** ESFAS Instrumentation logic for Functional Units 1 and 4 and for the P-12 interlock have been revised.	ESFAS logic change has been shown to be acceptable by non-LOCA analyses. Section 3.3

TABLE 1.2-1 (continued)
SUMMARY OF TECHNICAL SPECIFICATION CHANGES

PAGE	SECTION	DESCRIPTION OF CHANGE	BASIS
B 3/4 4-1	3/4 4.1	**DNBR Limit** The DNBR limit specified in Bases 3/4 4.1 is no longer applicable. "1.69" replaced with "the safety analysis limit"	No basis required.
3/4 4-4 3/4 4-5	3.4.2 3.4.3	**Safety Valve Lift Setting** Pressurizer code safety valve lift setting pressure tolerance increased to 3%	Relaxation based on evaluation or analysis of several limiting events. Sections 1.1& 3.3
3/4 1-11 3/4 1-12 3/4 5-5	3.1.2.3 3.1.2.4 3.5.2	**ECCS Pump Flows** Centrifugal charging, safety injection and residual heat removal pumps' test acceptance criteria relaxed. CCP - 10% RHR & SI - 15%	Relaxed pump curves have been verified to be acceptable for all applicable accidents. Section 3.3.4.7 & 3.10
3/4 1-15 3/4 1-16 3/4 5-11 B 3/4 5-3	3.1.2.7 3.1.2.8 3.5.5 B 3.5.5	**RWST Temperature** Minimum RWST temperature reduced from 80°F to 70°F	The 70°F minimum temperature is acceptable based on the LOCA and non-LOCA analyses. Section 3.1.1, W- letter 92AE*-G-074, dated June 12, 1992

TABLE 1.2-1 (continued)
SUMMARY OF TECHNICAL SPECIFICATION CHANGES

PAGE	SECTION	DESCRIPTION OF CHANGE	BASIS
B 3/4 6-2	3.6.1.4	**Internal Pressure** Thus, the value in the Bases for LCO 3.6.1.4 (11.89 psig), Intemal Pressure, is being revised to reflect the analysis results.	The results of the containment integrity analyses performed for the SGTP program resulted in a maximum calculated containment pressure of 11.49 psig. Section 3.5
3/4 8-3 3/4 8-5 3/4 8-7 3/4 3-27 3/4 3-28 3/4 3-29 3/4 3-30	3.8.1.1 3.3.2 Table 3.3-5	**EDG Start Time/ ESFAS Response Times** EDG start time relaxed to 30 seconds. ESFAS response times affected by EDG start time relaxation revised.	The increase of 20 seconds in the EDG start time has been shown to be acceptable for limiting accidents. Sections 3.3 & 3.5
5-5	5.4.2	**RCS Volume** Design Feature, RCS volume reduced.	Change based on 30% SGTP limit. Section 3.10

1.2 CURRENT LICENSE BASIS AND FUNCTION OF IDENTIFIED TECHNICAL SPECIFICATIONS AND DESCRIPTION OF PROPOSED CHANGE

The proposed changes to the Donald C. Cook Nuclear Plant Unit 1 Technical Specifications are summarized in Table 1.2-1. These changes reflect the impact on the design, analytical methodology, and safety analysis assumptions outlined in this amendment request. The proposed Technical Specification changes are included in Appendix A of this report. A brief overview of the significant Technical Specification changes follows.

The changes are based on analyses and evaluations associated with the SGTP Program. Since new analyses and evaluations were required to establish the acceptability of the SGTP level, several related Technical Specification relaxations were verified.

Core Safety Limits

Technical Specification Figure 2.1-1, Reactor Core Safety Limits, shows the loci of points of thermal power, RCS pressure and average temperature below which the calculated DNBR is no less than the design DNBR value and the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. The figure is based on the enthalpy hot channel factor.

Figure 2.1-1 has been replaced with a new figure based on the latest analyses, reflecting 30% tube plugging, reduced rated thermal power, reduced RCS flow, etc.

OPΔT/OTΔT Setpoints

Technical Specification Table 2.2-1 lists the reactor protection system instrumentation trip setpoints for the various trip functions. The reactor trip setpoint limits specified in Table 2.2-1 are the nominal values at which the reactor trips are set for each functional unit.

The Thermal Overpower ΔT (OPΔT) trip function provides assurance of fuel integrity (e.g., no fuel melting and less than 1% cladding strain) under all possible conditions, limits the required range for Thermal Overtemperature ΔT (OTΔT) protection, and provides a backup to the High Neutron Flux trip.

The OTΔT trip function provides sufficient core protection to preclude departure from nucleate boiling (DNB) over a range of operating and transient conditions. The setpoint is automatically varied with temperature, pressure, and the axial power distribution. The F(ΔI) penalty function adjusts the trip setpoint for axial peaks greater than design.

Revisions to the limiting safety system settings for the OTΔT and OPΔT trip functions (Table 2.2-1, Notes 1, 2, 3, and 4) are proposed to maintain consistency with the non-LOCA Accident Analysis. These trip functions provide primary protection against DNB and fuel centerline melting (excessive kw/ft) during postulated transients. The proposed settings have

2.2 NSSS DESIGN TRANSIENTS

The NSSS design transients evaluation for the Donald C. Cook Nuclear Plant Unit 1 SGTP Program was completed and confirmed that the NSSS design transients developed as part of the Donald C. Cook Nuclear Plant Units 1 and 2 Rerating Program continue to apply to Donald C. Cook Nuclear Plant Unit 1 at the increased SGTP conditions. The evaluation consisted of a comparison of the NSSS performance parameters for the SGTP Program with the parameters for the Rerating Program. The comparison concluded that the SGTP Program parameters that have the potential to impact the NSSS design transients (i.e., temperatures, pressures, and power levels) are bounded by the parameters used in the Rerating Program. The NSSS performance parameters that were not bounded in this manner (i.e., SGTP level and RCS flow) were evaluated and determined to not have a significant impact on the NSSS design transients. Overall, the evaluation concluded that the NSSS design transients developed as part of the Rerating Program continue to apply to Donald C. Cook Nuclear Plant Unit 1.

Program). These parameters were used for selected analyses, where high primary temperatures were limiting, and sensitive to RCS loop flow.

TABLE 2.1-1
COOK NUCLEAR PLANT UNIT 1 NSSS PERFORMANCE PARAMETERS
FOR SGTP PROGRAM

Parameter	(Unit 1, Original) Case 1
NSSS Power, MWt	3250
Core Power, MWt	3250
RCS Flow,(gpm/loop)*	88,500
Minimum Measured Flow, (total gpm)	361,600
RCS Temperatures, °F	
Core Outlet	602.0
Vessel Outlet	599.3
Core Average	570.5
Vessel Average	567.8
Vessel/Core Inlet	536.3
Steam Generator Outlet	536.3
Zero Load	547.0
RCS Pressure, psia	2250
Steam Pressure, psia	758
Steam Flow, (10^6 lb/hr.tot.)	14.12
Feedwater Temperature, °F	434.8
% SG Tube Plugging	0

Flow Definitions:

* RCS Flow (Thermal Design Flow) - The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based on this flow.

TABLE 2.1-1 (continued)
COOK NUCLEAR PLANT UNIT 1 NSSS PERFORMANCE PARAMETERS
FOR SGTP PROGRAM

Parameter	(Revised) Case 2	(Revised) Case 3	(Revised) Case 4	(Revised) Case 5
NSSS Power, MWt	3262	3262	3262	3262
Core Power, MWt	3250	3250	3250	3250
RCS Flow, (gpm/loop)*	83,200	83,200	79,000	79,000
Minimum Measured Flow, (total gpm)**	339,100	339,100	339,100	339,100
RCS Temperatures, °F				
Core Outlet	589.7	611.9	591.5	613.6
Vessel Outlet	586.8	609.1	588.5	610.8
Core Average	555.8	579.4	556.0	579.7
Vessel Average	553.0	576.3	553.0	576.3
Vessel/Core Inlet	519.2	543.5	517.5	541.8
Steam Generator Outlet	518.9	543.2	517.2	541.6
Zero Load	547.0	547.0	547.0	547.0
RCS Pressure, psia	2250 or 2100	2250 or 2100	2250 or 2100	2250 or 2100
Steam Pressure, psia	595	749	589	742
Steam Flow, (10^6 lb/hr.tot.)	14.12	14.17	14.12	14.17
Feedwater Temperature, °F	434.8	434.8	434.8	434.8
% SG Tube Plugging	30	30	30	30

Flow Definitions:

* RCS Flow (Thermal Design Flow) - The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based on this flow.

** Minimum Measured Flow - The flow specified in the Tech. Specs. which must be confirmed or exceeded by the flow measurements obtained during plant startup and is the flow used in reactor core DNB analyses for plants applying the Revised Thermal Design Procedure. MMF based on a flow measurement uncertainty of 1.9%. Analyses also bound a MMF of 341,100 gpm total which reflects a flow measurement uncertainty of 2.5%.

2.3 CONTROL/PROTECTION SYSTEM SETPOINTS

Control Systems were evaluated and found to be bounded by the analyses performed as part of the Rerating Program. These analyses reflected the objective of optimizing control parameters, primarily with respect to two aspects of plant behavior: stability of the control systems and operating margins to the various reactor protection system trips.

The flexibility identified during the Rerating Program to adjust the full load vessel average temperature and primary pressure as necessary on a cycle-to-cycle basis remains applicable to the SGTP Program. Control systems setpoints are selected for each fuel cycle from those analyzed for the Rerating Program. Therefore, the plant will be adequately stable for all SGTP Program operating conditions and operates with adequate margin to reactor protection system setpoints.

rod in the three phases. The Revised PAD Fuel Thermal Safety Model, described in References 13, generates the initial fuel rod conditions input to LOCBART.

SATAN-VI calculates the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown, information on the state of the system is transferred to the REFILL code which performs the calculation of the refill period to bottom of core recovery time. Once the vessel has refilled to the bottom of the core, the reflood portion of the transient begins. The BASH code is used to calculate the thermal-hydraulic simulation of the RCS for the reflood phase.

Information concerning the core boundary conditions is taken from all of the above codes and input to the LOCBART code for the purpose of calculating the core fuel rod thermal response for the entire transient. From the boundary conditions, LOCBART computes the fluid conditions and heat transfer coefficient for the full length of the fuel rod by employing mechanistic models appropriate to the actual flow and heat transfer regimes. Conservative assumptions ensure that the fuel rods modeled in the calculation represent the hottest rods in the entire core.

The large break analysis was performed with the December 1981 version of the Evaluation Model modified to incorporate the BASH (Reference 7) computer code.

Input Parameters and Initial Conditions:

The analysis presented in this section was performed with a reactor vessel upper head temperature equal to the RCS hot leg temperature and a uniform SGTP level of 30%. The analysis is also based on plant operation with the RHR cross-tie valves closed, and an EDG start time of 30 seconds which results in a safety injection delay time of 47 seconds. A list of plant input parameters used in the large break LOCA analysis is provided in Table 3.1-2.

A range of reactor operating temperatures was analyzed in order to justify plant operation at a reactor power level of 3250 MWt between 609.1°F to 586.8°F in the hot legs and 543.5°F and 519.2°F in the cold legs. In addition to the temperature range analyzed, initial RCS pressure was also varied to justify plant operation at 2250 and 2100 psia. A full spectrum break analysis was performed for nominal RCS conditions (initial RCS pressure of 2250 psia and hot leg temperature of 609.1°F) from which the limiting break size was determined. The limiting break was then reanalyzed for the reduced hot leg temperature of 586.8°F and nominal RCS pressure of 2250 psia. The limiting break was also reanalyzed for the nominal hot leg

2.0 BASIS FOR EVALUATIONS/ANALYSES PERFORMED

The purpose of the SGTP Program was to perform the necessary NSSS-related efforts to support an increase in the level of SGTP to as high as 30% and continue operational flexibility in terms of primary temperature and pressure. In addition to the change in parameters associated with the increased SGTP level, additional changes were incorporated into the analyses, as described in Section 1.0 (e.g., EDG delay time, pressurizer safety valve tolerance, etc.).

Previously, AEPSC submitted a report for NRC review in October 1988, which provided the necessary analysis, documentation, and licensing effort to support operation at reduced primary temperatures and pressures. These analyses were performed in an effort to reduce the propensity for the initiation and propagation of corrosion in the Cook Nuclear Plant Unit 1 Series 51 steam generator tubes. The Westinghouse input for this submittal was provided in WCAP 11902 (Reference 1). The efforts performed for WCAP 11902 supported 100% thermal power operation (3250 MWt core power) in the range of vessel average temperatures between 547°F and 576.3°F, at primary pressure values of 2100 psia and 2250 psia. The primary pressures were intended as two discrete values; the program was not structured to support a continuous range of primary pressures. The intent of the reduced primary pressure value is to minimize the primary to secondary pressure drop across the steam generator tubes at reduced temperature operation. In addition, the analyses and evaluations performed support a maximum average tube plugging level of 10%, with a peak steam generator tube plugging level of 15%.

Subsequently, a supplement to WCAP 11902 was issued as the Westinghouse input for a second submittal to the NRC to summarize the additional efforts performed to support a rerating of Cook Nuclear Plant Unit 1 and to provide part of the support for a Unit 2 rerating (Reference 2). The impact of this document on Cook Nuclear Plant Unit 1 is to support the licensing of a power uprating (in addition to supporting the range of operating conditions described above) to 3425 MWt NSSS. Only the reduced temperature and pressure portion of this program and associated operational improvements have been approved and implemented at this time. AEPSC currently selects the desired operating conditions from within the range addressed in the Rerating Program on a cycle-to-cycle basis. The efforts documented in WCAP-11902 and Supplement are referred to throughout this report as the "Rerating Program".

The RCS temperatures of the SGTP Program were chosen to be within the bounds of the Rerating Program. The two primary pressure values of 2100 psia and 2250 psia were evaluated. The maximum average and peak SGTP level was increased to 30%. A corresponding reduction in thermal design flow (TDF) and also a 5% loop flow asymmetry were also evaluated. Because the range of NSSS parameters was chosen to be within the bounds of the Rerating Program, many of the analyses performed for the Rerating Program (Reference 1 and 2) remain applicable to the SGTP Program. Upon approval of the analyses

and evaluations in this report, AEPSC will select the desired operating conditions from within the range addressed in the SGTP Program on a cycle to cycle basis.

References

1. WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report", October 1988

2. WCAP-11902, Supplement, "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 and 2 Licensing Report," September 1989

TABLE 3.1-14

SMALL-BREAK LOCA CALCULATIONS

±3% MAIN STEAM SAFETY VALVE SETPOINT TOLERANCE ANALYSIS

RESULTS

	Reduced Pressure, Reduced Temperature*	
	3-Inch	2-Inch
NOTRUMP Peak Clad Temperature (°F)	1951	1833
Peak Clad Temperature Location (ft)	12.0	12.0
Peak Clad Temperature Time (sec)	1890	4042
Local Zr/H_20 Reaction Maximum (%)	5.06	3.75
Local Zr/H_20 Reaction Location (ft)	12.0	12.0
Total Zr/H_20 Reaction (%)	< 1.0	< 1.0
Rod Burst	None	None
Burst and Blockage Penalty (°F)	117	15
Total Peak Clad Temperature (°F)	2068	1848

* Main steam safety valve setpoint tolerance increase case at 3250 MWt core power.

3.0 SAFETY EVALUATIONS/ANALYSES PERFORMED

3.1 LOSS OF COOLANT ACCIDENT (LARGE BREAK AND SMALL BREAK)

3.1.1 Large Break LOCA

A large break LOCA analysis was performed for Donald C. Cook Nuclear Plant Unit 1 to support an increase in the steam generator tube plugging level to 30%, while maintaining the operational flexibility of the plant by analyzing a range of initial RCS temperature conditions and two discrete RCS pressures. The large break analysis was performed with the 1981 version of the Westinghouse ECCS Evaluation Model using the BASH computer code. Analysis assumptions included 30% steam generator tube plugging, operation at a reactor power level of 3250 MWt with the RHR cross-tie valves closed, a total peaking factor of 2.15, and a hot channel enthalpy rise peaking factor of 1.55. Safety injection flows were based on pump head degradation of 15% for the high head safety injection pumps and RHR pumps, and 10% for the centrifugal pumps. The EDG start time was also increased to 30 seconds.

The analysis assumed a range of operating temperatures in order to justify plant operation between 609.1°F and 586.8°F in the hot legs and 543.5°F and 519.2°F in the cold legs. These temperature ranges represent the Unit 1 power capability parameters for 30% peak uniform steam generator tube plugging displayed in Table 2.1-1. Initial RCS pressure was also varied to justify plant operation at 2100 and 2250 psia. A full spectrum break analysis was performed for nominal RCS conditions (initial RCS pressure of 2250 psia and hot leg temperature of 609.1°F) from which the limiting break discharge coefficient was determined. The limiting break was then reanalyzed for the reduced hot leg temperature and nominal RCS pressure of 2250 psia, and also for the nominal hot leg temperature and RCS pressure of 2100 psia. The above cases were all analyzed with minimum safety injection flow. The limiting break was also analyzed with maximum safety injection flow.

The limiting break size was determined to be $C_D = 0.4$ at the nominal hot leg temperature ($T_{HOT} = 609.1°F$) and a pressure of 2100 psia with minimum safety injection flow. The peak cladding temperature was calculated to be 2164°F which is less than the 2200°limit in 10 CFR 50.46. A detailed description of the large break LOCA analysis is presented below.

<u>Identification of Causes and Frequency Classification</u>

A LOCA is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, 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 an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of Cook Nuclear Plant Unit 1, but is postulated as a conservative design basis.

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 (10 CFR 50.46 and Appendix K of 10 CFR 50, 1974 - 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 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, as required by the long-lived radioactivity remaining in the core.

These criteria were established to provide a significant margin in emergency core cooling system (ECCS) performance following a LOCA. WASH-1400 (USNRC 1975), Reference 2, presents a study in regards to the probability of occurrence of RCS pipe ruptures.

Sequence of Events and Systems Operations

Should a major break occur, depressurization of the RCS 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 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 supplement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. No credit is taken in the LOCA analysis for the boron content of the injection water. However, an average RCS/sump mixed boron concentration is calculated to ensure that the core remains subcritical. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis.

2. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

In the present Westinghouse design, the large break single-failure is the loss of one RHR (low head) pump. This means that credit could be taken for two charging pumps, two safety injection pumps, and one low head pump. The following is a discussion of the modeling

procedure for the minimum safeguards and the flow spilling from a break of an ECCS branch injection line (i.e., the spilling line assumptions).

The current procedure for large break analyses assumes that at least one train of ECCS is available for delivery of water to the RCS. Although the single failure is an RHR pump, only one pump in each subsystem is assumed to deliver to the primary loops. However, both EDGs are assumed to start in the modeling of the containment deck fans and sprays. Modeling full containment heat removal systems operation is required by Branch Technical Position CSB 6-1 and is conservative for the large break LOCA. The charging pump starts and delivers flow through the injection lines (one per loop) with one branch injection line spilling to the containment backpressure. To minimize delivery to the reactor, the branch line chosen to spill is selected as the one with the minimum resistance. When one safety injection pump and one low head residual heat removal pump start, flow is delivered to the RCS through the accumulator injection lines. Again, one line, with the minimum resistance, is assumed to spill to containment backpressure. In addition, the safety injection pump and low head RHR pump performance curves were degraded by 15%. For the charging pumps, the performance curves were degraded by 10% and a 25 gpm flow imbalance was assumed.

Therefore, in the large break ECCS analysis performed by Westinghouse, single failure is conservatively accounted for via the loss of an ECCS train, and the spilling of the minimum resistance injection line despite full containment active heat removal system operation (i.e., two EDGs).

The time sequence of events following a large break LOCA is presented in Table 3.1-1.

Before the break occurs, the unit 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 RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10 CFR 50 (Reference 1). Thereafter, the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes uncovered, both turbulent and laminar forced convection and radiation are considered as core heat transfer mechanisms.

The heat transfer between the RCS 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, the 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 signal actuates a feedwater isolation signal which isolates normal feedwater flow by closing the main feedwater isolation

valves, and also initiates emergency feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is made that accumulator water injected bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10 CFR 50. Since loss of offsite power (LOOP) is assumed, the RCPs 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 RCS pressure (initial values with uncertainty assumed to be 2317 psia or 2033 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowdown, the mechanisms that are responsible for the emergency core cooling water bypassing the core are calculated not to be effective. At this time (called end-of-bypass) refill of the reactor vessel lower plenum begins. Refill is completed 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 latter stage of blowdown and then the beginning-of-reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, 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 low head and high head safety injection pumps aid in the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been 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 (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold leg recirculation phase of operation in which spilled borated water is drawn from the engineered safety features (ESF) containment sumps by the low head safety injection (residual heat removal) pumps and returned to the RCS cold legs. The containment spray system continues to operate to further reduce containment pressure.

Approximately 12 hours after the initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel. Long-term cooling includes long-term criticality control. Criticality control is achieved by determining the RWST and accumulator concentration necessary to maintain subcriticality without credit for RCCA insertion. The necessary RWST and accumulator concentration is a

function of each core design and is checked each cycle. The current Technical Specification value is 2400 ppm to 2600 ppm boron (Reference 3).

Core and System Performance

Mathematical Model:

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50 (Federal Register 1974), Reference 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 RCS, the pressure and temperature transient within the containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

A description of the various aspects of the LOCA analysis methodology is given by Bordelon, Massie, and Zordan (Reference 4). This document describes 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, BASH and LOCBART codes, which are used in the LOCA analysis, are described in detail by Bordelon et al. (1974)[(5)]; Kelly et al. (Reference 6); Young et al. (Reference 7); and Bordelon et al. (Reference 4). Code modifications are specified in References 8, 9, 10 and 11. It is noted that the WREFLOOD code, which was previously used to calculate the RCS behavior during vessel lower plenum refill, has been replaced by the REFILL code as reported in Reference 18. The REFILL code is identical to the section of the WREFLOOD code that modeled the refill phase.

These codes assess the core heat transfer geometry and 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 RCS during blowdown and the REFILL computer code calculates this transient during the refill phase of the accident. The BASH code is used to determine the system response during the reflood phase of the transient. The LOTIC computer code, described by Hsieh and Raymund in WCAP-8355 and WCAP-8345 (Reference 12), calculates the containment pressure transient.

The containment pressure transient is input to BASH for the purpose of calculating the reflood transient. The LOCBART computer code calculates the thermal transient of the hottest fuel

temperature of 609.1°F and RCS pressure of 2100 psia. The cases analyzed are identified in Table 3.1-1.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from extensive sensitivity studies (Westinghouse 1974 (Reference 14); Salvatori 1974 (Reference 15); Johnson, Massie, and Thompson 1975 (Reference 16). In addition, 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 which were 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, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated.

Another input parameter that affects LOCA analysis results is the assumed axial power shape at the beginning of the accident. Large break LOCA analyses have been traditionally performed using a symmetric, chopped cosine axial power shape. Recent calculations have shown that there was a potential for top-skewed power distributions to result in peak cladding temperatures (PCT) greater than those calculated with a chopped cosine axial power distribution. Westinghouse has developed a process, which was applied to the Cycle 13 and 14 reloads for Cook Nuclear Plant Unit 1, that reasonably ensures that the cosine remains the limiting power distribution, by defining appropriate power distribution surveillance data. This process, called the power shape sensitivity model (PSSM), is described in a topical report (WCAP-12909-P), Reference 19, and further clarified in ET-NRC-91-3633, Reference 20, which are currently under NRC review. With implementation of the PSSM in the reload design process, top skewed axial power distributions that are potentially more limiting than the power distribution used in the ECCS analysis are reasonably precluded from occurring by the design and surveillance data provided to monitor the power distribution.

A meeting was held at the Westinghouse Licensing Office in Bethesda on December 17, 1981, between members of the U. S. Nuclear Regulatory Commission and members of the Westinghouse Nuclear Safety Department to discuss the impact of maximum safety injection on the large break ECCS analysis on a generic basis. Further discussion of this issue is provided in a letter from E. P. Rahe, Manager of Westinghouse Nuclear Safety Department, to Robert L. Tedesco of the U. S. Nuclear Regulatory Commission (Reference 17). A brief description of this issue is given below.

Westinghouse ECCS analyses currently assume minimum safeguards for the safety injection flow, which minimizes the amount of flow to the RCS by assuming maximum injection line resistances, degraded ECCS pump performance, and the loss of one RHR pump as the most limiting single failure. This is conservatively modeled as a loss of one train of safety injection, including RHR pump, safety injection pump and centrifugal charging pump. Both containment spray pumps are assumed operable. This is the limiting single failure assumption when offsite

power is unavailable for most Westinghouse plants. However, for some Westinghouse plants, the current nature of the Appendix K ECCS evaluation models is such that it may be more limiting to assume the maximum possible ECCS flow delivery. In that case, maximum safeguards which assume minimum injection line resistances, enhanced ECCS pump performance, and no single failure, result in the highest amount of flow delivered to the RCS. Therefore, the worst break for Cook Nuclear Plant Unit 1 (C_D = 0.4 for nominal hot leg temperature of 609.1°F and RCS pressure of 2100 psia) was reanalyzed assuming maximum safeguards.

Results:

Based on the results of the LOCA sensitivity studies (Westinghouse 1974 (Reference 14); Salvatori 1974 (Reference 15); Johnson, Massie, and Thompson 1975 (Reference 16) the limiting large break was found to be the double-ended cold leg (DECL) guillotine. Therefore, only the DECL guillotine break is considered in the large break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. The results of these calculations are summarized in Table 3.1-1.

The containment data used to generate the LOTIC backpressure transient are shown in Table 3.1-3. The mass and energy release data used for the limiting minimum safeguards case are shown in Table 3.1-4. Nitrogen release rates to the containment are given in Table 3.1-5.

Figures 3.1-1a through 3.1-19 present the results of the cases analyzed for the large break LOCA. The alpha designation in the figure number corresponds to the cases as described in Table 3.1-1.

<u>Figures 3.1-1a to 1f</u>	The system pressure shown is the calculated core pressure.
<u>Figures 3.1-2a to 2f</u>	The flow rate from the break is plotted as the sum of both ends of the guillotine break.
<u>Figures 3.1-3a to 3f</u>	The core pressure drop shown is from the lower plenum, near the core, to the upper plenum at the core outlet.
<u>Figures 3.1-4a to 4f</u>	The core flow is shown during the blowdown phase of the transient.
<u>Figures 3.1-5a to 5f</u>	The accumulator flow during blowdown is plotted as the sum of that injected into the intact cold legs.

Figures 3.1-6a to 6f	The core and downcomer collapsed liquid water level, and the core quench front are plotted during the reflood phase of the transient.
Figures 3.1-7a to 7f	The core inlet flow is shown as it is calculated during the reflood phase.
Figures 3.1-8a to 8f	The total accumulator and pumped ECCS flow injected into the intact cold legs during reflood is shown.
Figures 3.1-9a to 9f	The integral of the core inlet flow during reflood as calculated with BASH is plotted.
Figures 3.1-10a to 10f	The mass flux is plotted at the hot spot (the node which produced the peak clad temperature) on the hot rod.
Figures 3.1-11a to 11f	The heat transfer coefficient is plotted at the hot spot on the hot rod.
Figures 3.1-12a to 12f	The vapor temperature at the hot spot on the hot rod is plotted.
Figures 3.1-13a to 13f	The clad temperature at the hot spot is shown for the hot rod.
Figure 3.1-14	The containment pressure transient used in the analysis is provided for the minimum SI case.
Figures 3.1-15 to 18	These figures show the heat removal rates of the heat sinks found in the lower and upper compartment and the heat removal by the sump and lower compartment spray.
Figure 3.1-19	This figure shows the temperature transients in both the lower and upper compartments of containment.

As shown in Table 3.1-1, the limiting case for Cook Nuclear Plant Unit 1 is Case E (C_D = 0.4 for nominal hot leg temperature of 609.1°F and RCS pressure of 2100 psia). The maximum clad temperature calculated for a large break is 2164°F, which is less than the Acceptance Criteria limit of 2200°F. The maximum local metal-water reaction is 14.30 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46. The total core metal-water reaction for all breaks is less than the 1 percent criterion of 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

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3.1.2 Small Break LOCA

A small break LOCA analysis has been performed for the Donald C. Cook Unit 1 Nuclear Plant to support an increase in the steam generator tube plugging level to 30%, while maintaining the operational flexibility of the plant by demonstrating that the 10 CFR 50.46 Acceptance Criteria can be met for a range of initial RCS pressure and temperature conditions. The small break LOCA analysis was performed with the Westinghouse small break LOCA ECCS Evaluation Model using the NOTRUMP code[1][2], including the recent changes in Addendum 2[8] to incorporate modeling of safety injection into the broken loop and the COSI condensation model.

The key analysis input assumptions included 30% peak uniform steam generator tube plugging, operation at a reactor power level of 3250 MWt with the HHSI cross-tie discharge valves closed, a total peaking factor of 2.32, and a hot channel enthalpy rise peaking factor of 1.55. Also incorporated in the analysis are a reduced hot assembly average power and a power shape based on a reduced axial offset of +20%. Safety injection flows are based on pump head degradation of 15% for the high head safety injection pumps and 10% for the centrifugal charging pumps, and the emergency diesel generator start time was increased to 30 seconds.

The analysis was performed in order to bound plant operation within the range of RCS temperatures specified in the Unit 1 power capability parameters for 30% uniform steam generator tube plugging in Table 2.1-1, and at RCS pressures of 2100 and 2250 psia. A single break size analysis was performed at the previously-limiting break size of three inches. The calculation used the reduced temperature, reduced pressure operating condition which was previously demonstrated to be the limiting operating condition for the small break analysis. Based on an evaluation of the break spectrum and the range of operating conditions, it was concluded that the analyzed case would remain bounding with respect to peak clad temperature. The peak cladding temperature was calculated to be 1443°F which is less than the 2200°F limit in 10 CFR 50.46.

A detailed description of the analysis is presented in the following sections. Since the analysis to support 30% steam generator tube plugging is an extension of previous small break LOCA analyses performed for Cook Nuclear Plant Unit 1, the description also includes a discussion of the previous analyses. These include the Rerating Program analyses currently in the FSAR which were performed for a reactor power level of 3588 MWt, and the analysis performed for a reactor power level of 3250 MWt to support an increase in the main steam safety valve (MSSV) setpoint tolerance to ±3%.

3.1.2.1 Rerating Program Analysis

Analysis of Effects and Consequences

Method of Analysis

For loss-of-coolant accidents due to small breaks less than one square foot, the NOTRUMP computer code[1,2] 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 incorporating a number of advanced features. Among these are 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 NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants."

The reactor coolant system model is nodalized into volumes interconnected by flow paths. The broken loop is modelled explicitly, while the three intact loops are lumped into a second loop. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multinode capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations. Detailed descriptions of the NOTRUMP code and the evaluation model are provided in References 1 and 2.

Safety injection systems consist of gas pressurized accumulator tanks and pumped injection systems. Minimum emergency core cooling system availability is assumed for the analysis. Assumed pumped safety injection characteristics as a function of RCS pressure used as boundary conditions in the Rerating Program analysis are shown in Figure 3.1-20 and in Table 3.1-6. The safety injection flow rates presented are based on pump performance curves degraded 10 percent from the design head and are consistent with closure of the high head safety injection system cross-tie valve. The effect of flow from the RHR pumps is not considered in the small break analyses since their shutoff head is lower than the RCS pressure during the time portion of the transient considered here. Safety injection is delayed 27 seconds after the occurrence of the injection signal to account for diesel generator startup and emergency power bus loading in case of a loss of offsite power coincident with an accident.

Peak clad temperature calculations are performed with the LOCTA-IV[3] code using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture height as boundary conditions. Figure 3.1-21 depicts the hot rod axial power shape used to perform the small break analysis for the Rerating Program. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core. Such a distribution is limiting for small-break LOCAs because it minimizes coolant level swell, while maximizing vapor superheating and fuel rod heat generation in the uncovered elevations. The small break LOCA analysis assumes the core continues to operate at full rated power until the control rods are completely inserted.

Results

This section presents results of the limiting break analysis (as determined by the highest calculated peak fuel rod clad temperature) for a range of break sizes and RCS pressures and temperatures for a reactor power level of 3588 MWt. The limiting break was found to be a 3-inch diameter cold leg break initiated at reduced RCS pressure and temperature conditions. The maximum temperature attained during the transient was 2122°F. A list of input assumptions used in the Rerating Program analysis for reduced pressure and temperature conditions is provided in Table 3.1-7. The results of a three break spectrum analysis performed at reduced RCS pressure and temperature conditions are summarized in Table 3.1-8, while the key transient event times are listed in Table 3.1-9. Figures 3.1-22 through 3.1-29 show the limiting three-inch break transient, respectively:

- RCS pressure,
- Core mixture level,
- Peak clad temperature,
- Core outlet steam flow,
- Hot spot rod surface heat transfer coefficient,
- Hot spot fluid temperature,
- Cold leg break mass flow rate, and
- Safety injection mass flow rate.

During the initial period of the small-break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor recirculation cooling pumps as they coast down. Normal upward flow is maintained through the core and core heat is adequately removed. At the low heat generation rates following shutdown, the fuel rods continue to be well cooled as long as the core is covered by a two-phase mixture level. From the clad temperature transient for the 3-inch break calculation shown in Figure 3.1-24, it is seen that the peak clad temperature occurs near the time at which the core is most deeply uncovered when the top of the core is steam cooled. This time is also accompanied by the highest vapor superheating above the mixture level. A comparison of the total break flow to containment shown in Figure 3.1-28 to the safety injection flow rate shown in Figure 3.1-29 shows that at

the time the transient was terminated, the safety injection flow being delivered to the RCS exceeded the mass flow out the break. Although the inner vessel core mixture level has not yet covered the entire core, there is no longer a concem of exceeding the 10 CFR 50.46 criteria since the pressure is gradually decaying and there is a net mass inventory gain. As the RCS inventory continues to gradually increase, the core mixture level will continue to increase and the fuel clad temperatures will continue to decline.

Rerating Program Break Spectrum Cases

Studies documented in Reference 4 determined that the limiting small-break size occurred for breaks less than 10 inches in diameter. To ensure that the 3-inch diameter break was limiting for the reduced temperature and pressure RCS conditions, calculations were also run with breaks of 2 inches and 4 inches. The results of these calculations are shown in the Results Table 3.1-8 and Sequence of Events Table 3.1-9. Plots of the following parameters are shown in Figures 3.1-30 through 37 for the 2-inch break, and Figures 3.1-38 through 43 for the 4-inch break.

- RCS pressure,
- Core mixture level,
- Peak clad temperature,
- Core outlet steam flow,
- Hot spot rod surface heat transfer coefficient,
- Hot spot fluid temperature,
- Cold leg break mass flow rate, (for the 2-inch case only), and
- Safety injection mass flow rate (for the 2-inch case only).

As seen in Table 3.1-8, the maximum clad temperatures were calculated to be less than that for the 3-inch break.

Additional Rerating Program Analyses

Calculations were also performed for Cook Nuclear Plant Unit 1 with the NOTRUMP [1,2] and LOCTA-IV[3] codes to examine the influence of initial loop fluid operating temperatures and operating pressures on small break LOCA peak clad temperature. These additional analyses confirmed that the most limiting PCT result was that from the reduced temperature and pressure limiting 3-inch diameter break described previously.

To support operation of the Cook Nuclear Plant Unit 1 at RCS pressures of 2100 psia and 2250 psia for a range of loop operating temperatures, two additional analyses were performed. Calculations were performed for a 3-inch diameter break for an initial RCS pressure of 2250 psia at initial loop fluid operating temperatures corresponding to T_{avg} program setpoints of 547°F and 578°F. The results of these calculations are shown in the Results Table 3.1-10

and the Sequence of Events Table 3.1-11. Plots of the following parameters are shown in Figures 3.1-44 through 51 for the reduced temperature and high pressure case, and Figures 3.1-52 through 59 for the high temperature and high pressure case.

- RCS pressure,
- Core mixture level,
- Peak clad temperature,
- Core outlet steam flow,
- Hot spot rod surface heat transfer coefficient,
- Hot spot fluid temperature,
- Cold leg break mass flow rate, and
- Safety injection mass flow rate.

As seen in Table 3.1-10, the maximum clad temperatures were calculated to be less than that for the 3-inch break initiated at reduced temperature and pressure conditions.

NUREG-0737[5], Section II.K.3.31, required plant-specific small break LOCA analysis using an Evaluation Model revised per Section II.K.3.30. In accordance with NRC Generic Letter 83-65[6], generic analyses using NOTRUMP[1,2] were performed and are presented in WCAP-11145[7]. Those results demonstrate that in a comparison of cold leg, hot leg and pump suction leg break locations, the cold leg break location is limiting.

3.1.2.2 Main Steam Safety Valve Setpoint Tolerance Relaxation Analysis

Additional small break LOCA analyses were performed at a reactor power level of 3250 MWt to support an increase in the MSSV lift setpoint tolerance from ±1% to ±3%. Prior to the analysis performed to support MSSV tolerance relaxation, this mode of operation was supported by an evaluation limiting core power to 3250 MWt. The MSSV analyses were performed for operation with the HHSI cross-tie valves closed and assuming a 25 gpm charging pump flow imbalance. This resulted in a reduction in the charging pump flow, and thus a reduction in the total safety injection flow rate relative to the Rerating Program analysis. The limiting 3-inch break for reduced pressure and reduced temperature operating conditions was analyzed, since the Rerating Program analysis demonstrates that this case results in the most limiting clad temperature. Since the basis for the limiting case determination remains valid, it was not necessary to analyze the full spectrum of cases. However, an analysis was also performed for a 2-inch break since a reduction in safety injection flow rate can potentially shift the limiting break to a smaller break size. The analysis for the 2-inch break confirmed that the limiting break did not shift to a smaller break size.

A list of the plant input parameters for the ±3% MSSV setpoint tolerance analysis is provided in Table 3.1-12. The results of the limiting 3-inch break analysis are presented in the Sequence of Events Table 3.1-13 and the Results Table 3.1-14. Results of the non-limiting 2-inch case are also provided in Tables 3.1-13 and 3.1-14.

Plots of the following parameters for the 3-inch break analysis are shown in Figures 3.1-60 through 3.1-67, and for the 2-inch break in Figures 3.1-69 through 3.1.2-76:

- RCS pressure
- Core mixture level
- Peak clad temperature
- Core outlet steam flow
- Hot spot rod surface heat transfer coefficient
- Hot spot fluid temperature
- Cold leg break mass flow rate, and
- Safety injection mass flow rate

Figure 3.1-68 contains the power shape for a core power level of 3250 MWt which is applicable to both cases.

The 3-inch break with HHSI cross-ties closed, initiated at reduced pressure and temperature operating conditions and a core power level of 3250 MWt, represents the licensing basis small break analysis for an increased MSSV setpoint tolerance of ±3%. Application of a burst and blockage penalty resulted in a peak clad temperature of 2068°F, which was less than the 2200°F limit.

3.1.2.3 30% Steam Generator Tube Plugging Analysis

An additional small break LOCA analysis has been performed to support an increase in steam generator tube plugging level from 15% to a maximum of 30% in each steam generator. The analysis was performed in order to bound plant operation between 609.1°F and 586.8°F in the hot legs and 543.5°F and 519.2°F in the cold legs. These temperature ranges are defined in the Unit 1 power capability parameters for 30% peak uniform steam generator tube plugging displayed in Table 2.1-1. The analysis also supports plant operation at RCS pressures of 2100 and 2250 psia. The analysis was performed for the limiting 3-inch break with HHSI cross-tie valves closed at reduced pressure and reduced temperature operating conditions and a core power level of 3250 MWt, which was previously demonstrated to result in the most limiting clad temperature. An evaluation of the basis for the limiting case determination was performed and it was concluded that it was not necessary to perform a full break spectrum for this case.

The analysis incorporates a 20 second increase in emergency diesel generator starting delay to 30 seconds, which results in a total safety injection delay of 47 seconds after the occurrence of the injection signal. The safety injection flow rates used in the analysis include a 5% increase in high head safety injection pump degradation, for a total of 15% degradation. For the high head charging pumps, the performance curve degradation remained at 10% and a 25 gpm flow imbalance was assumed. The analysis also includes a reduction in the maximum axial offset from +30% to +20% and a reduction in the maximum hot assembly peaking factor from 1.433 to 1.38, with a corresponding change in the axial power shape used in the analysis. The use of the revised axial offset and hot assembly power factor in the small break LOCA analysis are consistent with the current core design and operation limits. An evaluation of up to 5% RCS loop flow asymmetry was also performed to support the analysis. A list of the plant input parameters for the 30% SGTP analysis is provided in Table 3.1-15.

Previously, safety injection into the broken loop was not modeled in the Westinghouse small break LOCA analyses since it was assumed that the additional safety injection would be a benefit. Because recent studies have shown that the response to broken loop safety injection can result in an increase in the calculated PCT, modeling of safety injection into the broken loop has now been incorporated into the NOTRUMP small break evaluation model. A more realistic model for condensation of steam by pumped safety injection based on data from the COSI test facility has also been incorporated, which provides a benefit larger than the penalty for safety injection in the broken loop. The methodology for modeling safety injection to the broken loop in small break LOCA analyses and application of the COSI condensation model are presented in the NOTRUMP Small Break ECCS Evaluation Model, Addendum 2[8]. The analysis for 30% steam generator tube plugging modeled the pumped safety injection and an accumulator in the broken loop, and used the more realistic COSI condensation model in Reference 8.

The results of the 3-inch break analysis are presented in the Sequence of Events Table 3.1-16 and the Results Table 3.1-17. Plots of the following parameters for the 3-inch break analysis are shown in Figures 3.1-77 through 3.1-85:

- RCS pressure
- Core mixture level
- Hot Spot Clad Temperature
- Core Outlet Steam Flow
- Hot spot rod surface heat transfer coefficient
- Hot spot fluid temperature
- Cold leg break mass flow rate
- Broken loop safety injection mass flow rate, and
- Lumped intact loop safety injection mass flow rate

Figure 3.1-86 contains the power shape used in the analysis.

Due to the modeling of safety injection in the broken loop with the COSI condensation model change, in conjunction with the reduced peaking factors, the PCT for the 30% steam generator tube plugging small break LOCA analysis is lower than for previous small break analyses. Because no rod burst was calculated to occur and the beginning of life calculated peak clad temperature is low enough to preclude a Zr/H_2O reaction temperature excursion following burst, no burst and blockage penalty is applied. The resulting total peak clad temperature of 1443°F is less than the 2200°F limit.

Small Break LOCA Analysis Conclusions

Analyses presented in this section show that the high head portion of the emergency core cooling system, together with the accumulators, provide sufficient core flooding to keep the calculated peak clad temperatures below required limits of 10 CFR 50.46. Hence, adequate protection is afforded by the emergency core cooling system in the event of a small break loss-of-coolant accident.

References

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TABLE 3.1-1
LARGE BREAK LOCA RESULTS

	Case A C_D=0.4 T_{HOT}= 609.1°F P=2250 psia Min. SI	Case B C_D=0.6 T_{HOT}= 609.1°F P=2250 psia Min. SI	Case C C_D=0.8 T_{HOT}= 609.1°F P=2250 psia Min. SI	Case D C_D=0.4 T_{HOT}= 586.8°F P=2250 psia Min. SI	Case E C_D=0.4 T_{HOT}= 609.1°F P=2100 psia Min. SI	Case F C_D=0.4 T_{HOT}= 609.1°F P=2100 psia Max. SI
Peak Clad Temperature (°F)	2069	1993	1965	2036	2164	2149
Peak Clad Location (ft)	5.75	6.25	6.25	6.00	6.25	6.25
Local Zr/H_2O Reaction (Max %)	7.59	8.19	6.62	8.45	14.30	12.01
Local Zr/H_2O Location (ft)	5.75	6.00	6.00	6.00	6.25	6.25
Total Zr/H_2O Reaction (%)	<1.0	<1.0	<1.0	<1.0	<1.0	<1.0
Hot Rod Burst Time (s)	43.6	41.8	45.7	46.5	42.0	42.0
Hot Rod Burst Loc. (ft)	5.75	6.00	6.00	6.00	6.25	6.25

TABLE 3.1-1 (continued)
LARGE BREAK LOCA RESULTS

	Case A C_D=0.4 T_{HOT}= 609.1°F P=2250 psia Min. SI	Case B C_D=0.6 T_{HOT}= 609.1°F P=2250 psia Min. SI	Case C C_D=0.8 T_{HOT}= 609.1°F P=2250 psia Min. SI	Case D C_D=0.4 T_{HOT}= 586.8°F P=2250 psia Min. SI	Case E C_D=0.4 T_{HOT}= 609.1°F P=2100 psia Min. SI	Case F C_D=0.4 T_{HOT}= 609.1°F P=2100 psia Max. SI
Start	0.0	0.0	0.0	0.0	0.0	0.0
Reactor Trip Signal	0.64	0.64	0.63	0.55	0.49	0.49
Safety Injection Signal	4.80	4.60	4.50	4.40	4.10	4.10
Accumulator Injection	18.70	13.90	11.60	17.80	18.70	18.70
End of Blowdown	40.75	31.77	28.05	40.61	39.96	39.96
Pump Injection	51.80	51.60	51.50	51.50	51.10	51.10
Bottom of Core Recovery	54.30	44.60	41.80	55.30	54.20	54.00
Accumulator Empty	69.09	62.30	48.75	70.09	68.96	69.78

TABLE 3.1-2
PLANT INPUT PARAMETERS USED IN LARGE BREAK LOCA ANALYSIS

Core Power (MWt)	102% of 3250
Peak Linear Power (kW/ft)	102% of 14.434
Total Core Peaking Factor, F_Q	2.15
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.55
Maximum Assembly Average Power, $P_{\overline{HA}}$	1.38
Fuel Assembly Array	15 X 15 OFA
Steam Generator Tube Plugging Level (%)	30
Accumulator Water Volume (ft^3/tank)	946
Accumulator Tank Volume (ft^3/tank)	1350
Minimum Accumulator Gas Pressure (psia)	600
Accumulator Water Temperature (°F)	100
Refueling Water Storage Tank Temperature (°F)	70 - 105
Thermal Design Flowrate (gpm/loop)	83,200
RCS Loop Average Temperature (°F)	553.0 and 576.3
Nominal Initial RCS Pressure (psia)	2100 and 2250
Nominal Steam Pressure (psia)	595 and 749
Safety Injection Delay Time (sec)	47
RHR Pump Head Degradation (%)	15
HHSI Pump Head Degradation (%)	15
Charging Pump Head Degradation (%)	10
Charging Pump Flow Imbalance (gpm)	25
RHR Cross-Tie Valve Position	Closed

TABLE 3.1-3
LARGE BREAK CONTAINMENT DATA
(ICE CONDENSER CONTAINMENT)

NET FREE VOLUME		
(Includes Distribution Between Upper, Lower, and Dead-Ended Compartments)	UC LC DE IC	746,829 ft^3 249,446 ft^3 116,168 ft^3 163,713 ft^3
Initial Conditions		
Pressure		14.7 psia
Maximum Temperature for the Upper, Lower, and Dead-Ended Compartments	UC LC DE	100°F 120°F 120°F
Minimum Temperature for the Upper, Lower, and Dead-Ended Compartments	UC LC DE	60°F 60°F 60°F
RWST Temperature		70°F
Temperature Outside Containment		-22°F
Initial Spray Temperature		70°F
Spray System		
Runout Flow for a Spray Pump		3600 gpm
Number of Spray Pumps Operating		2
Post-Accident Initiation of Spray System		36 sec
Distribution of Spray Flow to the Upper and Lower Compartments	LC UC	2700 gpm 4500 gpm
Deck Fan		
Post-Accident Initiation of Deck Fans		480 sec
Flow Rate per Fan		43,890 cfm per fan
Assumed Spray Efficiency of Water from Ice Condenser Drains		100%

TABLE 3.1-3 (continued)
STRUCTURAL HEAT SINKS

Wall	Compartment	Area (ft^2)	Thickness (ft)	Material
1	LC	12,105	0.0469/2.0	Steel/concrete
2	LC	11,701	2.0	Concrete
3	LC	65,979	4.0	Concrete
4	LC	5,462	0.0833	Steel
5	LC	5,273	0.0103	Steel
6	LC	290	0.25	Lead
7	LC	14,896	0.0078	Steel
8	LC	4,515	0.1042	Steel
9	LC	5,775	0.009	Steel
10	LC	57,317	0.00833	Steel
11	LC	9,404	0.0313	Steel
12	LC	2,623	0.0313	Steel
13	UC	378	0.0365/0.1667	Steel/concrete
14	UC	34,895	0.0078	Steel
15	UC	8,060	0.0208	Steel
16	UC	420	0.0052	Steel
17	UC	29,332	2.0	Concrete
18	UC	34,125	0.0469/2.0	Steel/concrete
19	UC	420	0.0052	Steel

UC: Upper Compartment
LC: Lower Compartment
DE: Dead-Ended Compartment
IC: Ice Compartment

TABLE 3.1-4
MASS AND ENERGY RELEASE RATES, MINIMUM SI

Time (sec)	Mass (lbm/sec)	Energy (BTU/sec)
0	57910	$3.081(10^7)$
2	48870	$2.542(10^7)$
4	33500	$1.762(10^7)$
6	25260	$1.357(10^7)$
8	22660	$1.223(10^7)$
10	19580	$1.096(10^7)$
12	16980	$9.838(10^6)$
12.4	16000	$9.346(10^6)$
14	14530	$8.608(10^6)$
16	12140	$7.313(10^6)$
18	10410	$6.254(10^6)$
20	9170	$5.472(10^6)$
24	7010	$3.871(10^6)$
28	6750	$2.839(10^6)$
32	5640	$1.757(10^6)$
36	3580	$7.951(10^5)$
40	4390	$9.057(10^5)$
52	230	$1.267(10^4)$
65	280	$6.321(10^4)$
75	390	$2.073(10^5)$
86	810	$2.884(10^5)$
95	420	$2.464(10^5)$
124	400	$1.666(10^5)$
206	430	$1.452(10^5)$
294	330	$1.314(10^5)$

TABLE 3.1-5
NITROGEN MASS AND ENERGY RELEASE RATES

Time (sec)	Flow Rate (lbm/sec)
69.2	231.8
73.2	166.4
77.2	120.8
81.2	87.3
85.2	62.1
89.2	42.9
93.2	28.8
97.2	19.5
101.2	14.1
105.2	11.1
109.2	9.0
113.2	7.3
117.2	5.9
121.2	4.8
125.2	3.9
129.2	3.2
137.2	2.1
141.2	1.8
145.2	1.4
153.2	1.0
161.2	0.7
169.2	0.5
177.2	0.3

TABLE 3.1-6

SAFETY INJECTION FLOW RATE

RERATING PROGRAM ANALYSIS

RCS PRESSURE (psia)	HHSI FLOW (lb/sec)	CHARGING FLOW (lb/sec)	TOTAL FLOW (lb/sec)
415	20.48	37.32	57.80
515	19.38	35.29	54.67
615	18.18	33.29	51.47
715	16.89	31.26	48.15
815	15.54	29.06	44.60
915	14.09	26.80	40.89
1015	12.50	24.48	36.98
1115	10.46	22.16	32.62
1215	7.99	19.74	27.73
1315	4.48	17.27	22.05
1415	0.00	14.70	14.70
1515	0.00	12.02	12.02

TABLE 3.1-7

PLANT INPUT PARAMETERS USED IN SMALL BREAK LOCA ANALYSIS

RERATING PROGRAM ANALYSIS

Core Power	102% of 3588 MWt
Total Core Peaking Factor	2.32
Steam Generator Tube Plugging Level	15% (peak uniform)
Accumulator Conditions:	
Cover Gas Pressure	600 psia
Water Volume	946.0 ft^3
Total Volume	1350 ft^3
RCS Initial Conditions:	
Reduced Temperature, Reduced Pressure Case	
Loop Temperatures Consistent With T_{avg} Program Setpoint of,	547°F
Pressure	2100 psia
Vessel Flowrate	354000 gpm

TABLE 3.1-7

PLANT INPUT PARAMETERS USED IN SMALL BREAK LOCA ANALYSIS

RERATING PROGRAM ANALYSIS

Core Power	102% of 3588 MWt
Total Core Peaking Factor	2.32
Steam Generator Tube Plugging Level	15% (peak uniform)
Accumulator Conditions:	
Cover Gas Pressure	600 psia
Water Volume	946.0 ft^3
Total Volume	1350 ft^3
RCS Initial Conditions:	
Reduced Temperature, Reduced Pressure Case	
Loop Temperatures Consistent With T_{avg} Program Setpoint of,	547°F
Pressure	2100 psia
Vessel Flowrate	354000 gpm

TABLE 3.1-8

SMALL-BREAK LOCA CALCULATION

RERATING PROGRAM ANALYSIS

RESULTS

PARAMETER	VALUE Reduced Temperature, Reduced Pressure Break Size: 2-Inch	3-Inch	4-Inch
Peak clad temperature (°F)	1899	2122	1414
Elevation (ft)	12.00	12.00	11.25
Zr/H_2O cumulative reaction			
Maximum local (%)	7.16	7.70	0.25
Elevation (ft)	12.00	12.00	11.50
Total core (%)	< 0.3	< 0.3	< 0.3
Rod Burst	None	None	None

CALCULATION:

NSSS Power MWt 102% of	3588*
Peak Linear Power kW/ft 102% of	16.426
Hot Rod Power Distribution (kW/ft)	See Figure 3.1-21
Accumulator Water Volume, cu. ft.	946

* Does not include pump heat.

TABLE 3.1-8

SMALL-BREAK LOCA CALCULATION

RERATING PROGRAM ANALYSIS

RESULTS

PARAMETER	VALUE Reduced Temperature, Reduced Pressure Break Size: 2-Inch	3-Inch	4-Inch
Peak clad temperature (°F)	1899	2122	1414
Elevation (ft)	12.00	12.00	11.25
Zr/H_2O cumulative reaction			
Maximum local (%)	7.16	7.70	0.25
Elevation (ft)	12.00	12.00	11.50
Total core (%)	< 0.3	< 0.3	< 0.3
Rod Burst	None	None	None

CALCULATION:

NSSS Power MWt 102% of	3588*
Peak Linear Power kW/ft 102% of	16.426
Hot Rod Power Distribution (kW/ft)	See Figure 3.1-21
Accumulator Water Volume, cu. ft.	946

* Does not include pump heat.

TABLE 3.1-9

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

RERATING PROGRAM ANALYSIS

Small-Break Loss of Coolant Accident

Event	Time (s) Reduced Temperature, Reduced Pressure Break Size: 2-Inch	3-Inch	4-Inch
Break occurs	0	0	0
Reactor trip signal	25.37	11.24	6.85
Safety injection signal	36.54	17.10	10.74
Start of safety injection delivery	63.54	44.10	37.74
Loop seal venting	1634.4	652.1	420.4
Loop seal core uncovery	N/A	645.8	424.6
Loop seal core recovery	N/A	680.3	439.2
Boil-off core uncovery	2216.7	1045.7	696.5
Accumulator injection begins	N/A	1711.5	901.0
Peak clad temperature occurs	4143.8	1958.7	969.5
Top of core covered	N/A	N/A	1982.7
SI flow exceeds break flow	4587.5	2197.1	N/A

TABLE 3.1-9

TIME-SEQUENCE OF EVENTS FOR CONDITION III EVENTS

RERATING PROGRAM ANALYSIS

Small-Break Loss of Coolant Accident

Event	Time (s) Reduced Temperature, Reduced Pressure Break Size: 2-Inch	3-Inch	4-Inch
Break occurs	0	0	0
Reactor trip signal	25.37	11.24	6.85
Safety injection signal	36.54	17.10	10.74
Start of safety injection delivery	63.54	44.10	37.74
Loop seal venting	1634.4	652.1	420.4
Loop seal core uncovery	N/A	645.8	424.6
Loop seal core recovery	N/A	680.3	439.2
Boil-off core uncovery	2216.7	1045.7	696.5
Accumulator injection begins	N/A	1711.5	901.0
Peak clad temperature occurs	4143.8	1958.7	969.5
Top of core covered	N/A	N/A	1982.7
SI flow exceeds break flow	4587.5	2197.1	N/A

TABLE 3.1-10

SMALL-BREAK LOCA CALCULATION

RERATING PROGRAM ANALYSIS

RESULTS

PARAMETER	VALUE High Temp. High Pressure 3-Inch	 Reduced Temp. High pressure 3-Inch
Peak clad temperature (°F)	1756	1887
Elevation (ft)	11.75	11.75
Zr/H_2O cumulative reaction		
Maximum local (%)	1.99	4.64
Elevation (ft)	11.75	11.75
Total core (%)	< 0.3	< 0.3
Rod Burst	None	None

CALCULATION:

NSSS Power MWt 102% of	3588*
Peak Linear Power kW/ft 102% of	16.426
Hot Rod Power Distribution (kW/ft)	See Figure 3.1-21
Accumulator Water Volume, cu. ft.	946

* Does not include pump heat.

TABLE 3.1-10

SMALL-BREAK LOCA CALCULATION

RERATING PROGRAM ANALYSIS

RESULTS

PARAMETER	VALUE High Temp. High Pressure 3-Inch	VALUE Reduced Temp. High pressure 3-Inch
Peak clad temperature (°F)	1756	1887
Elevation (ft)	11.75	11.75
Zr/H_2O cumulative reaction		
Maximum local (%)	1.99	4.64
Elevation (ft)	11.75	11.75
Total core (%)	< 0.3	< 0.3
Rod Burst	None	None

CALCULATION:

NSSS Power MWt 102% of	3588*
Peak Linear Power kW/ft 102% of	16.426
Hot Rod Power Distribution (kW/ft)	See Figure 3.1-21
Accumulator Water Volume, cu. ft.	946

* Does not include pump heat.

TABLE 3.1-11

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

RERATING PROGRAM ANALYSIS

Small-Break Loss of Coolant Accident

Event	Time (s) High Temp. High Pressure	Time (s) Reduced Temp. High Pressure
	3-Inch	3-Inch
Break occurs	0	0
Reactor trip signal	19.03	15.97
Safety injection signal	29.74	20.95
Start of safety injection delivery	51.74	47.95
Loop seal venting	666.96	698.78
Loop seal core uncovery	N/A	691.54
Loop seal core recovery	N/A	726.85
Boil-off core uncovery	1070.4	1166.8
Accumulator injection begins	1672.0	1855.2
Peak clad temperature occurs	1793.7	1986.2
Top of core covered	N/A	N/A
SI flow exceeds break flow	2022.0	2282.7

TABLE 3.1-11

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

RERATING PROGRAM ANALYSIS

Small-Break Loss of Coolant Accident

Event	Time (s) High Temp. High Pressure	Time (s) Reduced Temp. High Pressure
	3-Inch	3-Inch
Break occurs	0	0
Reactor trip signal	19.03	15.97
Safety injection signal	29.74	20.95
Start of safety injection delivery	51.74	47.95
Loop seal venting	666.96	698.78
Loop seal core uncovery	N/A	691.54
Loop seal core recovery	N/A	726.85
Boil-off core uncovery	1070.4	1166.8
Accumulator injection begins	1672.0	1855.2
Peak clad temperature occurs	1793.7	1986.2
Top of core covered	N/A	N/A
SI flow exceeds break flow	2022.0	2282.7

TABLE 3.1-12

PLANT INPUT PARAMETERS USED IN SMALL BREAK LOCA ANALYSIS

±3% MAIN STEAM SAFETY VALVE SETPOINT TOLERANCE ANALYSIS

Core Power (MWt)	102% of 3250
Peak Linear Power (kW/ft)	102% of 14.921
Total Core Peaking Factor, F_Q	2.32
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.55
Maximum Assembly Average Power, $\bar{P}_{HA}$	1.433
Axial Offset (%)	+30
Fuel Assembly Array	15 X 15 OFA
Steam Generator Tube Plugging Level (%)	15
Accumulator Water Volume (ft^3/tank)	946
Accumulator Tank Volume (ft^3/tank)	1350
Minimum Accumulator Gas Pressure (psia)	600
Accumulator Water Temperature (°F)	130
Refueling Water Storage Tank Temperature (°F)	120
Thermal Design Flowrate (gpm/loop)	88,500
RCS Loop Average Temperature (°F)	547.0
Nominal Initial RCS Pressure (psia)	2100
Nominal Steam Pressure (psia)	607
Safety Injection Delay Time (sec)	27
HHSI Pump Head Degradation (%)	10
Charging Pump Head Degradation (%)	10
Charging Pump Flow Imbalance (gpm)	25
HHSI Cross-Tie Valve Position	Closed
Auxiliary Feedwater Total Flowrate (gpm)	750

TABLE 3.1-12

PLANT INPUT PARAMETERS USED IN SMALL BREAK LOCA ANALYSIS

±3% MAIN STEAM SAFETY VALVE SETPOINT TOLERANCE ANALYSIS

Core Power (MWt)	102% of 3250
Peak Linear Power (kW/ft)	102% of 14.921
Total Core Peaking Factor, F_Q	2.32
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.55
Maximum Assembly Average Power, $\bar{P}_{HA}$	1.433
Axial Offset (%)	+30
Fuel Assembly Array	15 X 15 OFA
Steam Generator Tube Plugging Level (%)	15
Accumulator Water Volume (ft^3/tank)	946
Accumulator Tank Volume (ft^3/tank)	1350
Minimum Accumulator Gas Pressure (psia)	600
Accumulator Water Temperature (°F)	130
Refueling Water Storage Tank Temperature (°F)	120
Thermal Design Flowrate (gpm/loop)	88,500
RCS Loop Average Temperature (°F)	547.0
Nominal Initial RCS Pressure (psia)	2100
Nominal Steam Pressure (psia)	607
Safety Injection Delay Time (sec)	27
HHSI Pump Head Degradation (%)	10
Charging Pump Head Degradation (%)	10
Charging Pump Flow Imbalance (gpm)	25
HHSI Cross-Tie Valve Position	Closed
Auxiliary Feedwater Total Flowrate (gpm)	750

TABLE 3.1-13

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

±3% MAIN STEAM SAFETY VALVE SETPOINT TOLERANCE ANALYSIS

Small-break Loss of Coolant Accident

Event	Time (s) Reduced Pressure, Reduced Temperature* 3-Inch	2-Inch
Break Occurs	0.0	0.0
Reactor trip signal	8.64	19.03
Safety injection signal	17.13	37.11
Start of safety injection	44.13	64.11
Start of auxiliary feedwater delivery	68.6	79.1
Loop seal venting	592	1390
Loop seal core uncovery	N/A	N/A
Loop seal core recovery	N/A	N/A
Boil-off core uncovery	984	2312
Accumulator injection begins	1680	N/A
Peak clad temperature occurs	1890	4042
Top of core covered	N/A	N/A
SI flow rate exceeds break flow rate	1890	4091

* Main steam safety valve setpoint tolerance increase case at 3250 MWt core power.

TABLE 3.1-13

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

±3% MAIN STEAM SAFETY VALVE SETPOINT TOLERANCE ANALYSIS

Small-break Loss of Coolant Accident

Event	Time (s) Reduced Pressure, Reduced Temperature* 3-Inch	2-Inch
Break Occurs	0.0	0.0
Reactor trip signal	8.64	19.03
Safety injection signal	17.13	37.11
Start of safety injection	44.13	64.11
Start of auxiliary feedwater delivery	68.6	79.1
Loop seal venting	592	1390
Loop seal core uncovery	N/A	N/A
Loop seal core recovery	N/A	N/A
Boil-off core uncovery	984	2312
Accumulator injection begins	1680	N/A
Peak clad temperature occurs	1890	4042
Top of core covered	N/A	N/A
SI flow rate exceeds break flow rate	1890	4091

* Main steam safety valve setpoint tolerance increase case at 3250 MWt core power.

TABLE 3.1-14

SMALL-BREAK LOCA CALCULATIONS

±3% MAIN STEAM SAFETY VALVE SETPOINT TOLERANCE ANALYSIS

RESULTS

	Reduced Pressure, Reduced Temperature*	
	3-Inch	2-Inch
NOTRUMP Peak Clad Temperature (°F)	1951	1833
Peak Clad Temperature Location (ft)	12.0	12.0
Peak Clad Temperature Time (sec)	1890	4042
Local Zr/$H_2$0 Reaction Maximum (%)	5.06	3.75
Local Zr/$H_2$0 Reaction Location (ft)	12.0	12.0
Total Zr/$H_2$0 Reaction (%)	< 1.0	< 1.0
Rod Burst	None	None
Burst and Blockage Penalty (°F)	117	15
Total Peak Clad Temperature (°F)	2068	1848

* Main steam safety valve setpoint tolerance increase case at 3250 MWt core power.

TABLE 3.1-15

PLANT INPUT PARAMETERS USED IN SMALL BREAK LOCA ANALYSIS

30% SGTP PROGRAM ANALYSIS WITH HHSI CROSS-TIES CLOSED

Core Power (MWt)	102% of 3250
Peak Linear Power (kW/ft)	102% of 14.12
Total Core Peaking Factor, F_Q	2.32
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.55
Maximum Assembly Average Power, $\overline{P}_{HA}$	1.38
Axial Offset (%)	+20
Fuel Assembly Array	15 X 15 OFA
Steam Generator Tube Plugging Level (%)	30
Accumulator Water Volume (ft^3/tank)	946
Accumulator Tank Volume (ft^3/tank)	1350
Minimum Accumulator Gas Pressure (psia)	600
Accumulator Water Temperature (°F)	130
Refueling Water Storage Tank Temperature (°F)	120
Thermal Design Flowrate (gpm/loop)	83,200
RCS Loop Average Temperature (°F)	553.0
Nominal Initial RCS Pressure (psia)	2100
Nominal Steam Pressure (psia)	595
Safety Injection Delay Time (sec)	47
HHSI Pump Head Degradation (%)	15
Charging Pump Head Degradation (%)	10
Charging Pump Flow Imbalance (gpm)	25
HHSI Cross-Tie Valve Position	Closed
Auxiliary Feedwater Total Flowrate (gpm)	750

TABLE 3.1-15

PLANT INPUT PARAMETERS USED IN SMALL BREAK LOCA ANALYSIS

30% SGTP PROGRAM ANALYSIS WITH HHSI CROSS-TIES CLOSED

Core Power (MWt)	102% of 3250
Peak Linear Power (kW/ft)	102% of 14.12
Total Core Peaking Factor, F_Q	2.32
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.55
Maximum Assembly Average Power, $\overline{P}_{HA}$	1.38
Axial Offset (%)	+20
Fuel Assembly Array	15 X 15 OFA
Steam Generator Tube Plugging Level (%)	30
Accumulator Water Volume (ft^3/tank)	946
Accumulator Tank Volume (ft^3/tank)	1350
Minimum Accumulator Gas Pressure (psia)	600
Accumulator Water Temperature (°F)	130
Refueling Water Storage Tank Temperature (°F)	120
Thermal Design Flowrate (gpm/loop)	83,200
RCS Loop Average Temperature (°F)	553.0
Nominal Initial RCS Pressure (psia)	2100
Nominal Steam Pressure (psia)	595
Safety Injection Delay Time (sec)	47
HHSI Pump Head Degradation (%)	15
Charging Pump Head Degradation (%)	10
Charging Pump Flow Imbalance (gpm)	25
HHSI Cross-Tie Valve Position	Closed
Auxiliary Feedwater Total Flowrate (gpm)	750

TABLE 3.1-16

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

30% SGTP PROGRAM ANALYSIS WITH HHSI CROSS-TIES CLOSED

Small-break Loss of Coolant Accident

Event	Time (s) Reduced Pressure, Reduced Temperature* 3-Inch
Break occurs	0.0
Reactor trip signal	8.8
Safety injection signal	17.4
Start of safety injection	64.4
Start of auxiliary feedwater delivery	88.8
Loop seal venting	528
Loop seal core uncovery	N/A
Loop seal core recovery	N/A
Boil-off core uncovery	1054
Accumulator injection begins	1648
Peak clad temperature occurs	1748
Top of core recovered	2995
Combined pumped SI flow rate exceeds break flow rate	1856

* 30% steam generator tube plugging case at 3250 MWt core power.

TABLE 3.1-16

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

30% SGTP PROGRAM ANALYSIS WITH HHSI CROSS-TIES CLOSED

Small-break Loss of Coolant Accident

Event	Time (s) Reduced Pressure, Reduced Temperature* 3-Inch
Break occurs	0.0
Reactor trip signal	8.8
Safety injection signal	17.4
Start of safety injection	64.4
Start of auxiliary feedwater delivery	88.8
Loop seal venting	528
Loop seal core uncovery	N/A
Loop seal core recovery	N/A
Boil-off core uncovery	1054
Accumulator injection begins	1648
Peak clad temperature occurs	1748
Top of core recovered	2995
Combined pumped SI flow rate exceeds break flow rate	1856

* 30% steam generator tube plugging case at 3250 MWt core power.

TABLE 3.1-17

SMALL-BREAK LOSS OF COOLANT ACCIDENT CALCULATIONS

30% SGTP PROGRAM ANALYSIS WITH HHSI CROSS-TIES CLOSED

RESULTS

	Reduced Pressure, Reduced Temperature* 3-Inch
NOTRUMP Peak Clad Temperature (°F)	1443°F
Peak Clad Temperature Location (ft)	11.5
Peak Clad Temperature Time (sec)	1748
Local Zr/H_2O Reaction Maximum (%)	< 1.0
Local Zr/H_2O Reaction Location (ft)	11.5
Total Zr/H_2O Reaction (%)	< 1.0
Rod Burst	None
Burst and Blockage Penalty	None
Total Peak Clad Temperature (°F)	1443°F

* 30% steam generator tube plugging case at 3250 MWt core power.

TABLE 3.1-17

SMALL-BREAK LOSS OF COOLANT ACCIDENT CALCULATIONS

30% SGTP PROGRAM ANALYSIS WITH HHSI CROSS-TIES CLOSED

RESULTS

	Reduced Pressure, Reduced Temperature* 3-Inch
NOTRUMP Peak Clad Temperature (°F)	1443°F
Peak Clad Temperature Location (ft)	11.5
Peak Clad Temperature Time (sec)	1748
Local Zr/H_2O Reaction Maximum (%)	< 1.0
Local Zr/H_2O Reaction Location (ft)	11.5
Total Zr/H_2O Reaction (%)	< 1.0
Rod Burst	None
Burst and Blockage Penalty	None
Total Peak Clad Temperature (°F)	1443°F

* 30% steam generator tube plugging case at 3250 MWt core power.

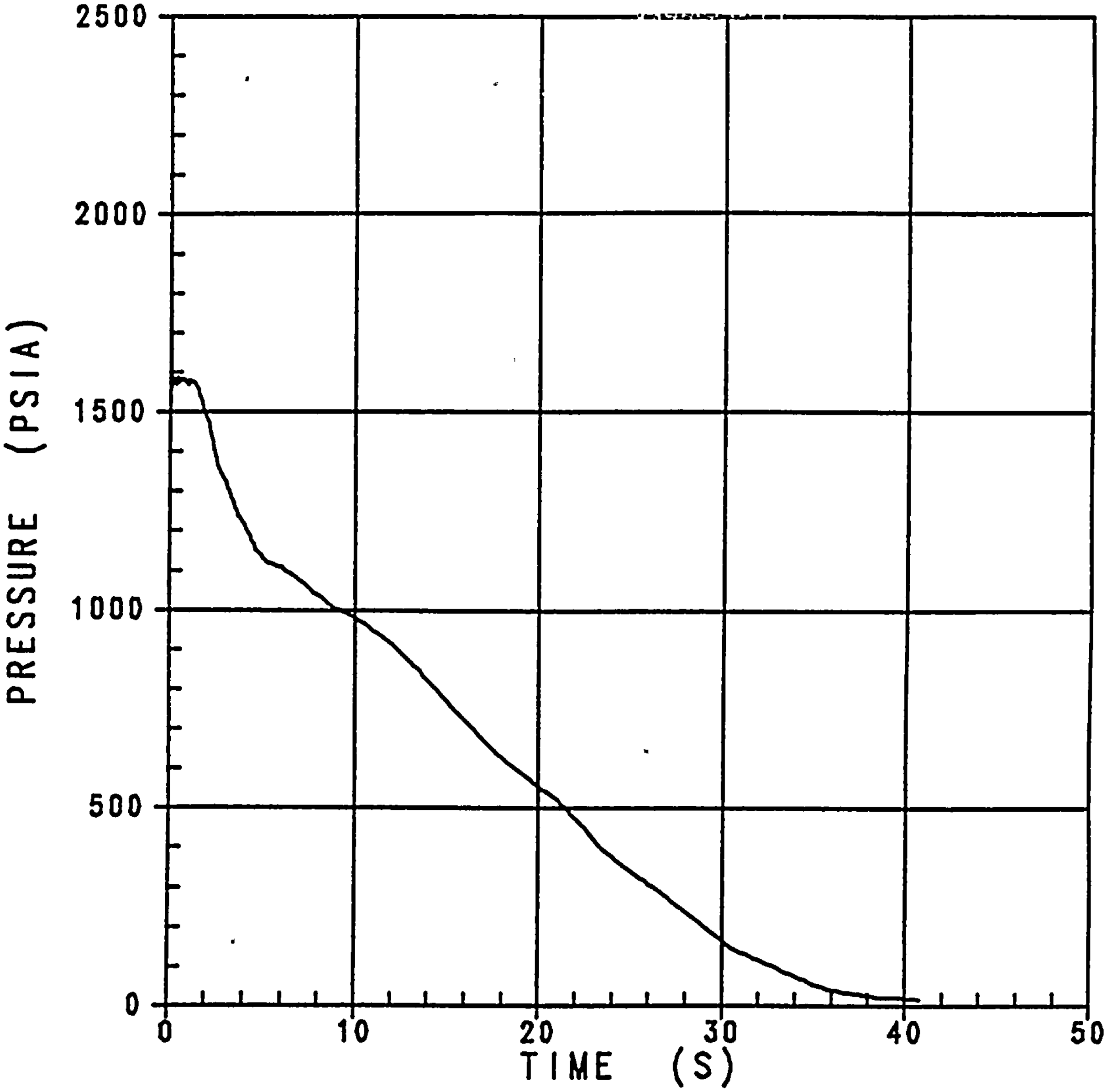

Figure 3.1-1a Reactor Coolant System Pressure
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

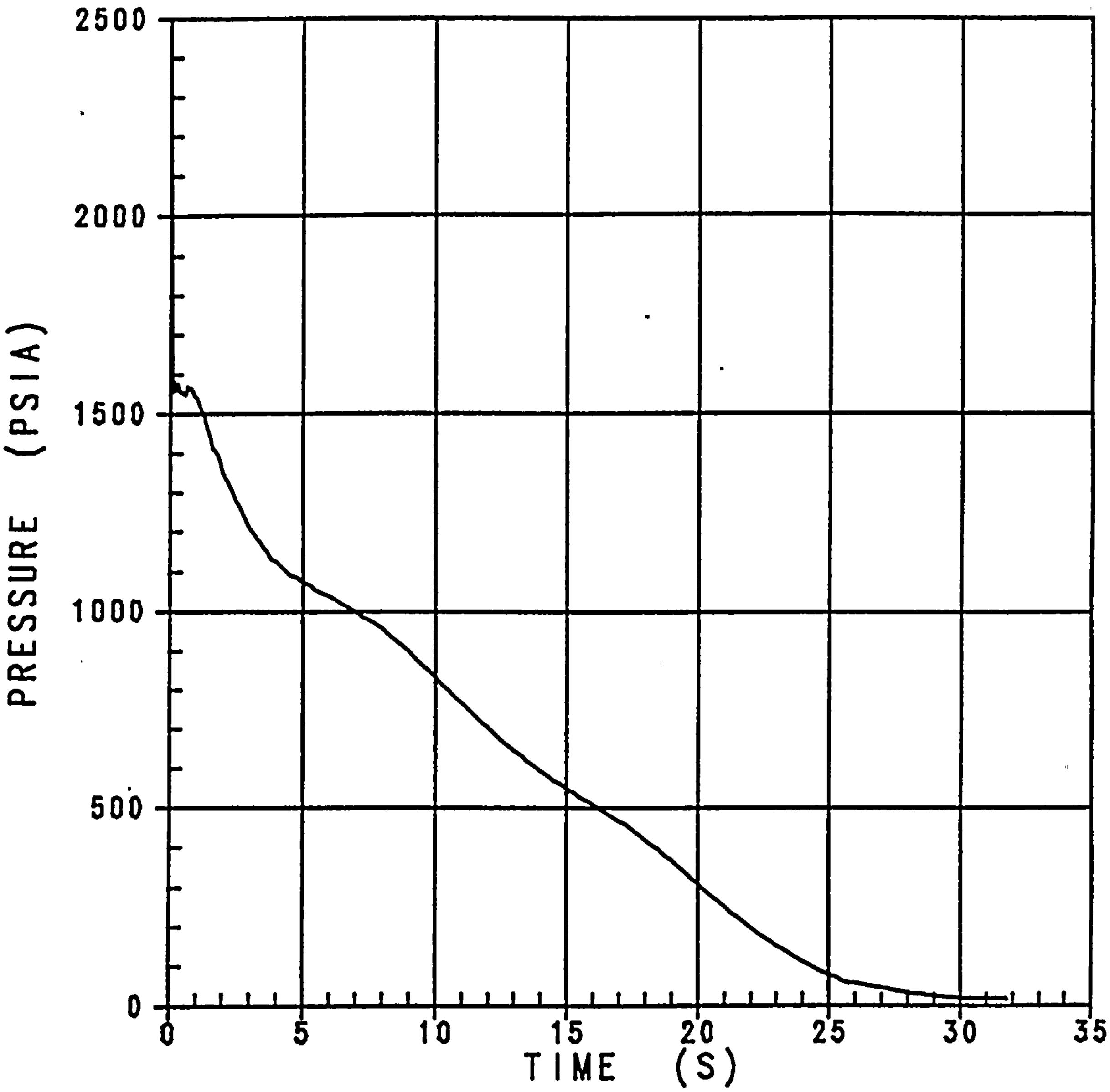

Figure 3.1-1b · Reactor Coolant System Pressure
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

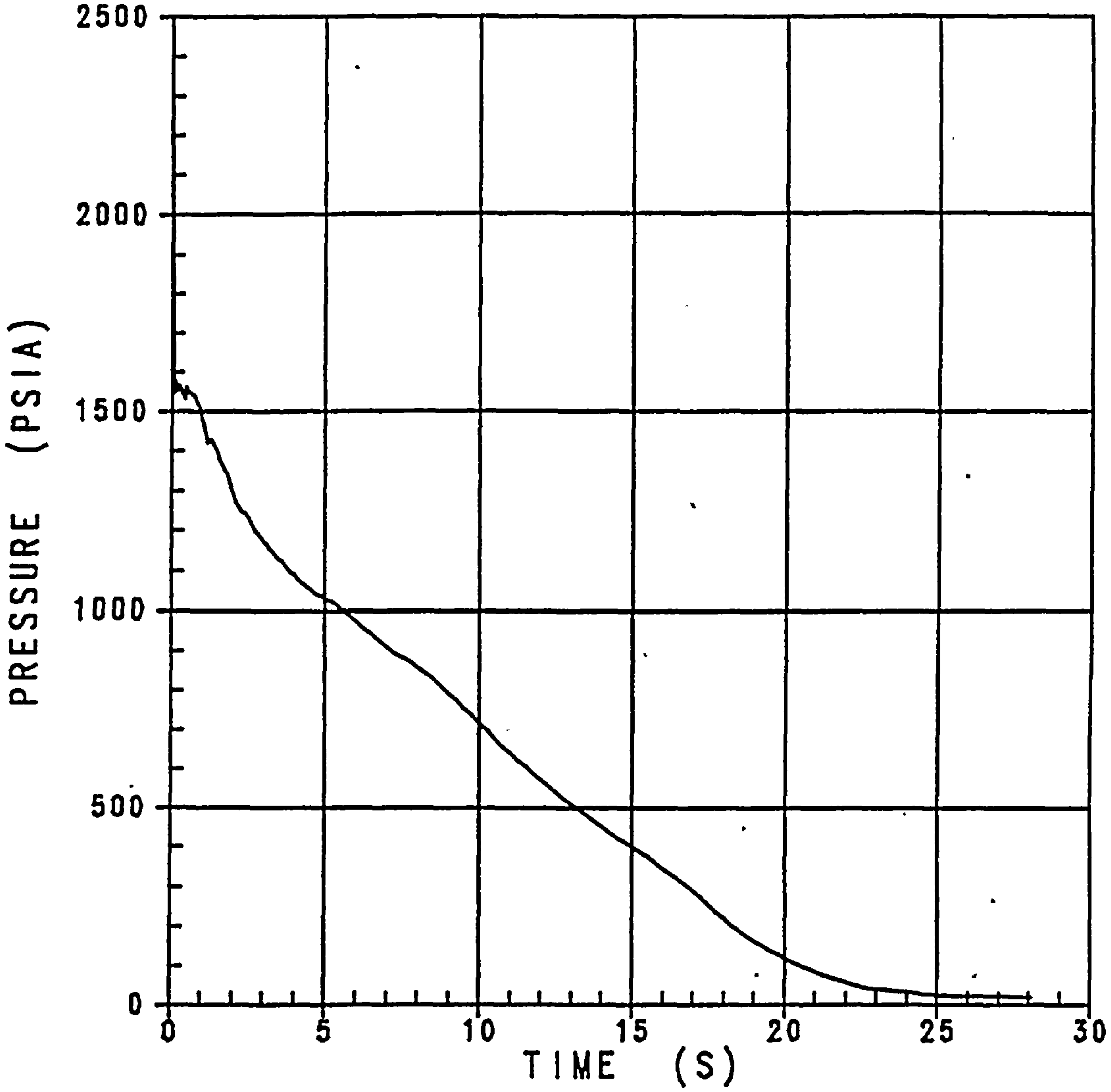

Figure 3.1-1c Reactor Coolant System Pressure
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

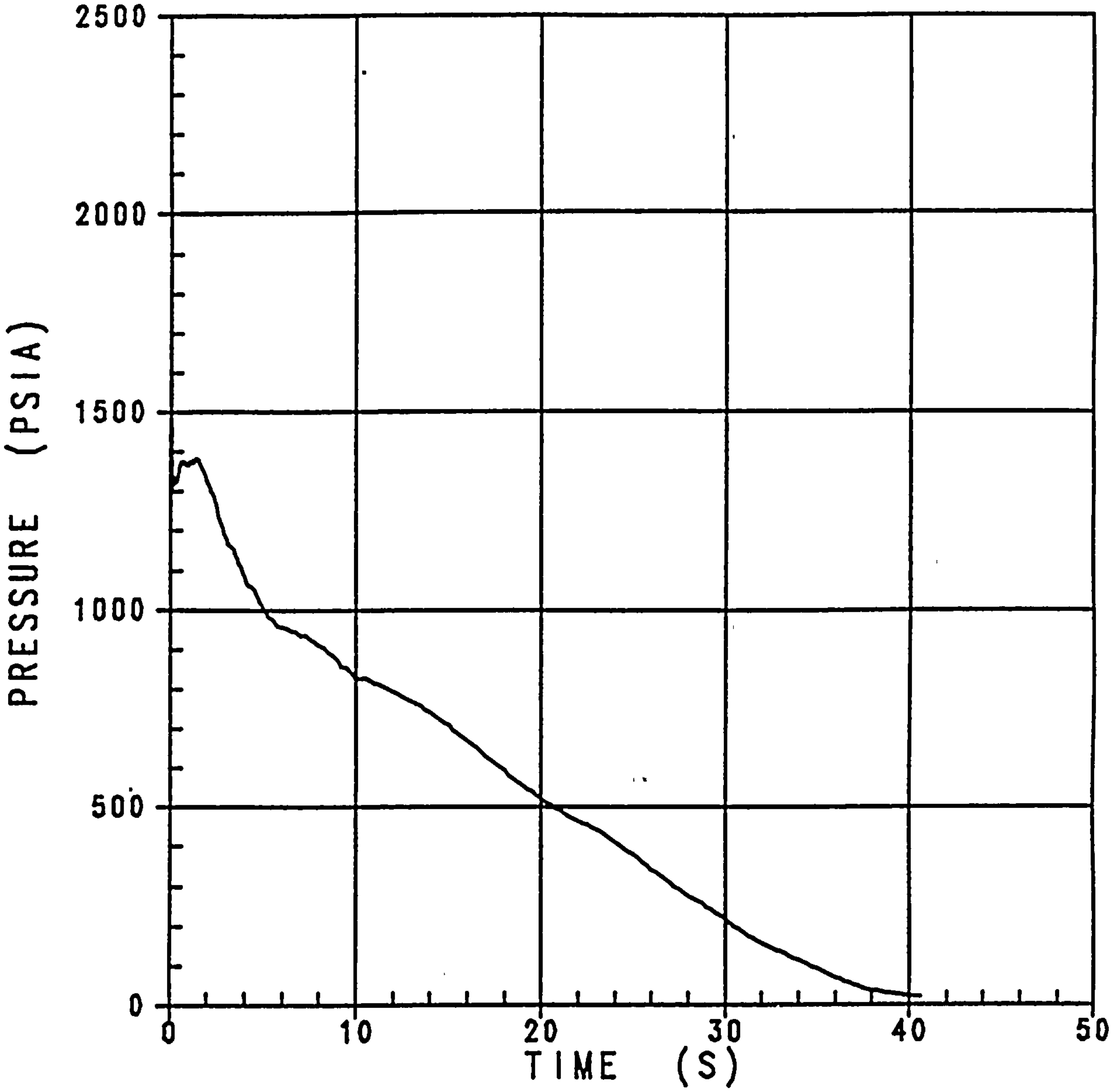

Figure 3.1-1d Reactor Coolant System Pressure
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

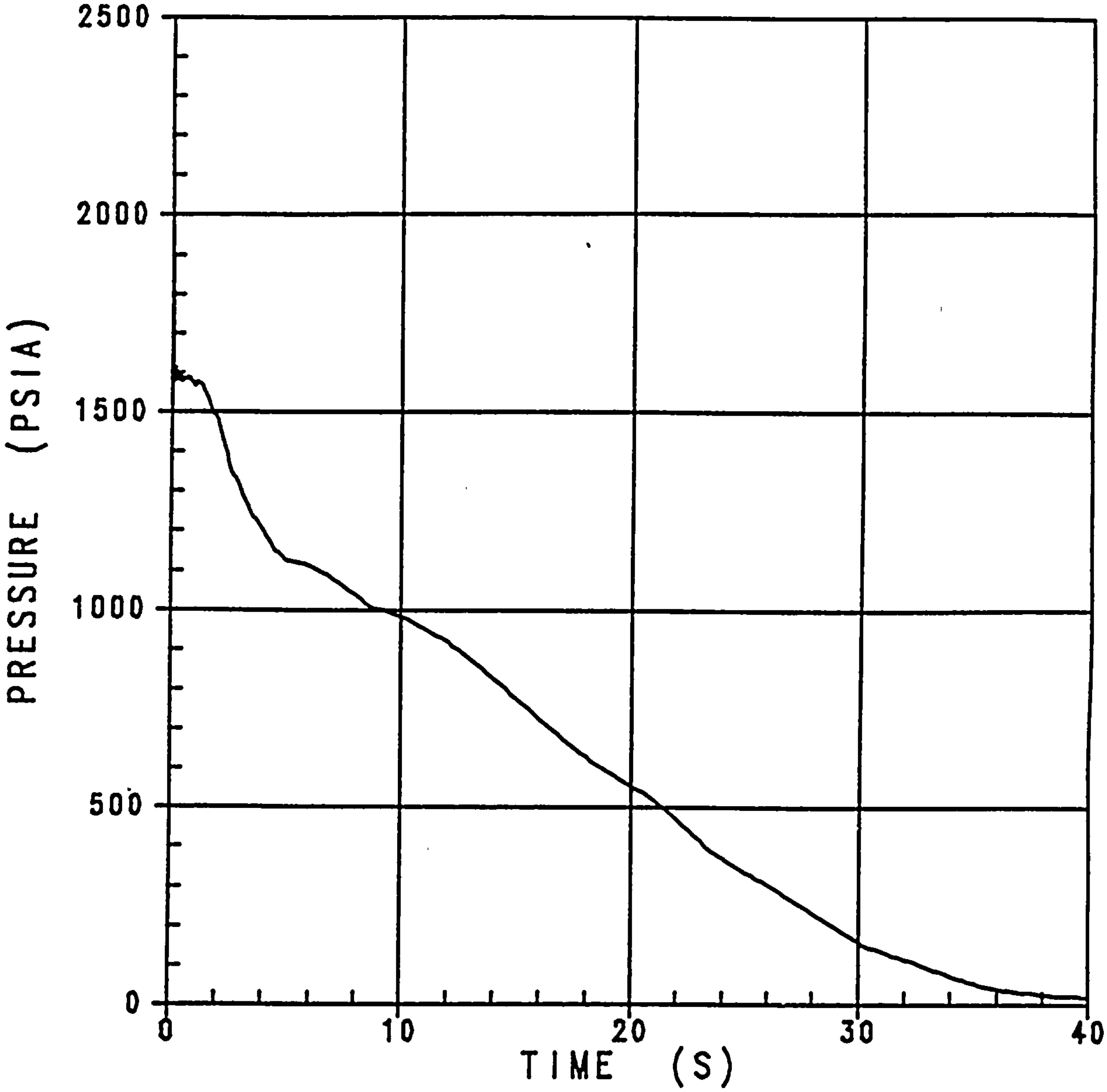

Figure 3.1-1e Reactor Coolant System Pressure
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

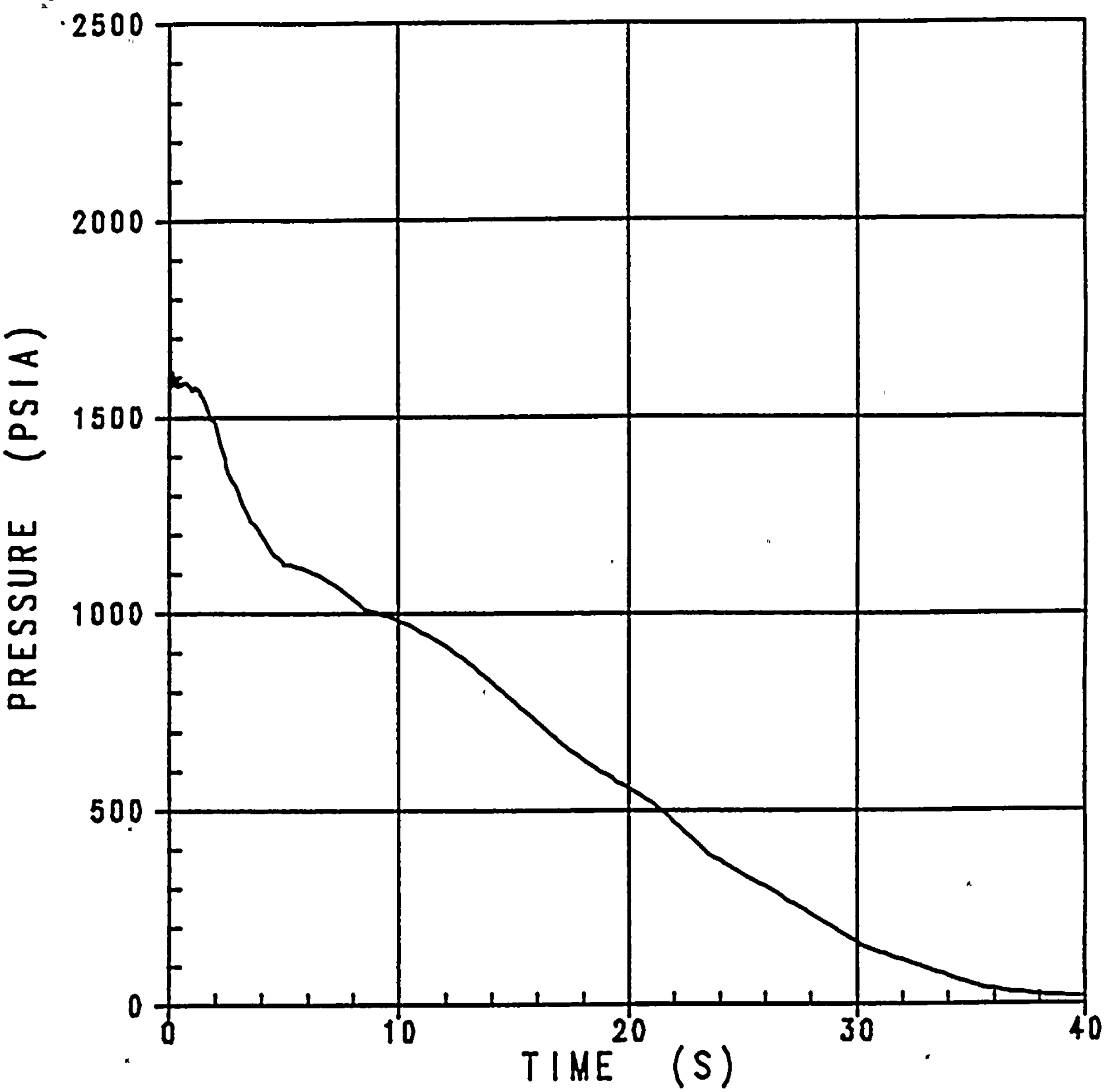

Figure 3.1-1f Reactor Coolant System Pressure
Case A, CD=0.4, Thot=609.1°F, P=2100 psia, max SI
Donald C. Cook Unit 1

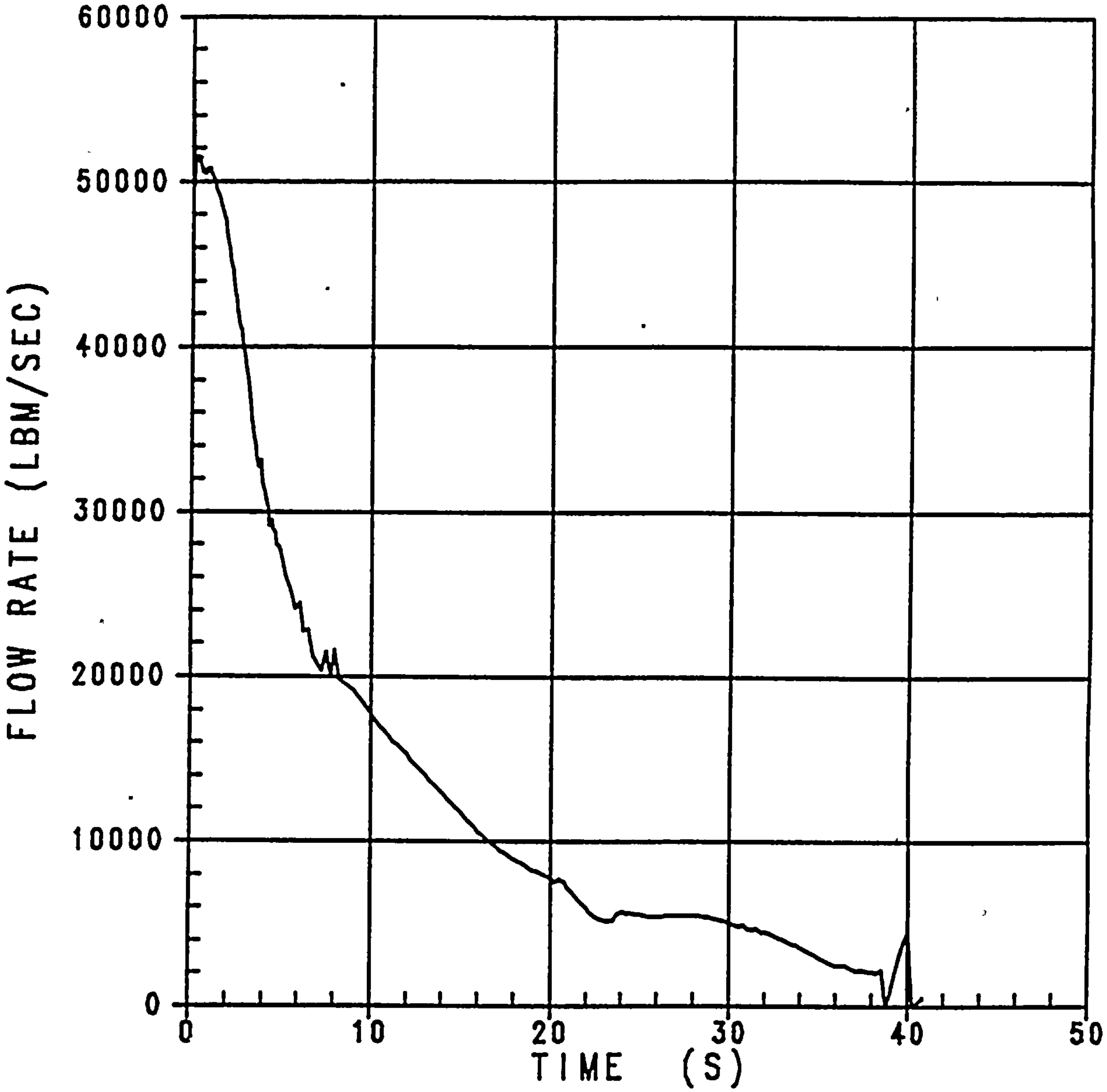

Figure 3.1-2a Break Flow During Blowdown
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

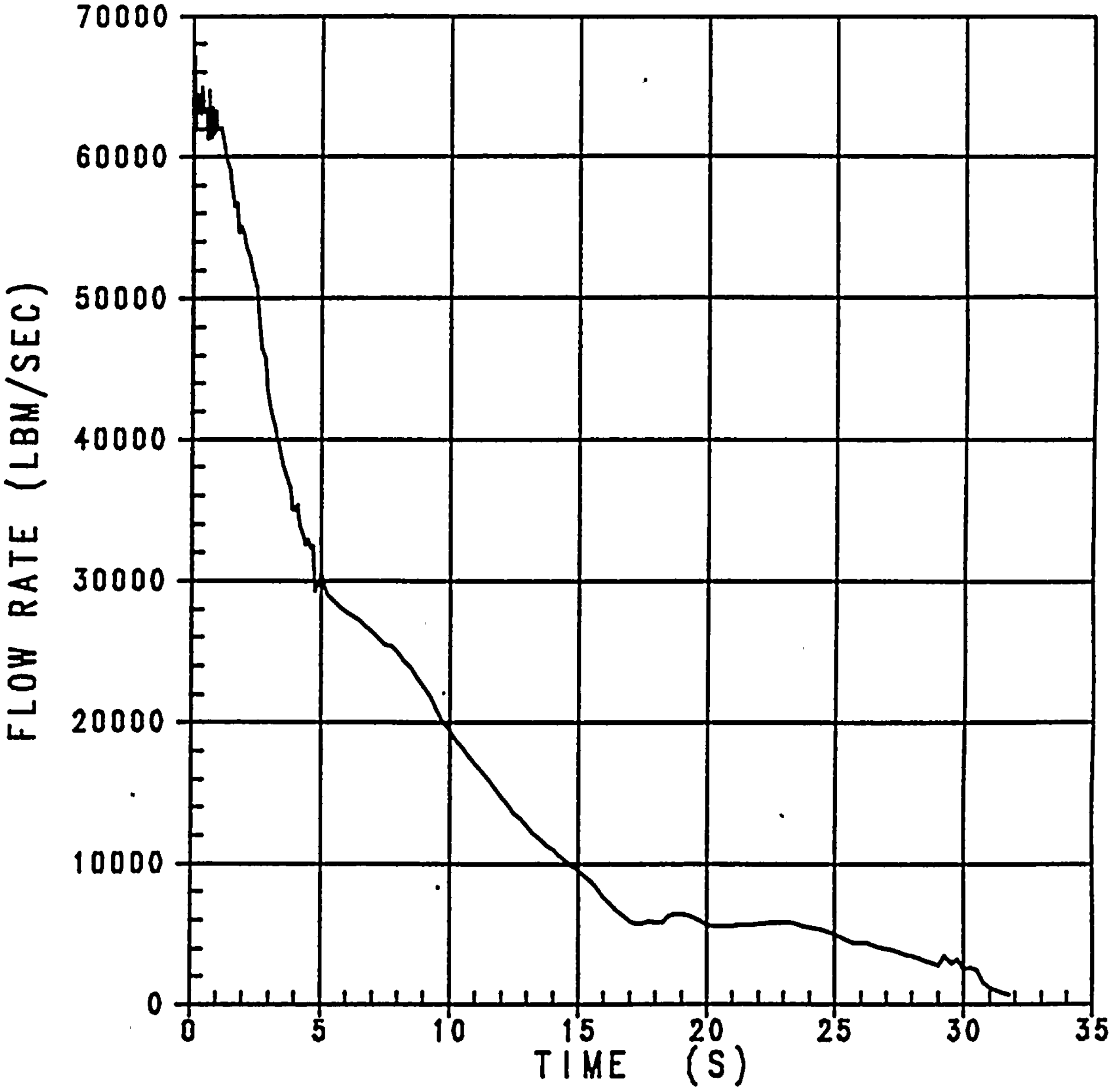

Figure 3.1-2b Break Flow During Blowdown
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

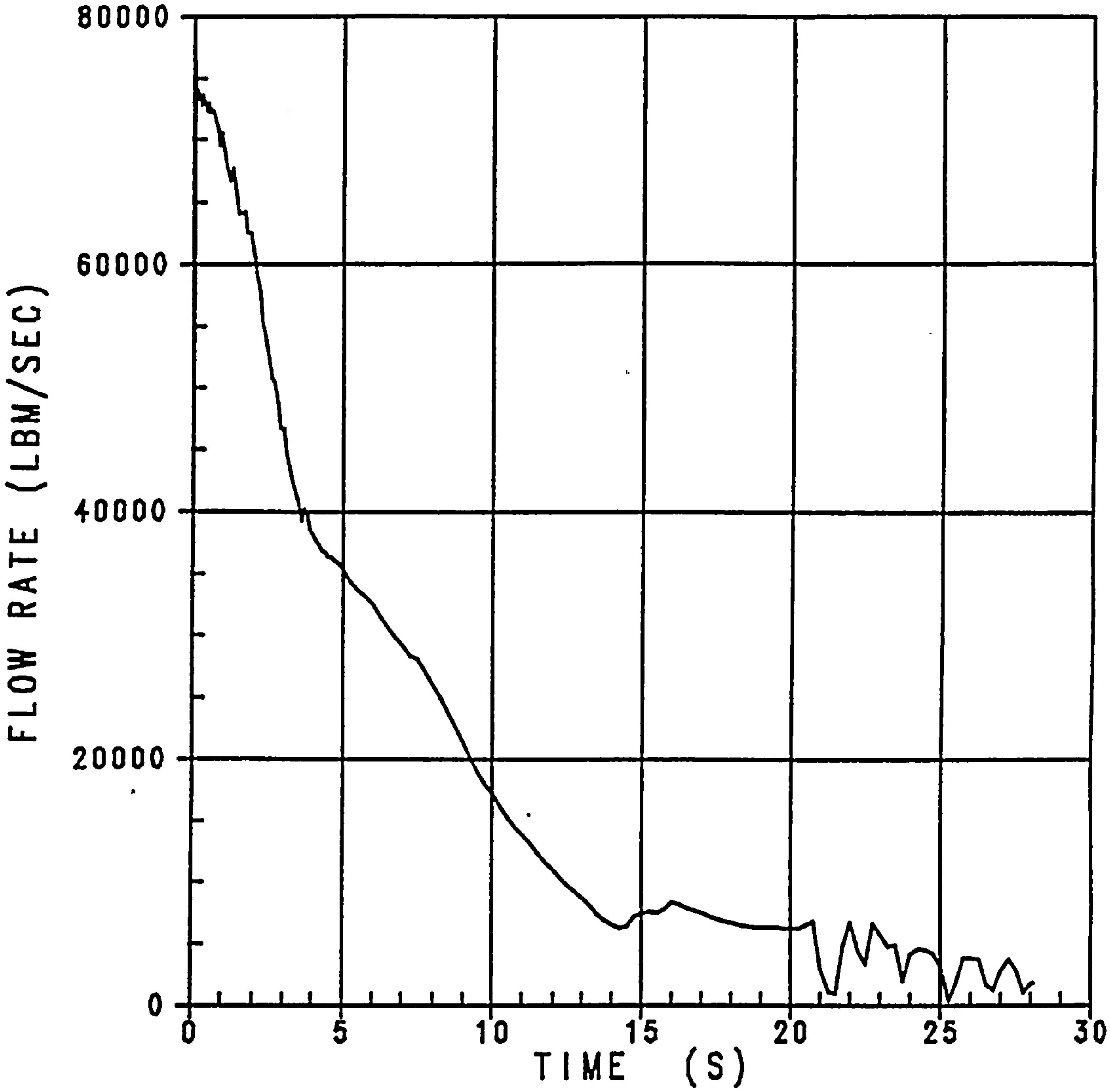

Figure 3.1-2c Break Flow During Blowdown
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

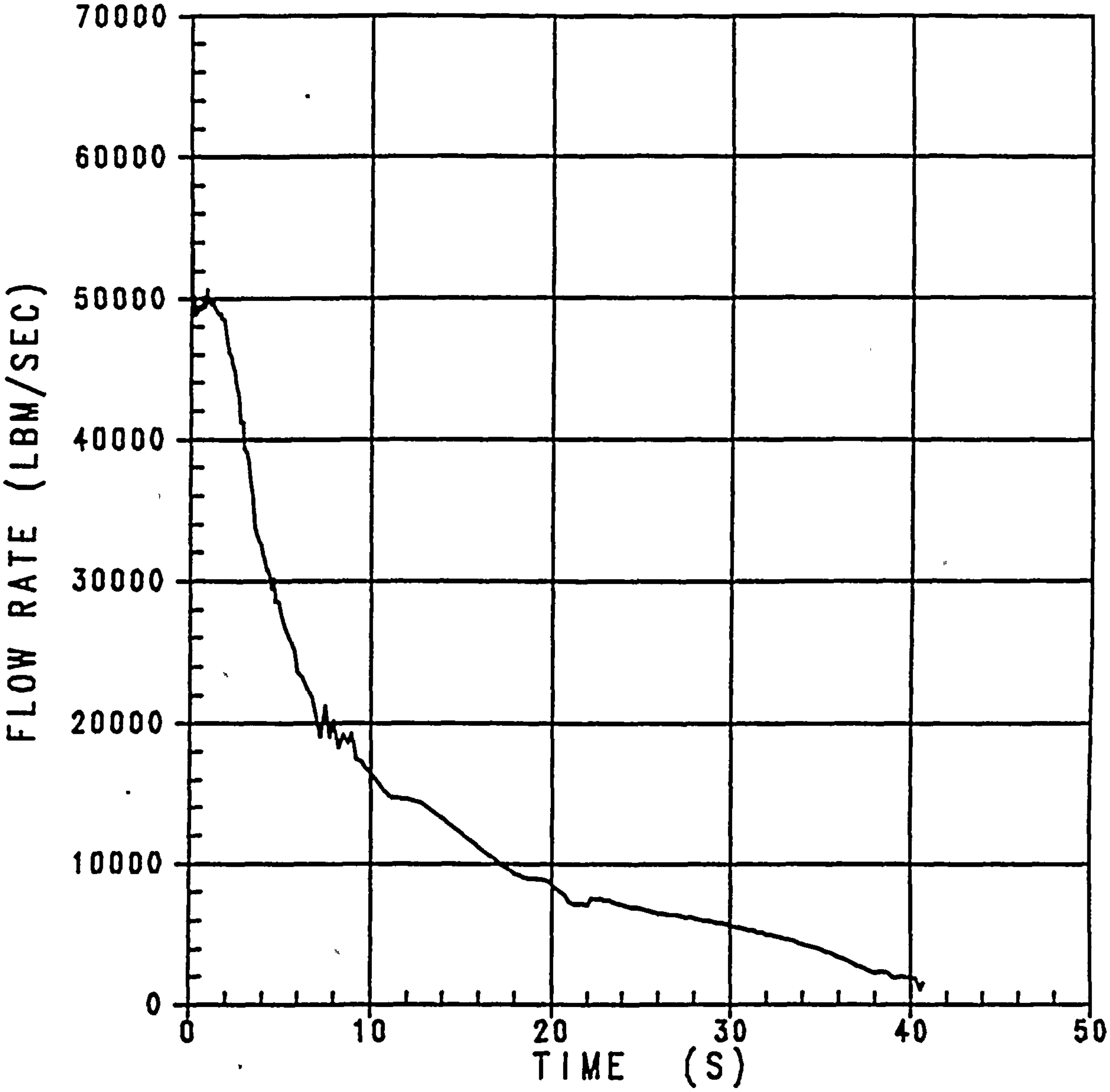

Figure 3.1-2d Break Flow During Blowdown
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

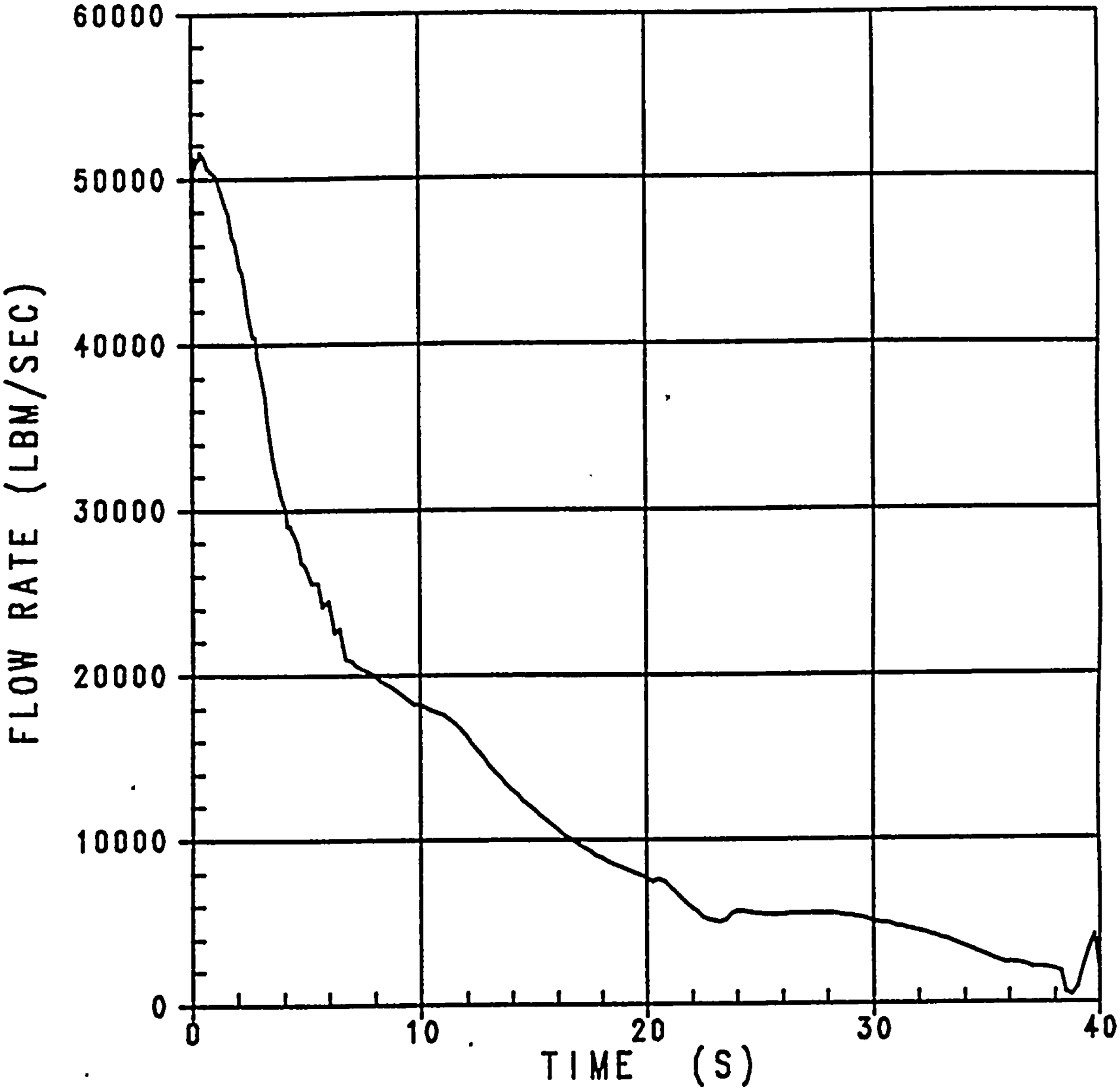

Figure 3.1-2e Break Flow During Blowdown
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

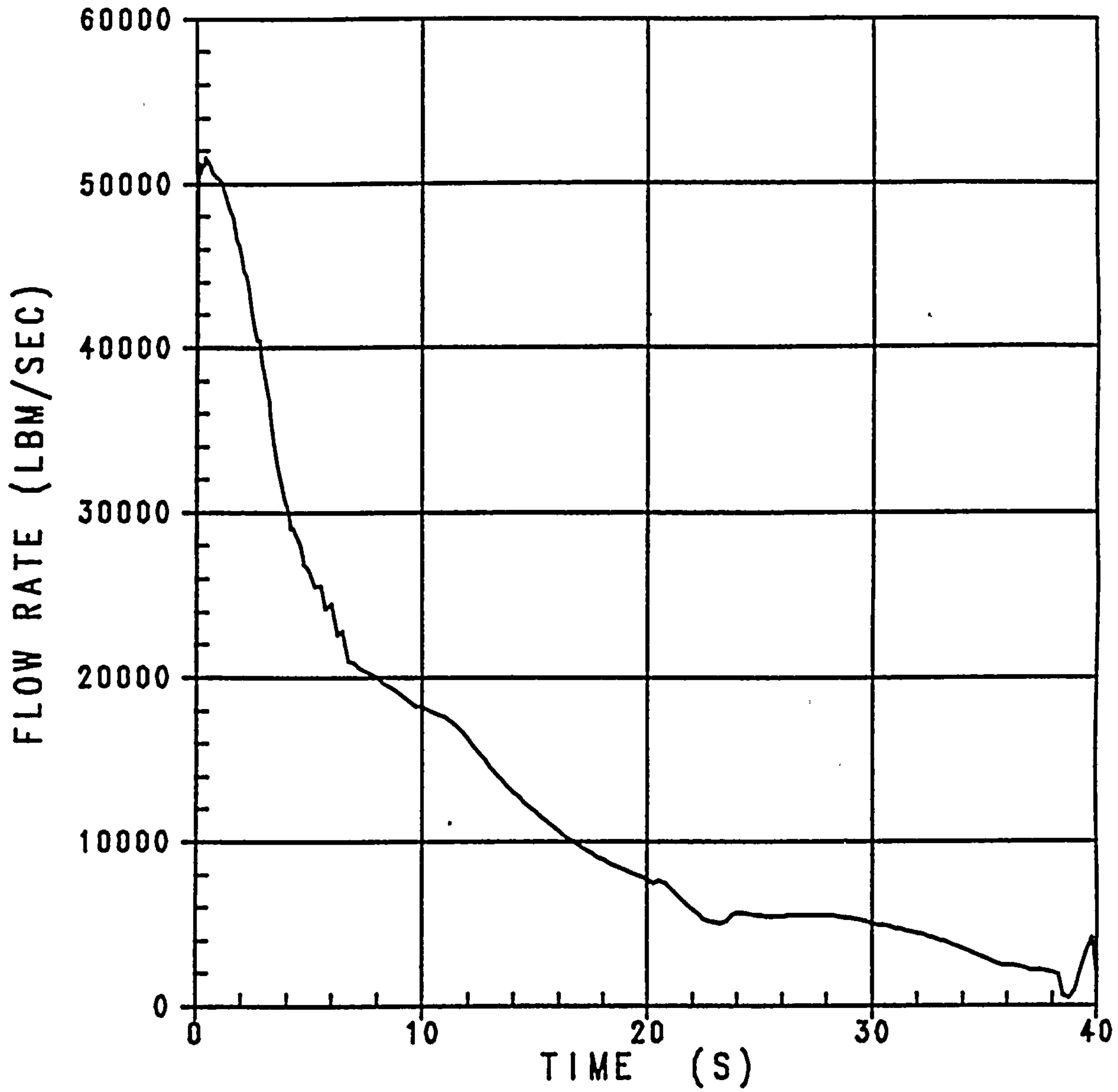

Figure 3.1-2f Break Flow During Blowdown
Case F, CD=0.4, Thot=609.1°F, P=2250 psia, max SI
Donald C. Cook Unit 1

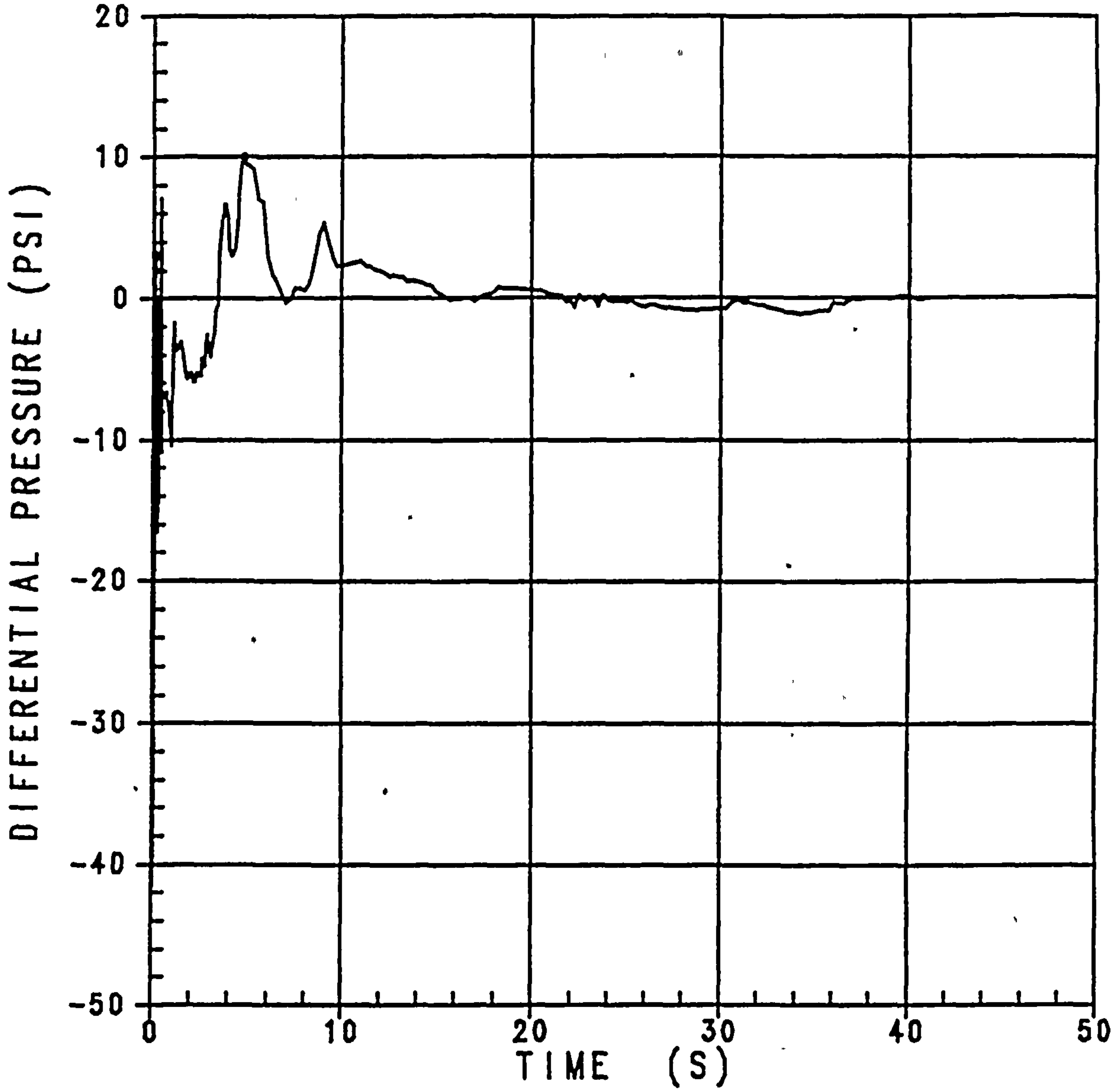

Figure 3.1-3a Core Pressure Drop
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

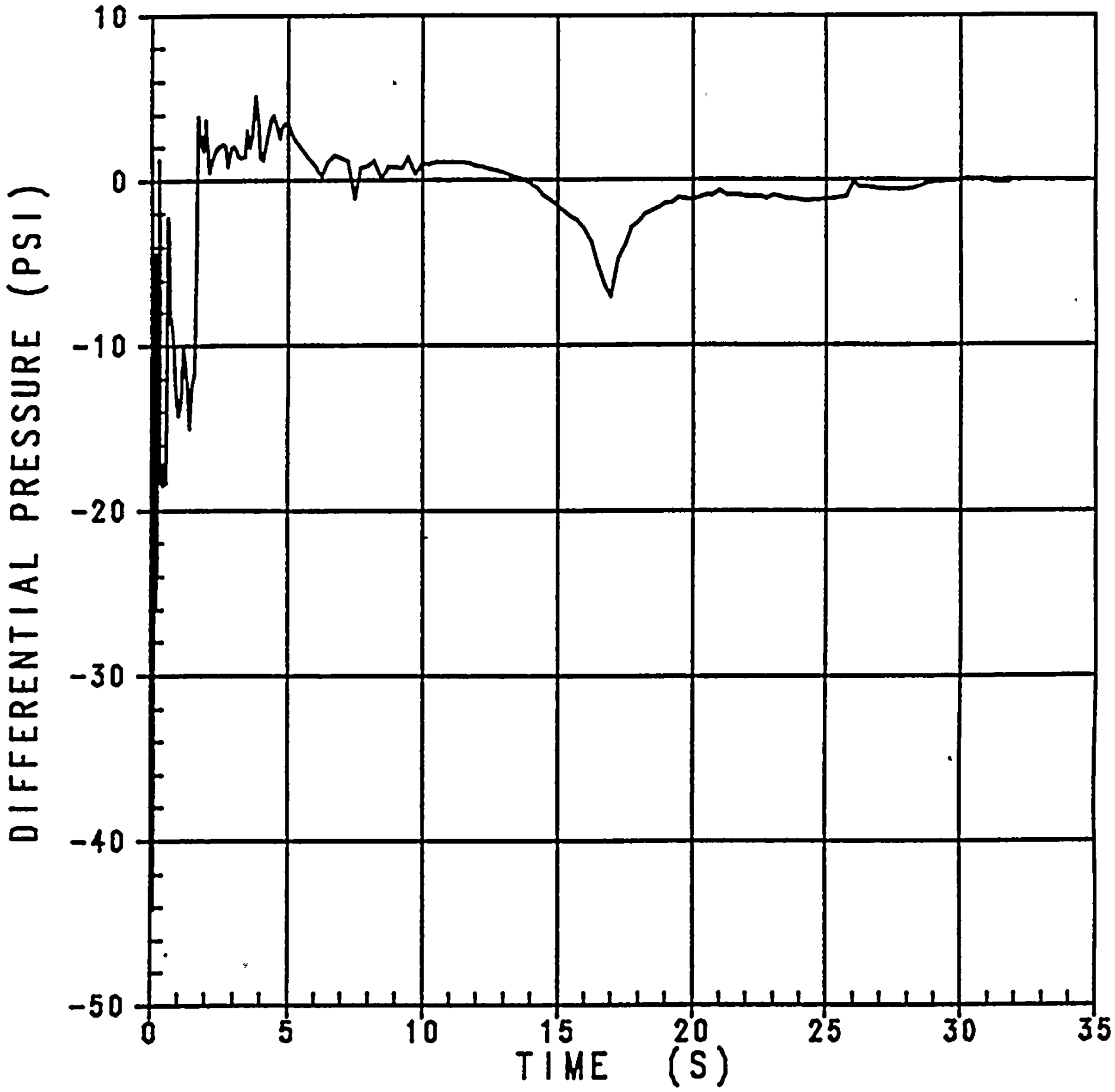

Figure 3.1-3b Core Pressure Drop
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

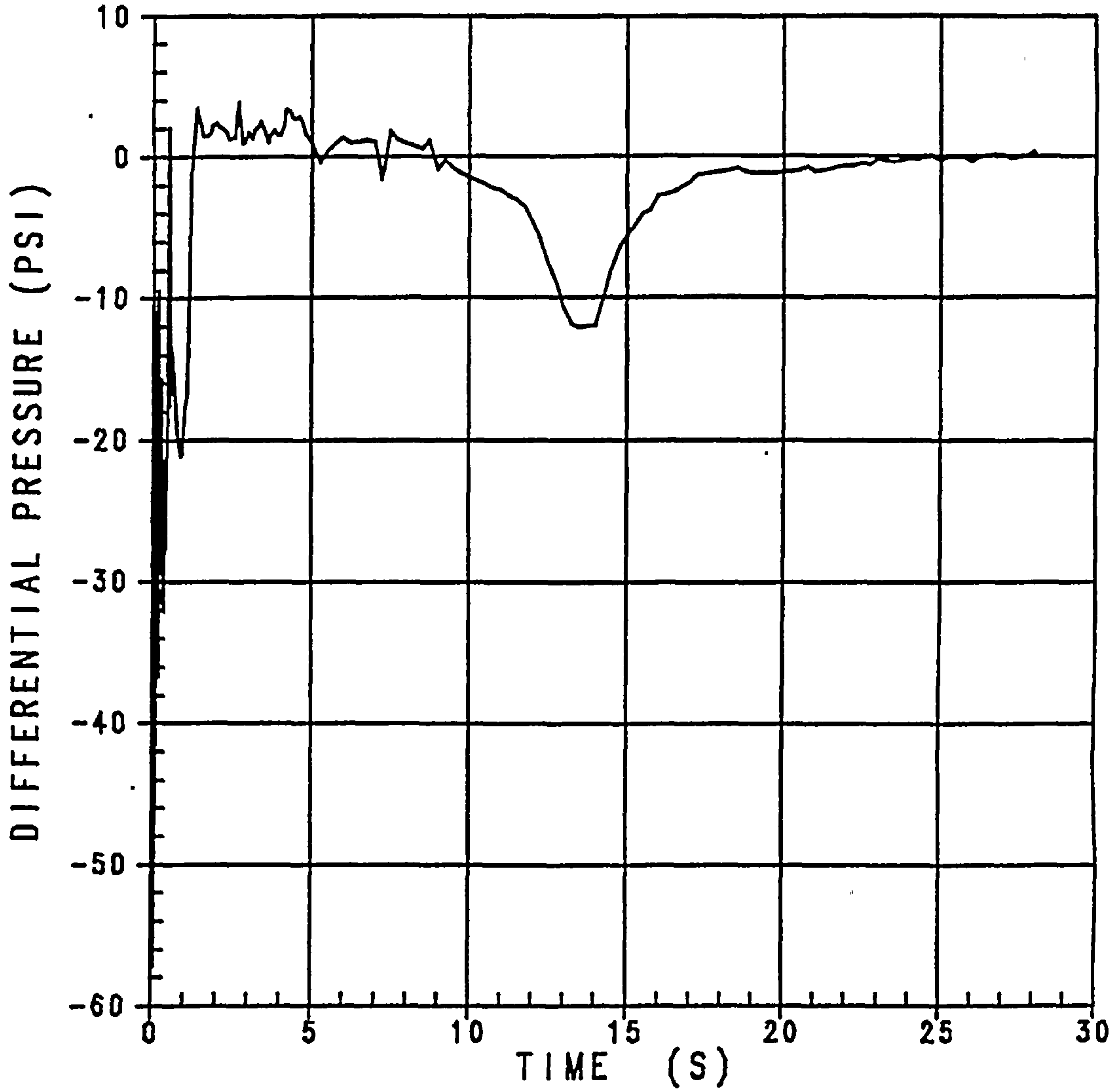

Figure 3.1-3c Core Pressure Drop
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

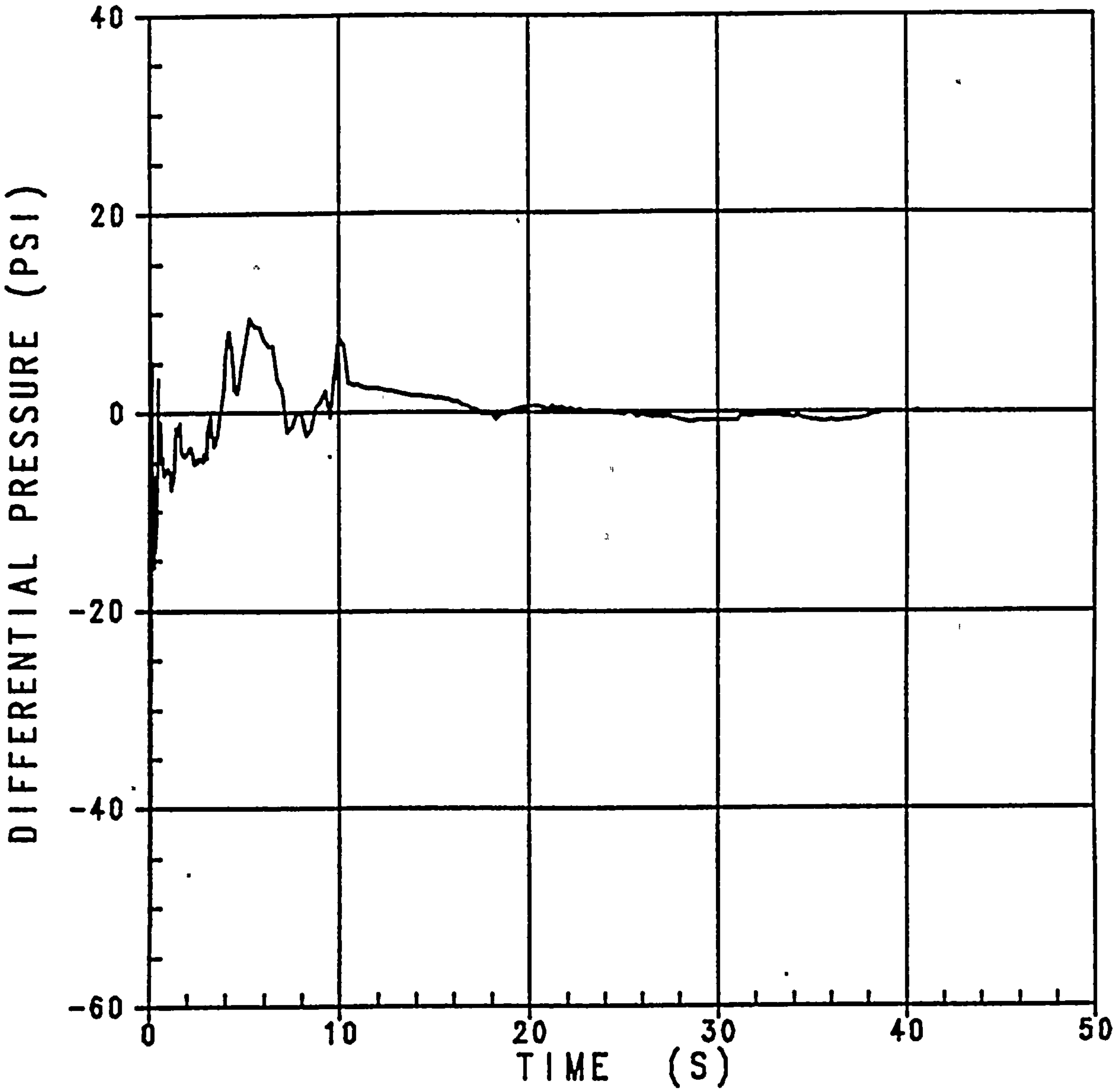

Figure 3.1-3d Core Pressure Drop
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

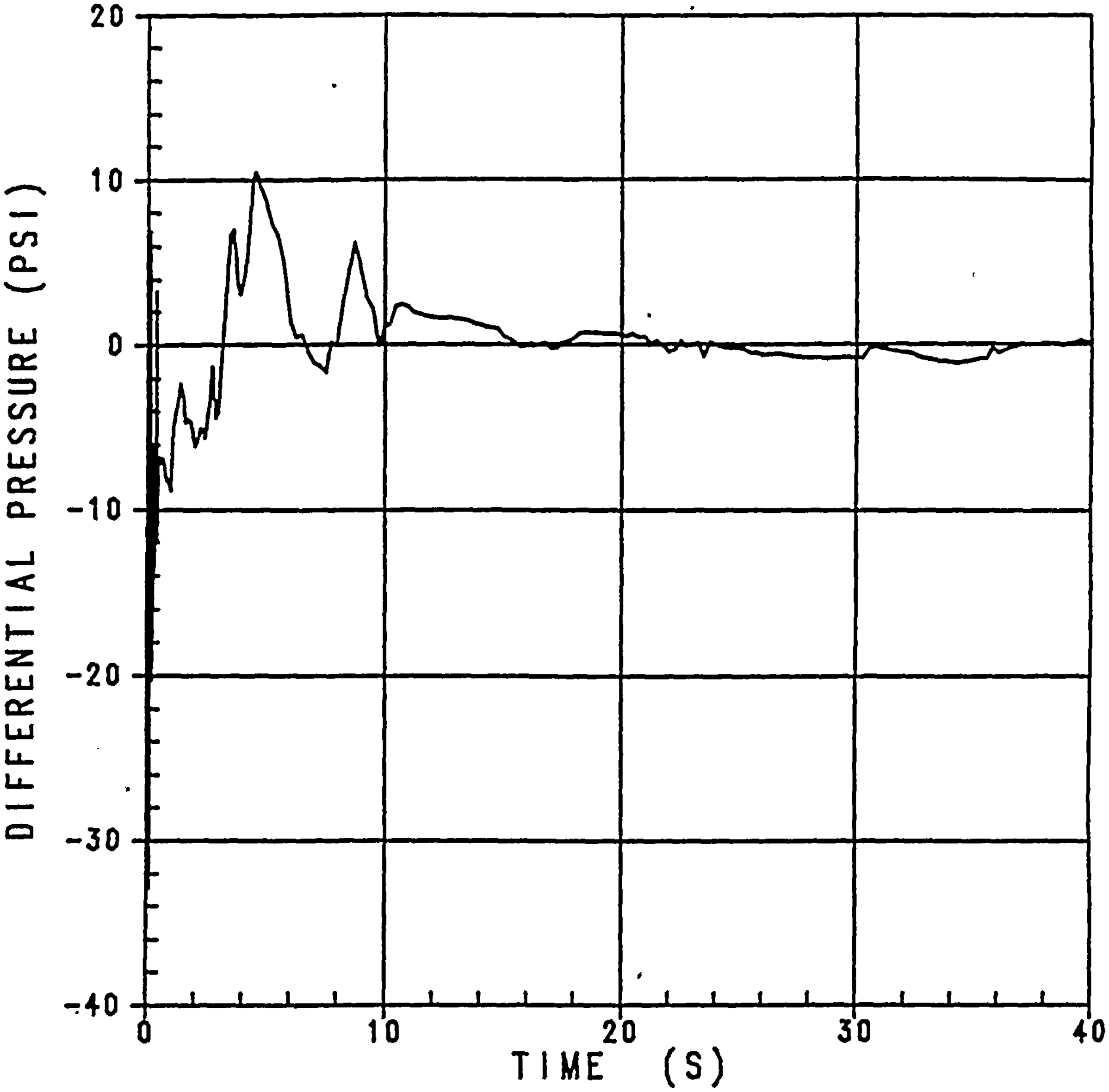

Figure 3.1-3e Core Pressure Drop
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

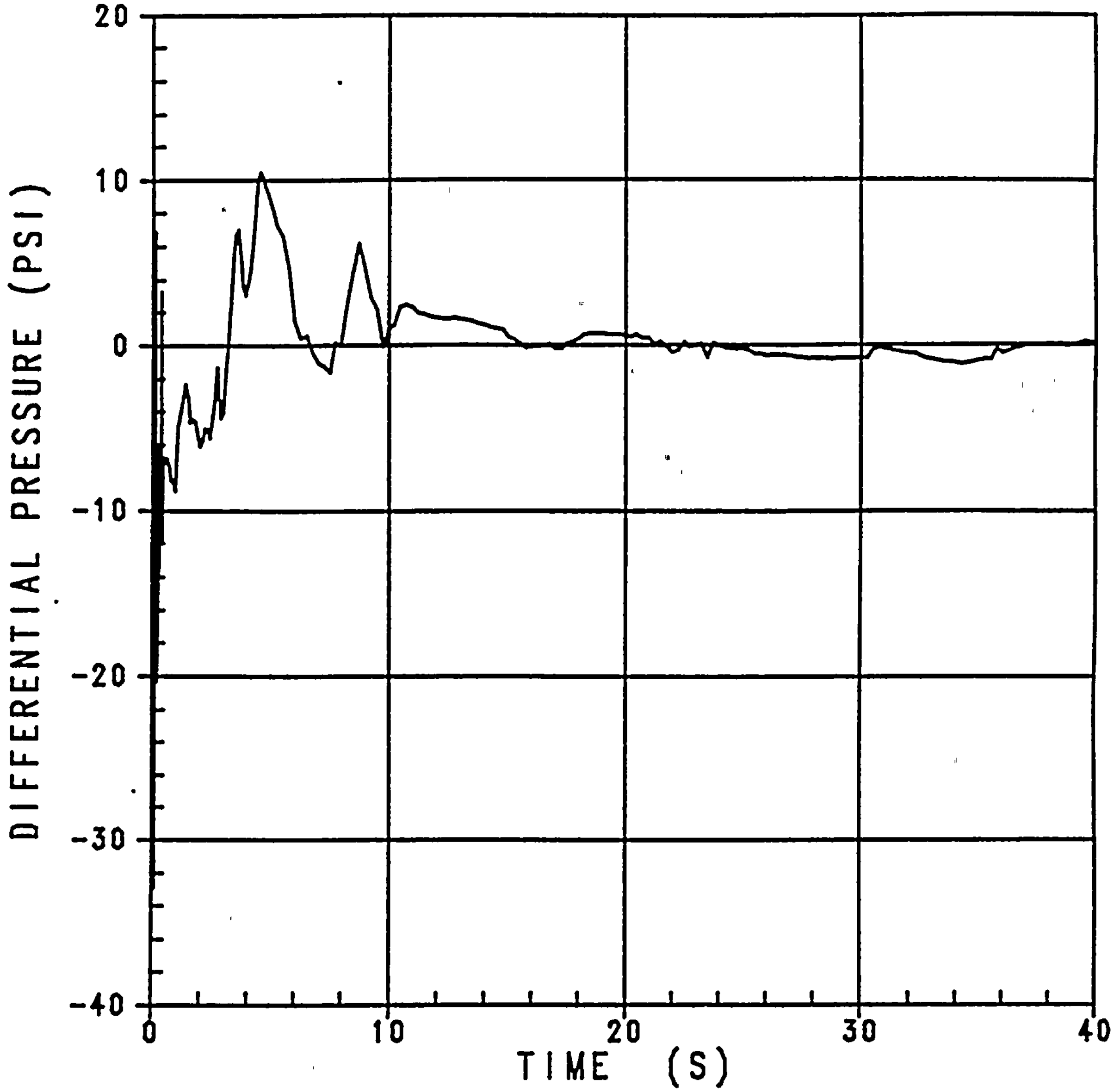

Figure 3.1-3f Core Pressure Drop
Case F, CD=0.4, Thot=609.1°F, P=2100 psia, max SI
Donald C. Cook Unit 1

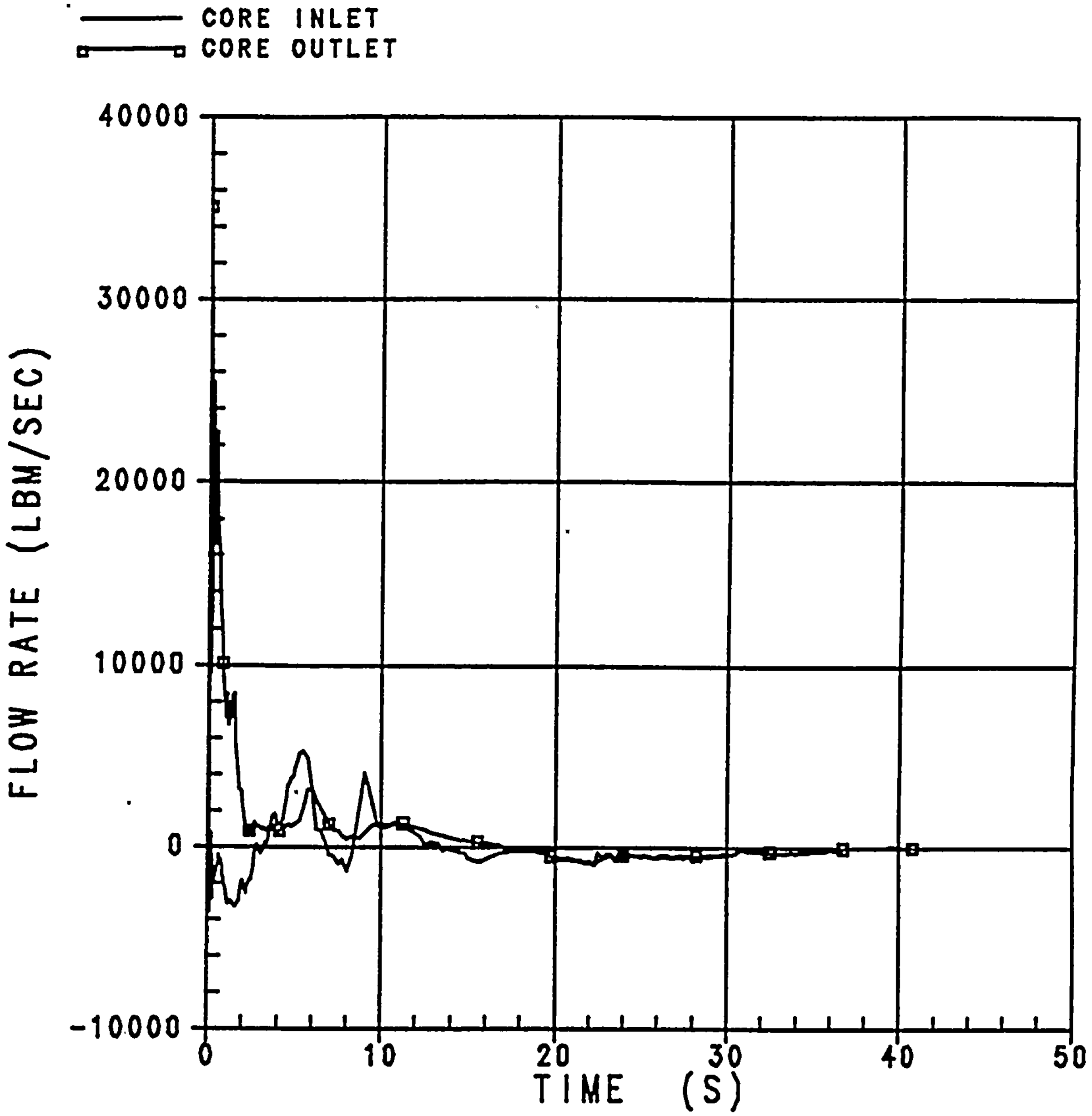

Figure 3.1-4a Core Flowrate
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

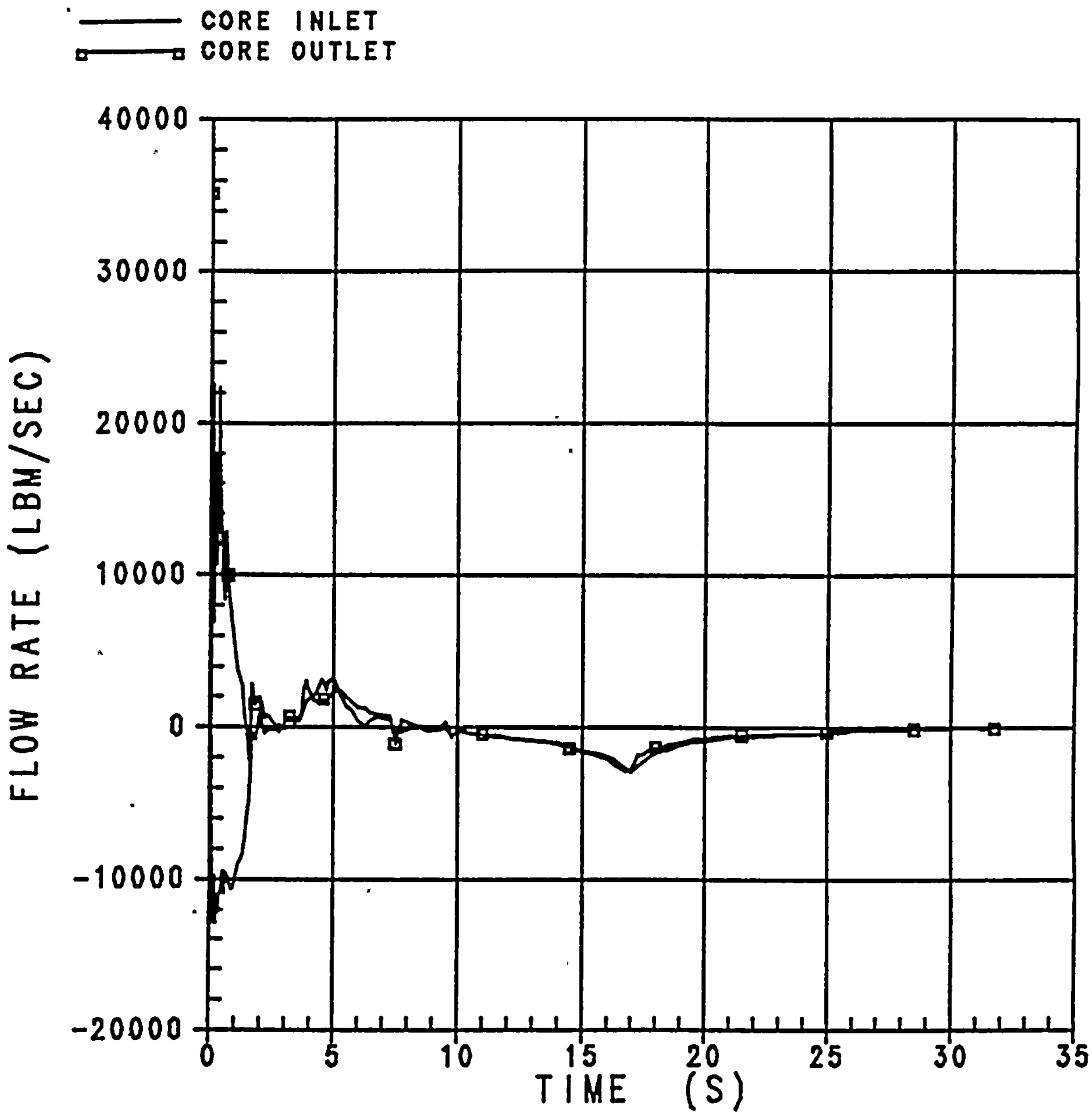

Figure 3.1-4b Core Flowrate
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

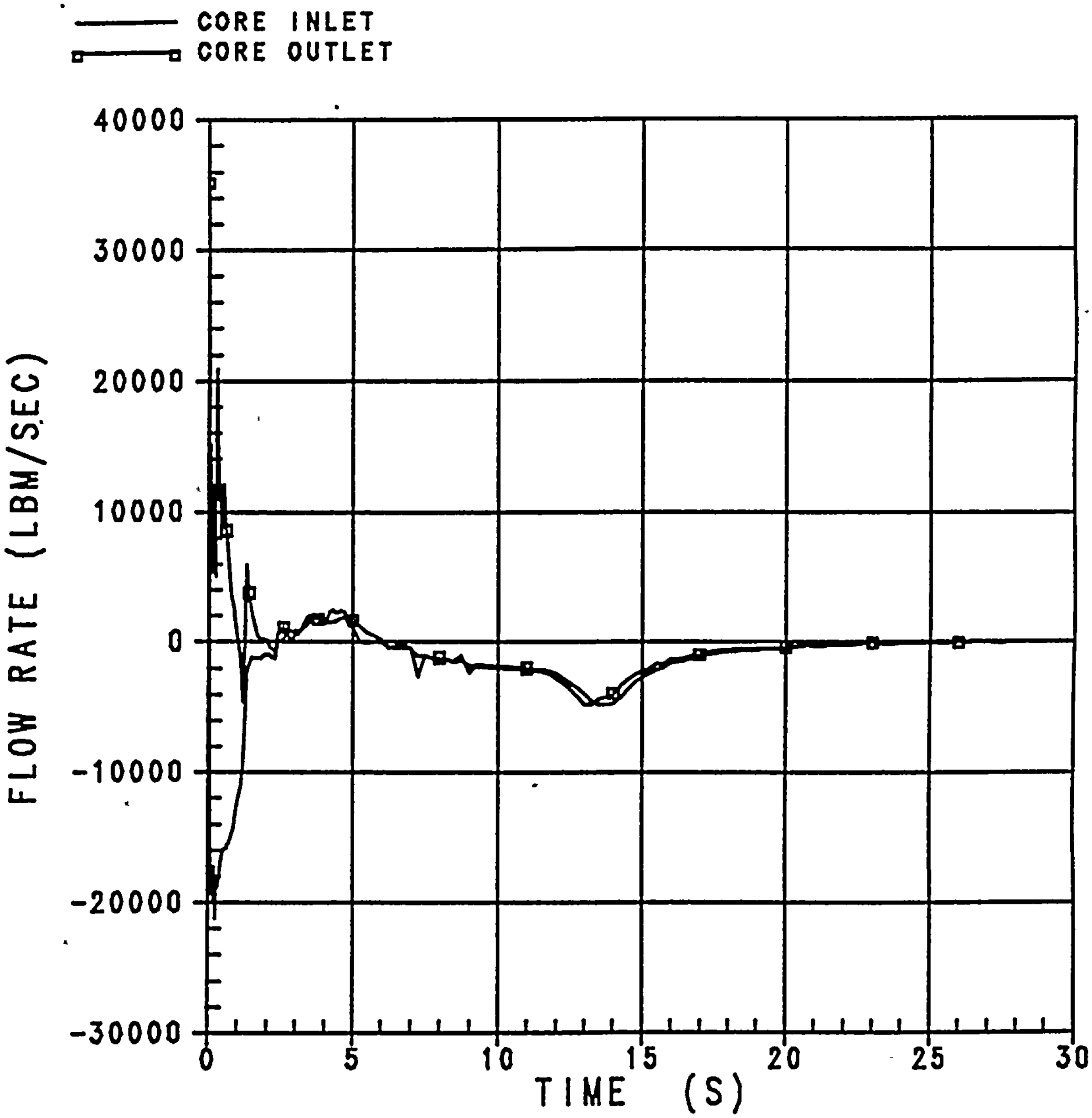

Figure 3.1-4c Core Flowrate
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

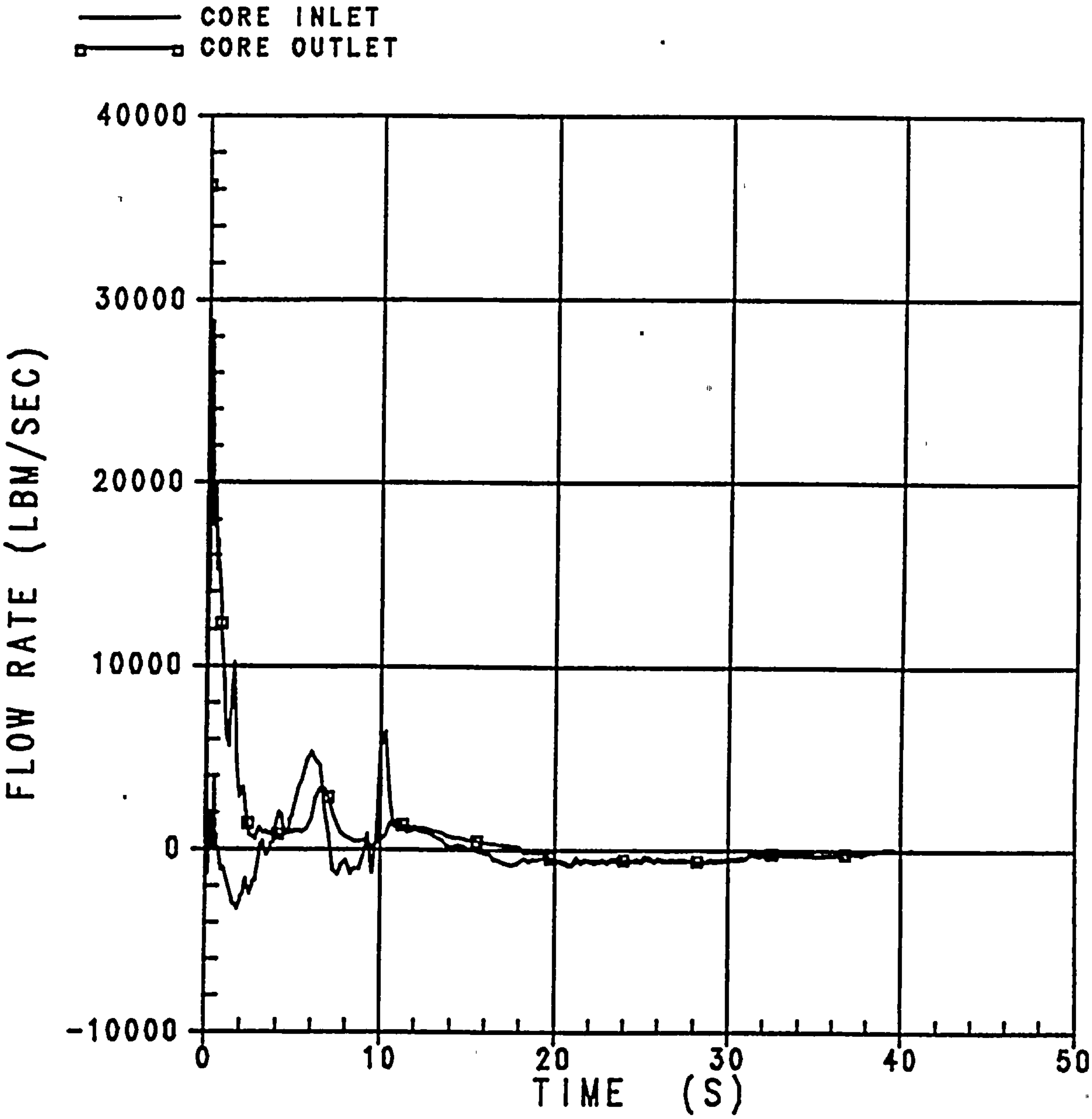

Figure 3.1-4d Core Flowrate
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

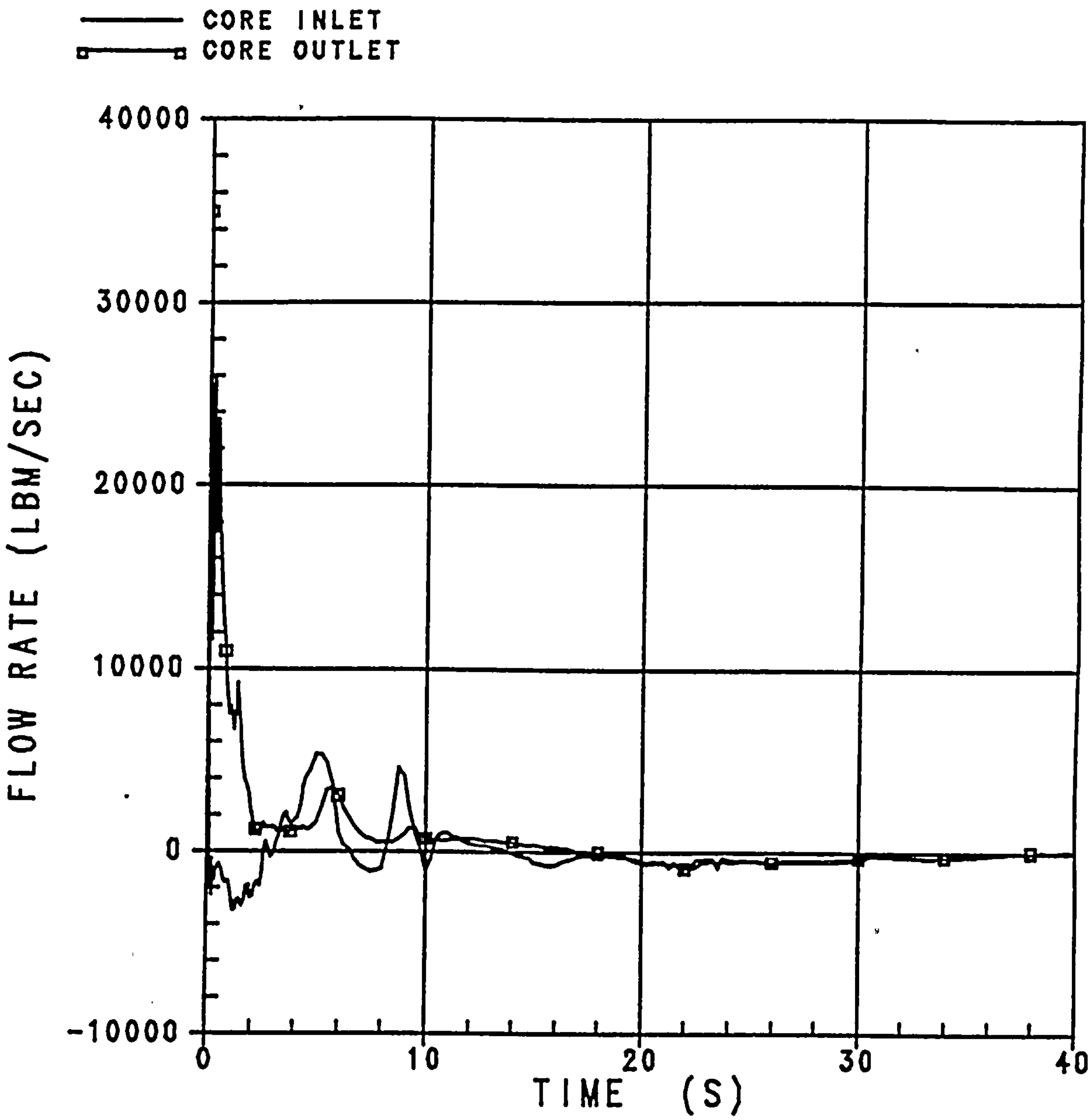

Figure 3.1-4e Core Flowrate
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

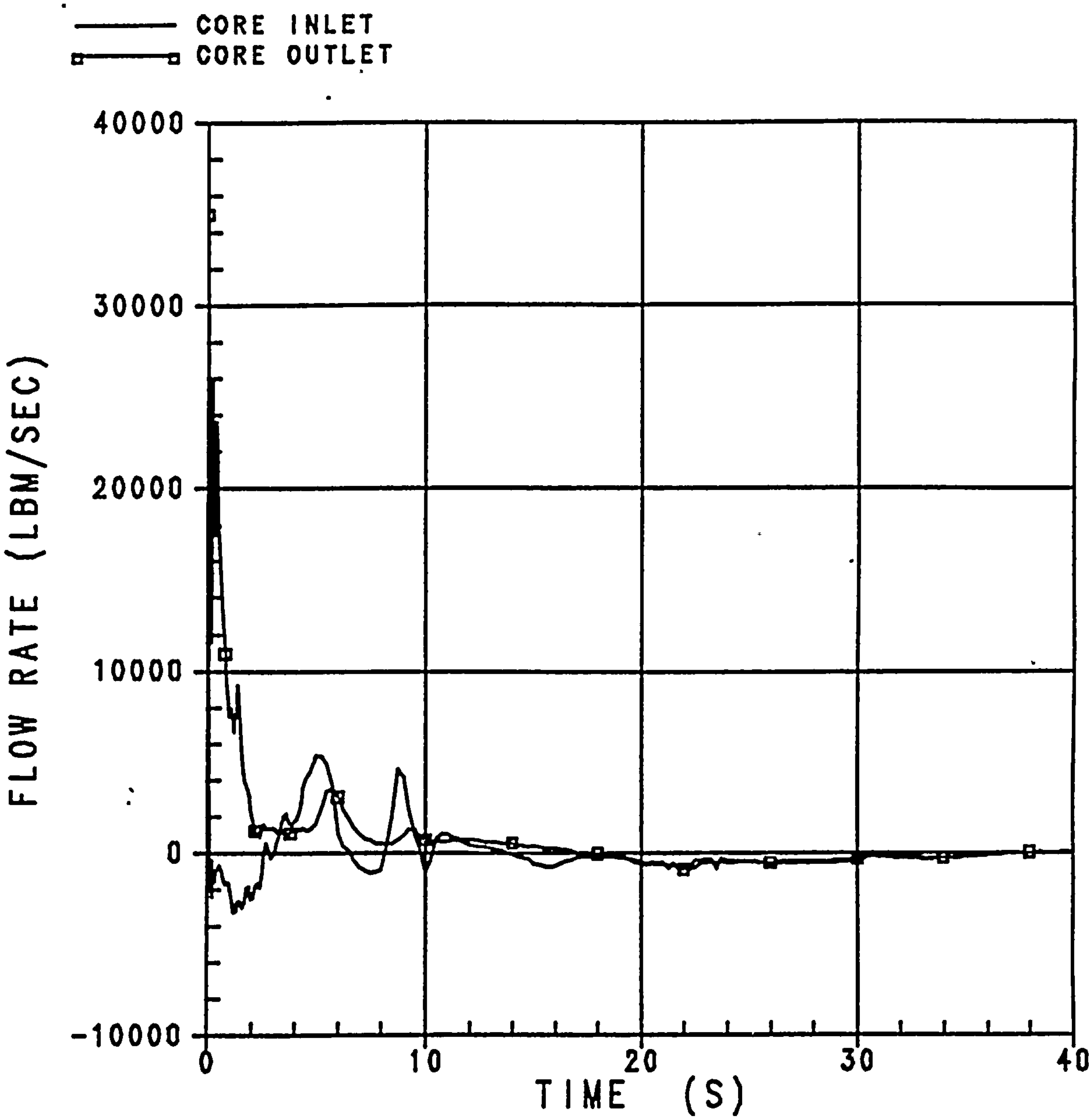

Figure 3.1-4f Core Flowrate
Case F, CD=0.4, Thot=609.1°F, P=2100 psia, max SI
Donald C. Cook Unit 1

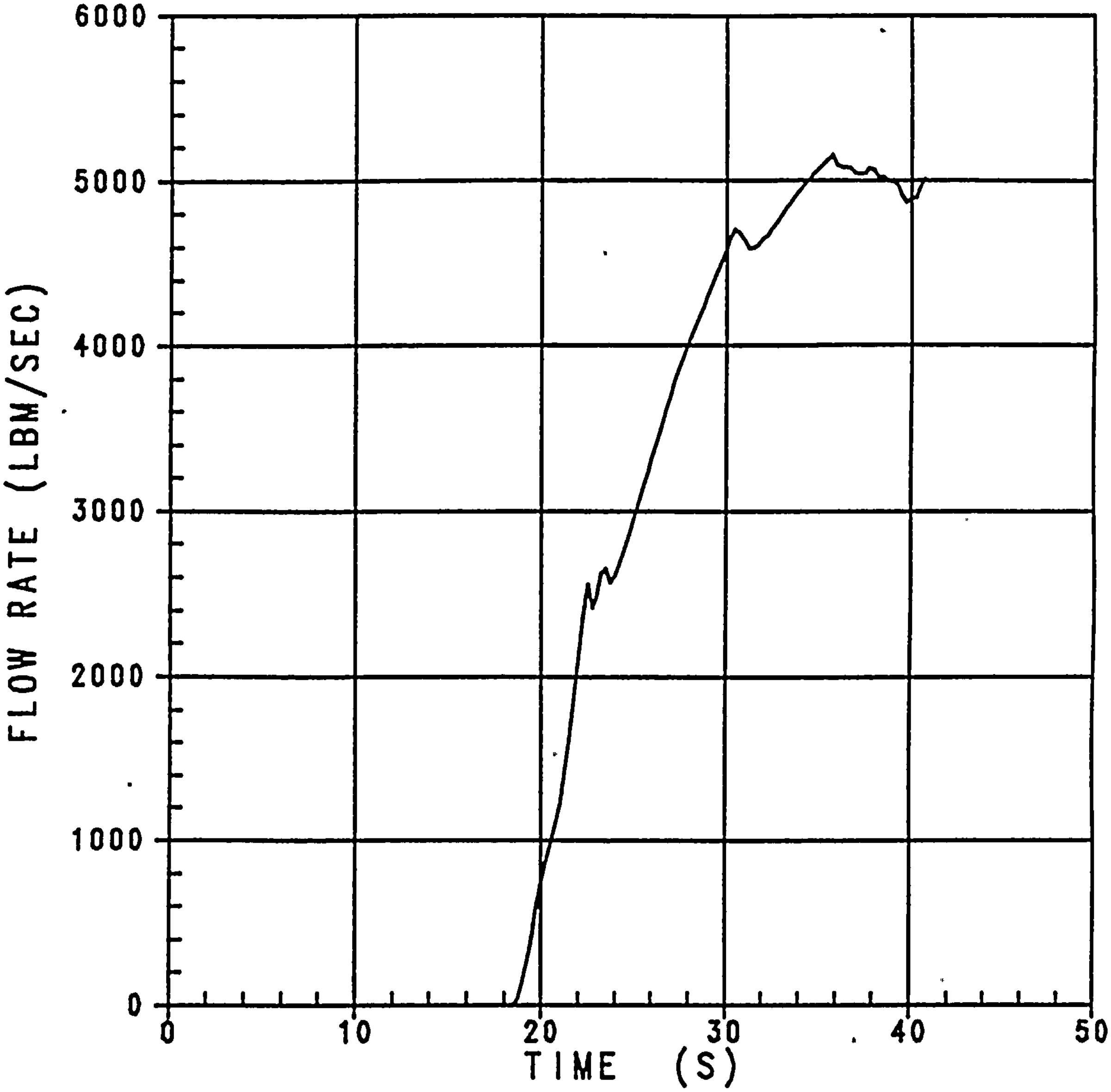

Figure 3.1-5a Accumulator Flow During Blowdown
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

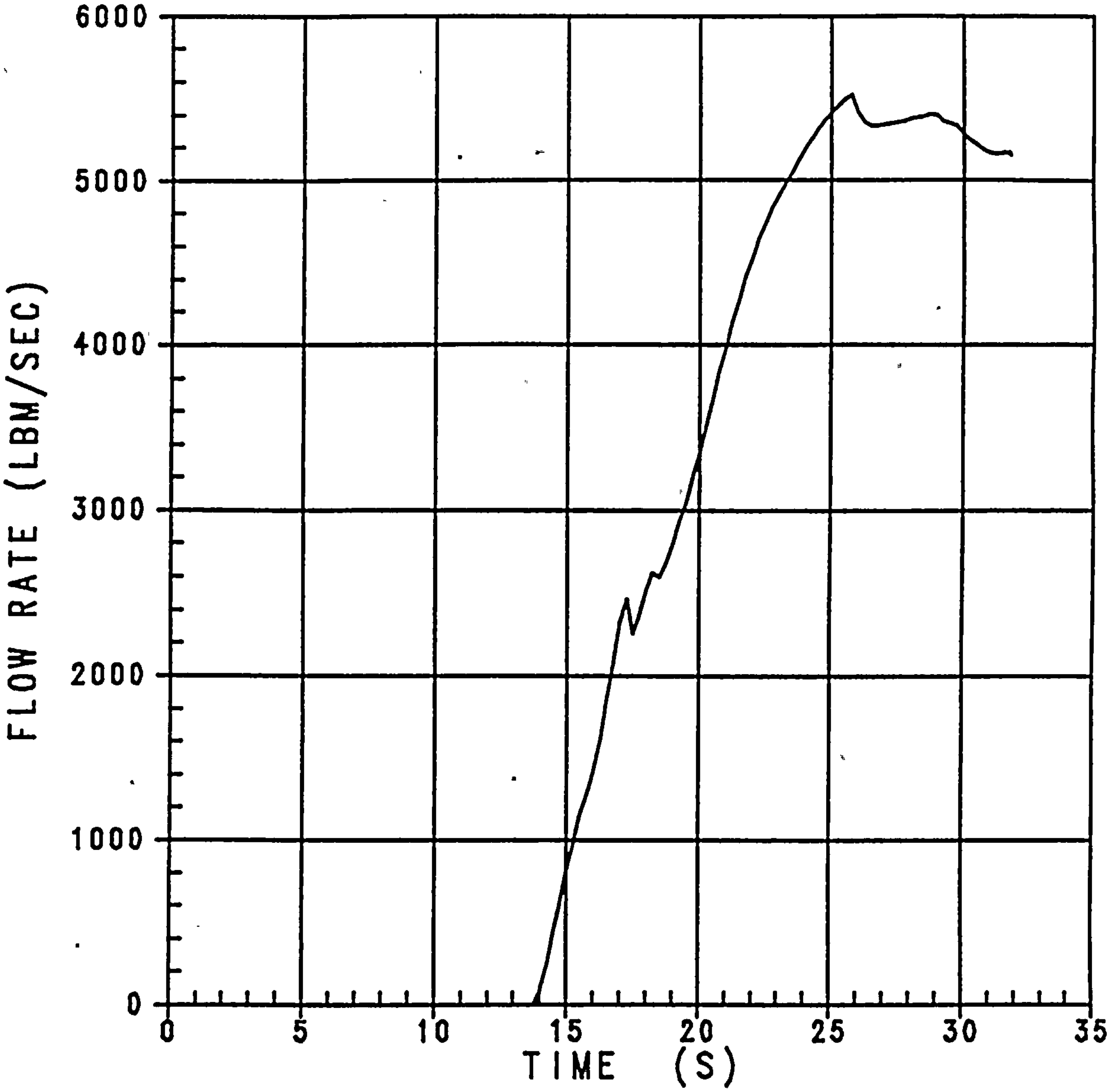

Figure 3.1-5b Accumulator Flow During Blowdown
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

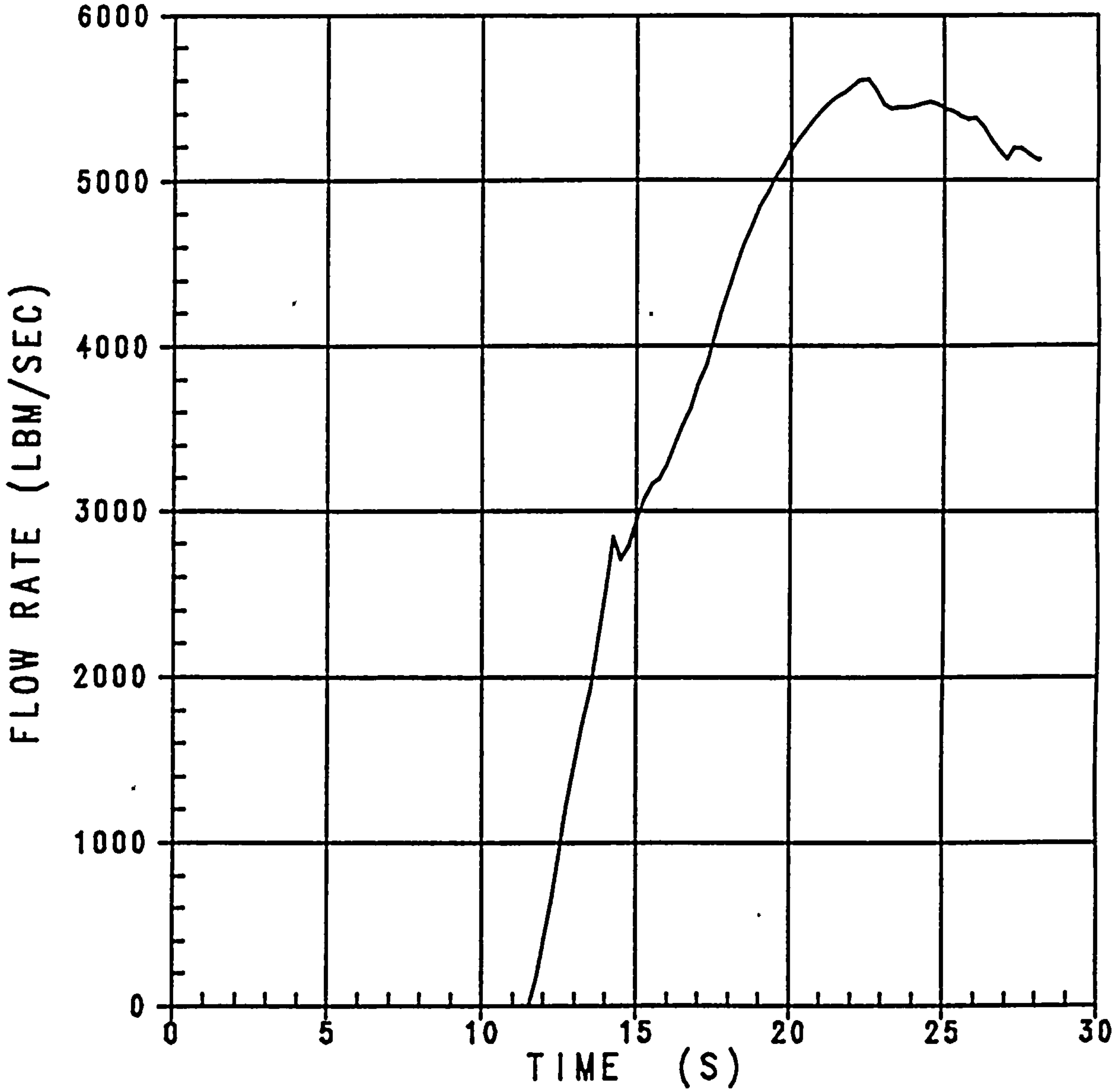

Figure 3.1-5c Accumulator Flow During Blowdown
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

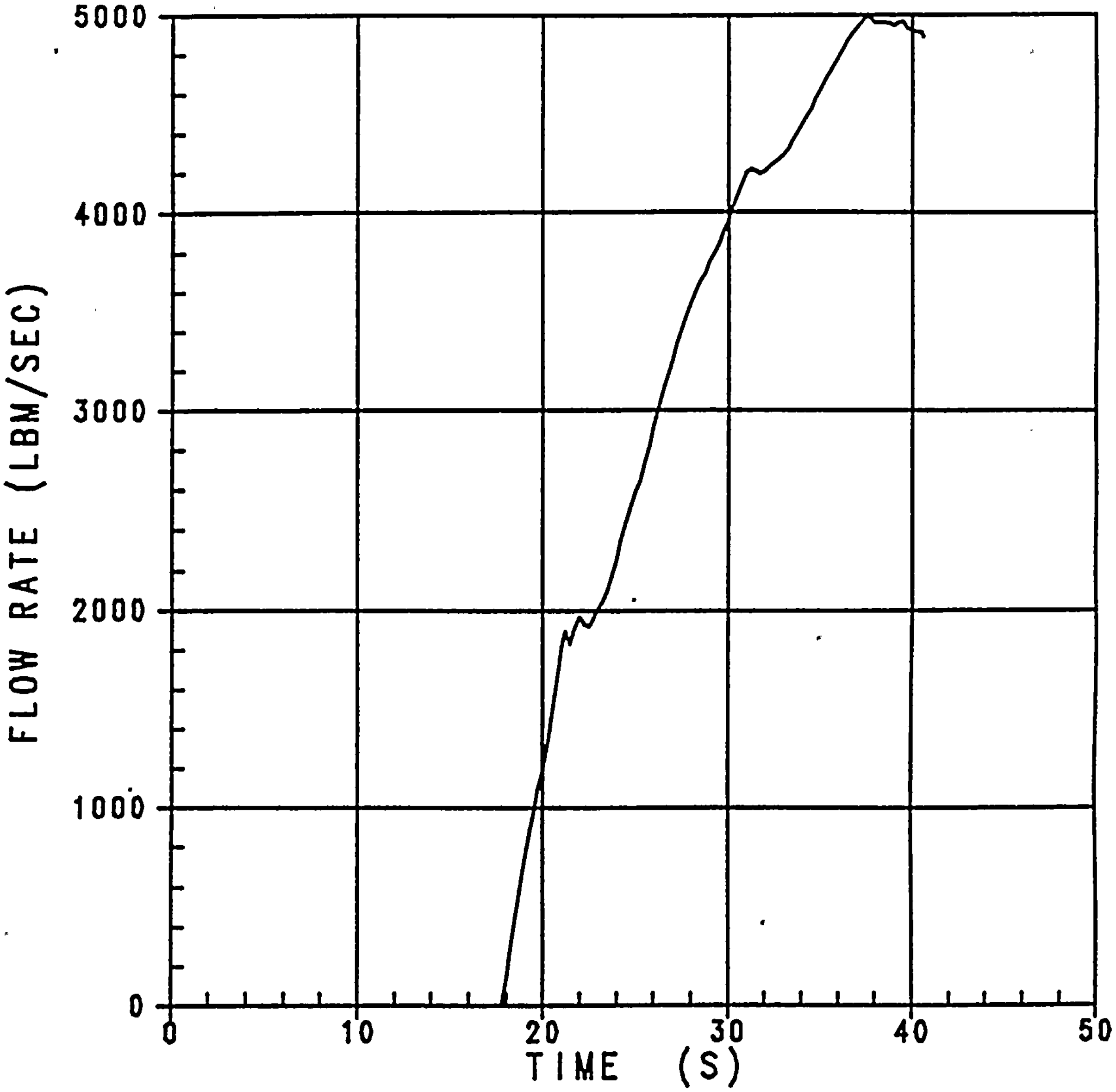

Figure 3.1-5d Accumulator Flow During Blowdown
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

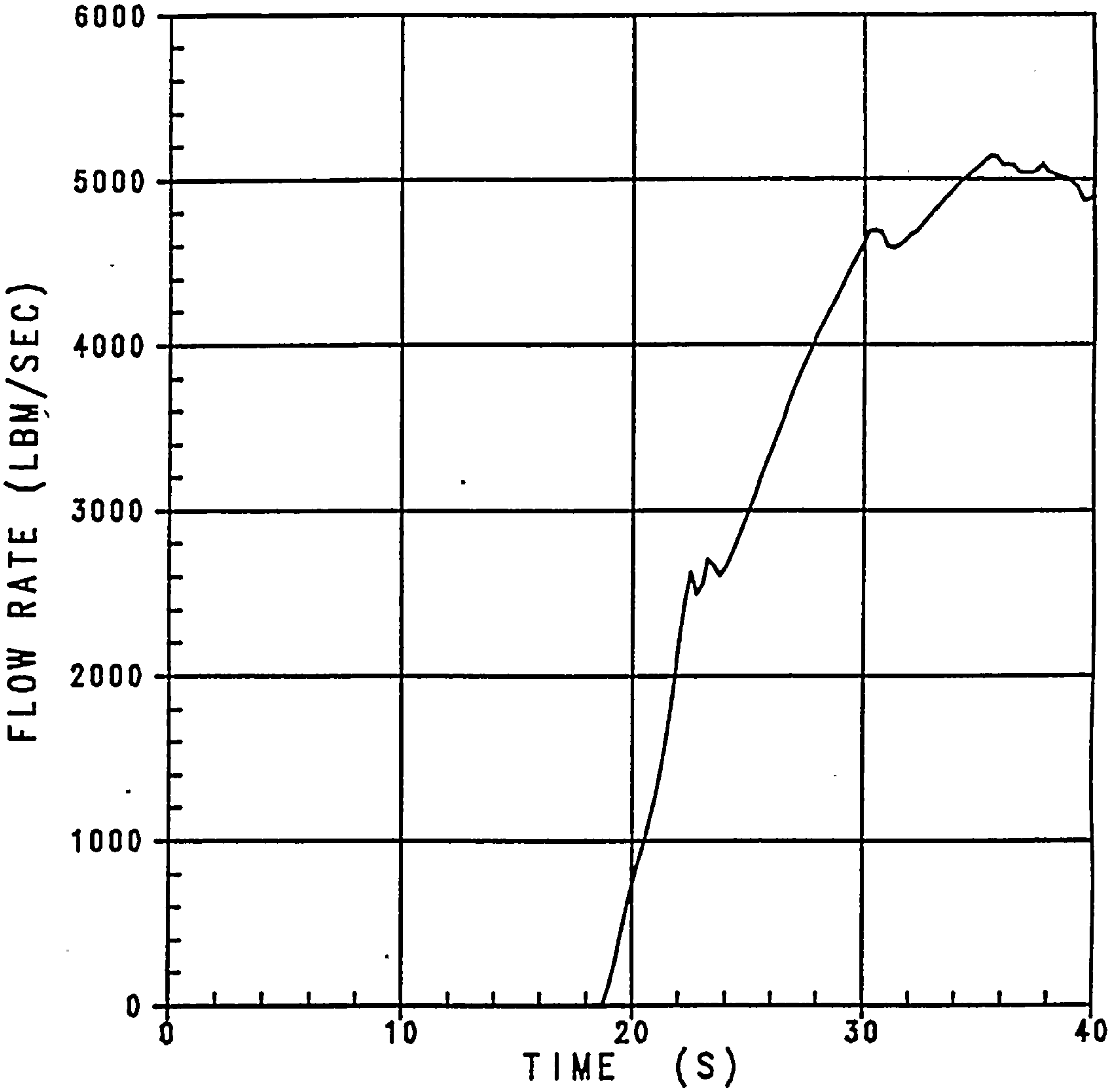

Figure 3.1-5e Accumulator Flow During Blowdown
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

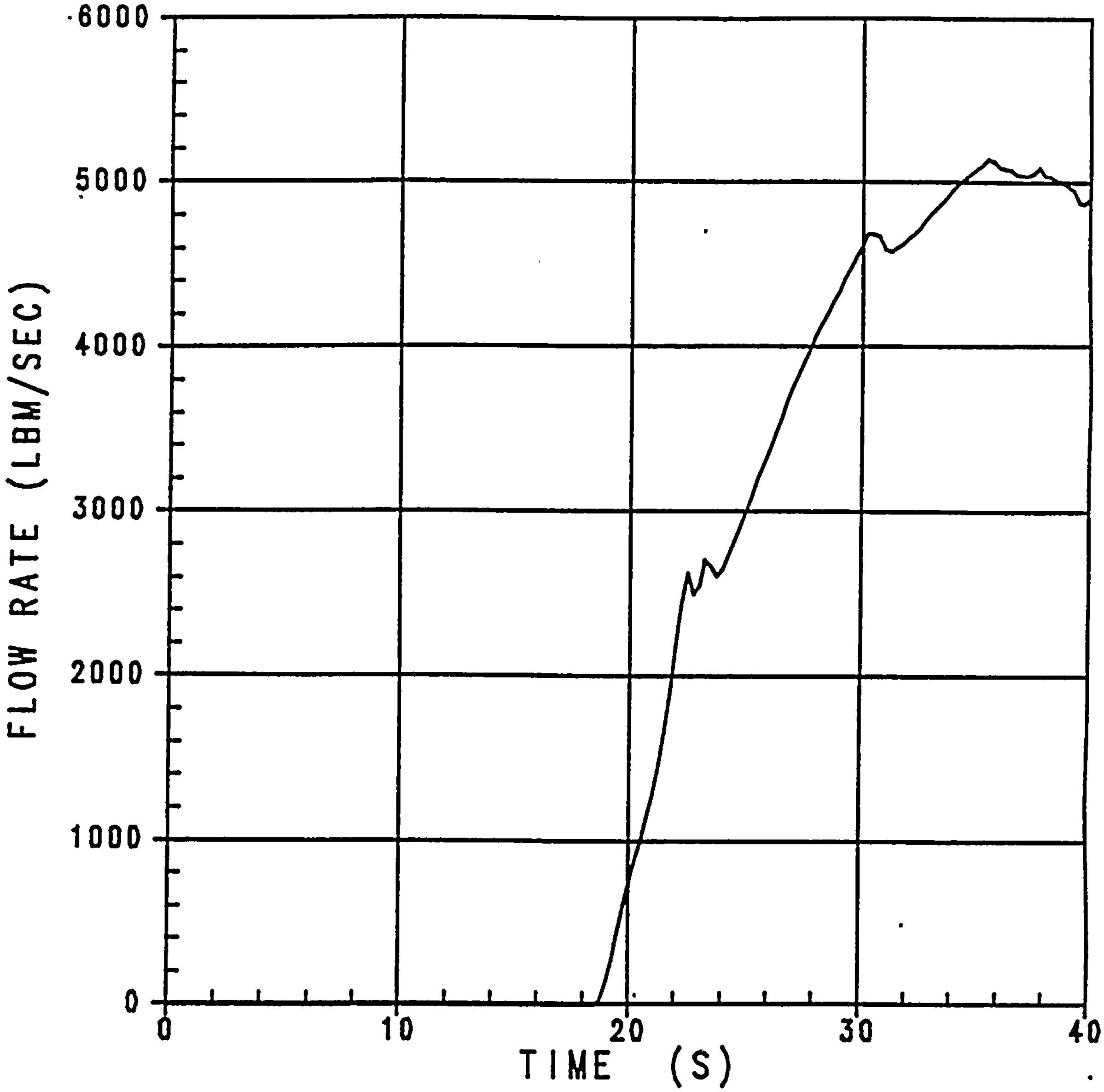

Figure 3.1-5f Accumulator Flow During Blowdown
Case F, CD=0.4, Thot=609.1°F, P=2100 psia, max SI
Donald C. Cook Unit 1

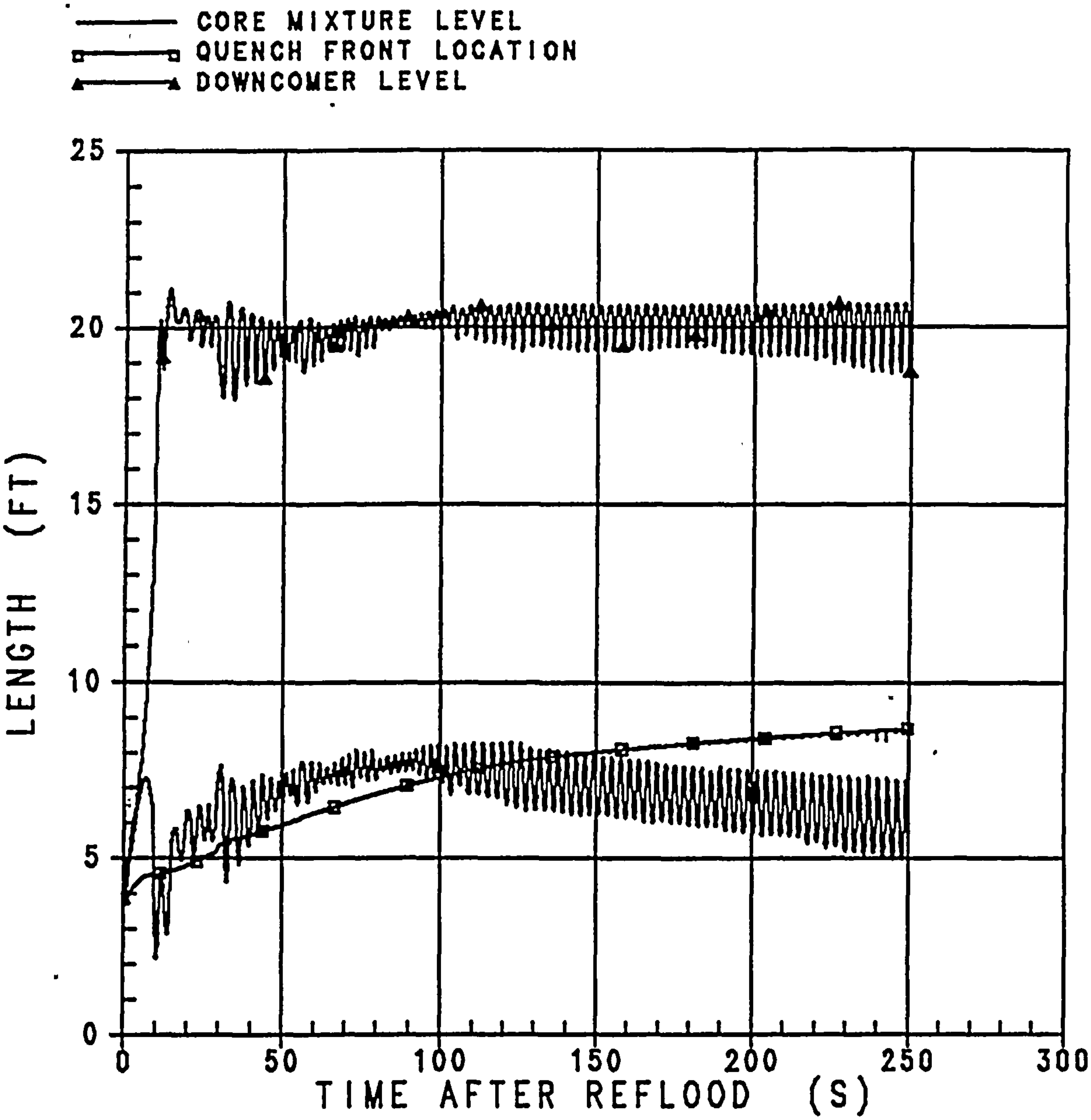

Figure 3.1-6a Vessel Liquid Levels During Reflood
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

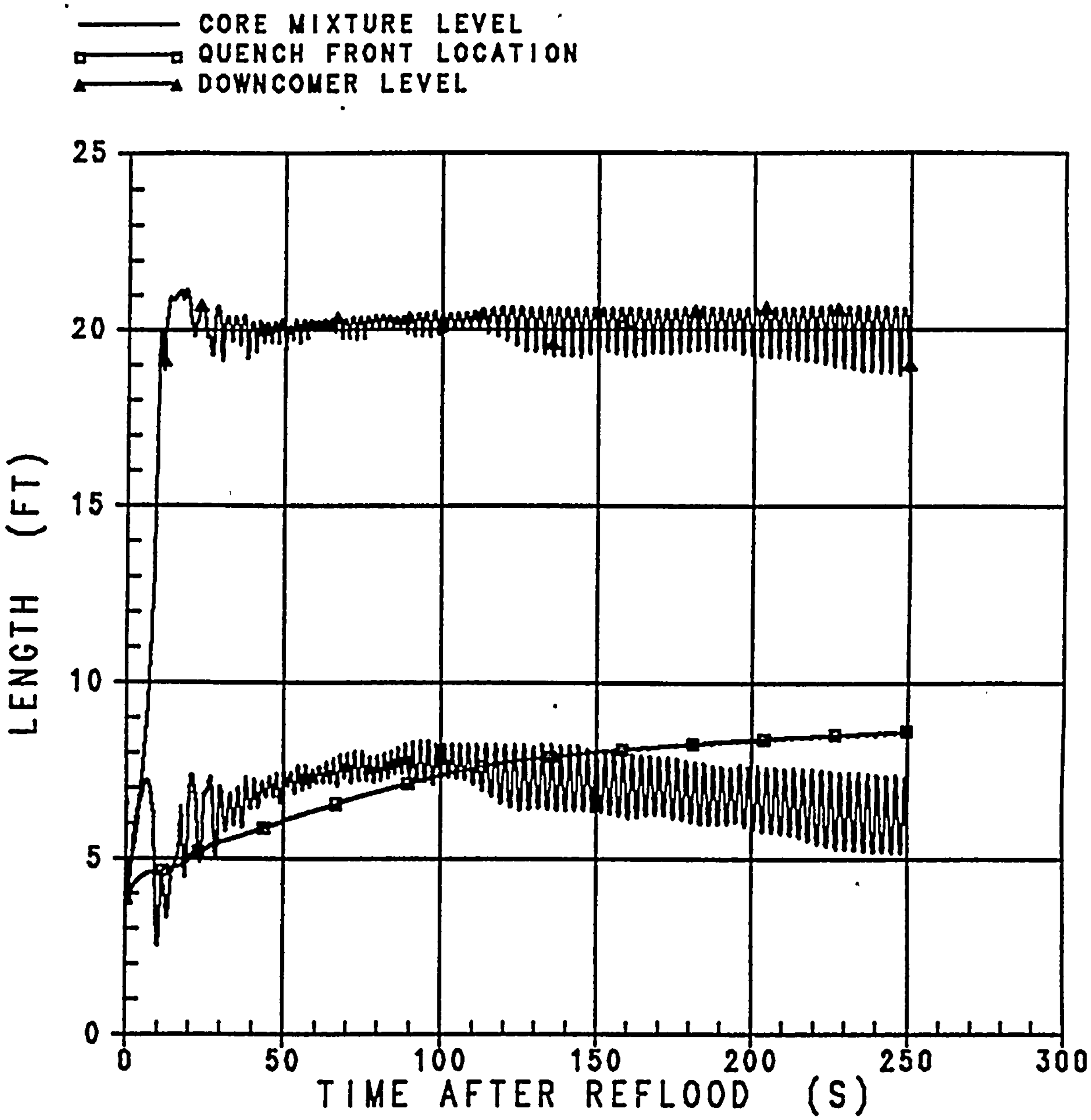

Figure 3.1-6b Vessel Liquid Levels During Reflood
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

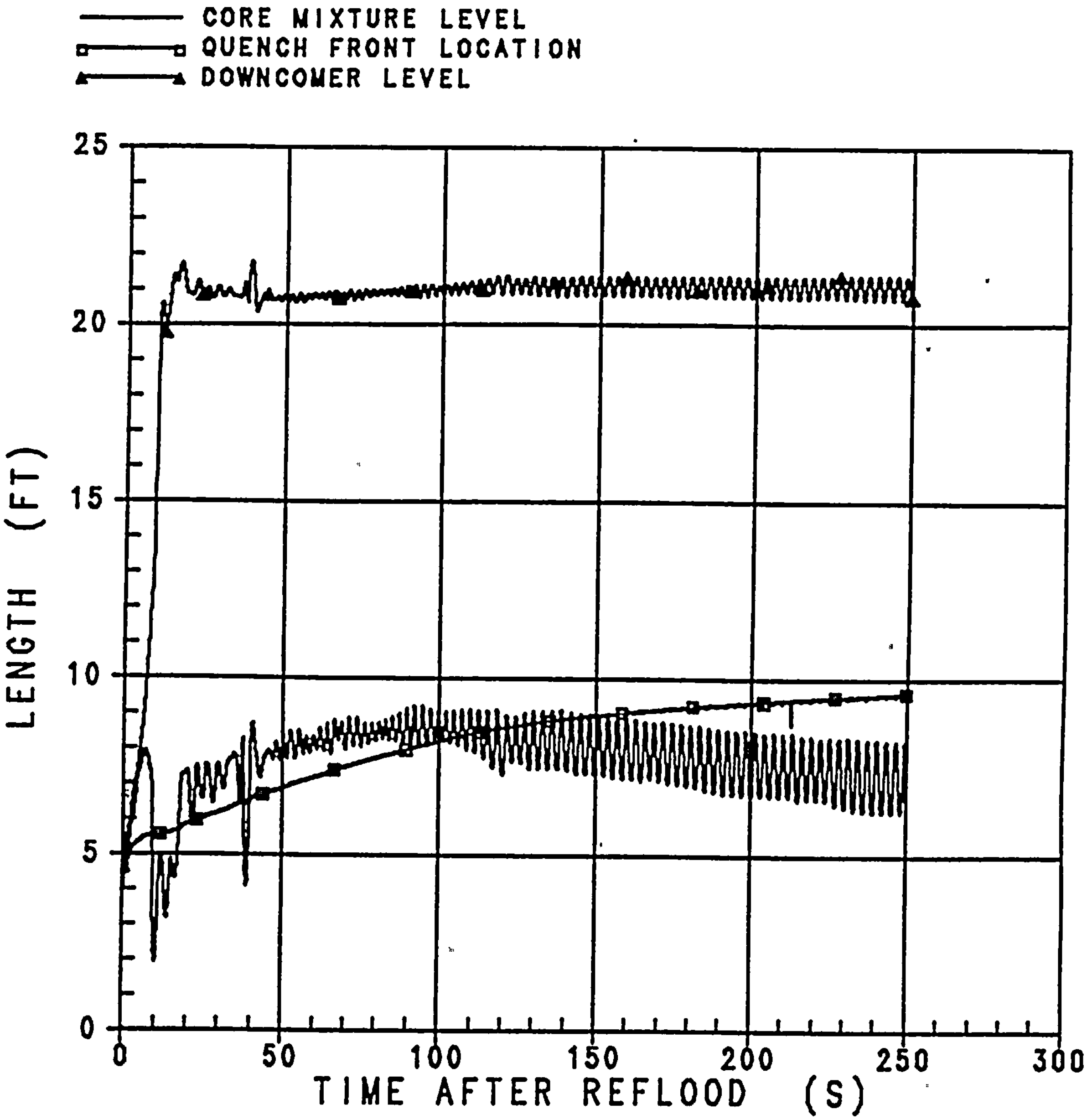

Figure 3.1-6c Vessel Liquid Levels During Reflood
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

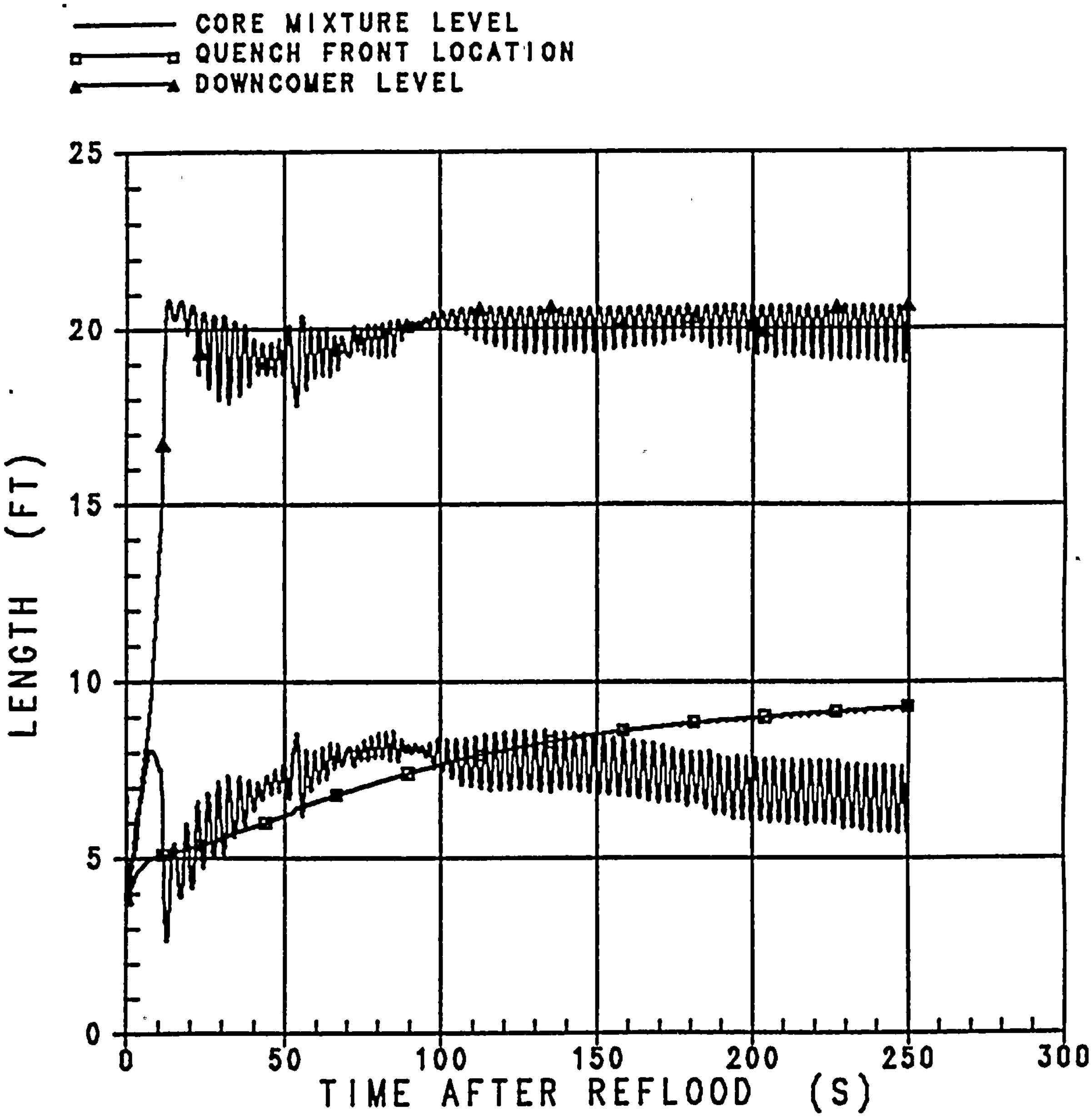

Figure 3.1-6d Vessel Liquid Levels During Reflood
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

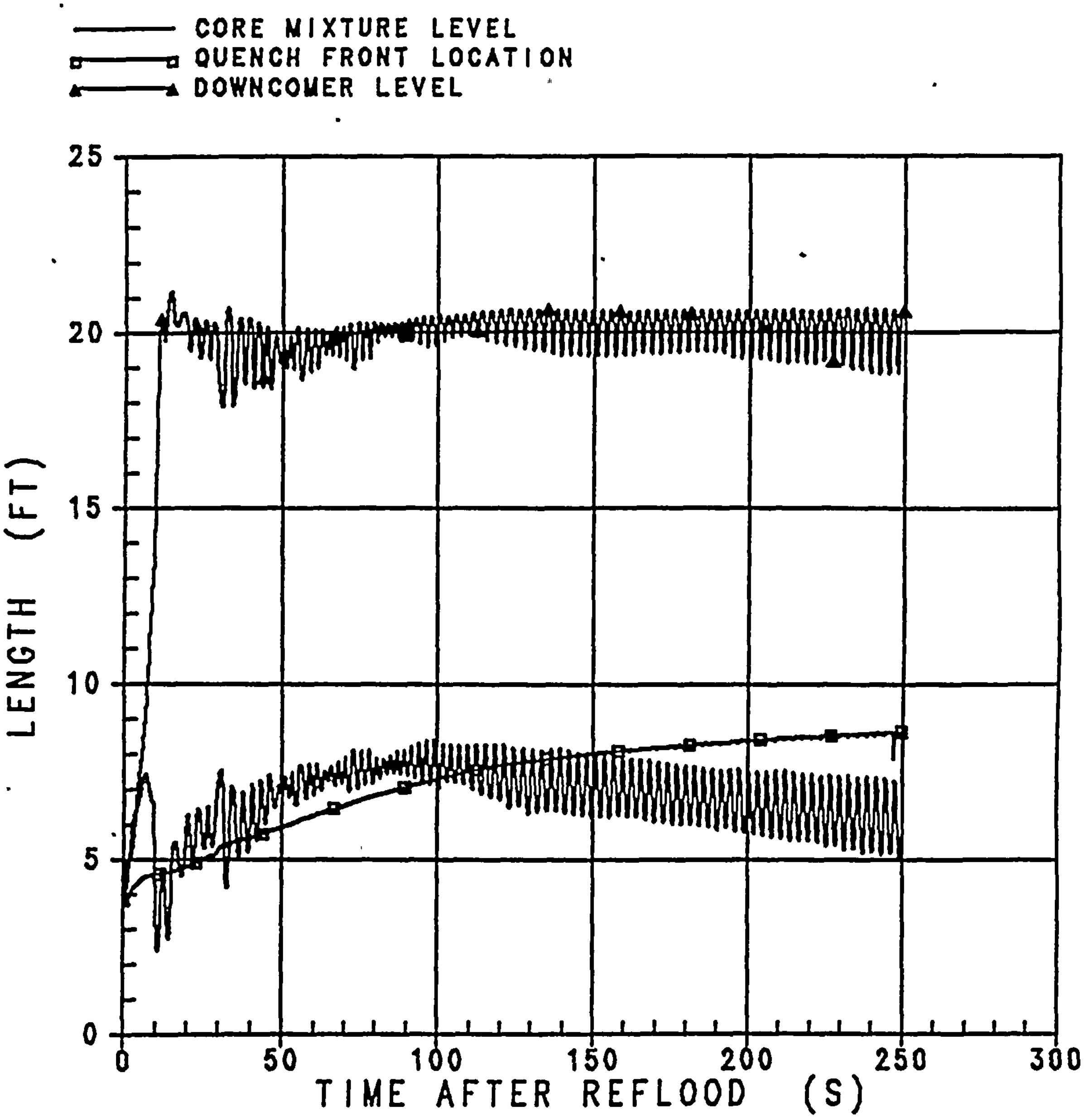

Figure 3.1-6e Vessel Liquid Levels During Reflood
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

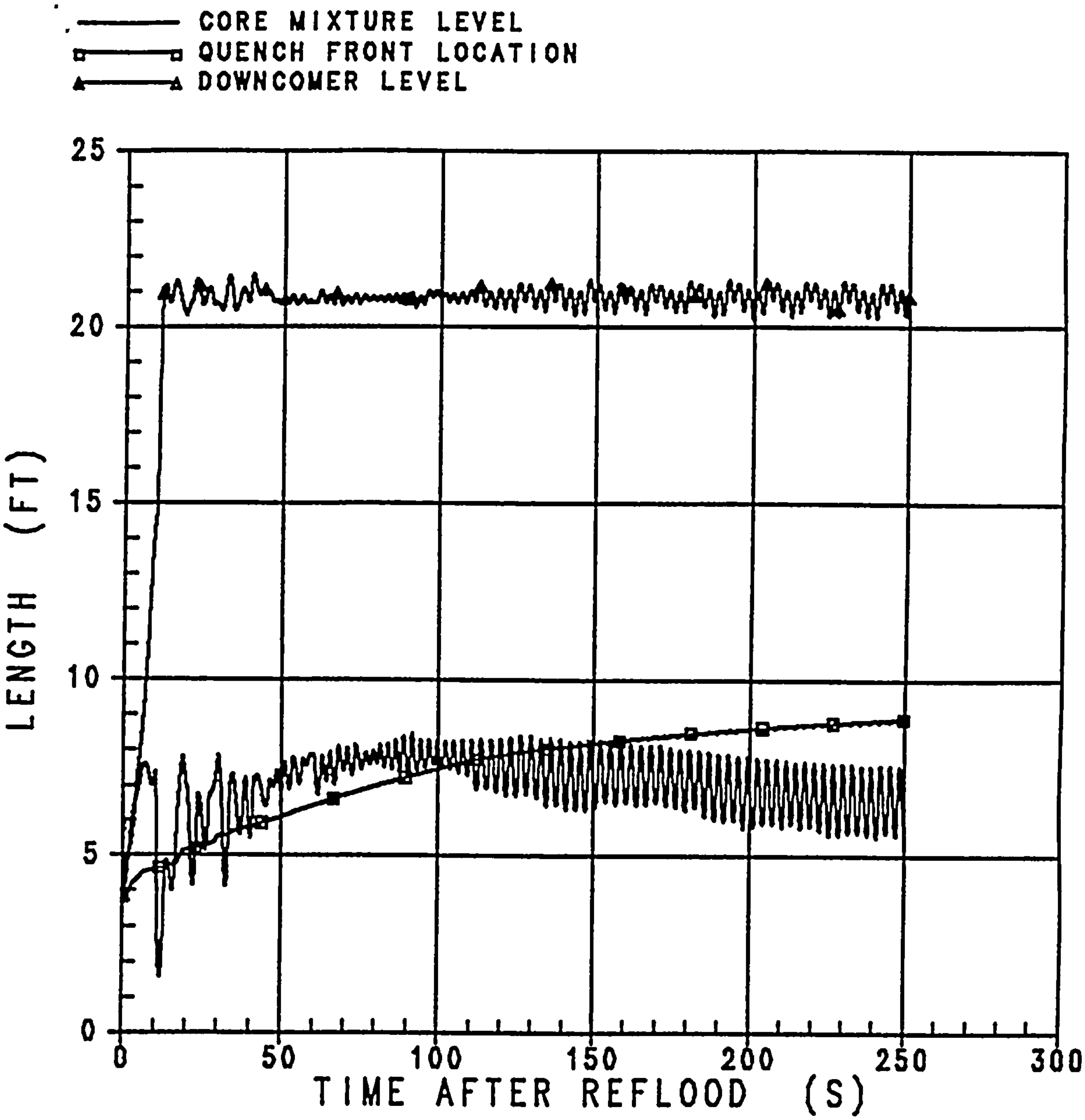

Figure 3.1-6f Vessel Liquid Levels During Reflood
Case F, CD=0.4, Thot=609.1°F, P=2100 psia, max SI
Donald C. Cook Unit 1

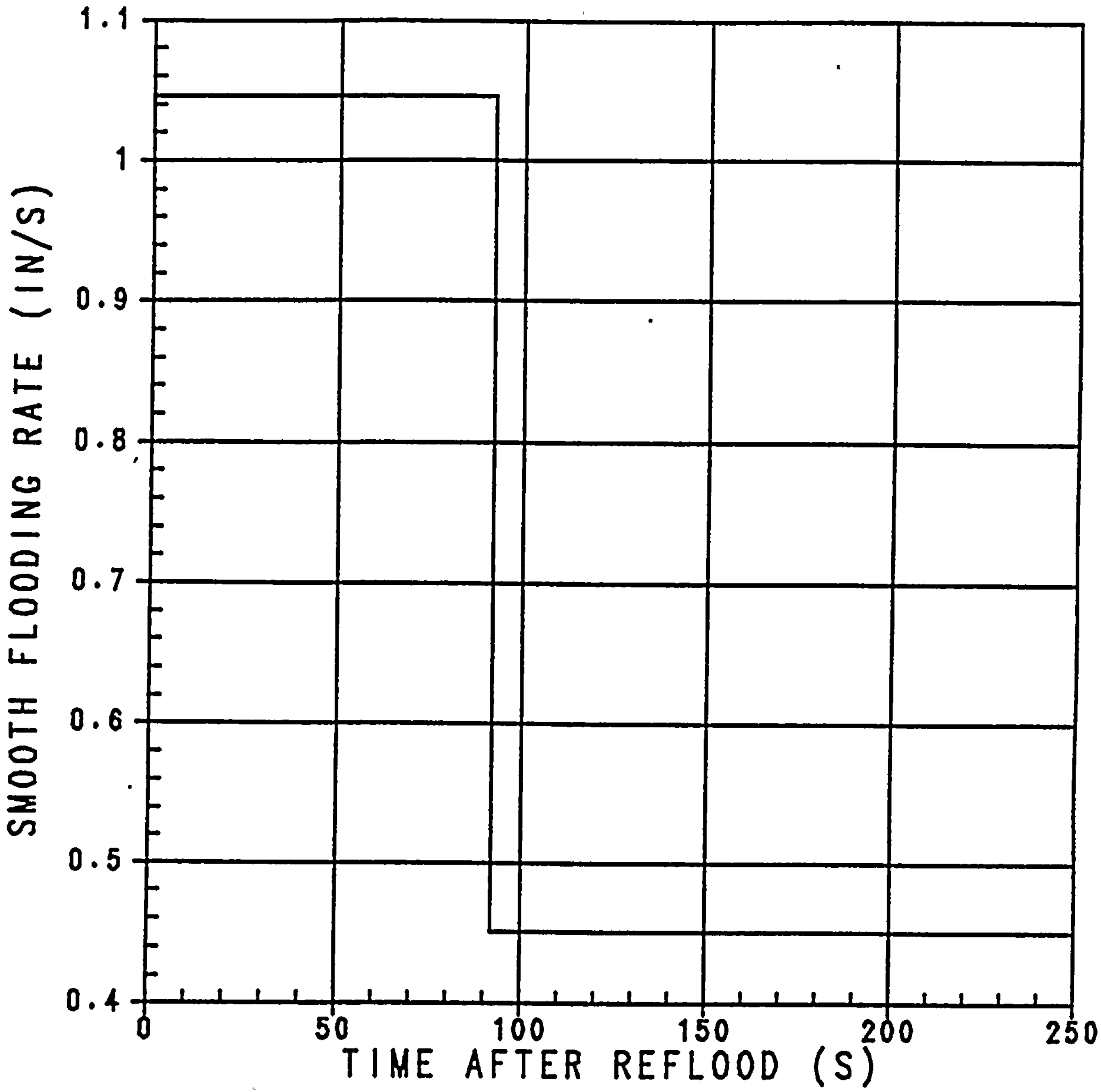

Figure 3.1-7a Core Inlet Flow During Reflood
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

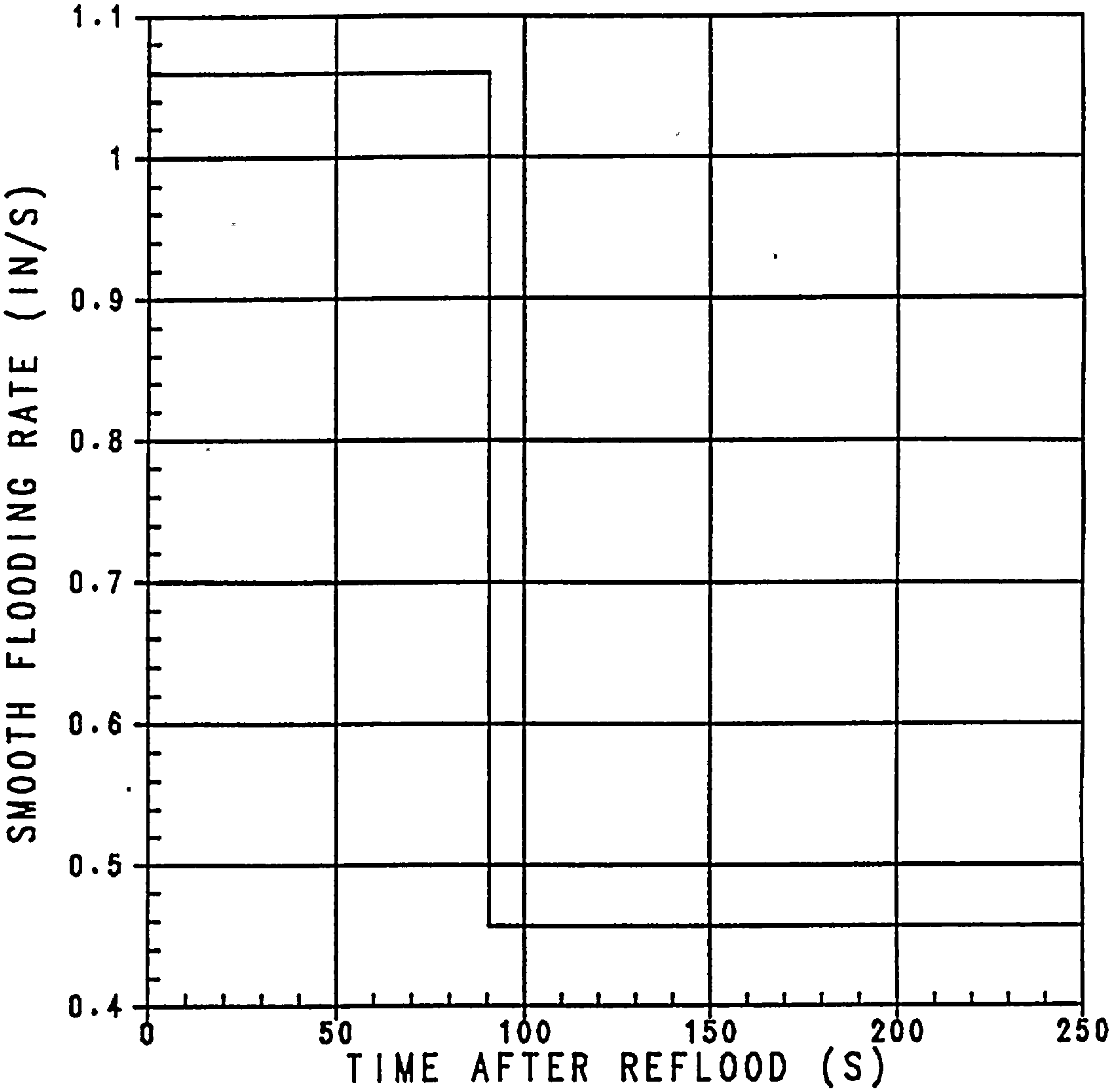

Figure 3.1-7b Core Inlet Flow During Reflood
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

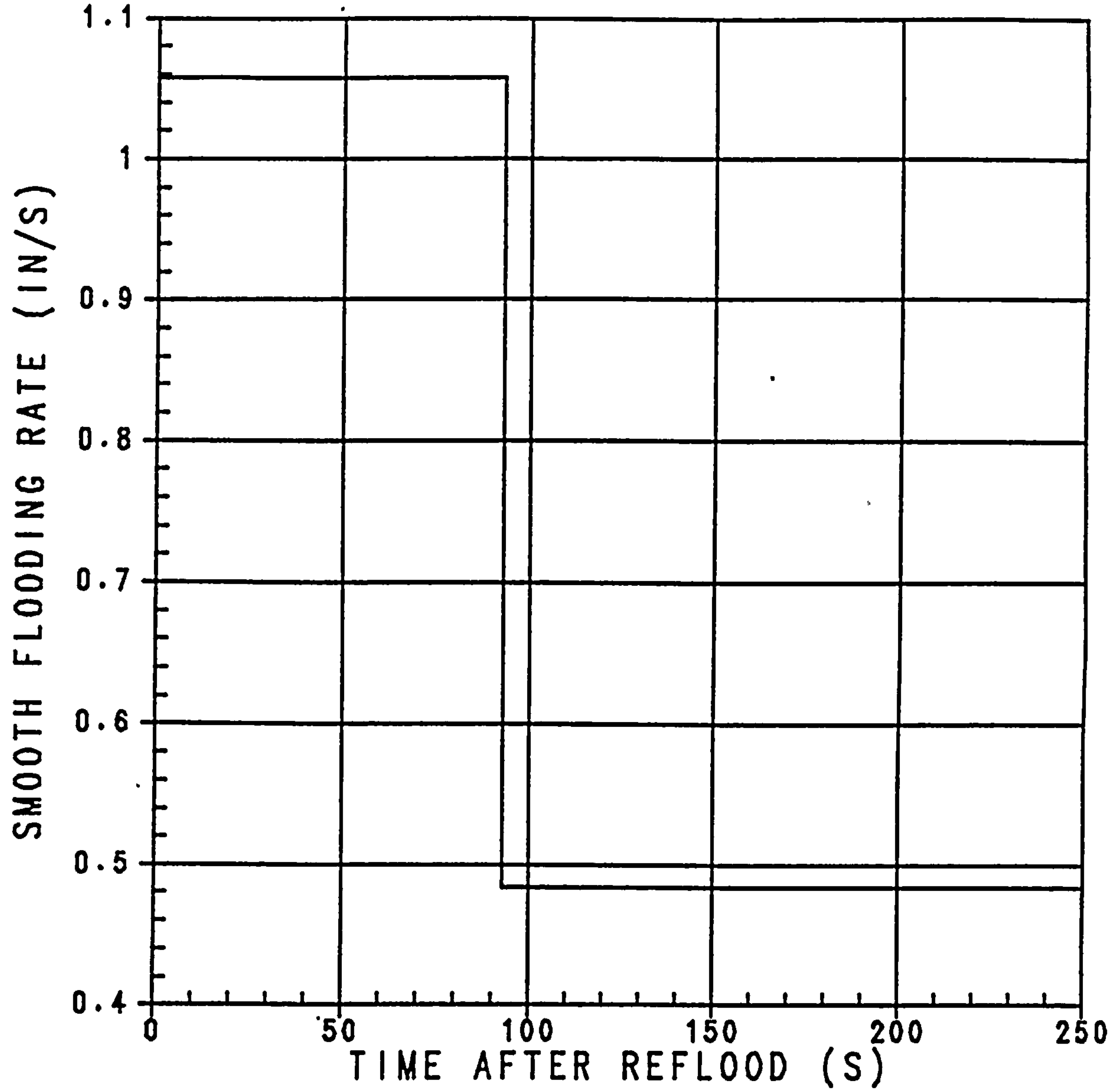

Figure 3.1-7c Core Inlet Flow During Reflood
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

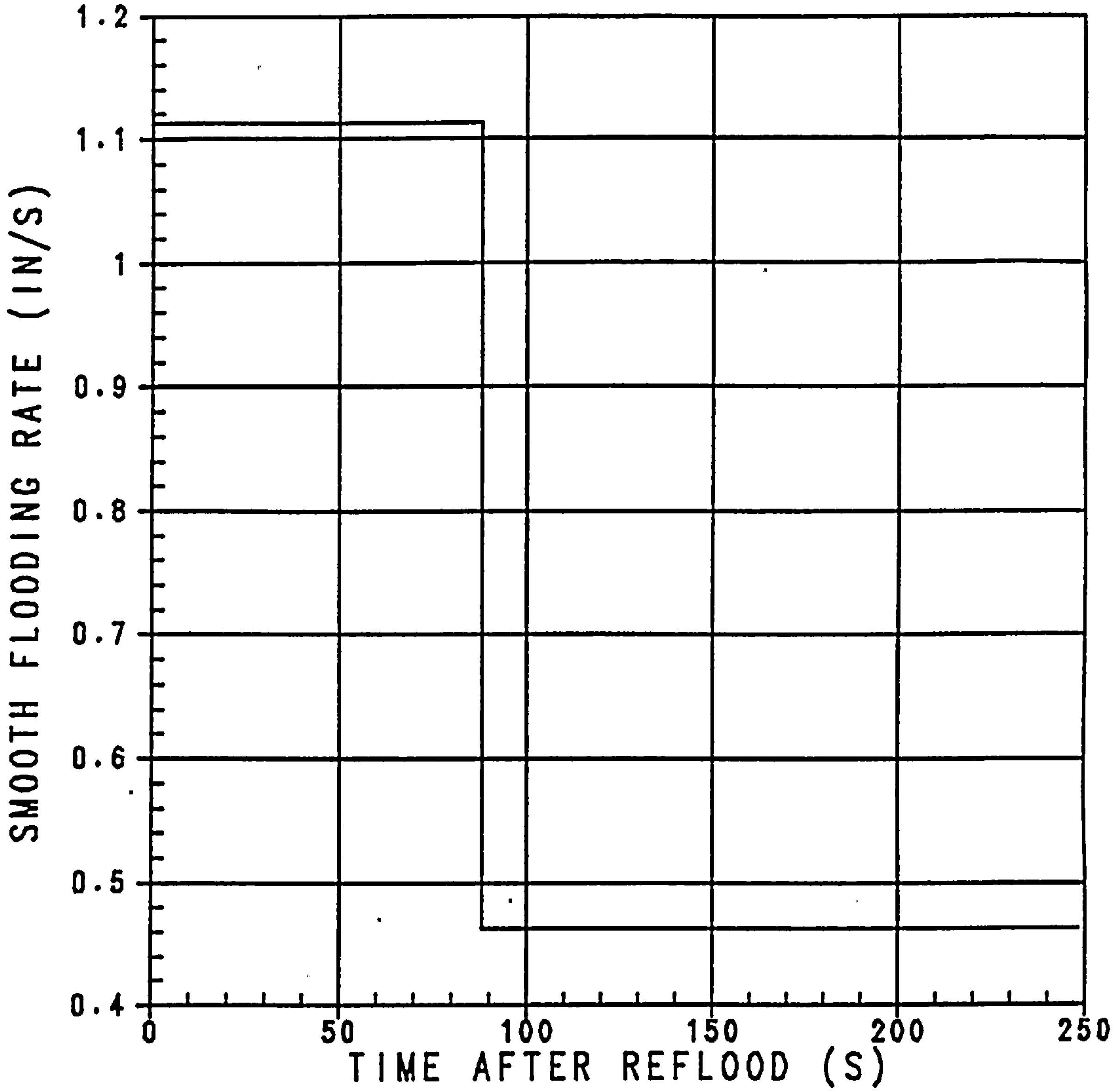

Figure 3.1-7d Core Inlet Flow During Reflood
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

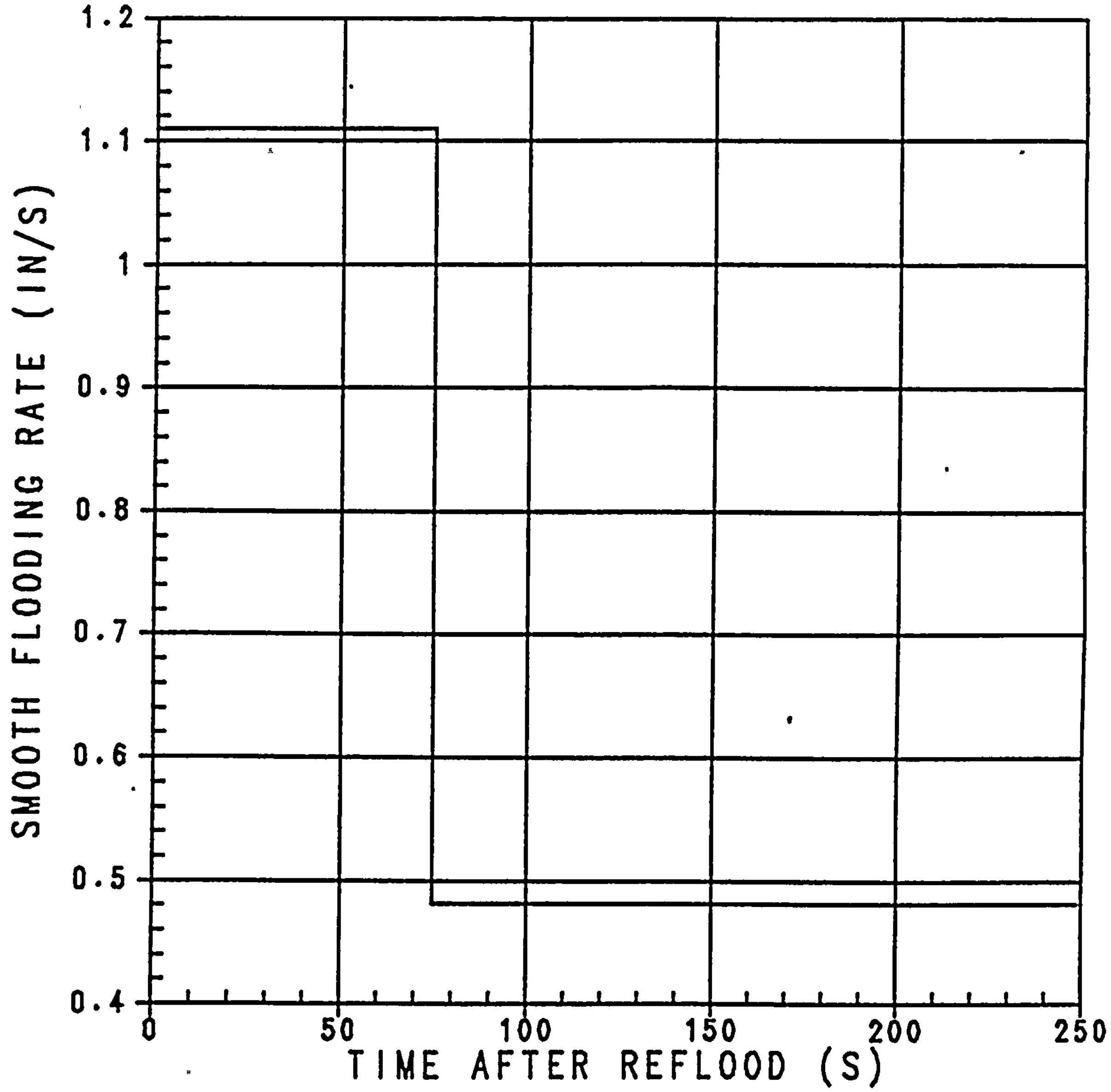

Figure 3.1-7e Core Inlet Flow During Reflood
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

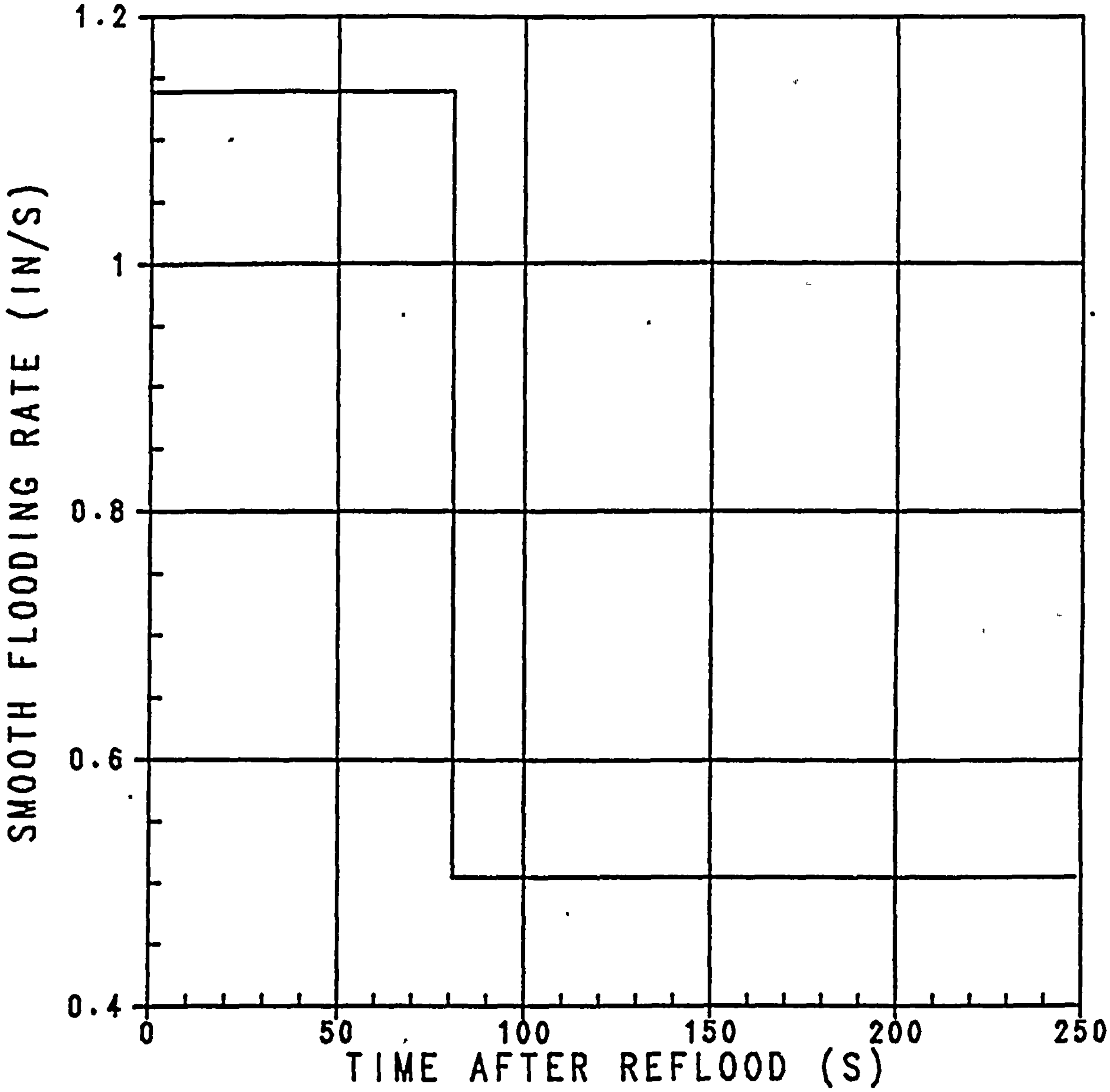

Figure 3.1-7f	Core Inlet Flow During Reflood
Case F, CD=0.4, Thot=609.1°F, P=2100 psia, max SI
Donald C. Cook Unit 1

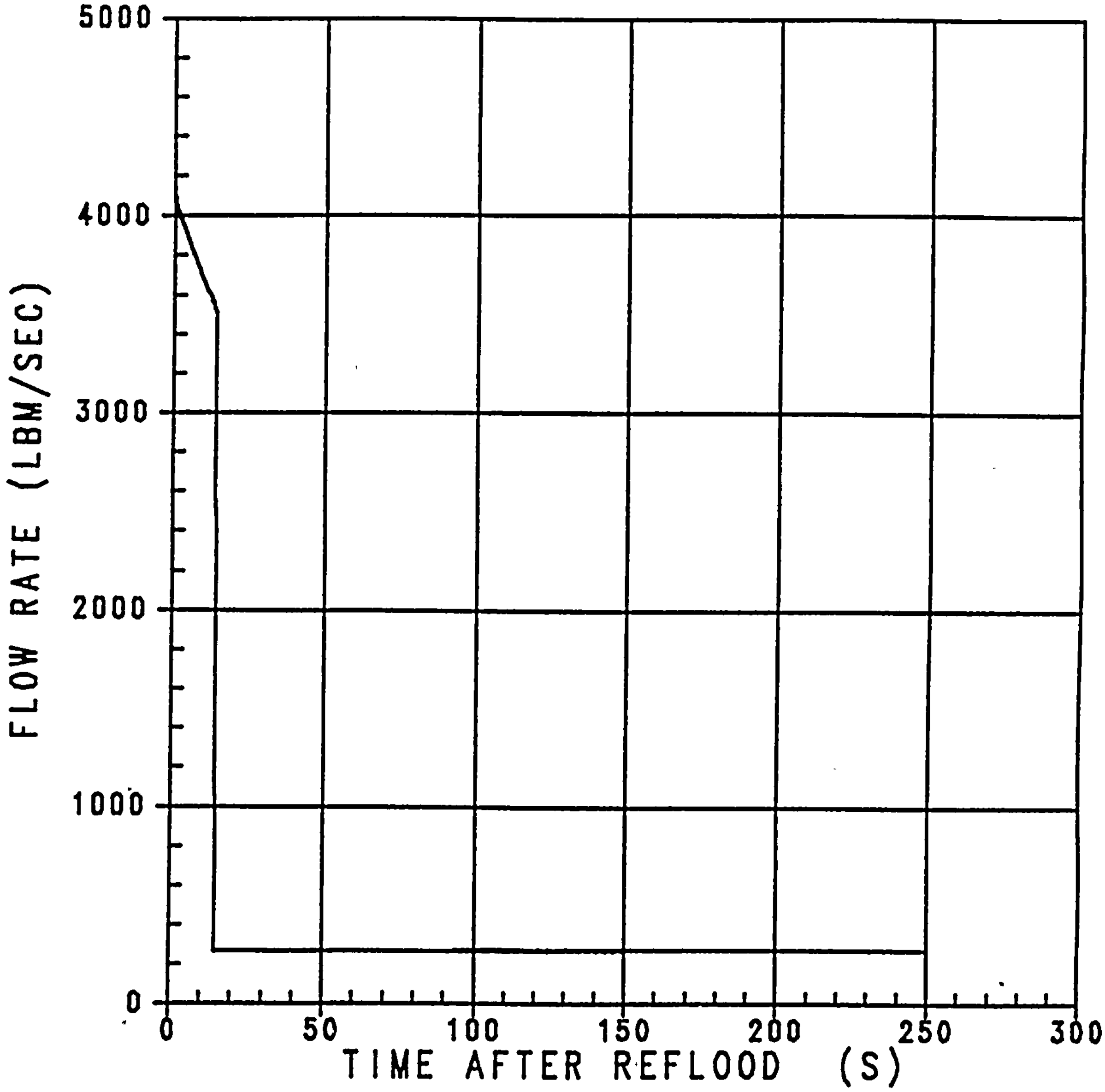

Figure 3.1-8a Accumulator and SI Flow During Reflood
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

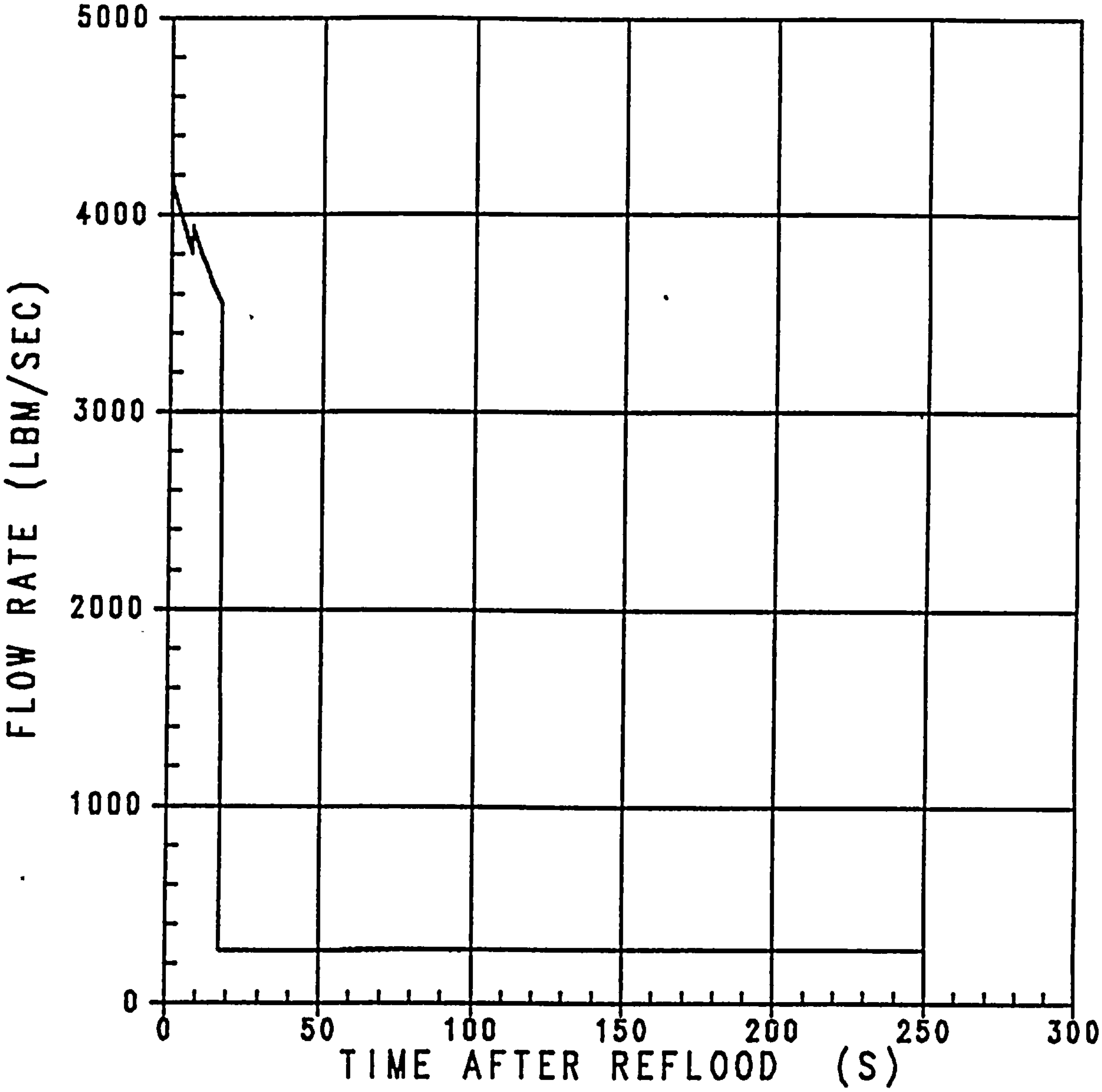

Figure 3.1-8b Accumulator and SI Flow During Reflood
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

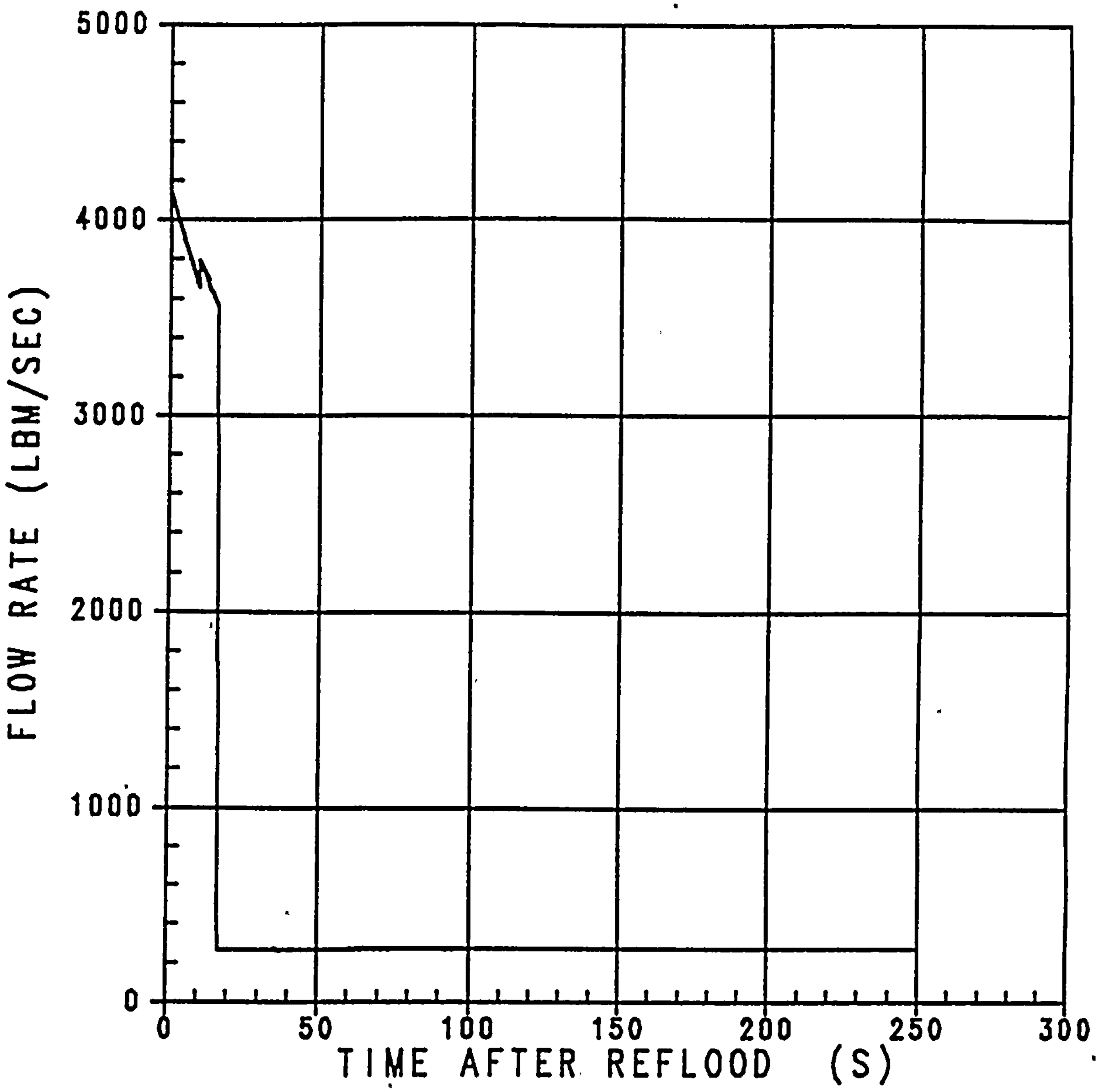

Figure 3.1-8c Accumulator and SI Flow During Reflood
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

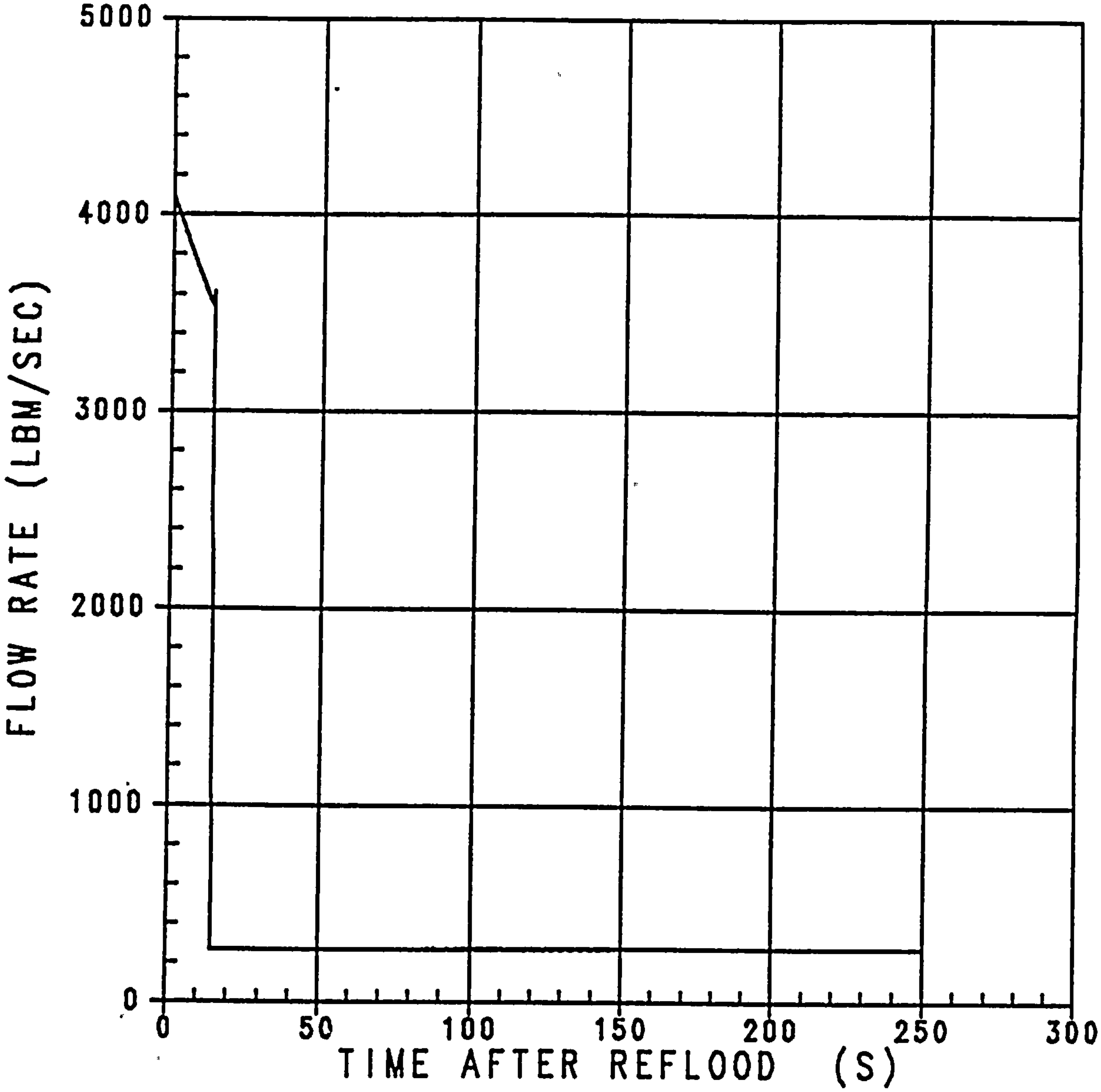

Figure 3.1-8d Accumulator and SI Flow During Reflood
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

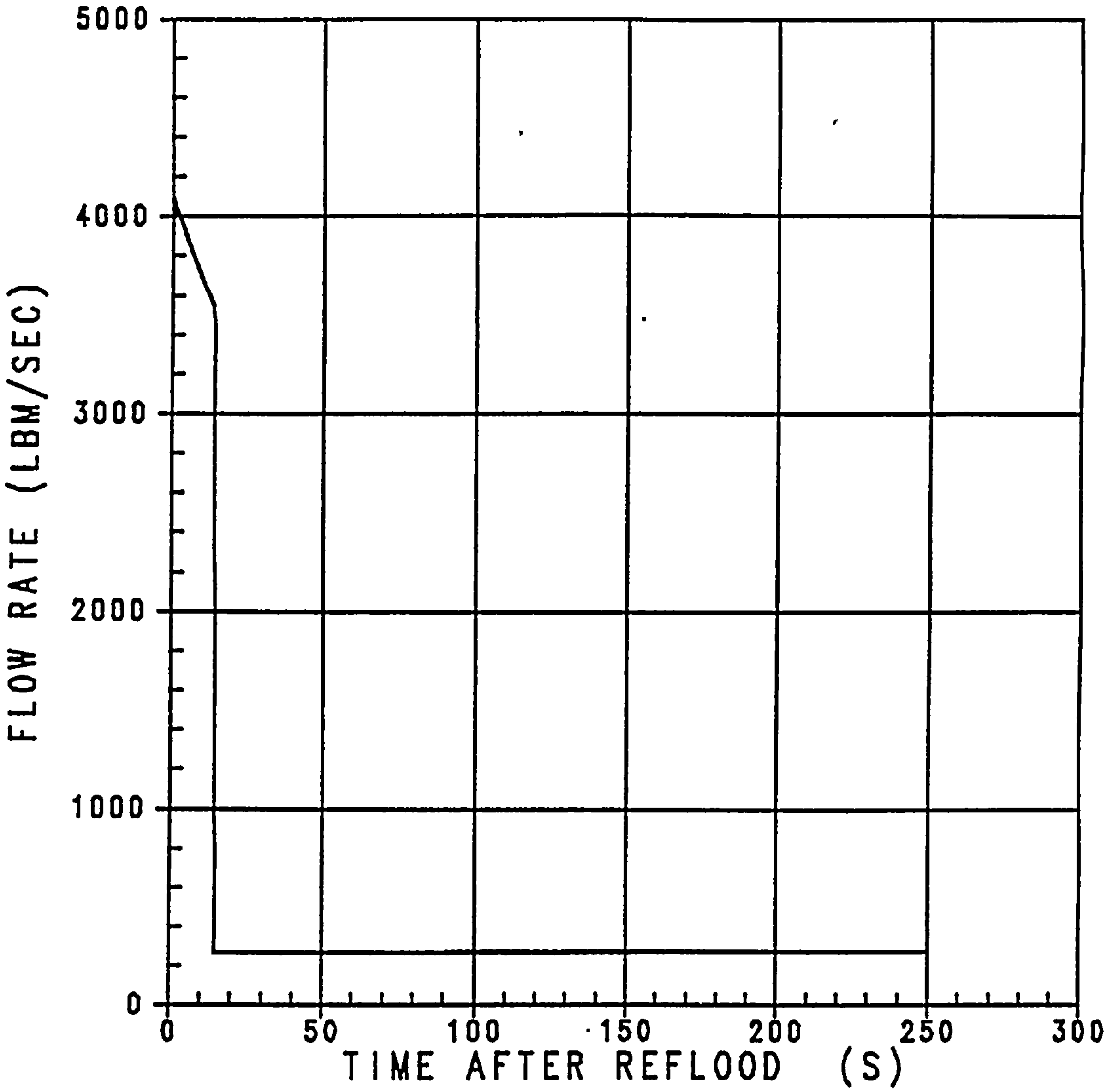

Figure 3.1-8e Accumulator and SI Flow During Reflood
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

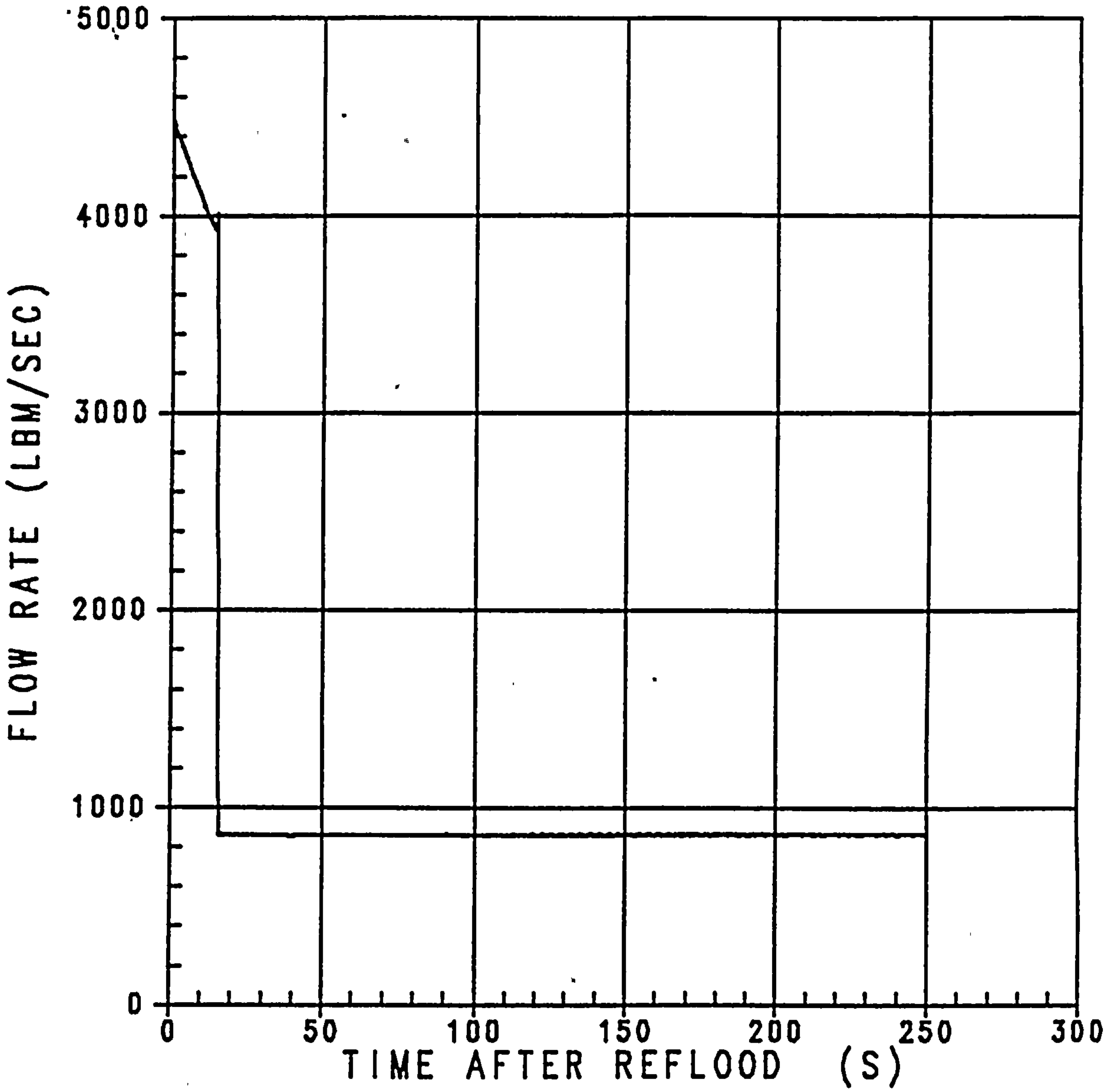

Figure 3.1-8f Accumulator and SI Flow During Reflood
Case F, CD=0.4, Thot=609.1°F, P=2100 psia, max SI
Donald C. Cook Unit 1

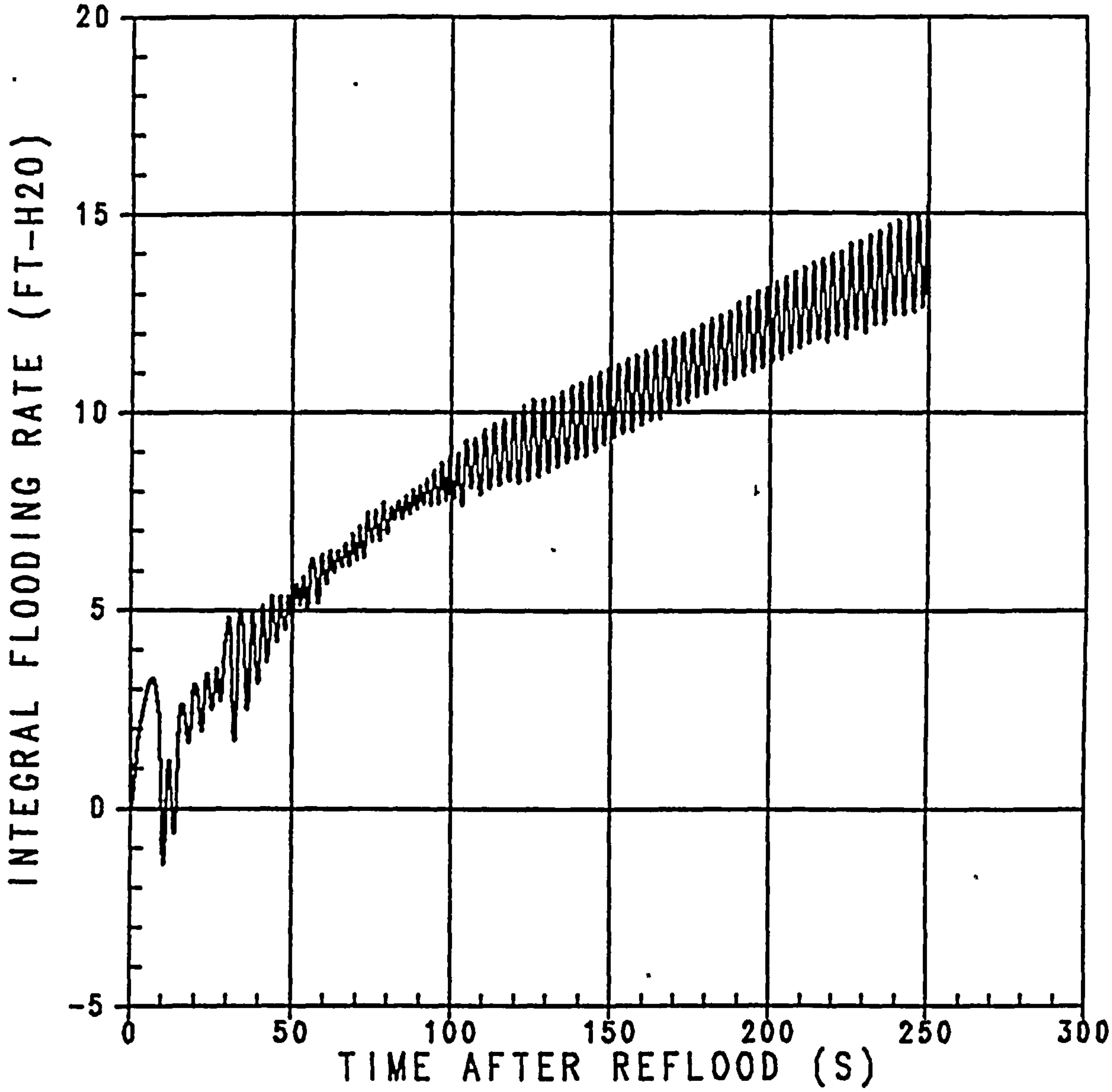

Figure 3.1-9a Integral of Core Inlet Flow
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

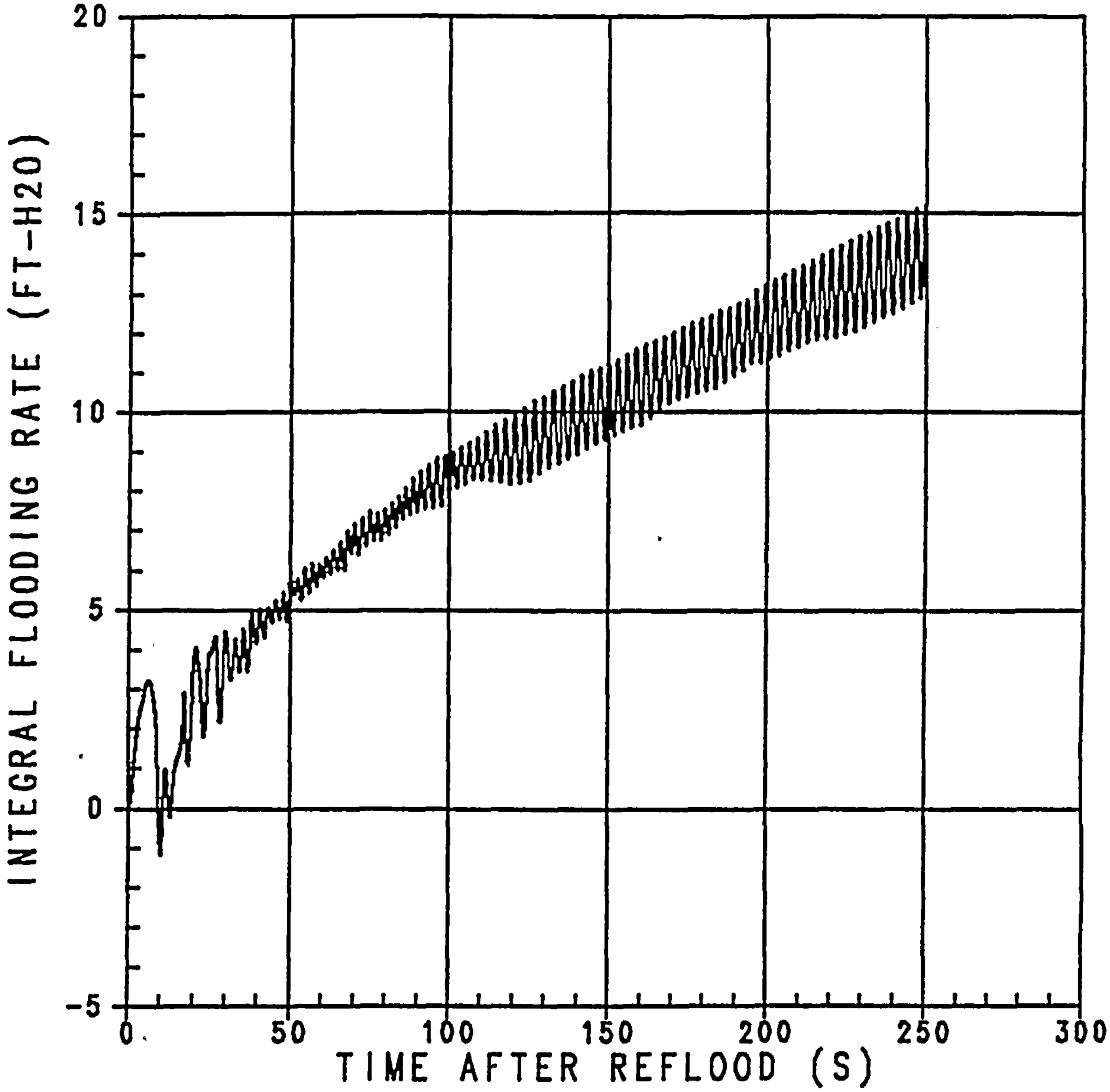

Figure 3.1-9b Integral of Core Inlet Flow
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

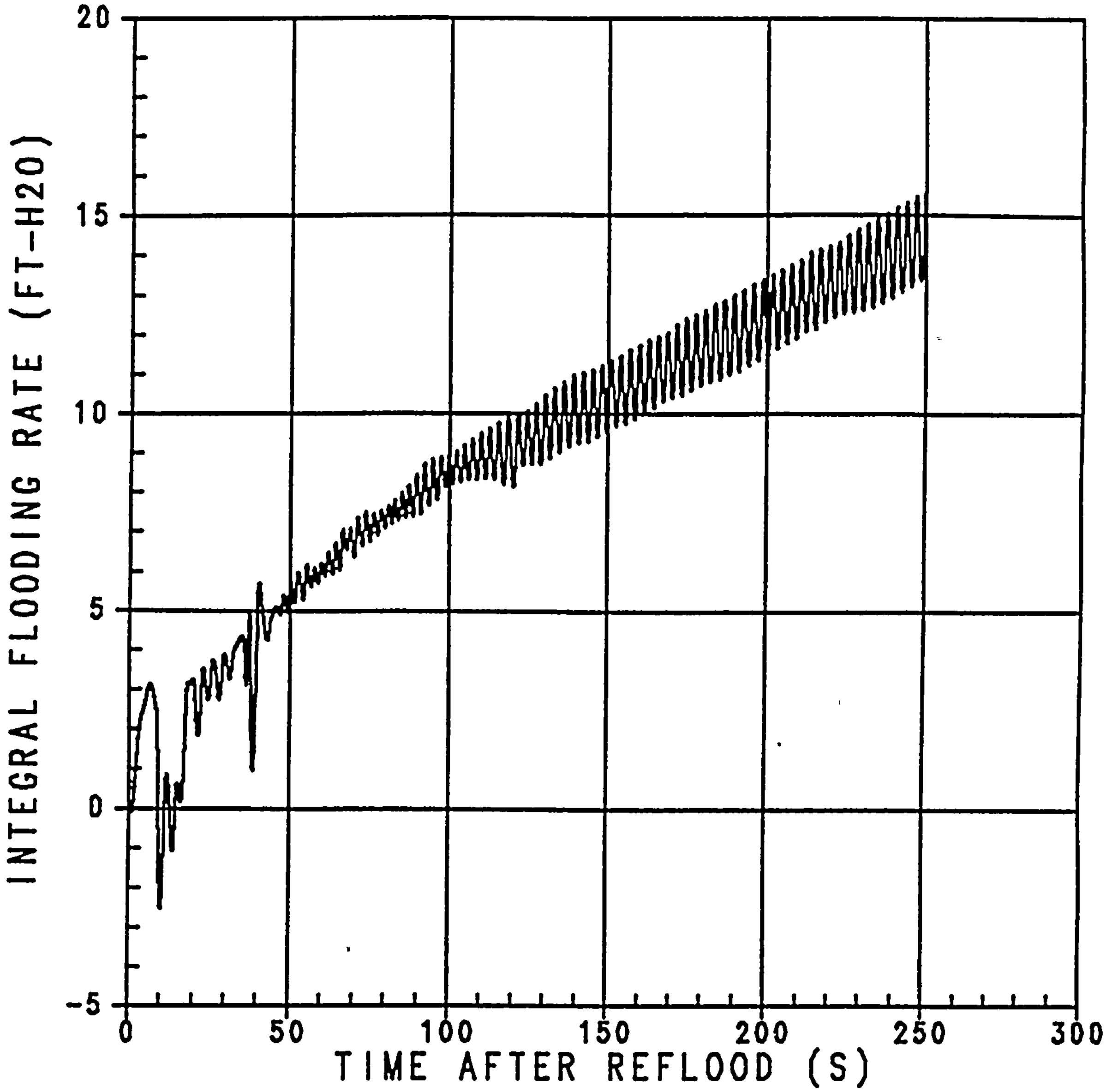

Figure 3.1-9c Integral of Core Inlet Flow
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

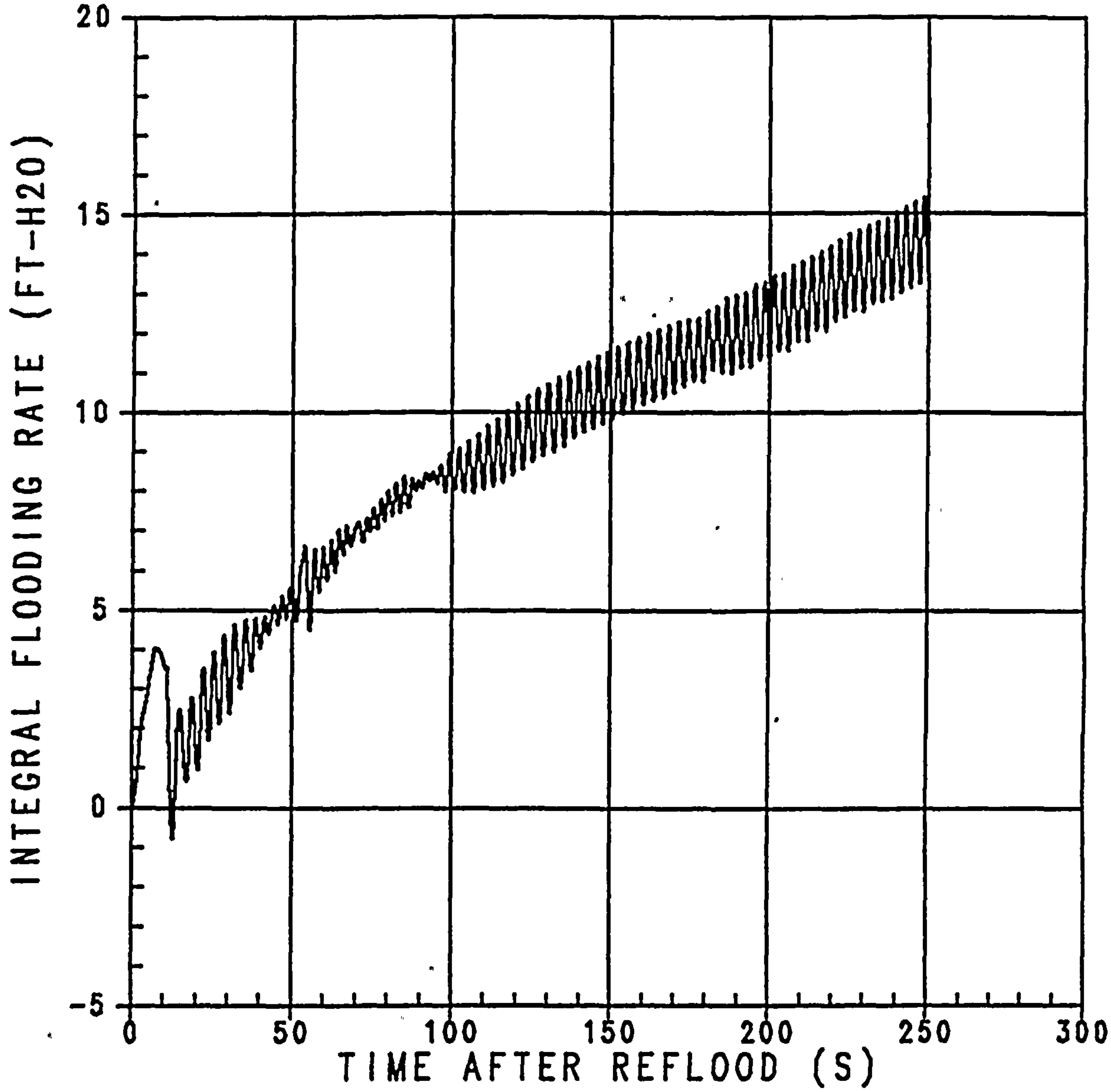

Figure 3.1-9d Integral of Core Inlet Flow
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

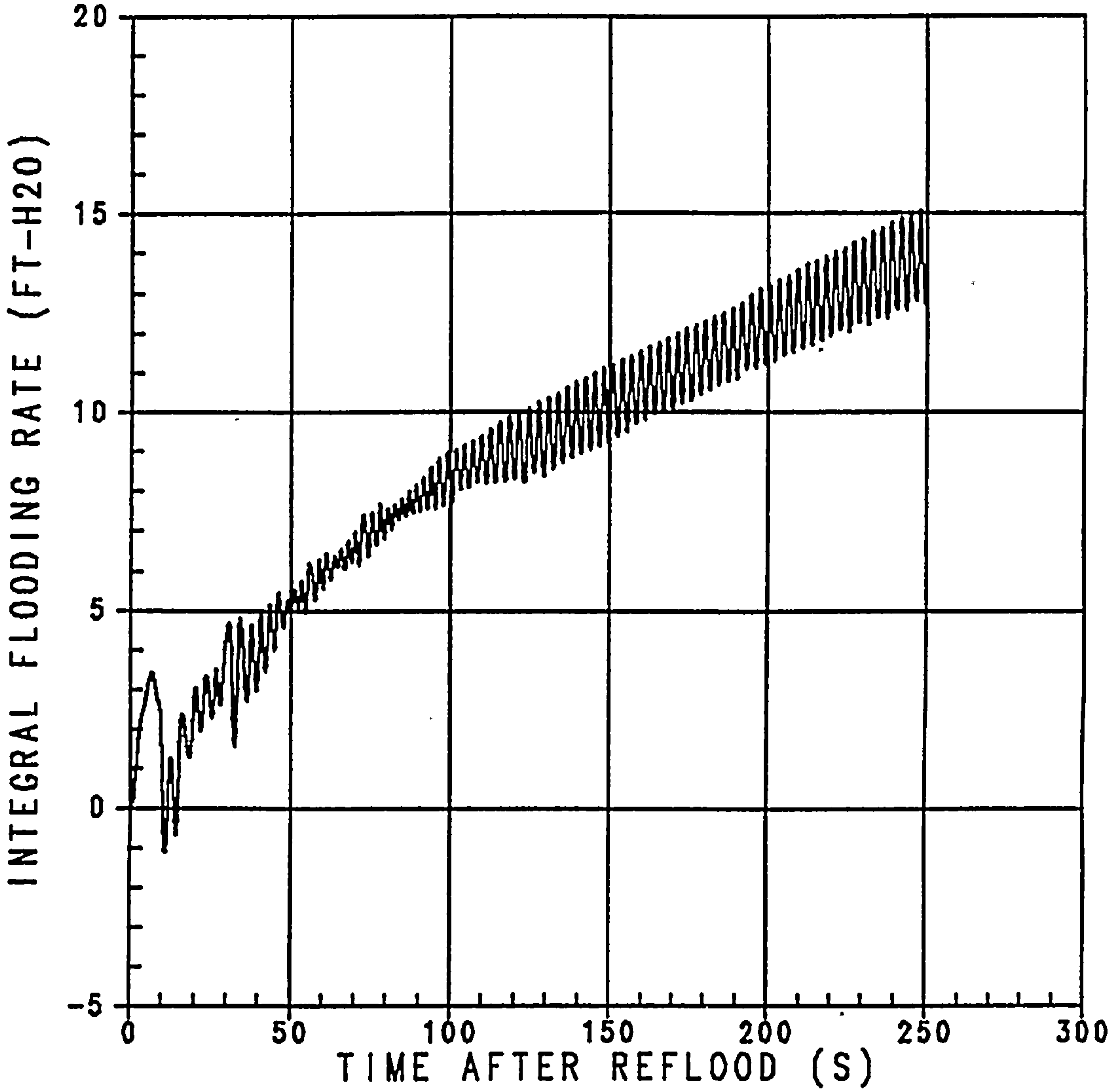

Figure 3.1-9e Integral of Core Inlet Flow
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

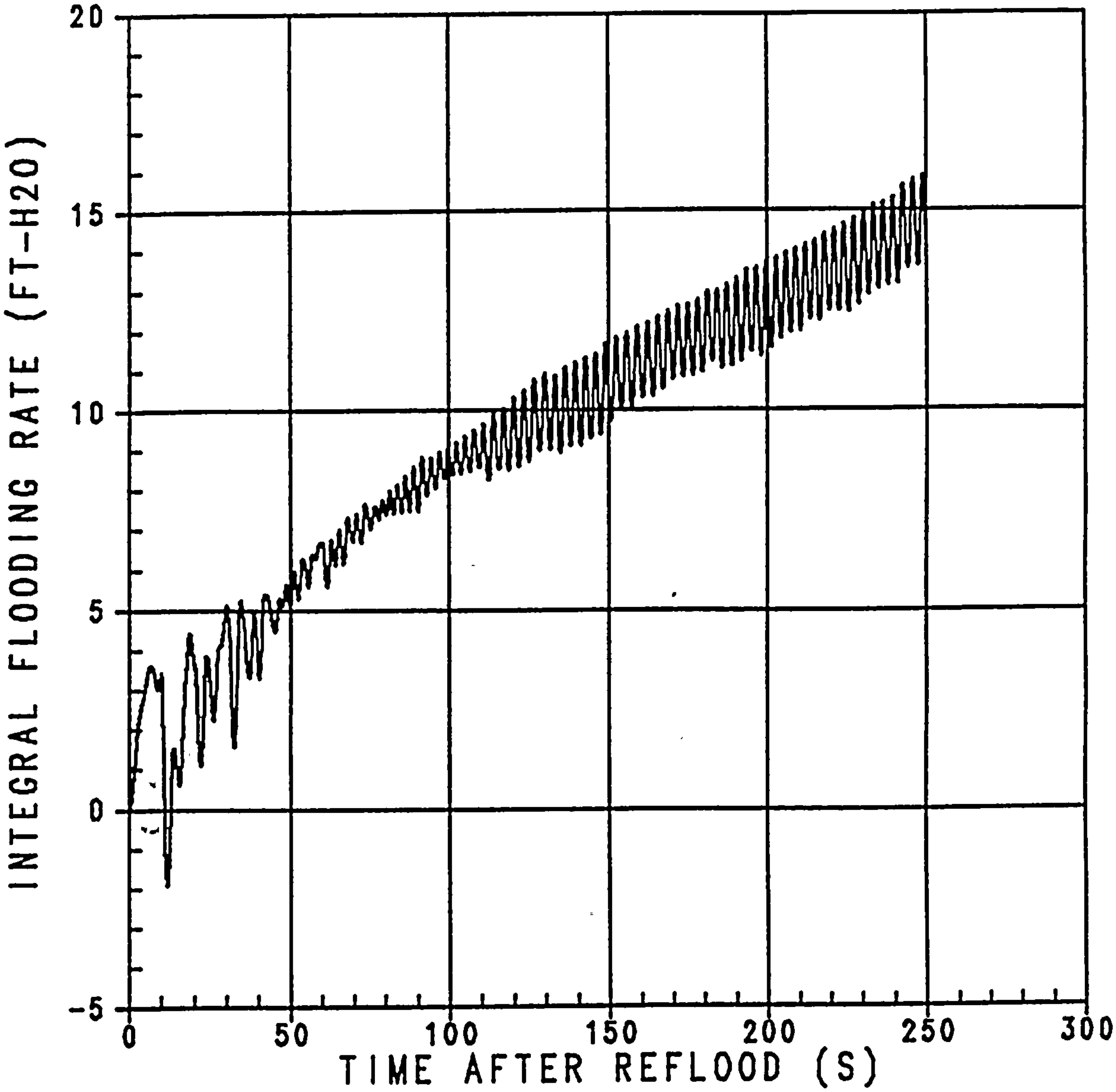

Figure 3.1-9f Integral of Core Inlet Flow
Case F, CD=0.4, Thot=609.1°F, P=2100 psia, max SI
Donald C. Cook Unit 1

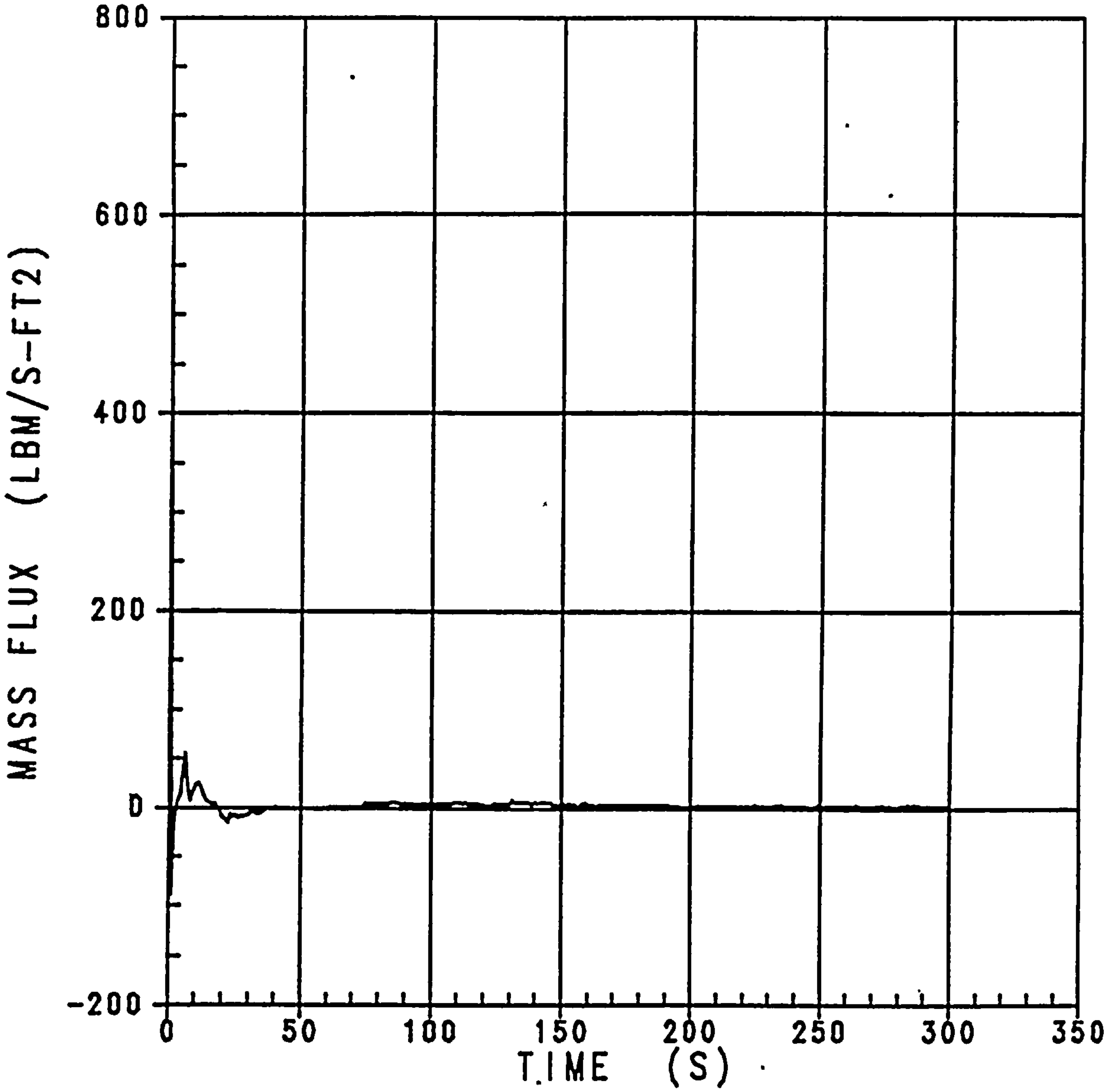

Figure 3.1-10a Mass Flux at Peak Temperature Elevation
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

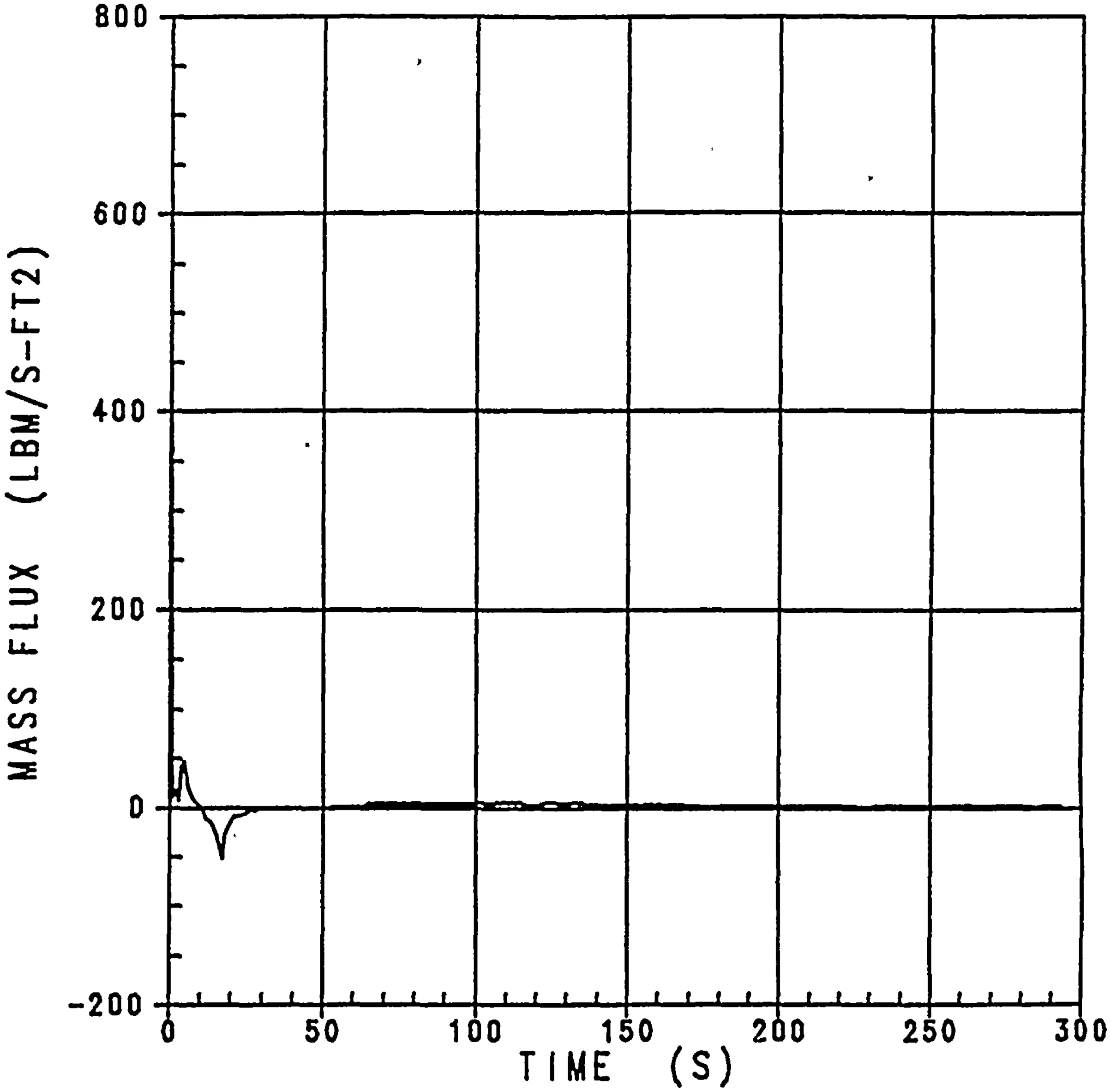

Figure 3.1-10b Mass Flux at Peak Temperature Elevation
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

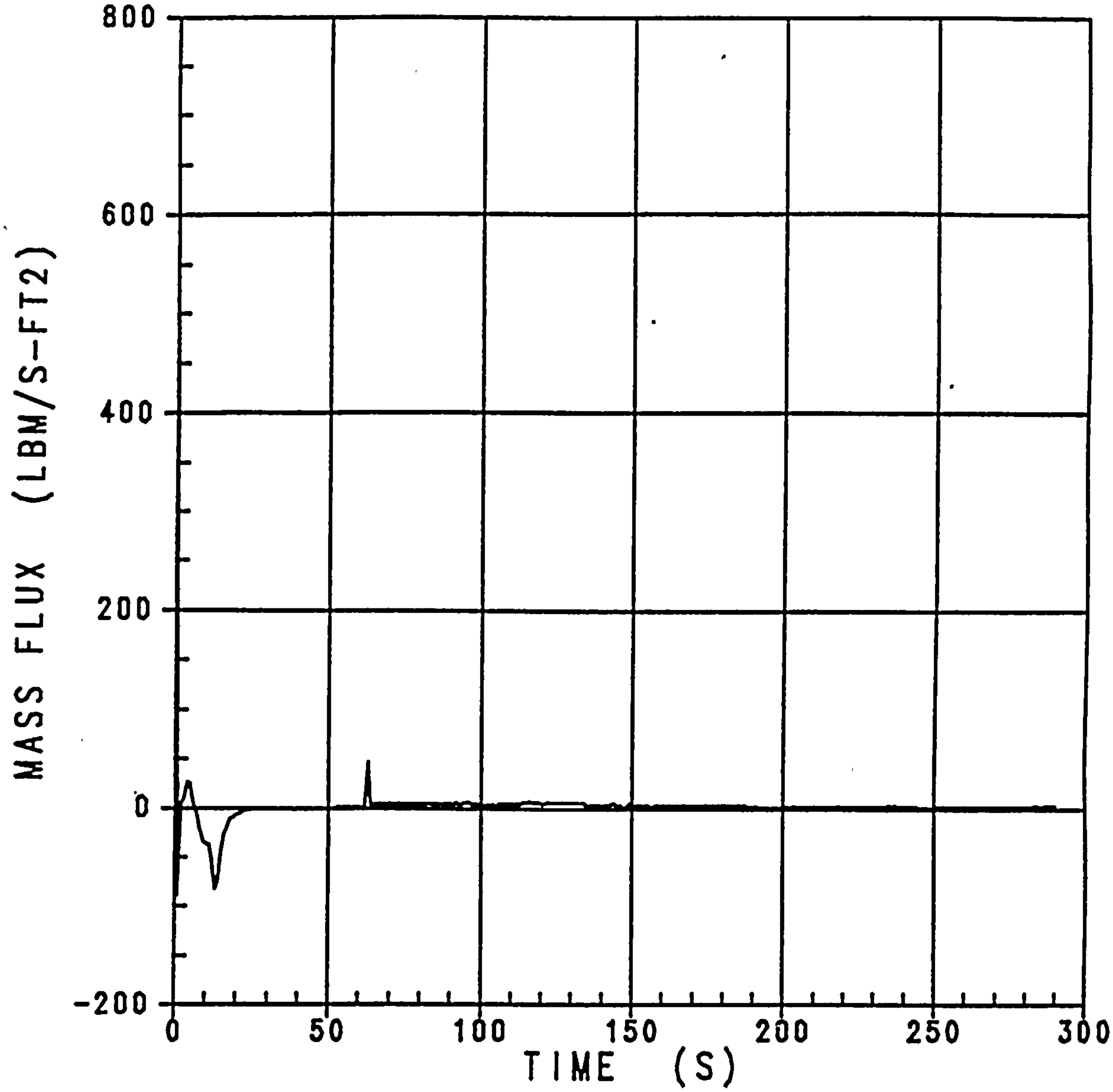

Figure 3.1-10c Mass Flux at Peak Temperature Elevation
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

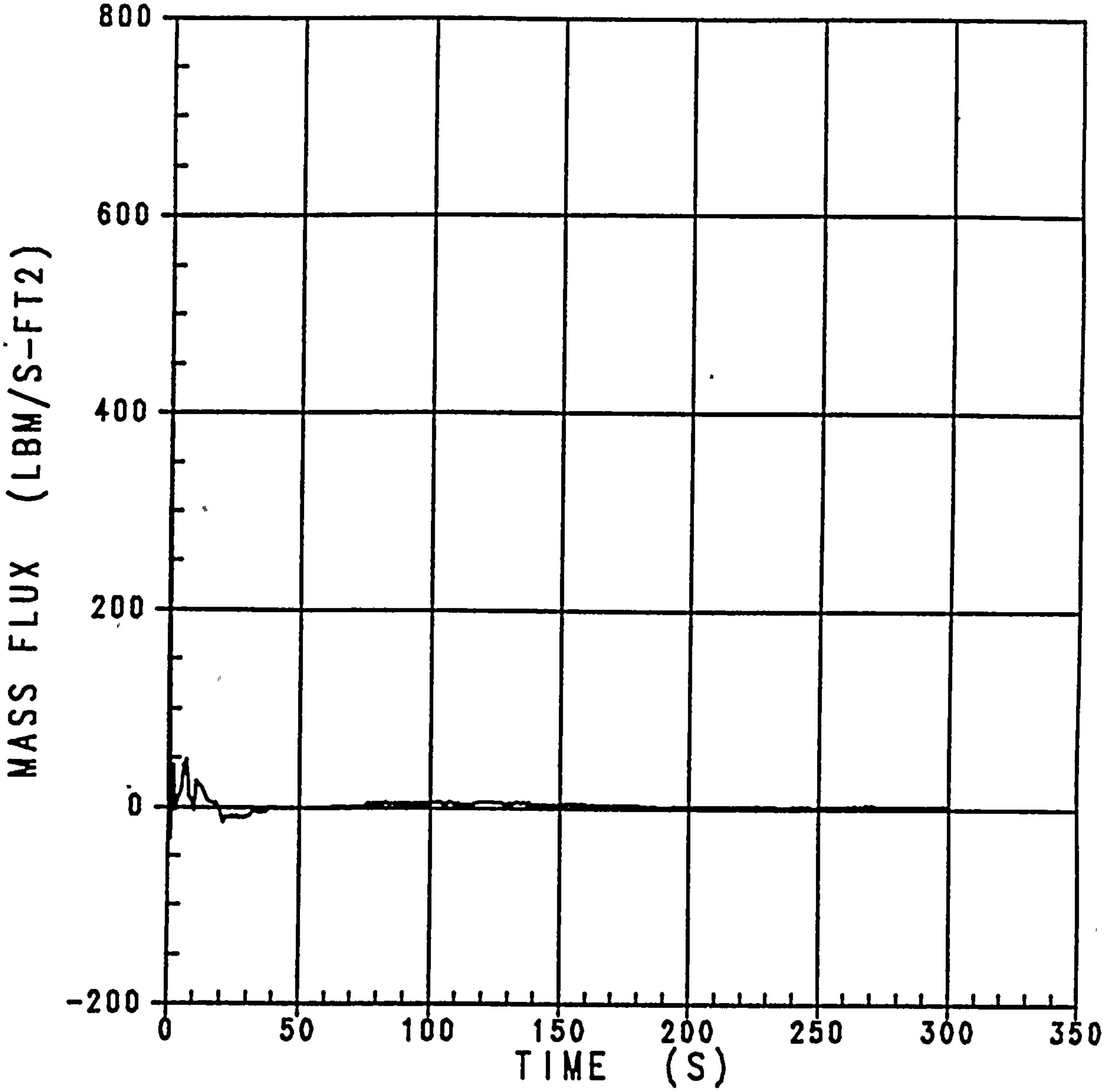

Figure 3.1-10d Mass Flux at Peak Temperature Elevation
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

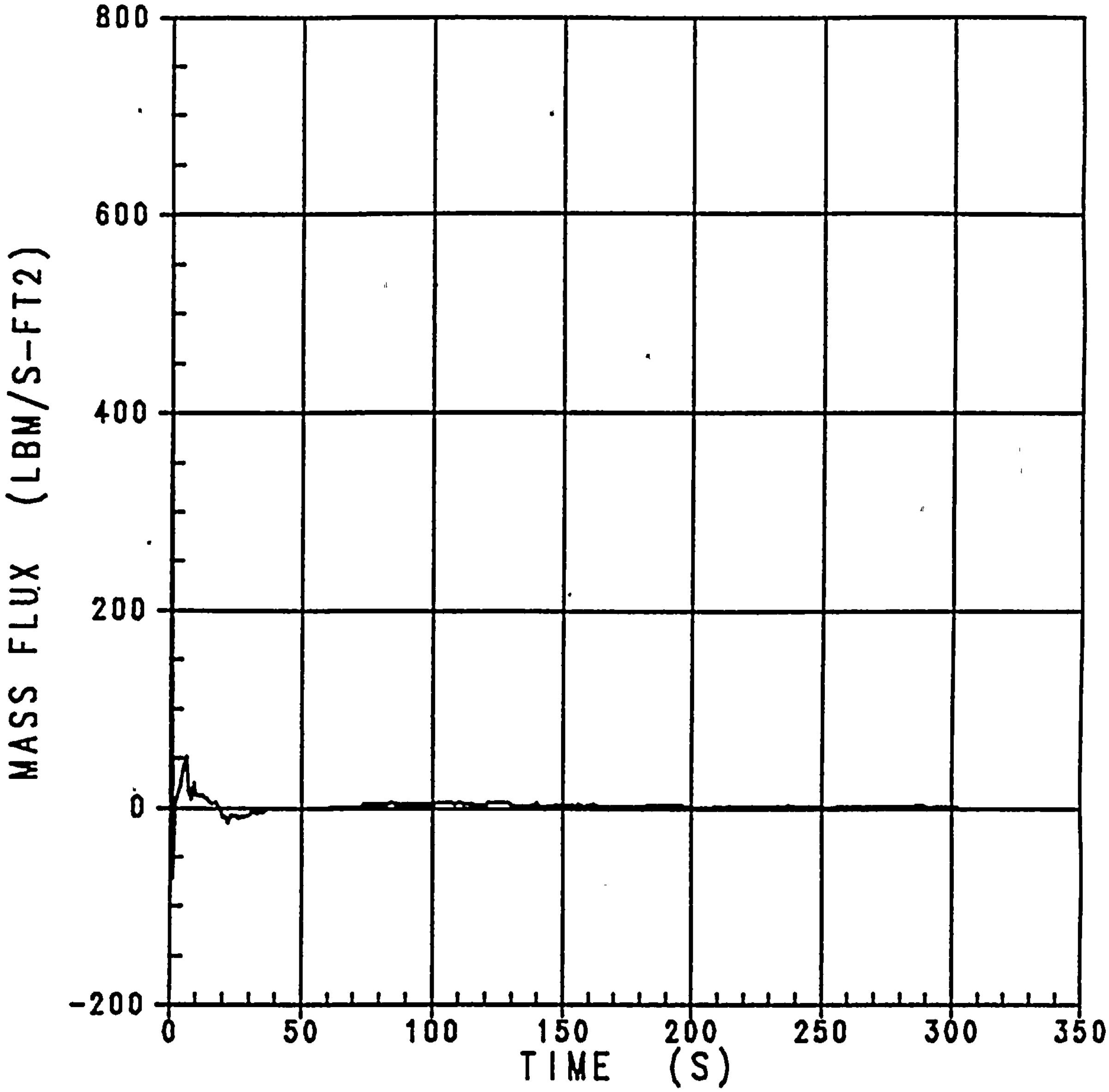

Figure 3.1-10e Mass Flux at Peak Temperature Elevation
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

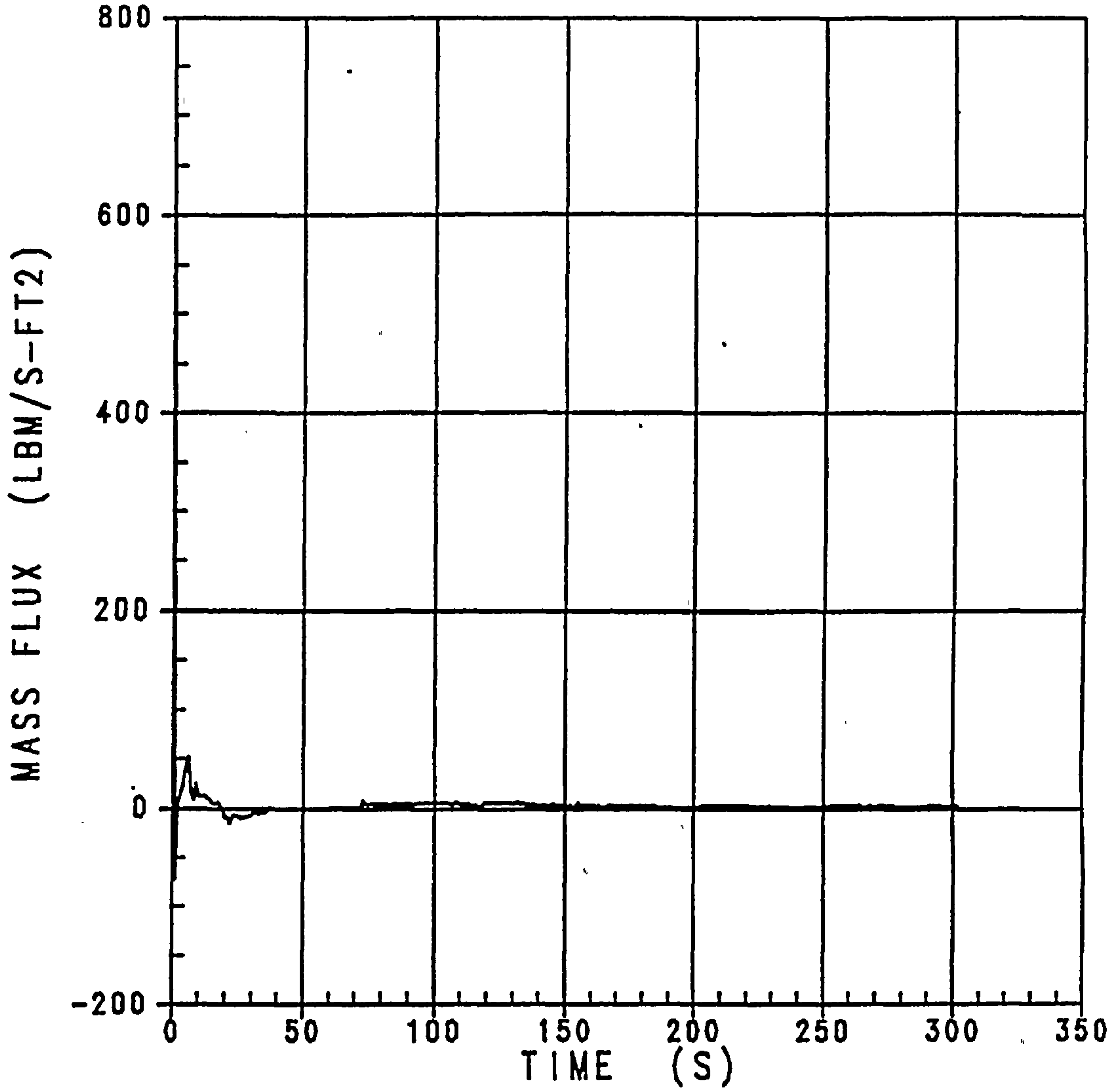

Figure 3.1-10f Mass Flux at Peak Temperature Elevation
Case F, CD=0.4, Thot=609.1°F, P=2100 psia, max SI
Donald C. Cook Unit 1

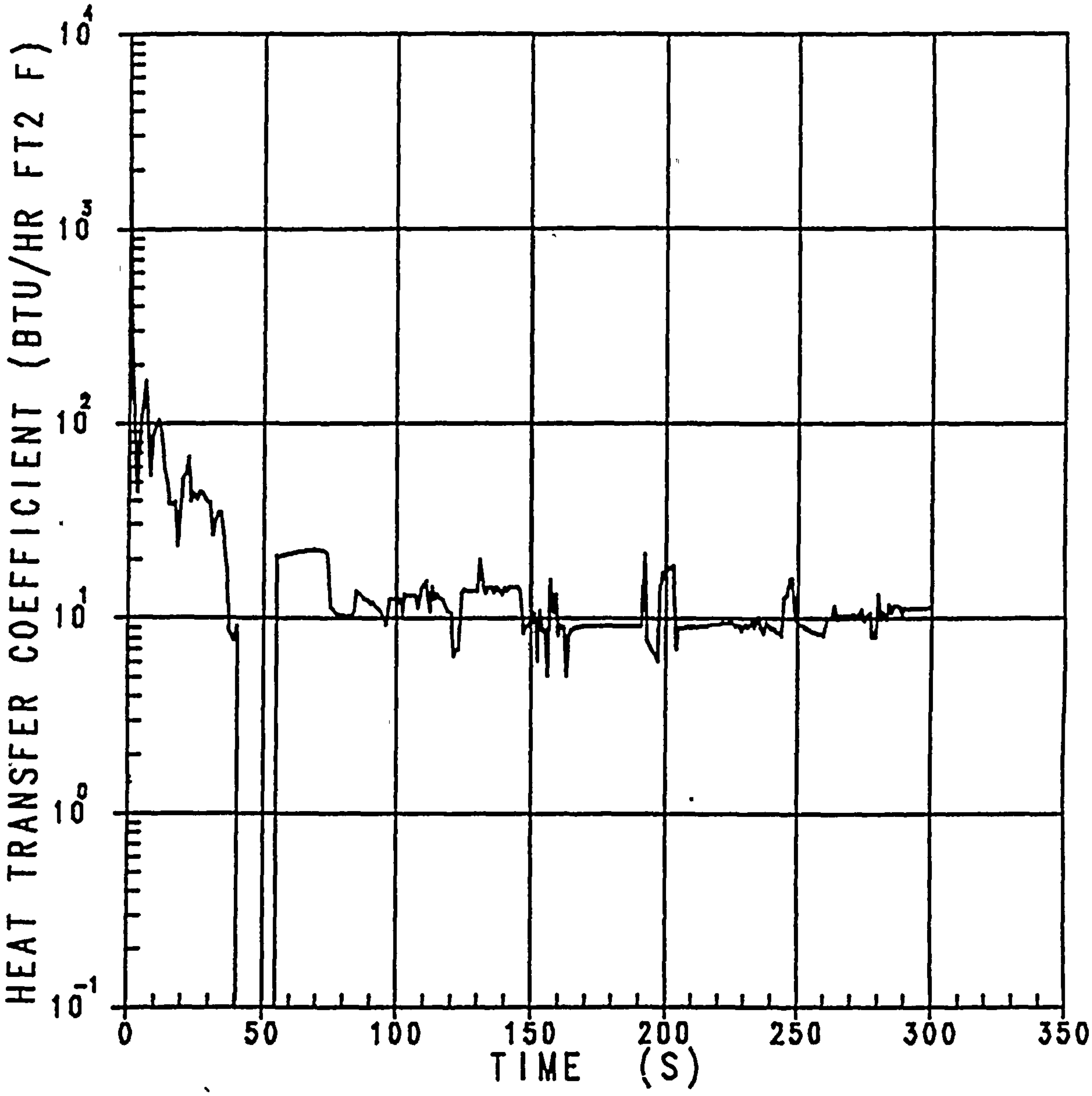

Figure 3.1-11a Rod H.T.C. at Peak Temperature Elevation
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

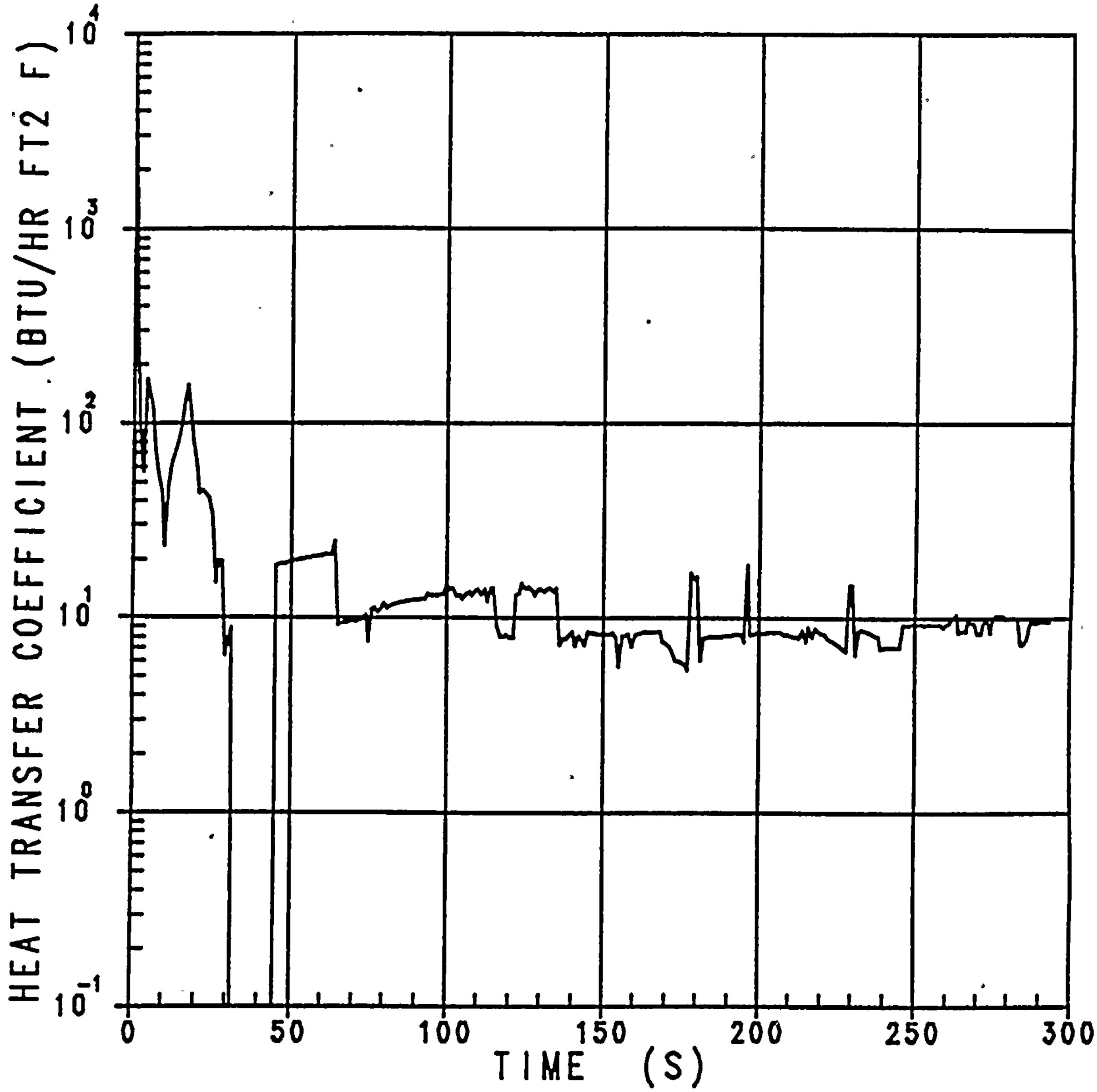

Figure 3.1-11b Rod H.T.C. at Peak Temperature Elevation
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

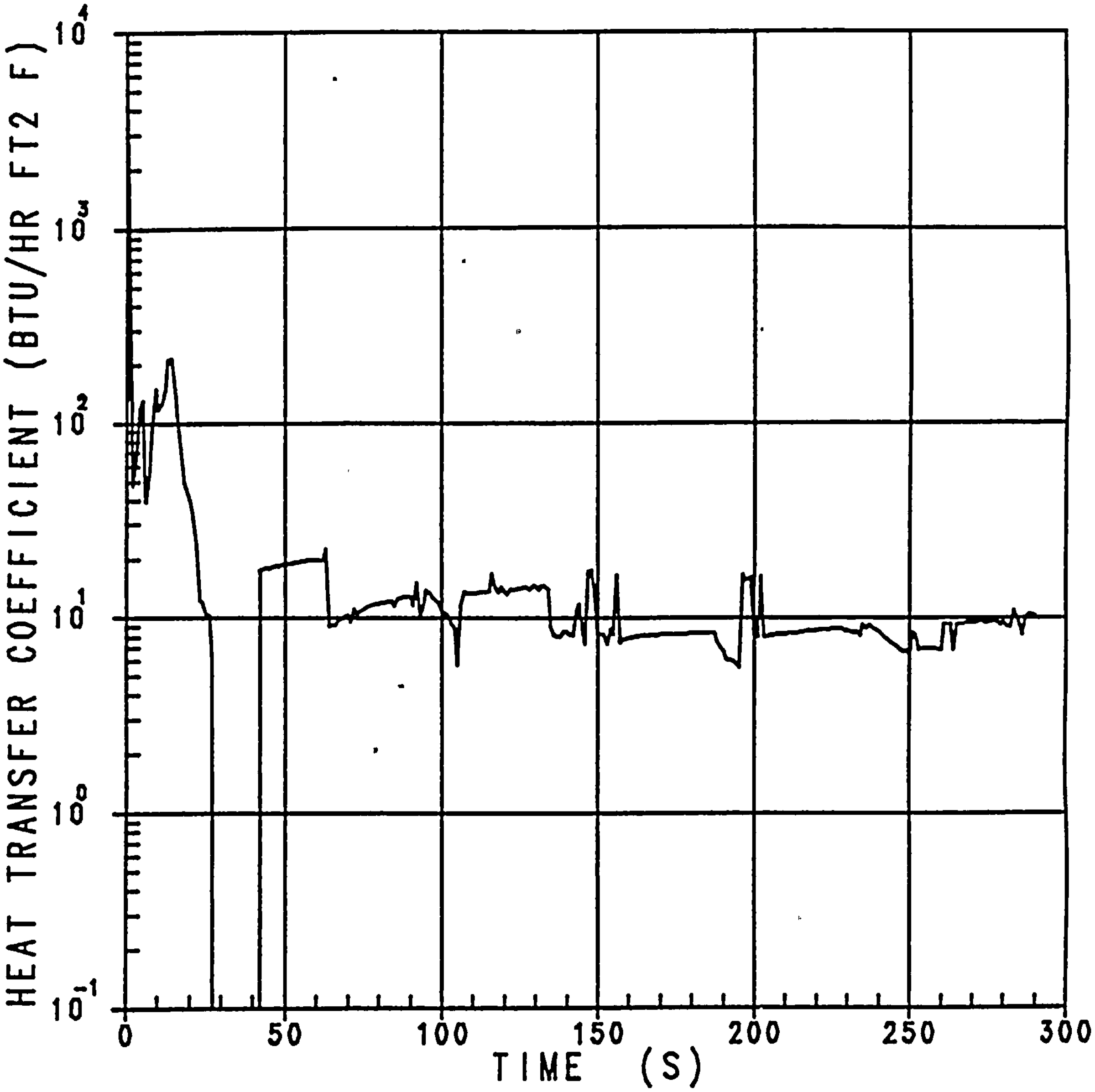

Figure 3.1-11c Rod H.T.C. at Peak Temperature Elevation
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

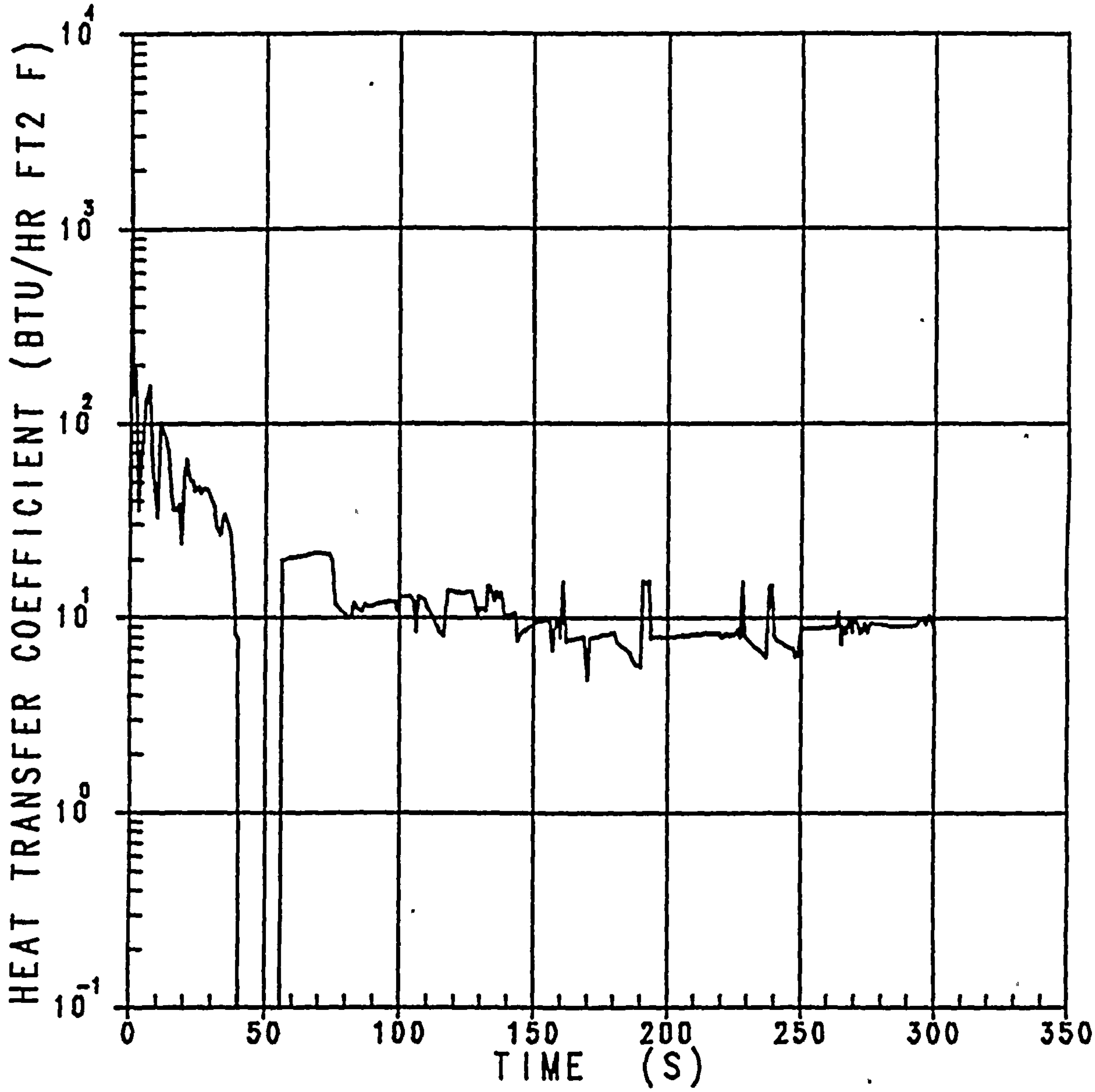

Figure 3.1-11d Rod H.T.C. at Peak Temperature Elevation
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

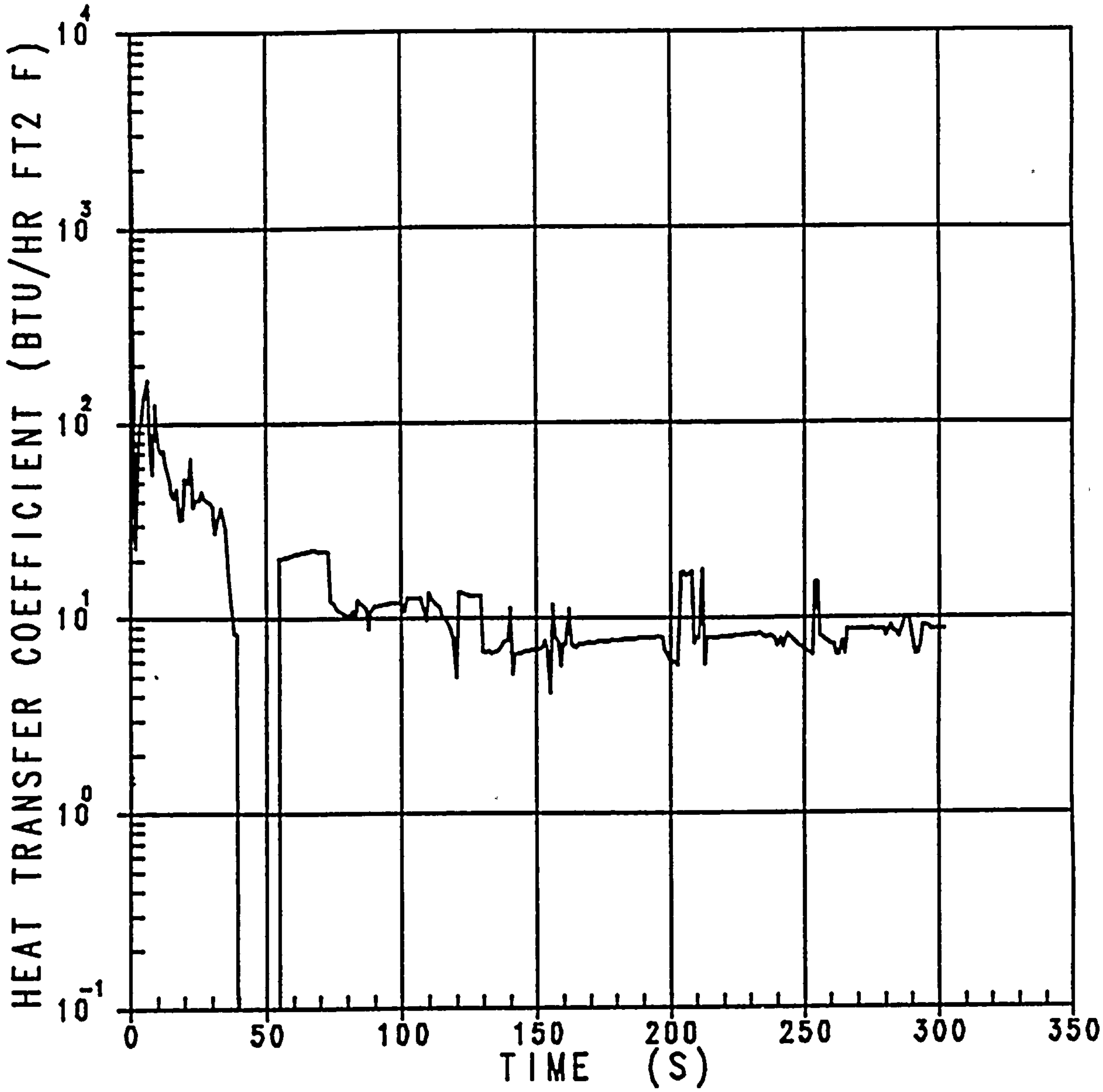

Figure 3.1-11e Rod H.T.C. at Peak Temperature Elevation
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

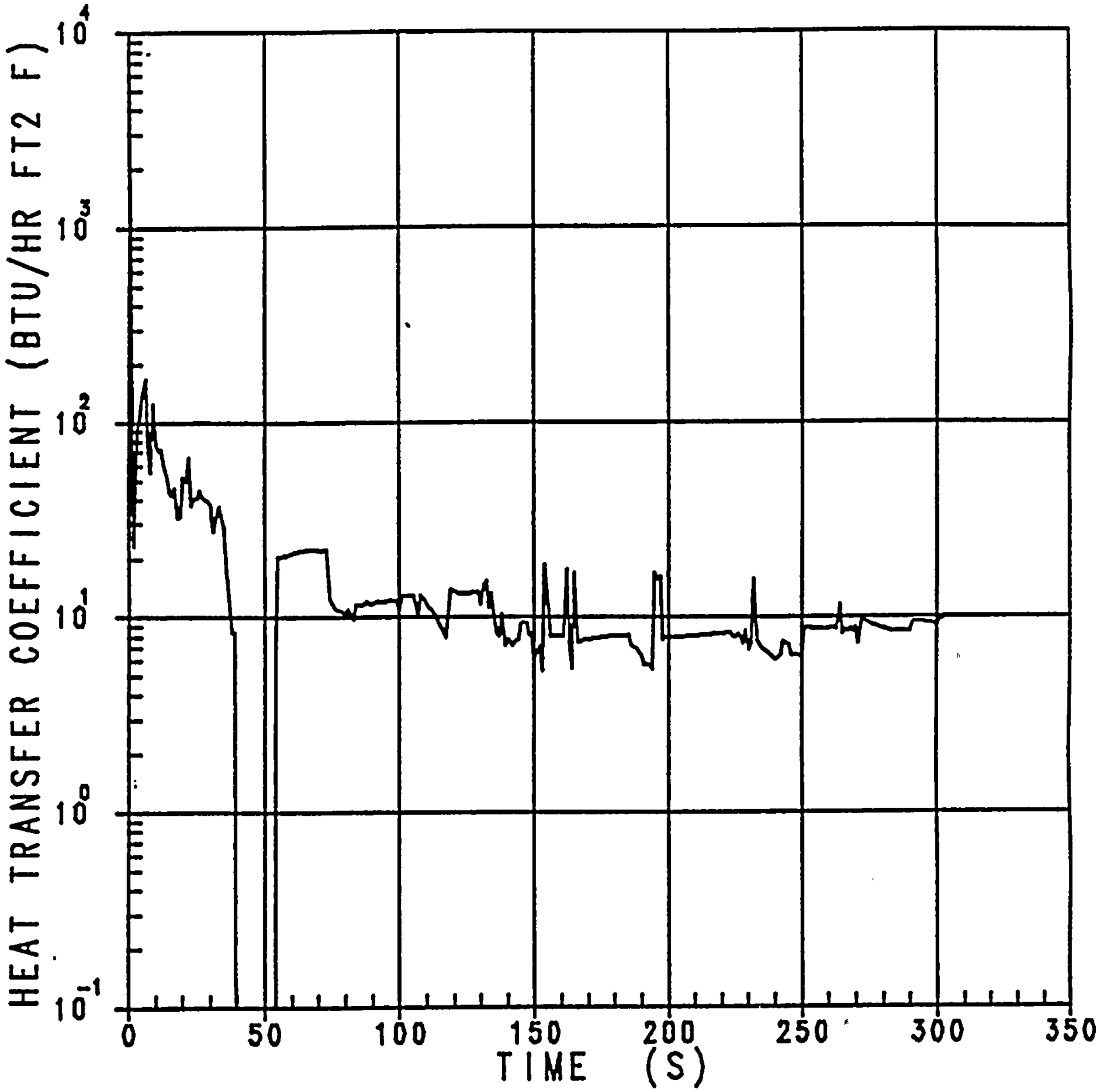

Figure 3.1-11f Rod H.T.C. at Peak Temperature Elevation
Case F, CD=0.4, Thot=609.1°F, P=2100 psia, max SI
Donald C. Cook Unit 1

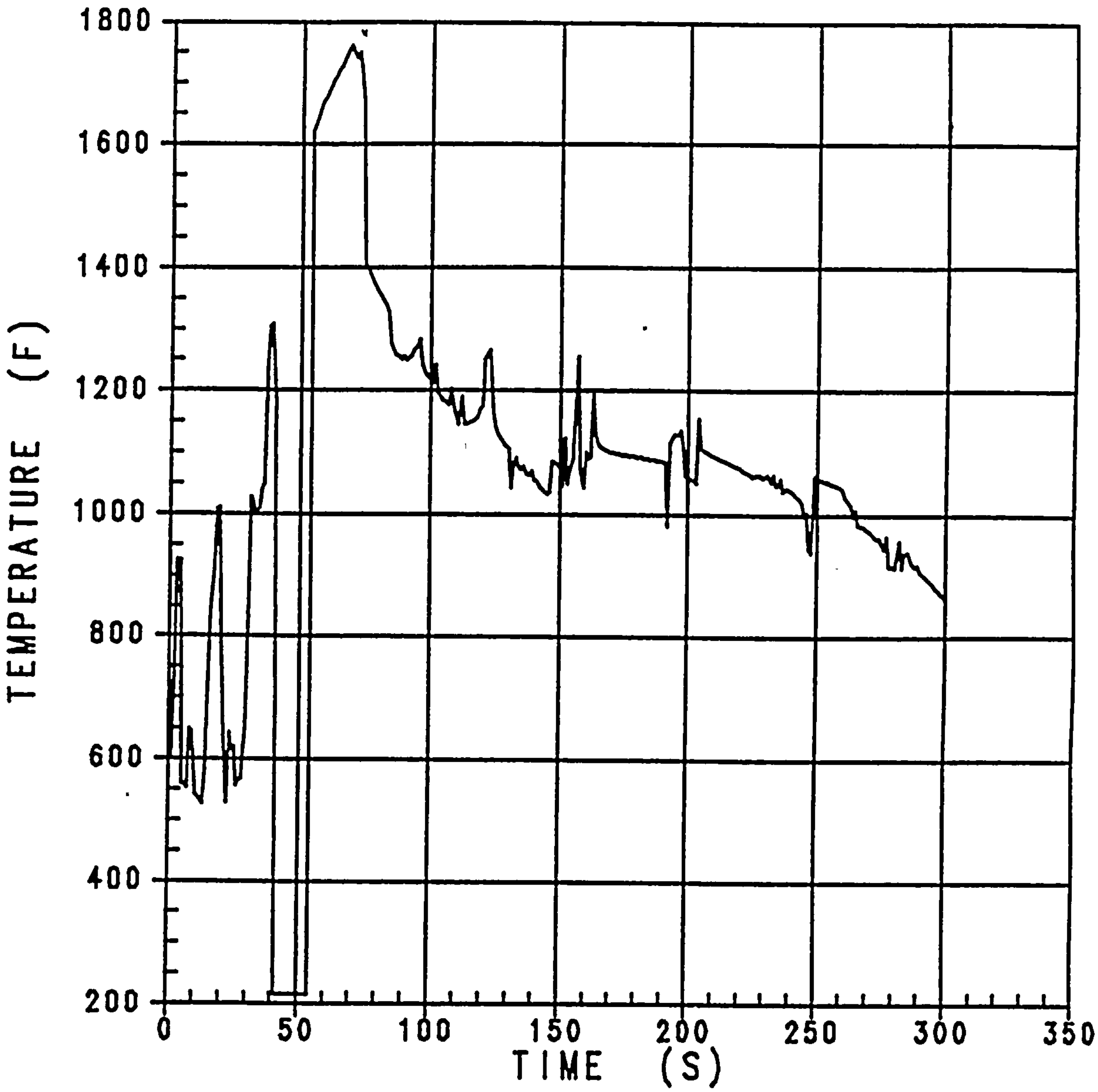

Figure 3.1-12a Vapor Temperature
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

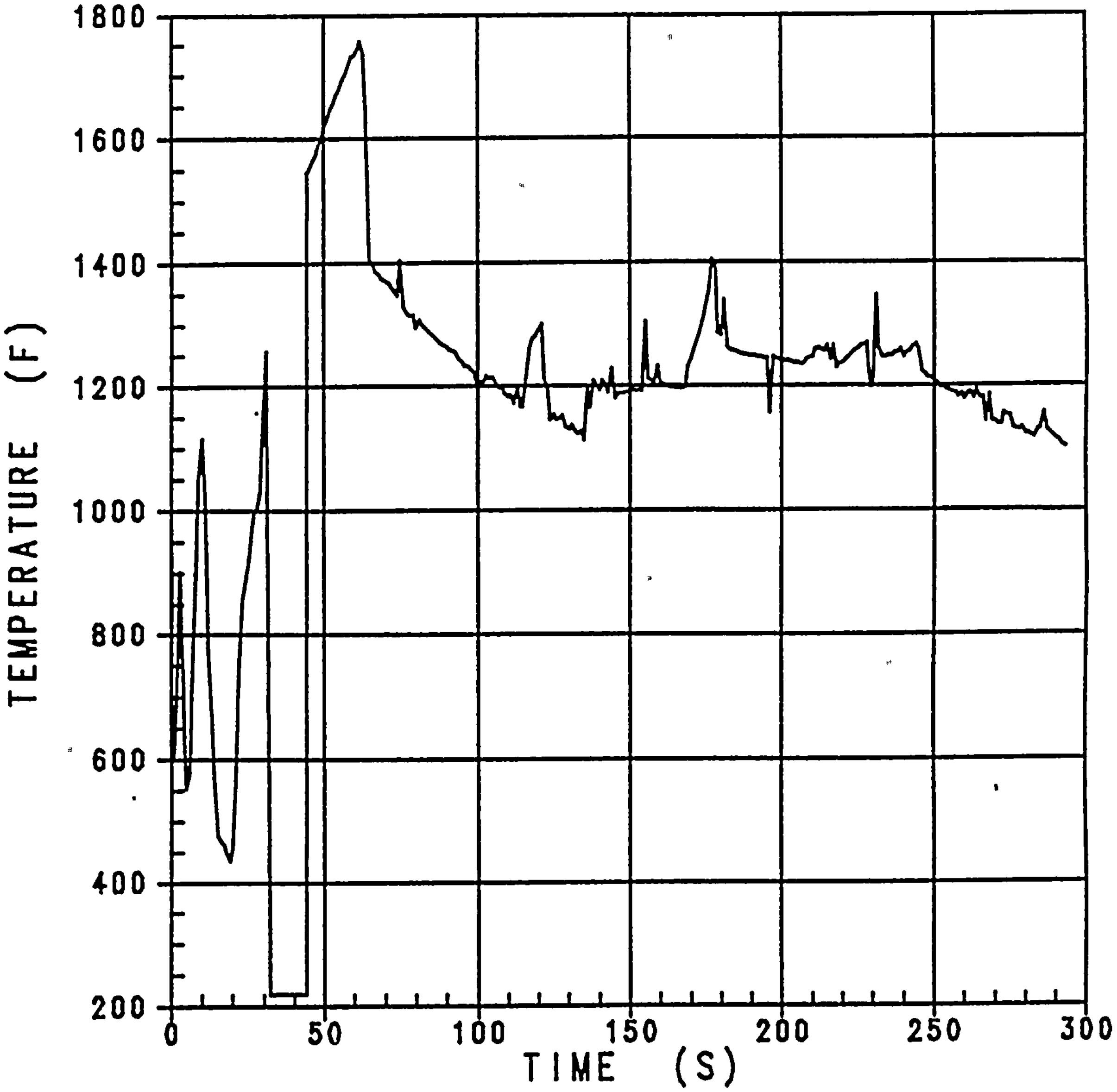

Figure 3.1-12b Vapor Temperature
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

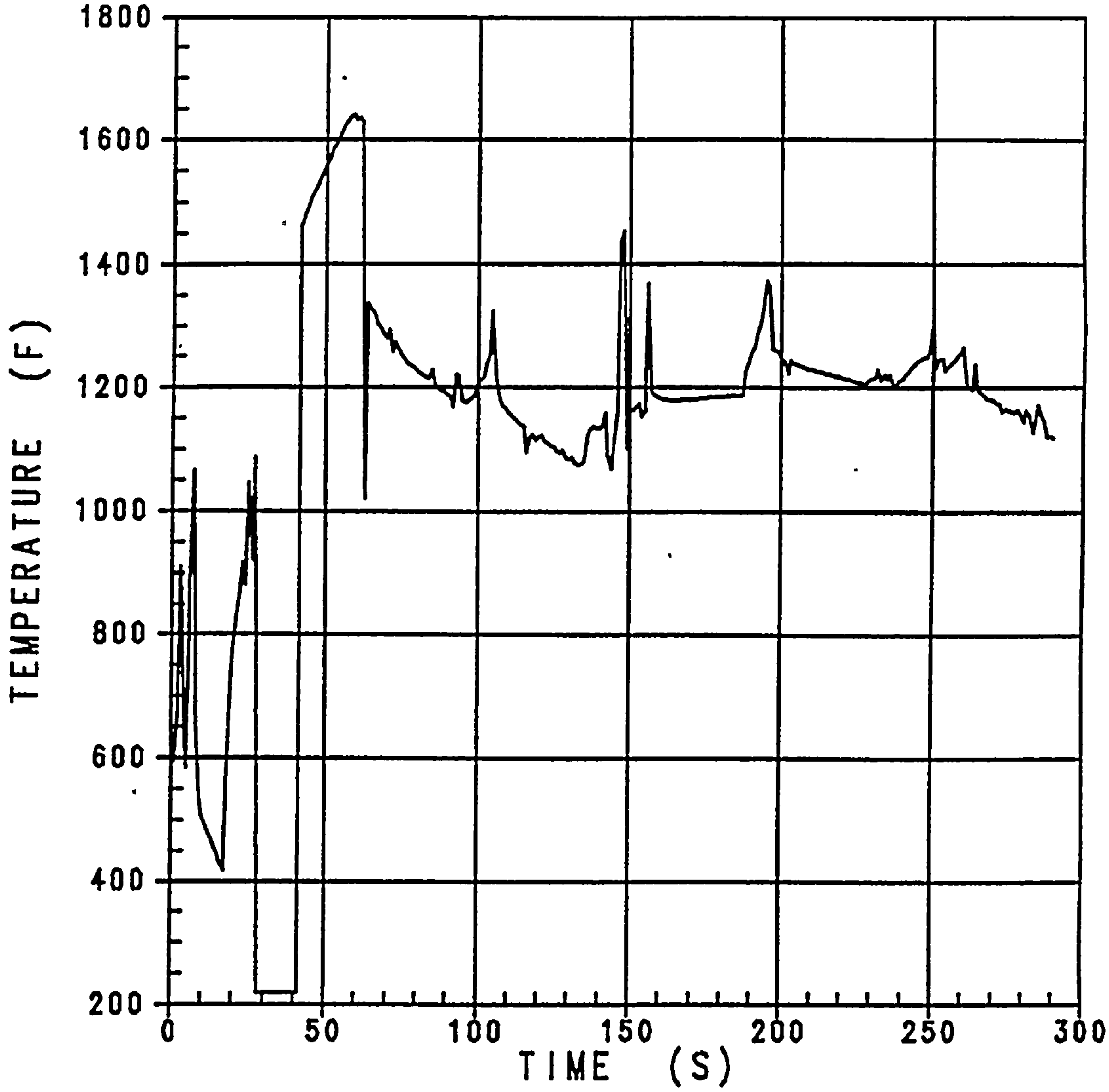

Figure 3.1-12c Vapor Temperature
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

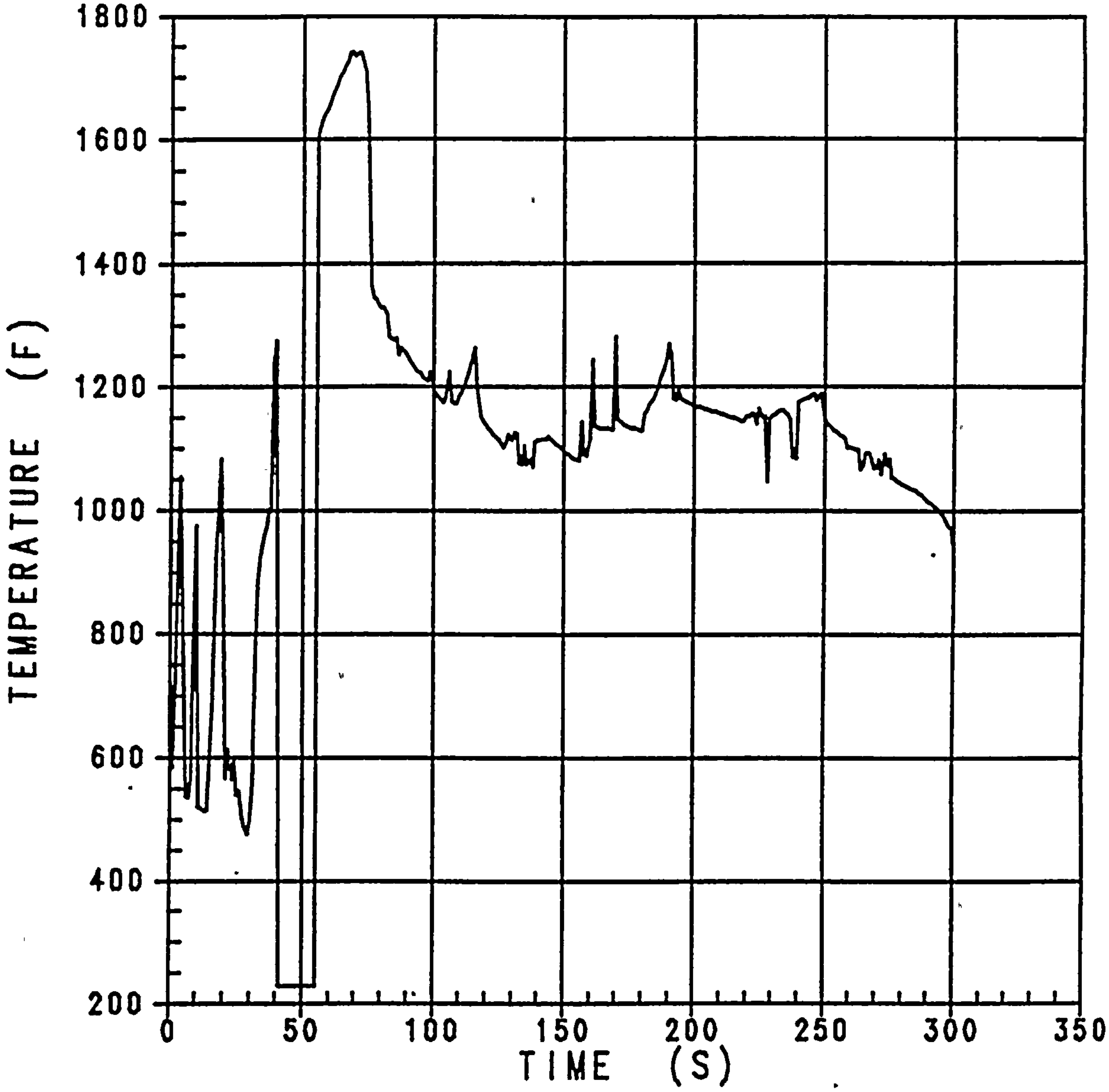

Figure 3.1-12d Vapor Temperature
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

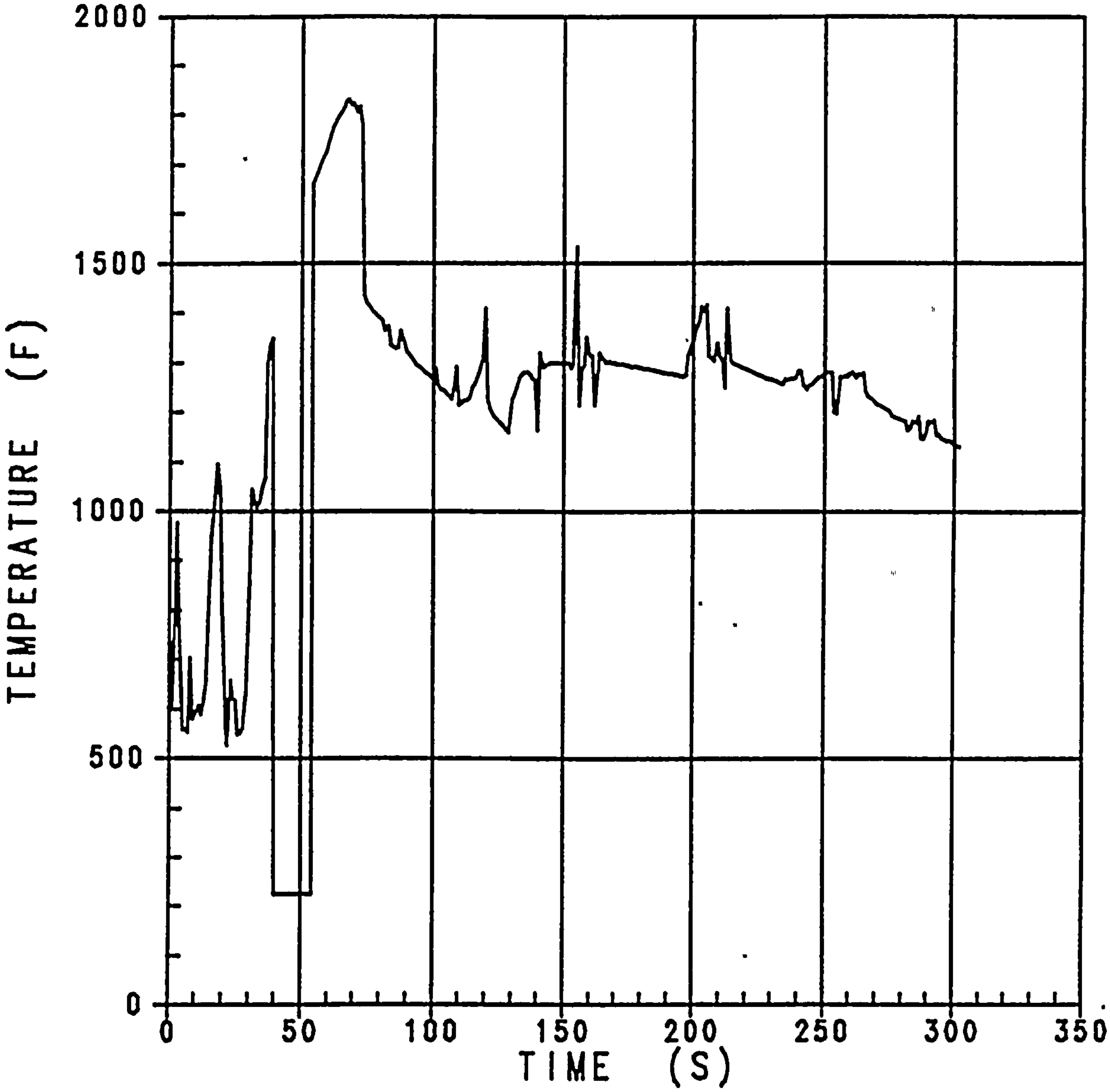

Figure 3.1-12e Vapor Temperature
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

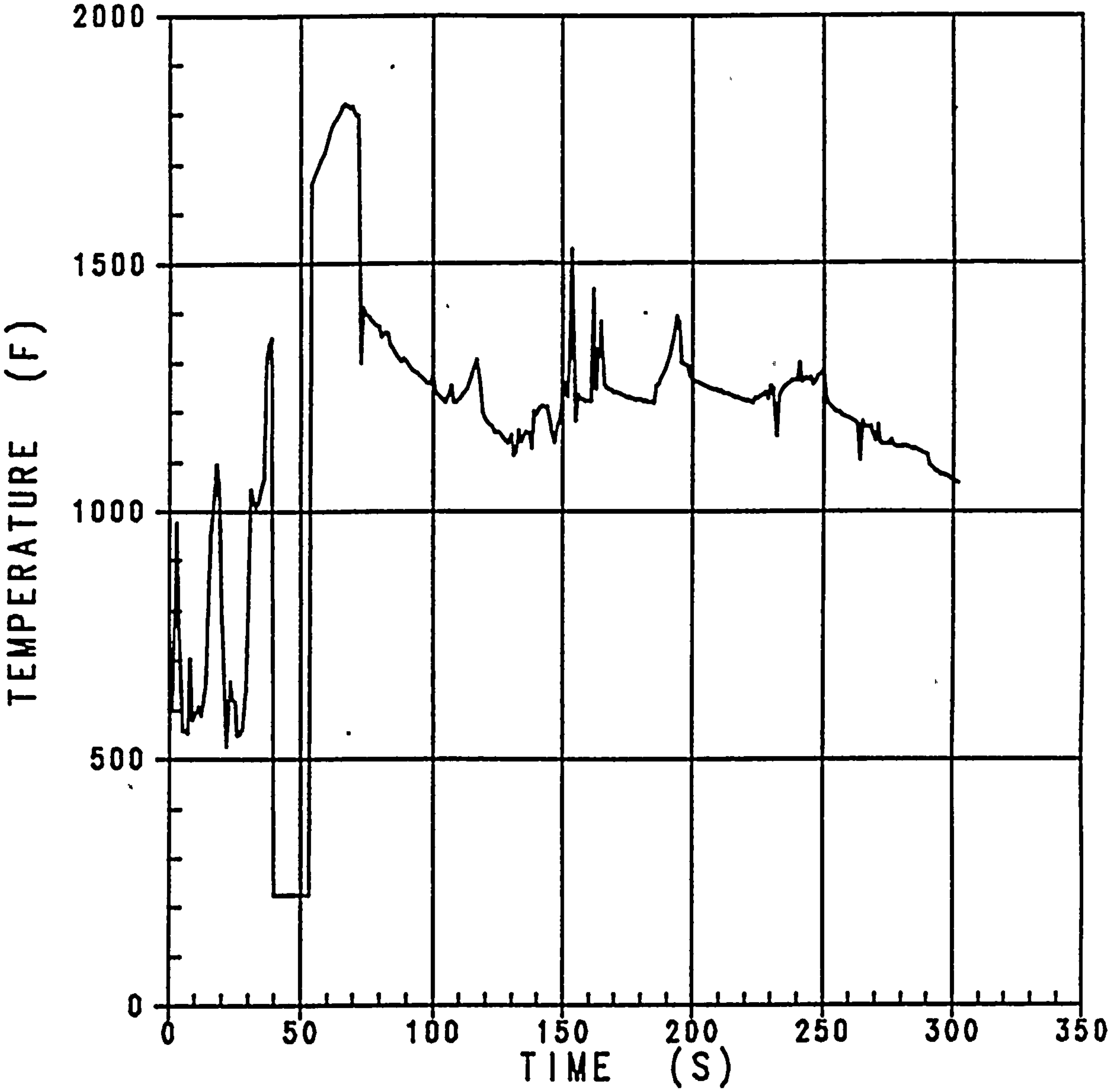

Figure 3.1-12f Vapor Temperature
Case F, CD=0.4, Thot=609.1°F, P=2100 psia, max SI
Donald C. Cook Unit 1

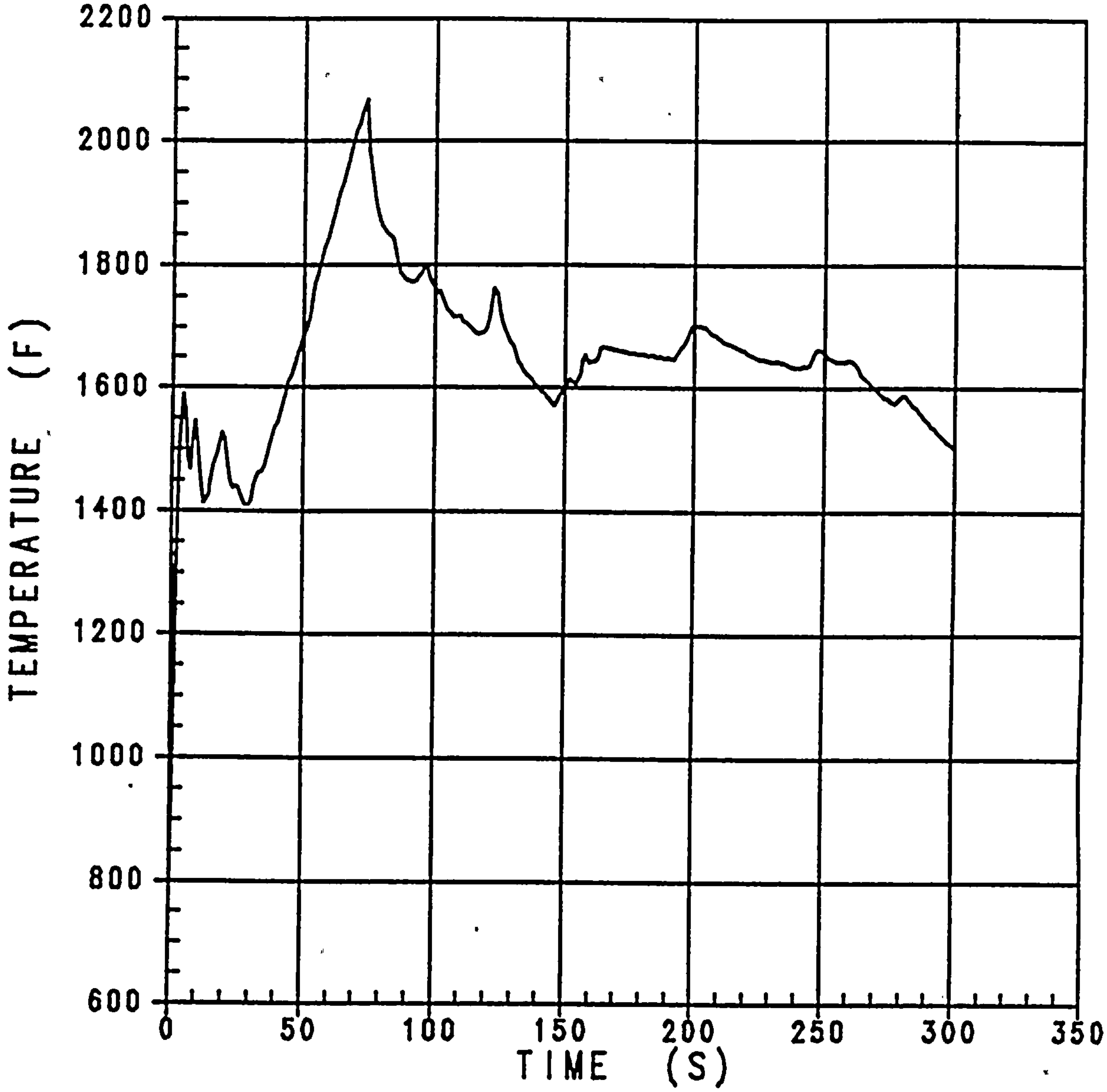

Figure 3.1-13a Fuel Rod Peak Clad Temperature
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

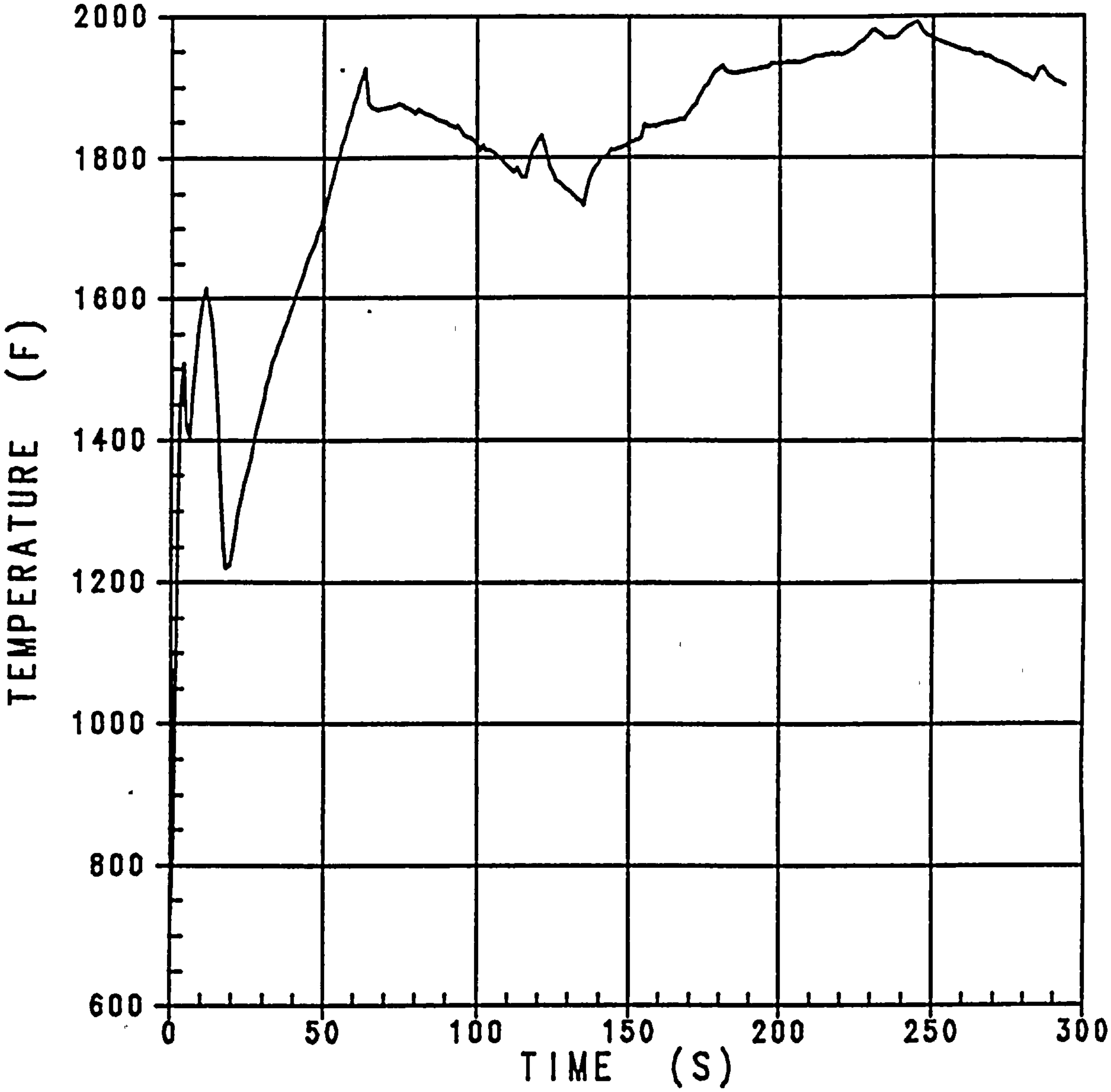

Figure 3.1-13b Fuel Rod Peak Clad Temperature
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

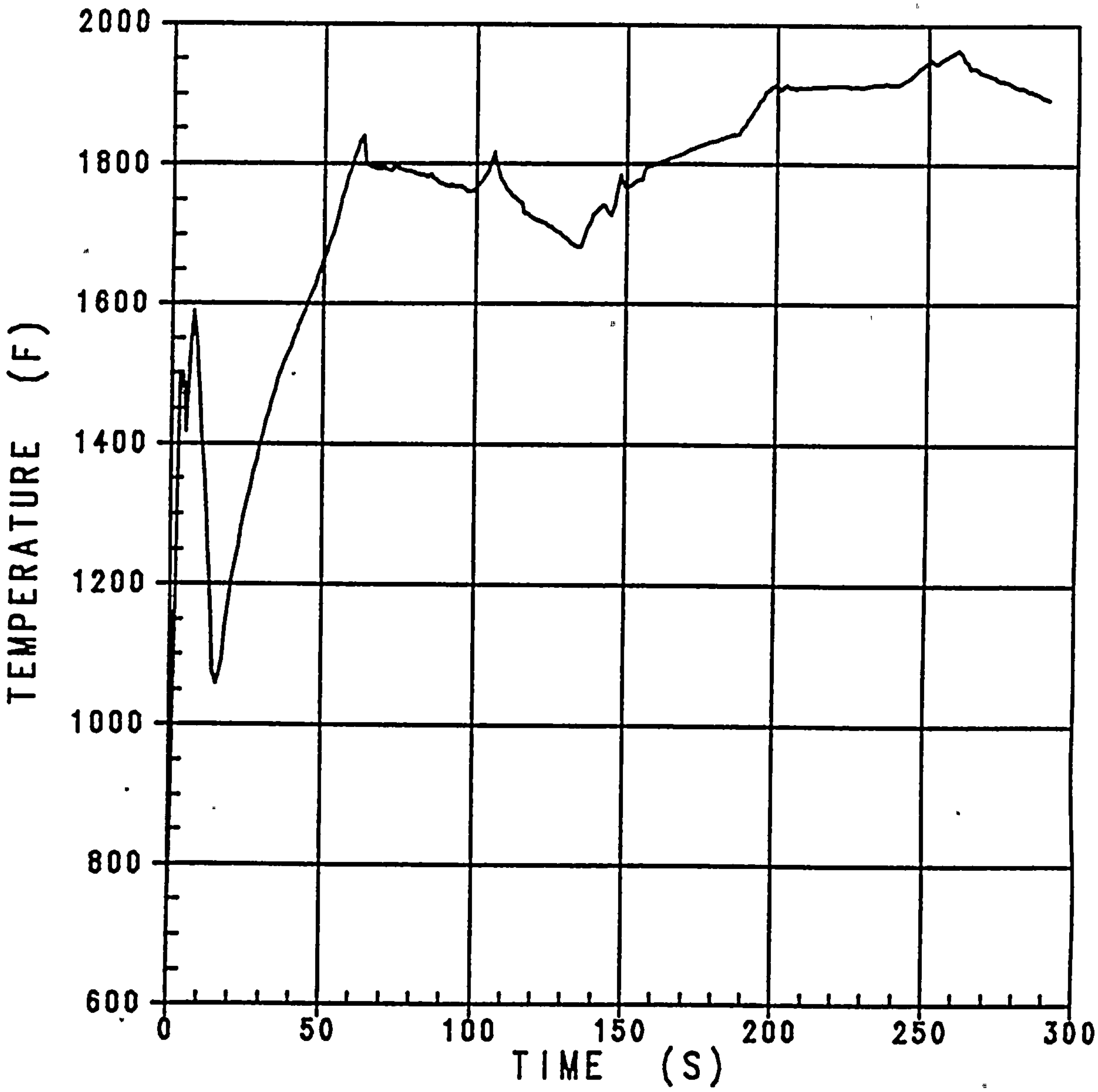

Figure 3.1-13c Fuel Rod Peak Clad Temperature
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

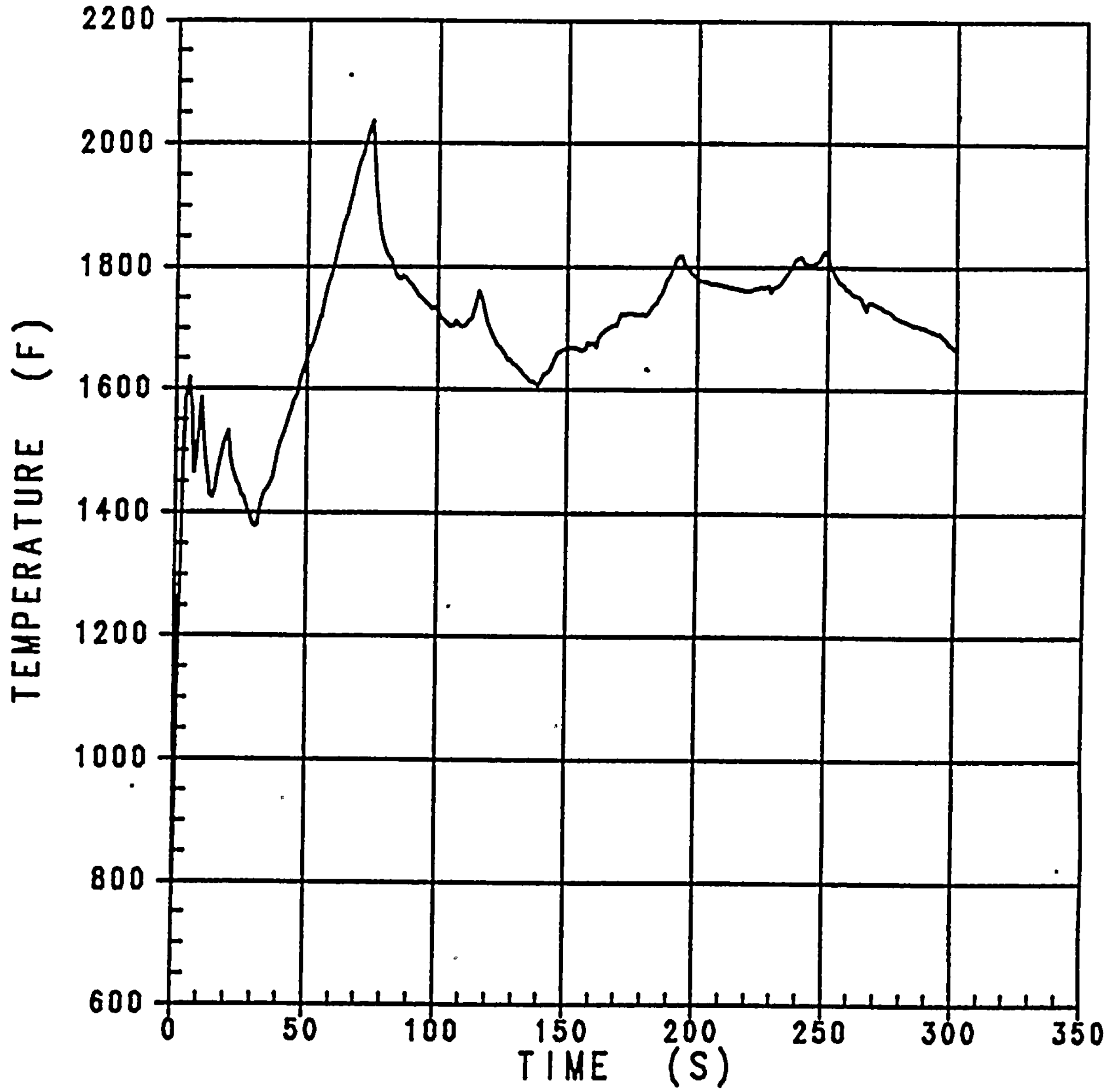

Figure 3.1-13d Fuel Rod Peak Clad Temperature
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

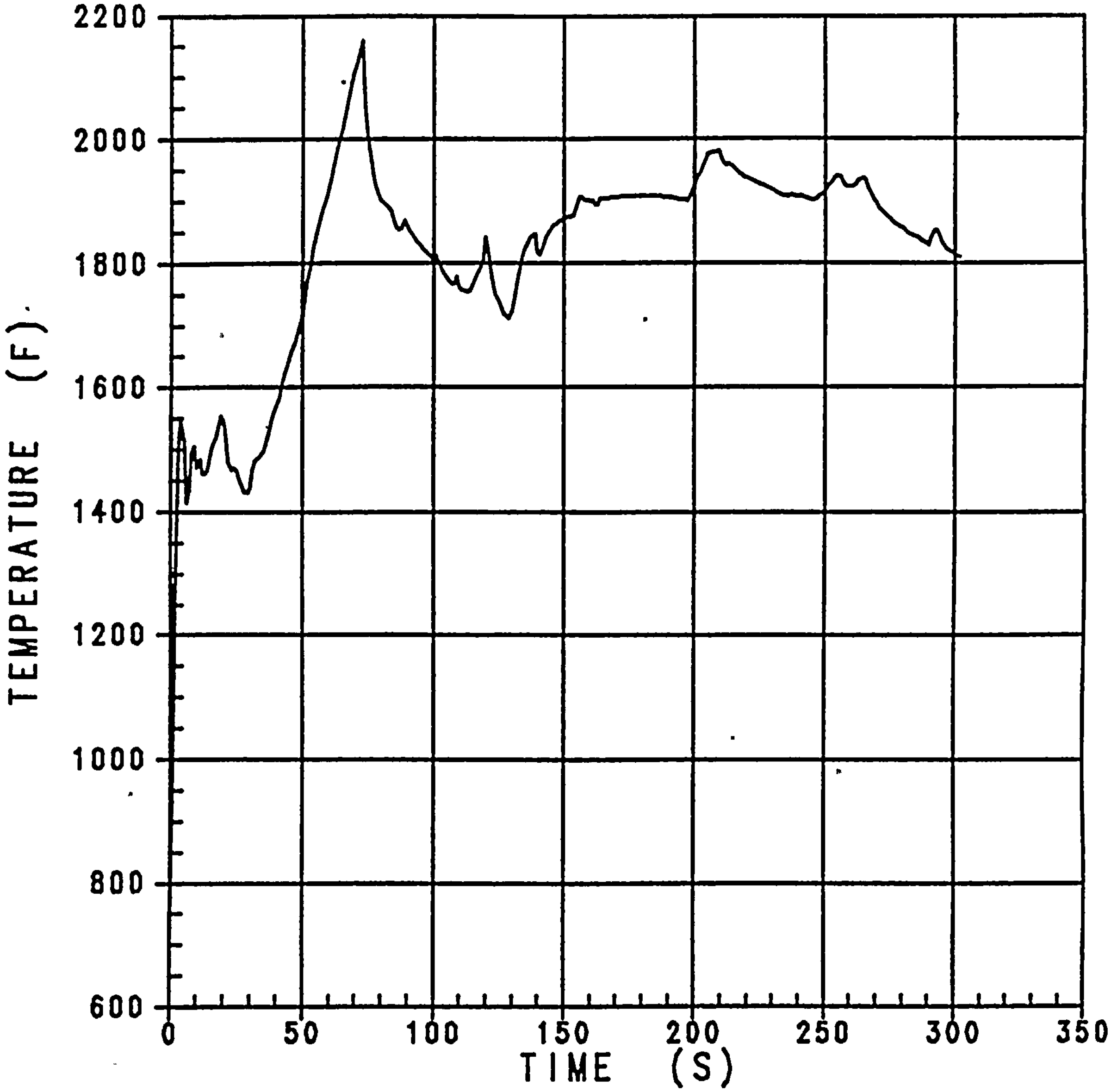

Figure 3.1-13e Fuel Rod Peak Clad Temperature
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

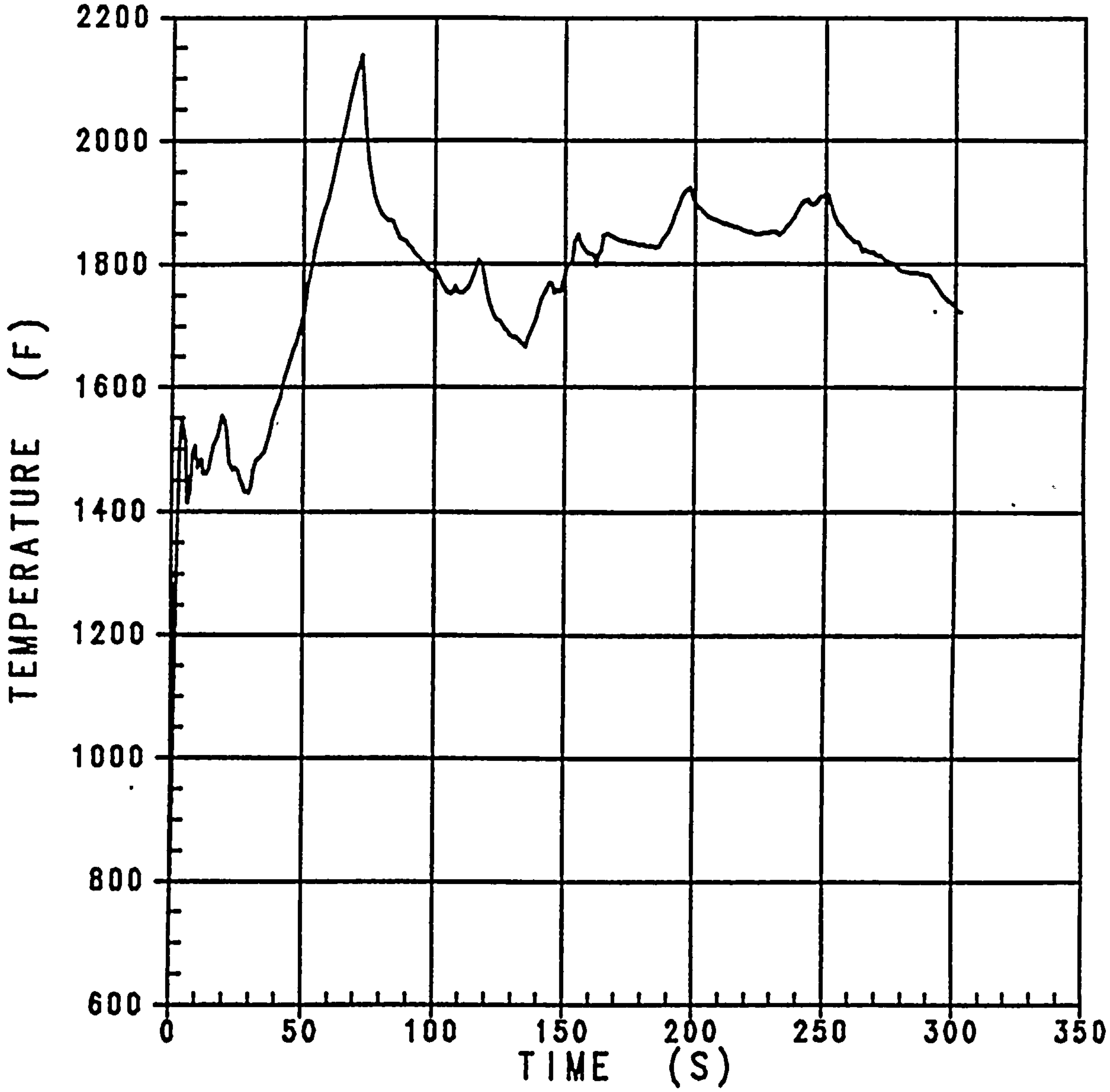

Figure 3.1-13f Fuel Rod Peak Clad Temperature
Case F, CD=0.4, Thot=609.1°F, P=2100 psia, max SI
Donald C. Cook Unit 1

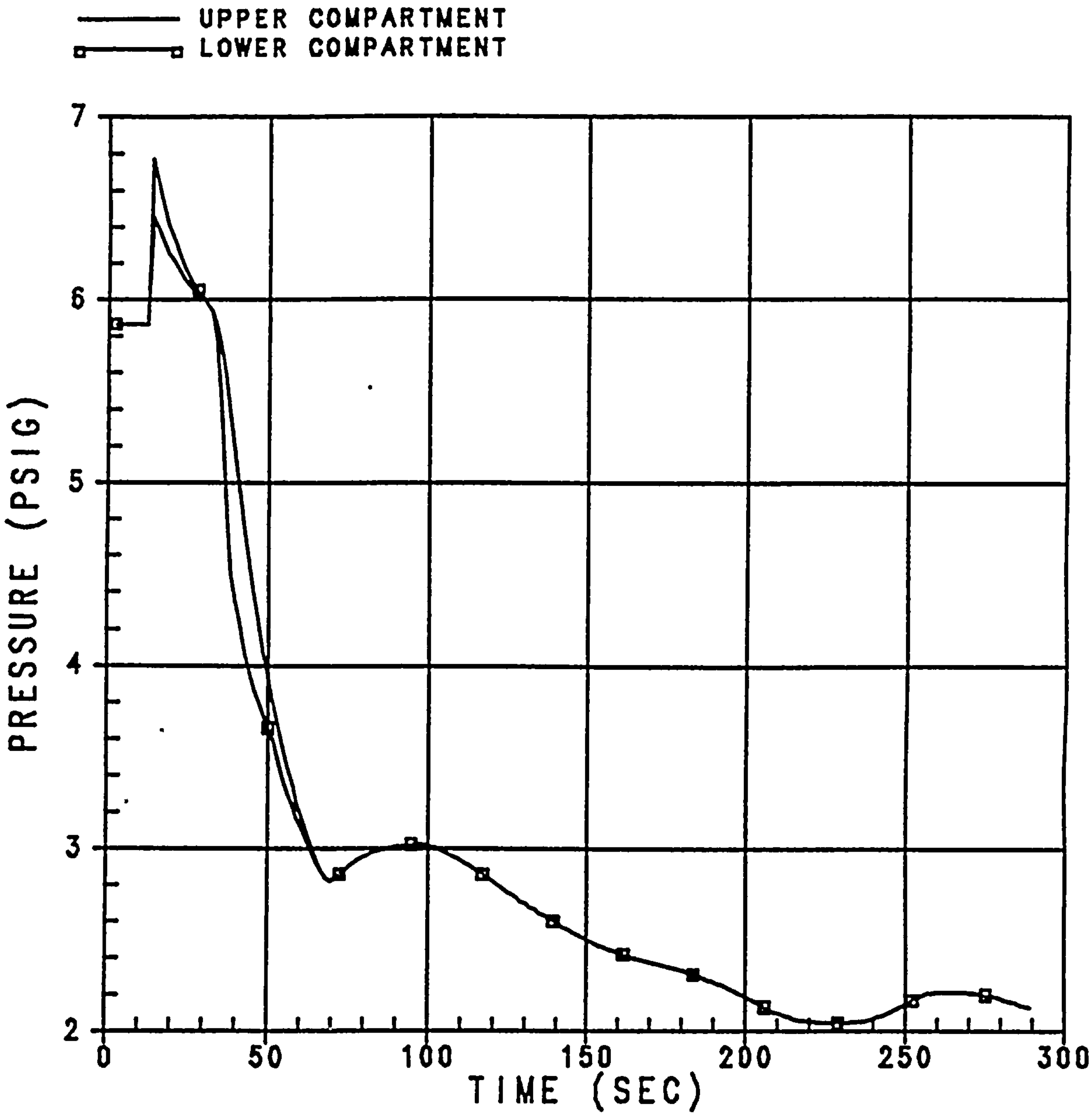

Figure 3.1-14 Containment Pressure
CD=0.4, Min SI
Donald C. Cook Unit 1

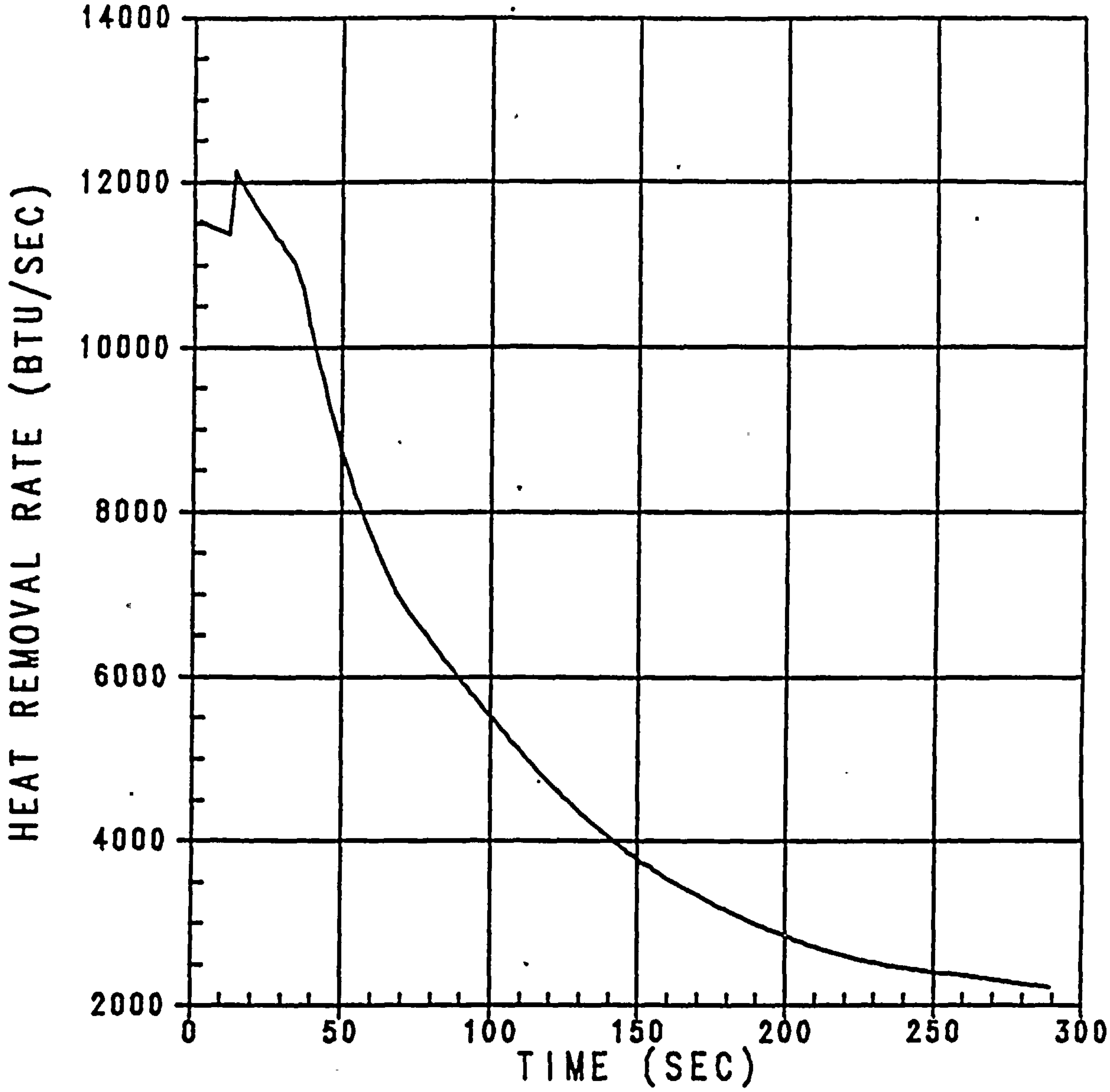

Figure 3.1-15 Upper Compartment Structural Heat Removal Rate
CD=0.4, Min SI
Donald C. Cook Unit 1

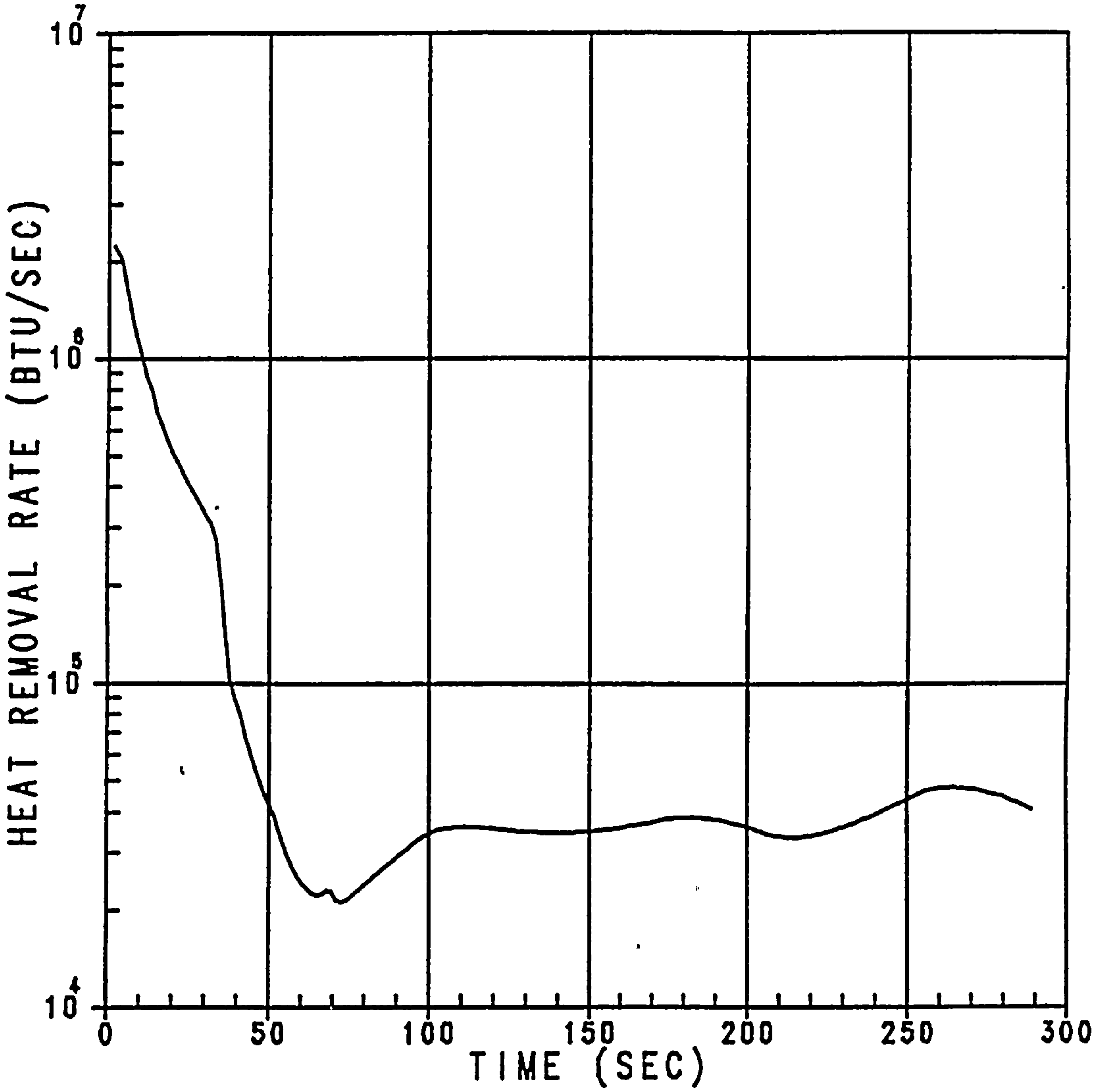

Figure 3.1-16 Lower Compartment Structural Heat Removal Rate
CD=0.4, Min SI
Donald C. Cook Unit 1

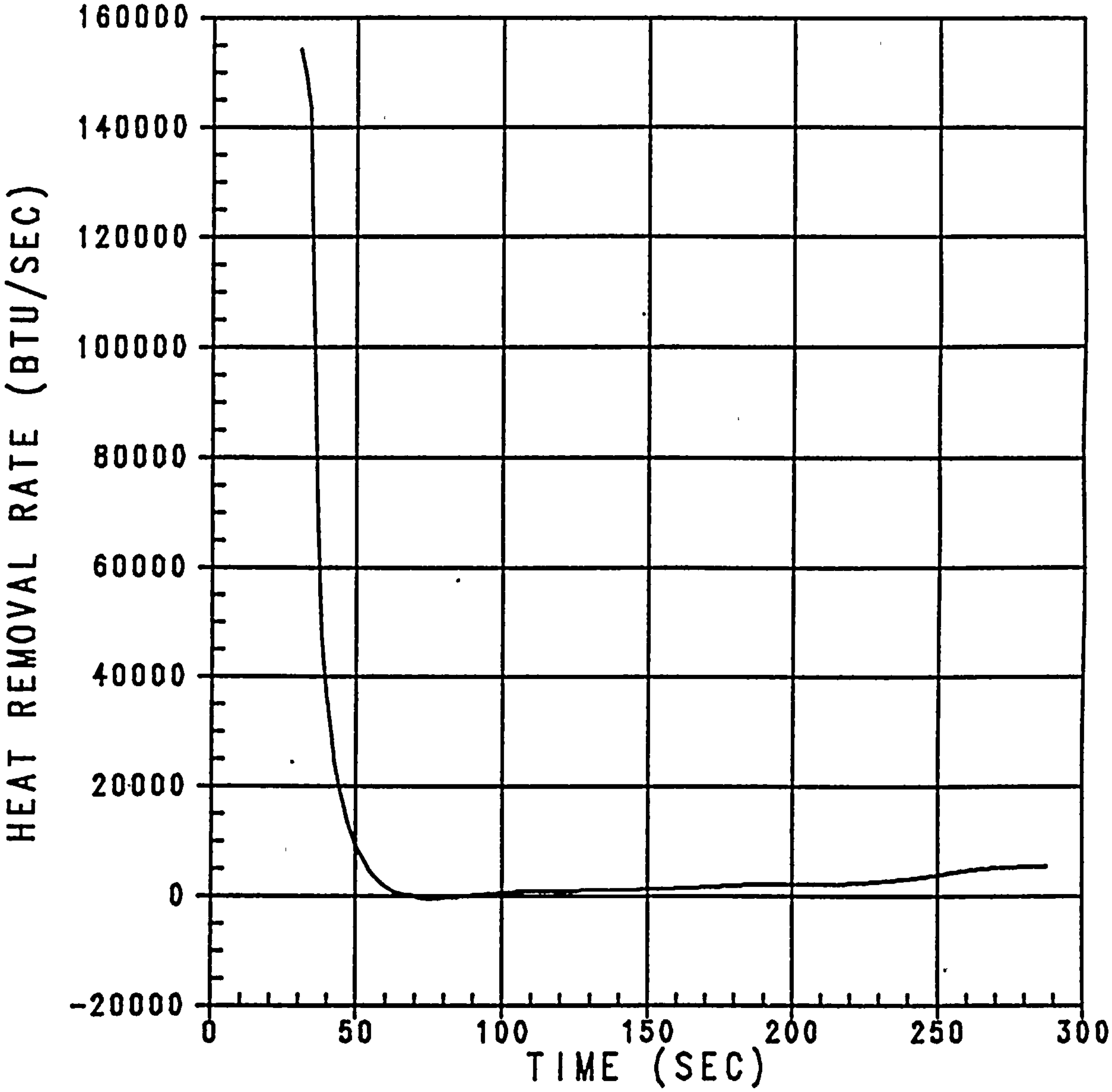

Figure 3.1-17 Heat Removal by Sump
CD=0.4, Min SI
Donald C. Cook Unit 1

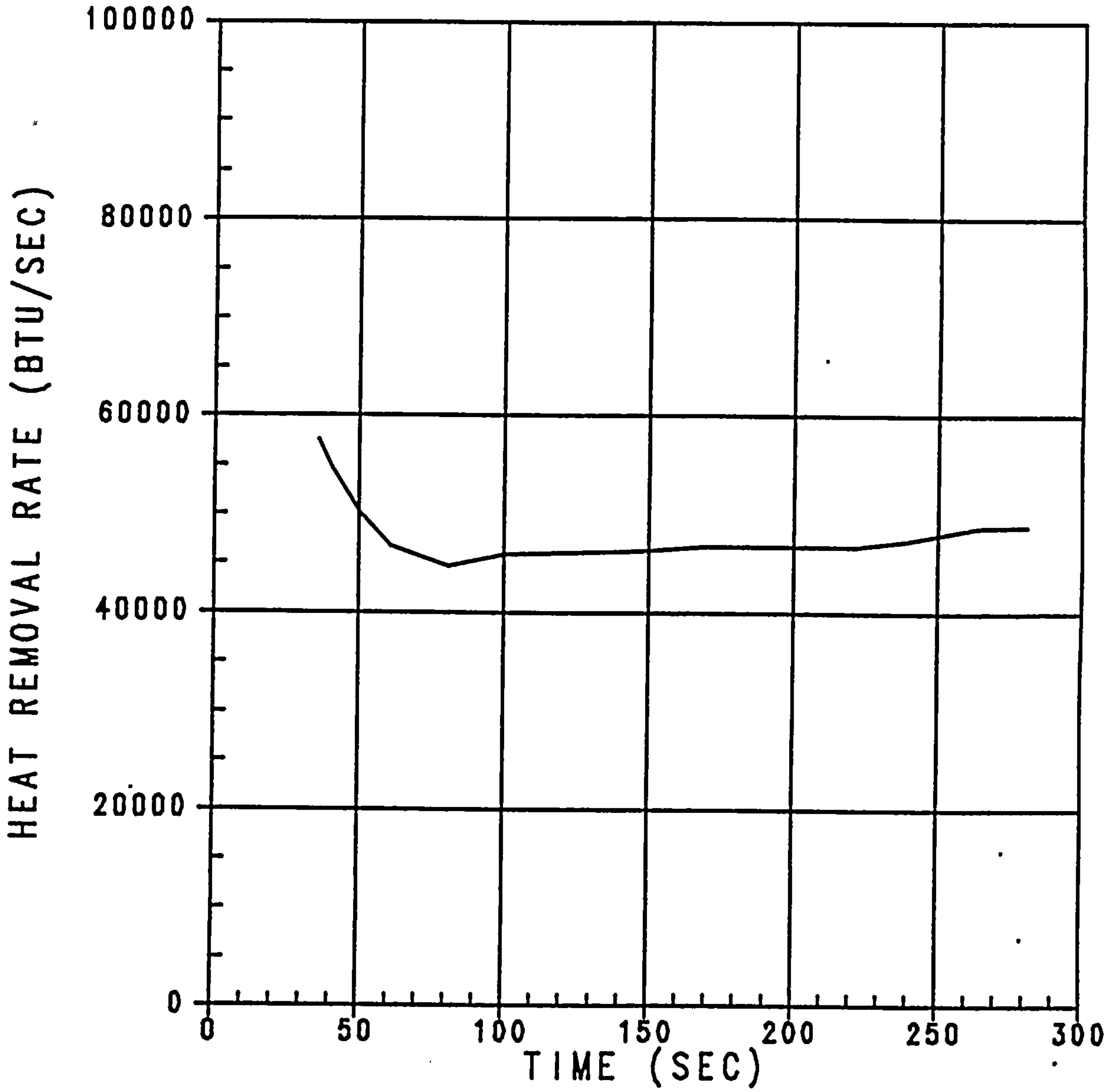

Figure 3.1-18 Heat REmoval by Lower Compartment Spray
CD=0.4, Min SI
Donald C. Cook Unit 1

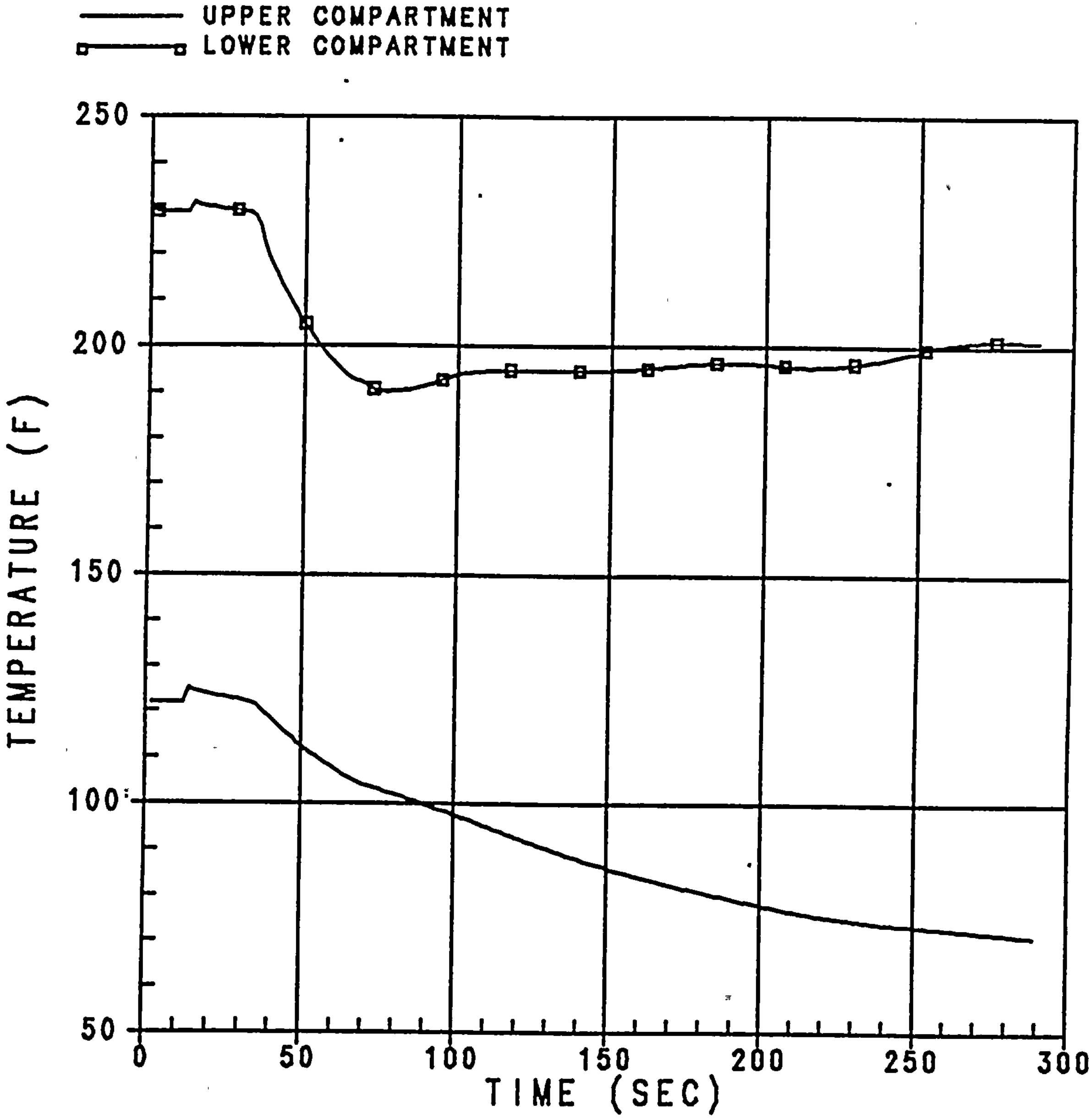

Figure 3.1-19 Containment Temperature
CD=0.4, Min SI
Donald C. Cook Unit 1

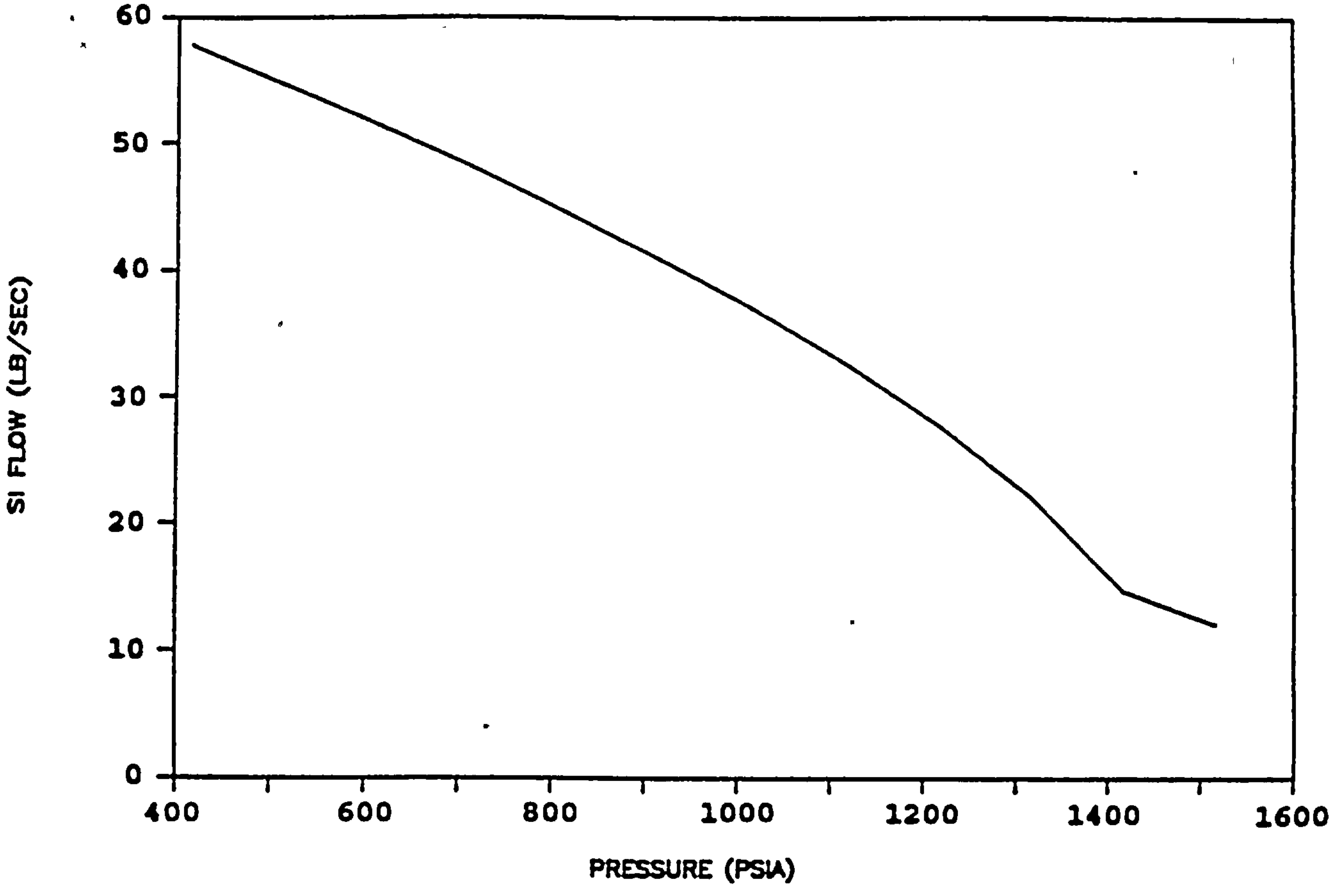

Figure 3.1-20 Safety Injection Flow Rate
Donald C. Cook Unit 1

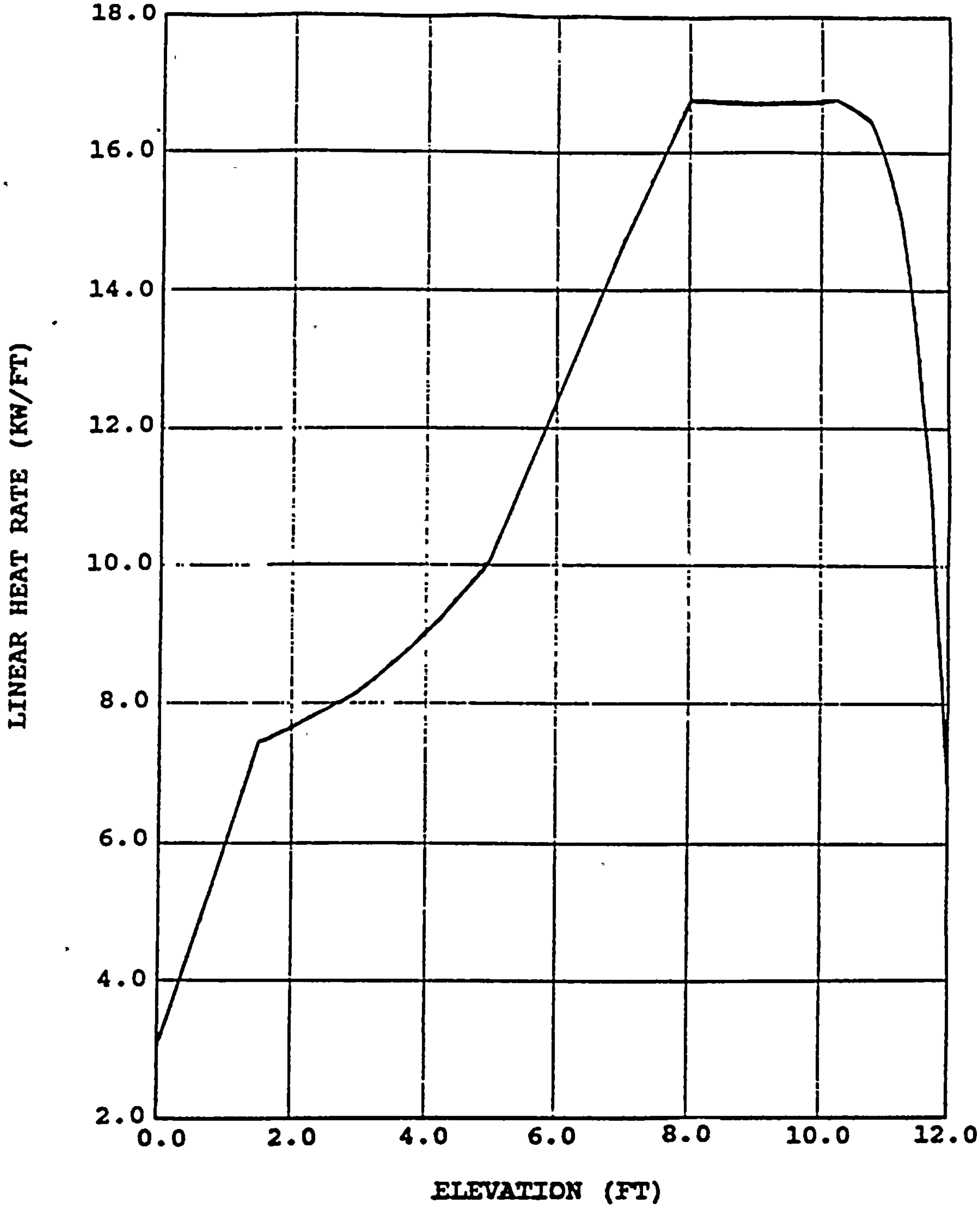

Figure 3.1-21 Hot Rod Power Distribution
Donald C. Cook Unit 1

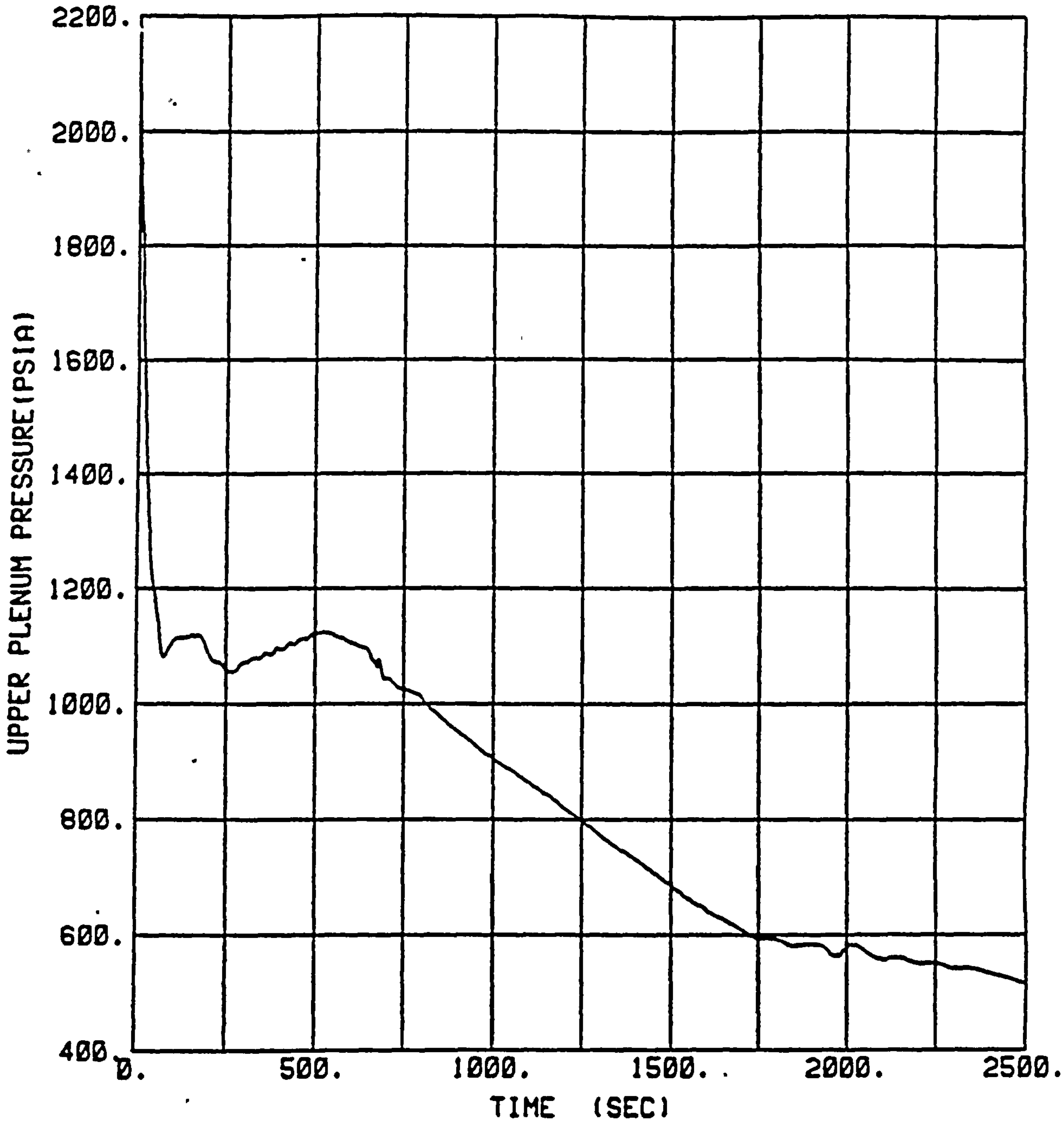

Figure 3.1-22 RCS Pressure (3 Inch) Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

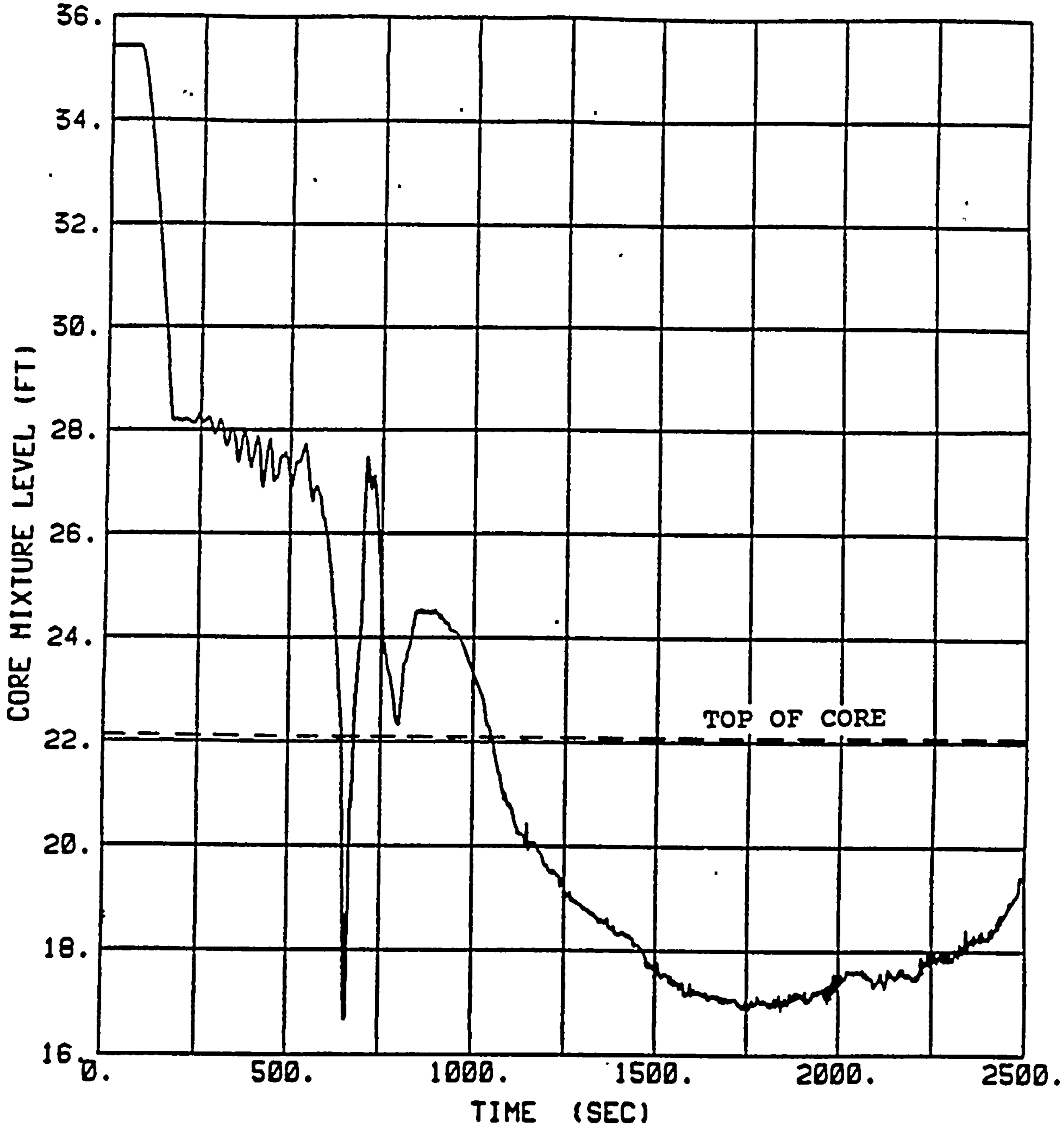

Figure 3.1-23 Core Mixture Height (3 Inch) Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

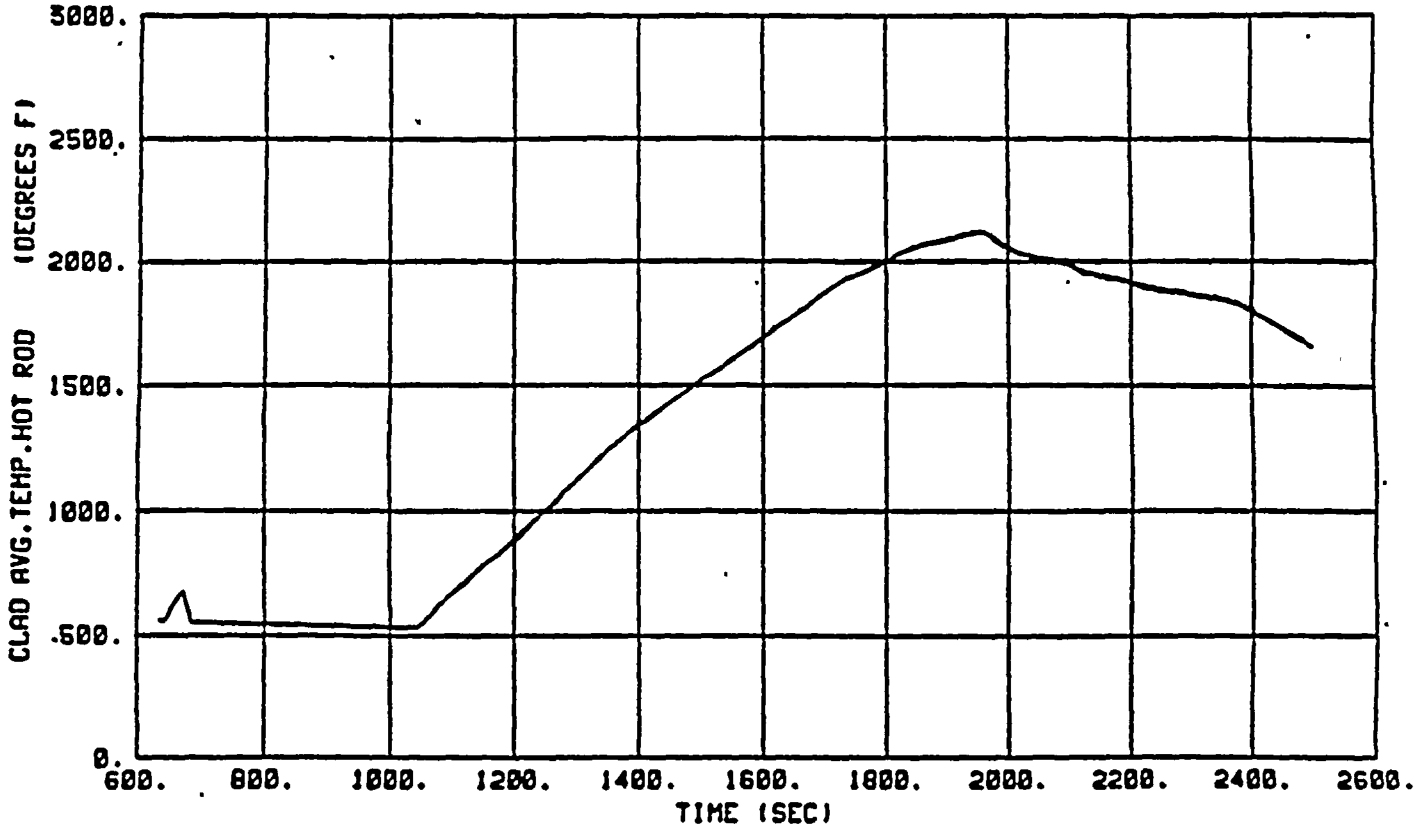

Figure 3.1-24 Hot Spot Clad Temperature (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

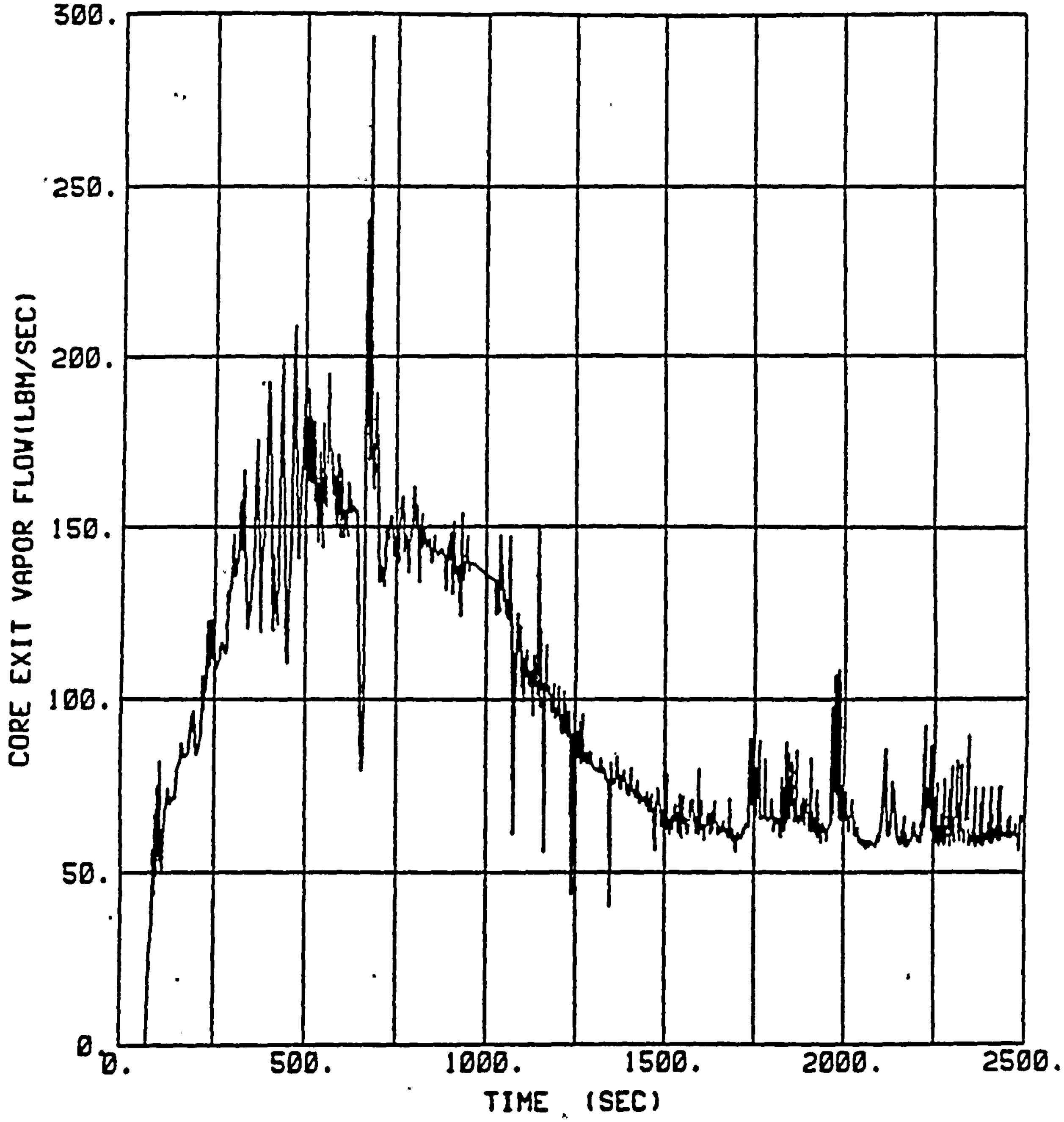

Figure 3.1-25 Core Steam Flowrate (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

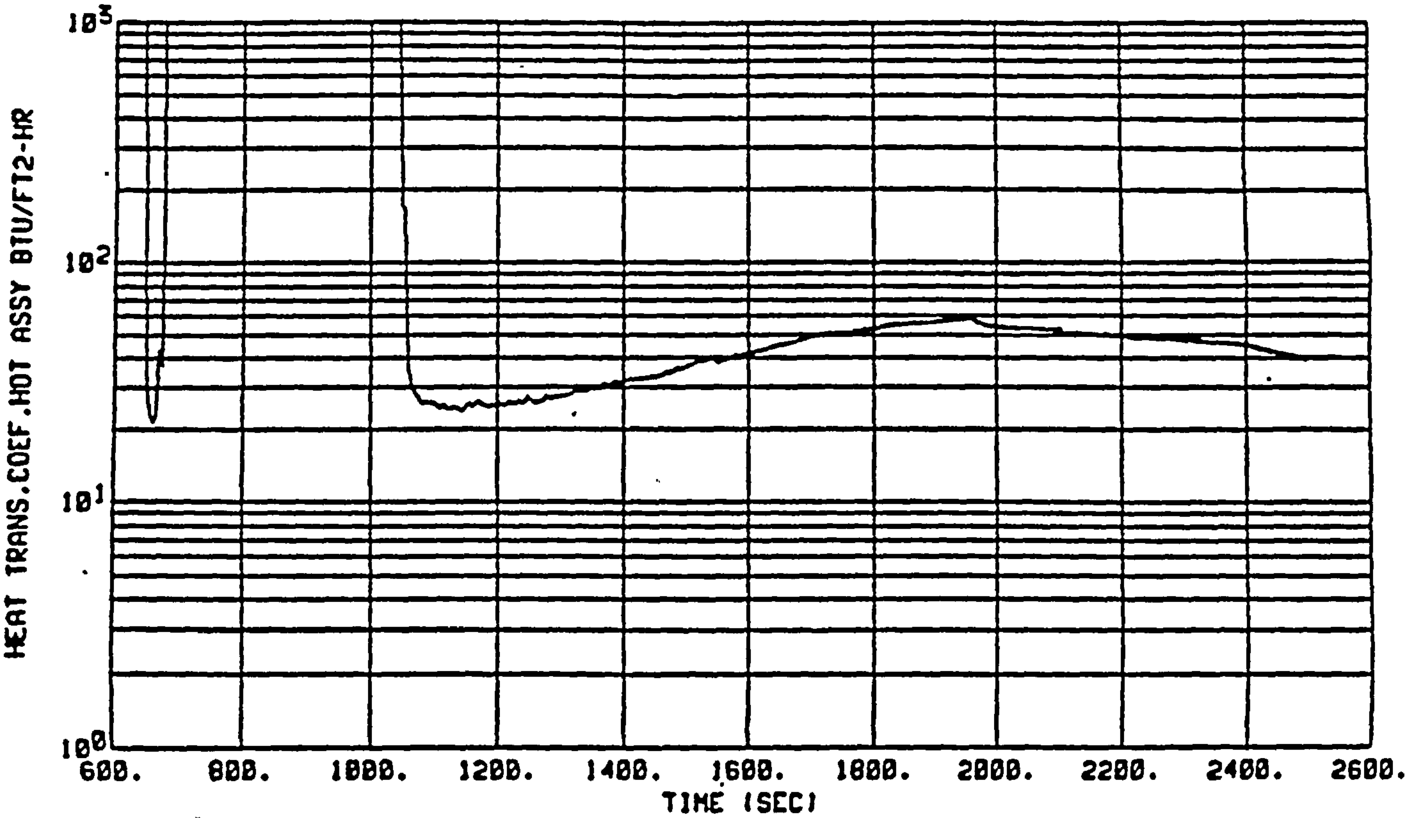

Figure 3.1-26 Hot Spot Heat Transfer Coefficient (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

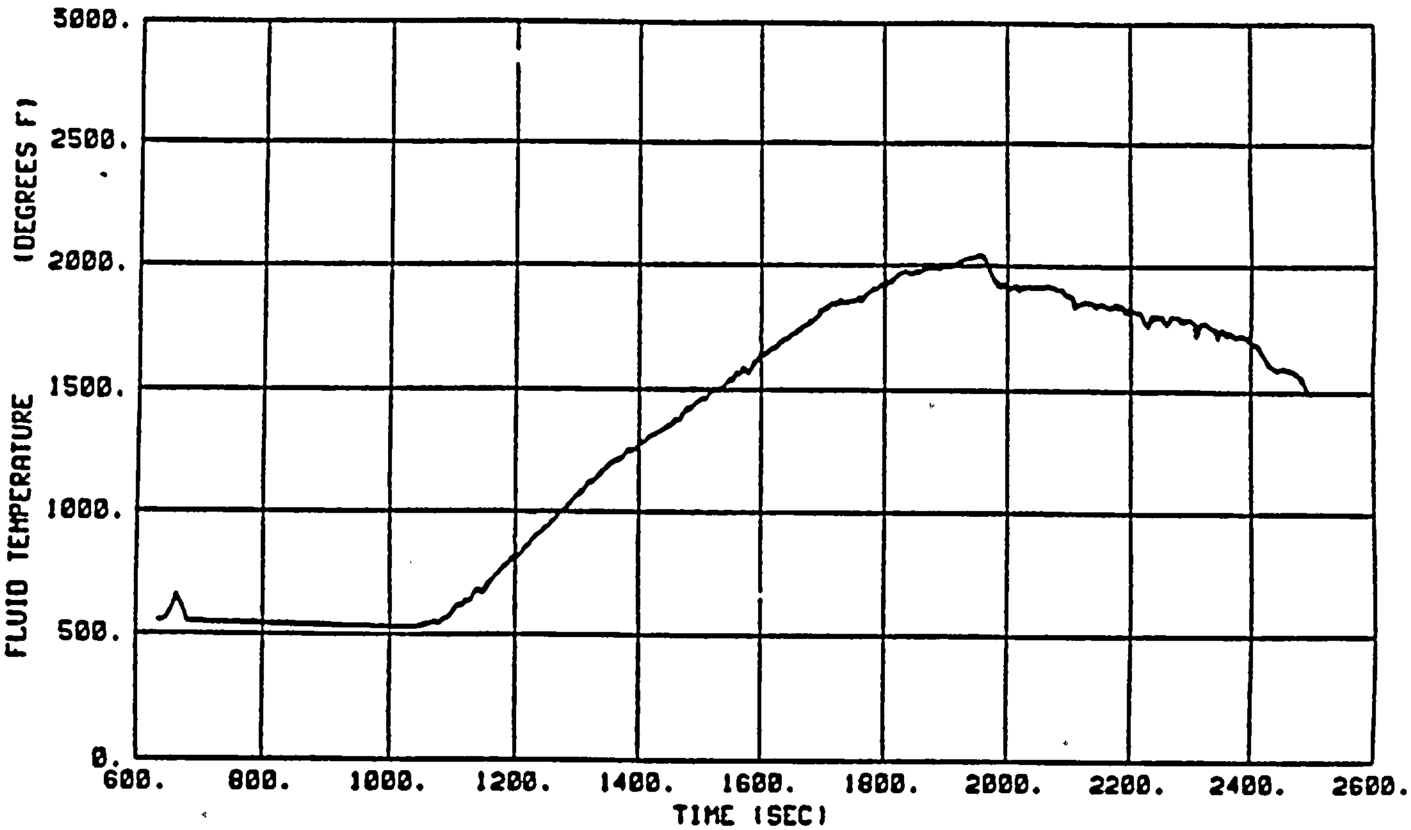

Figure 3.1-27 Hot Spot Fluid Temperature (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

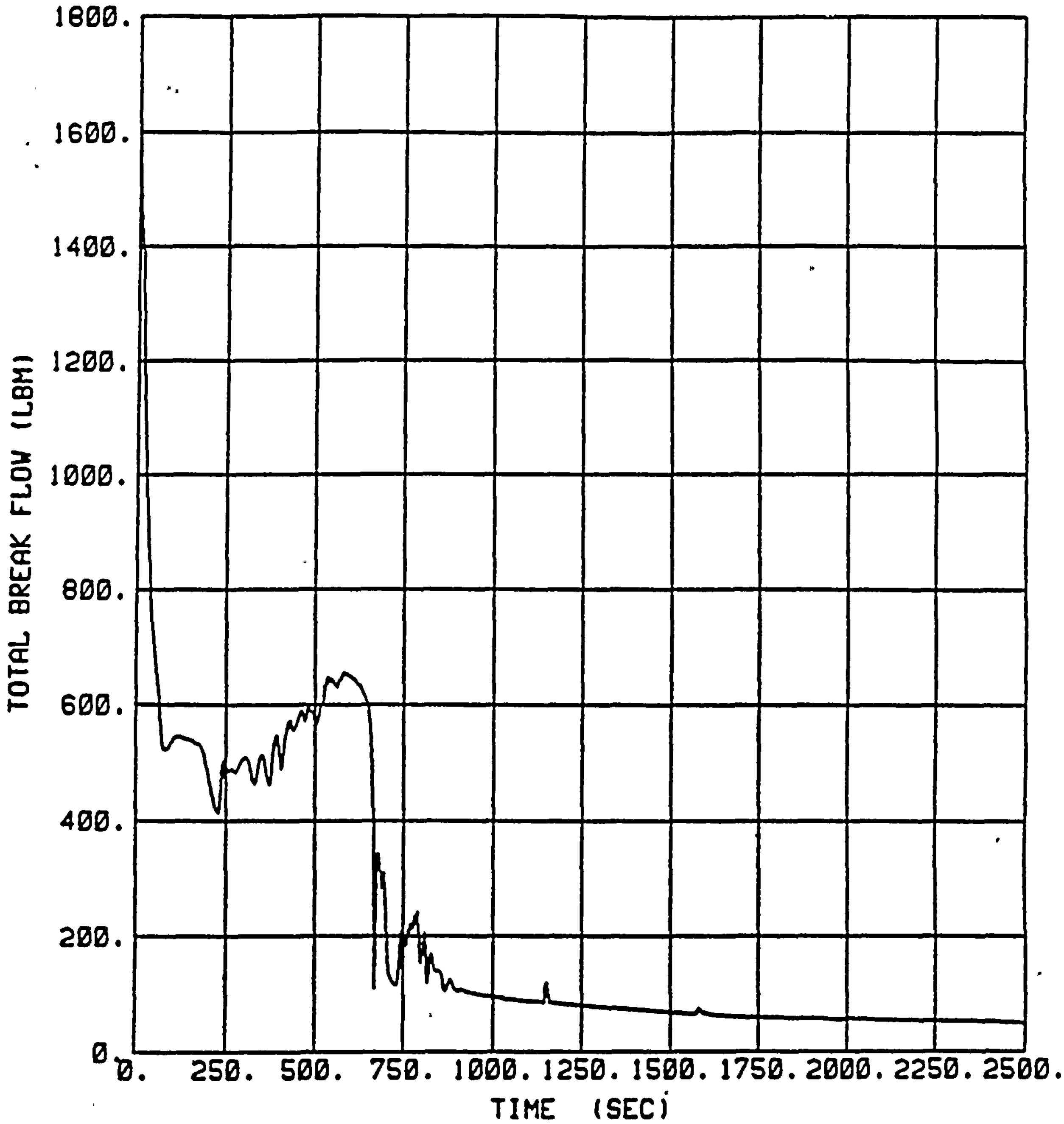

Figure 3.1-28 Total Break Flow (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

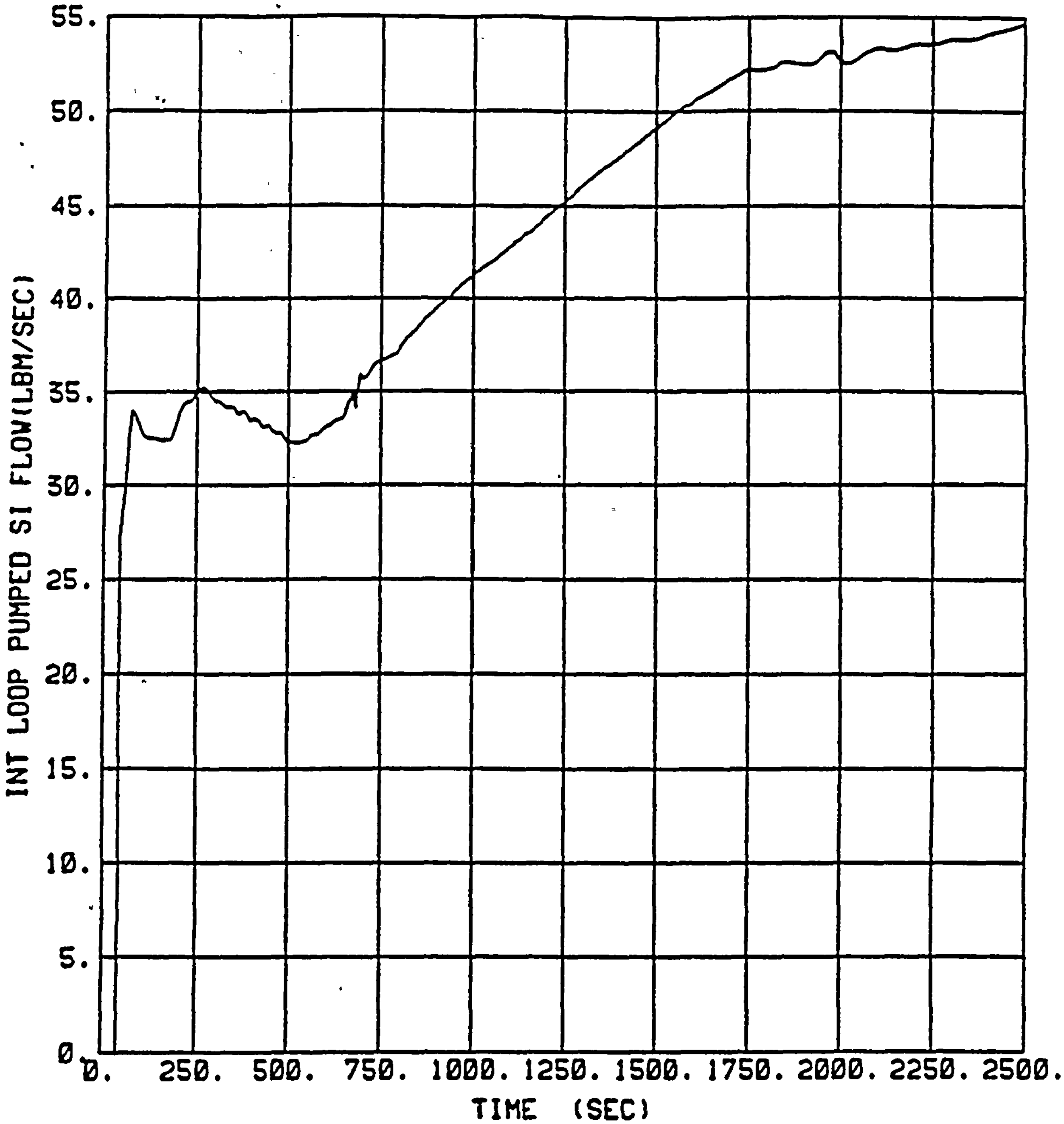

Figure 3.1-29 Intact Loop Pumped SI Flow (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

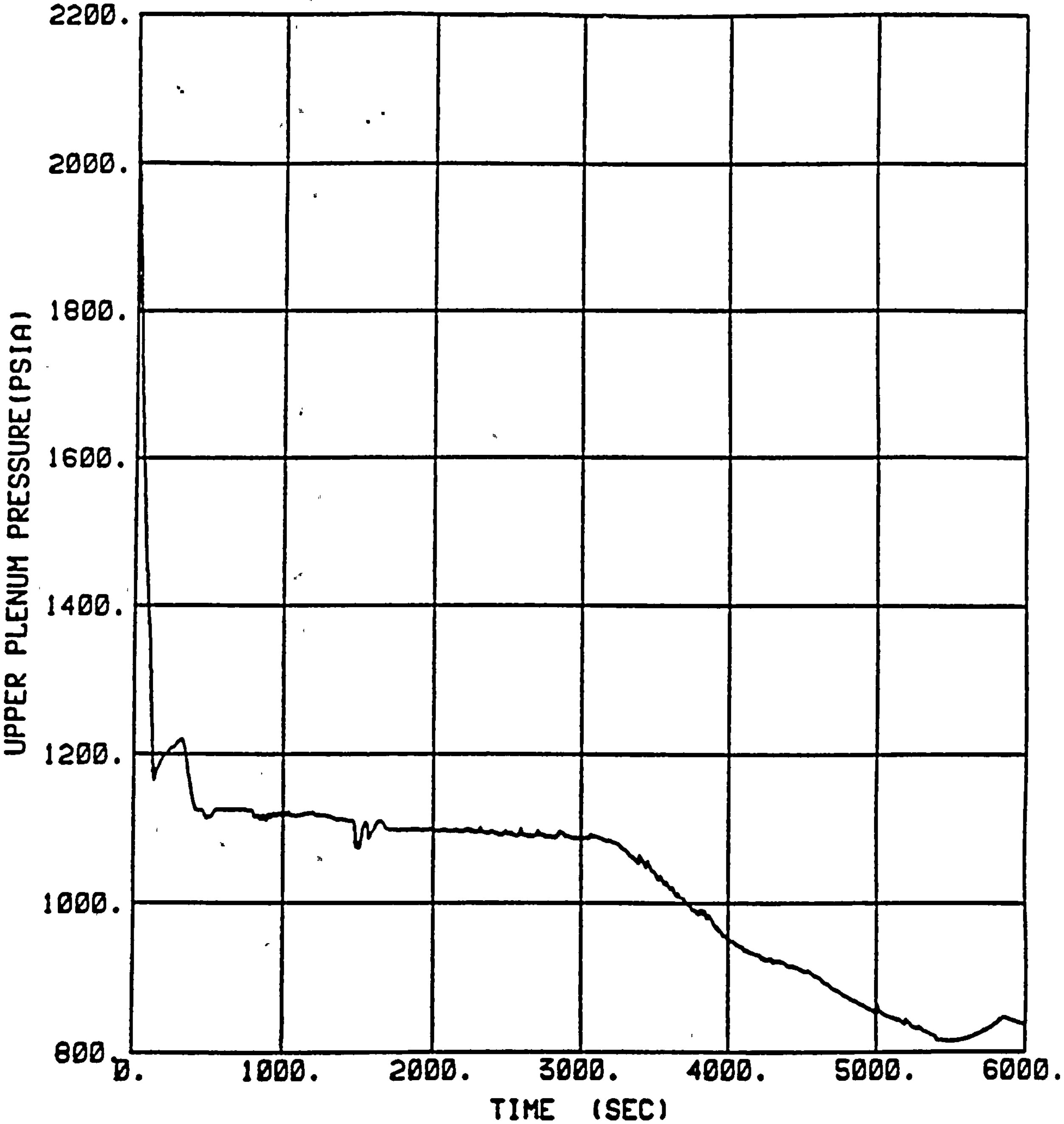

Figure 3.1-30 RCS Pressure (2 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

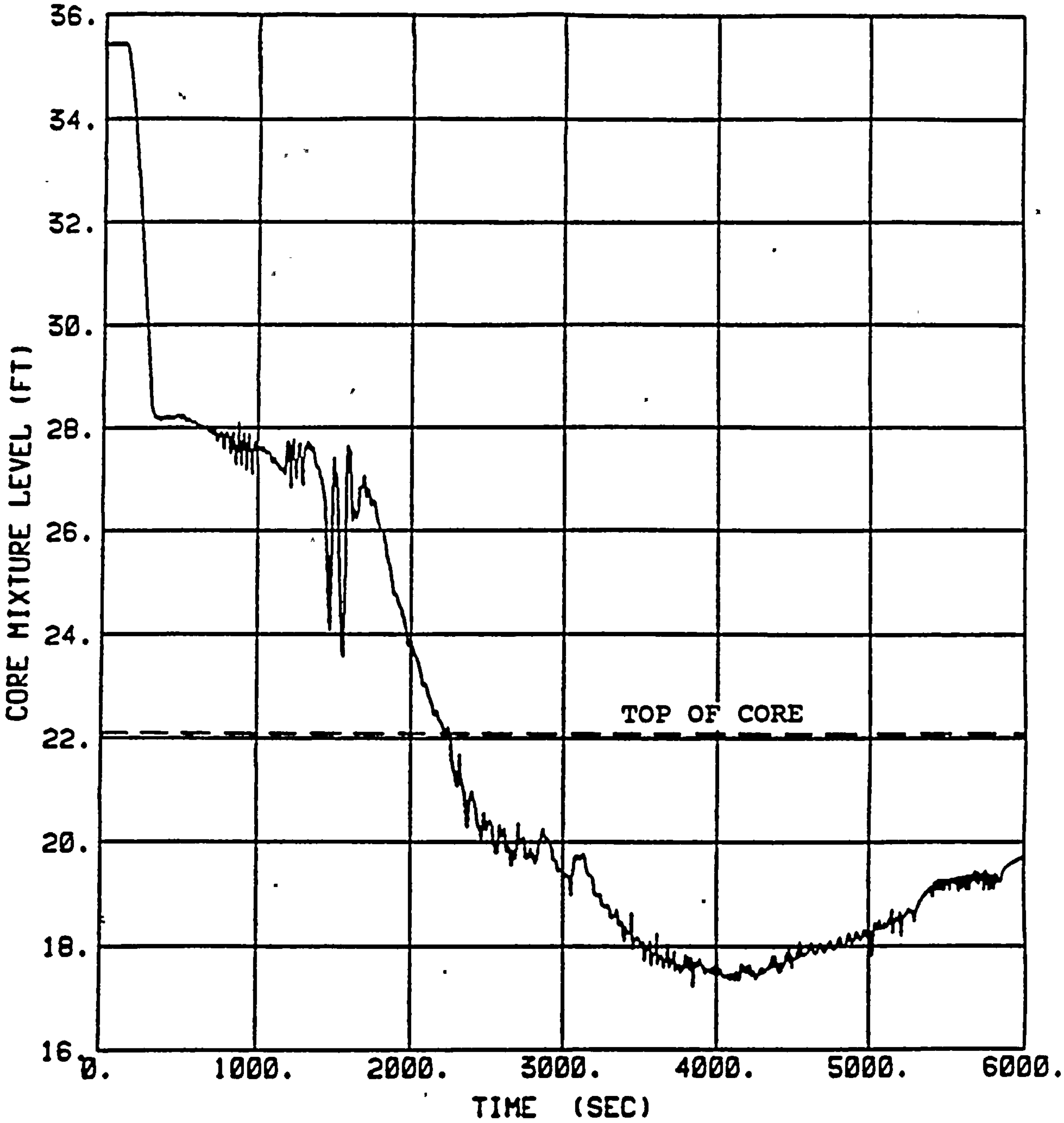

Figure 3.1-31 Core Mixture Height (2 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

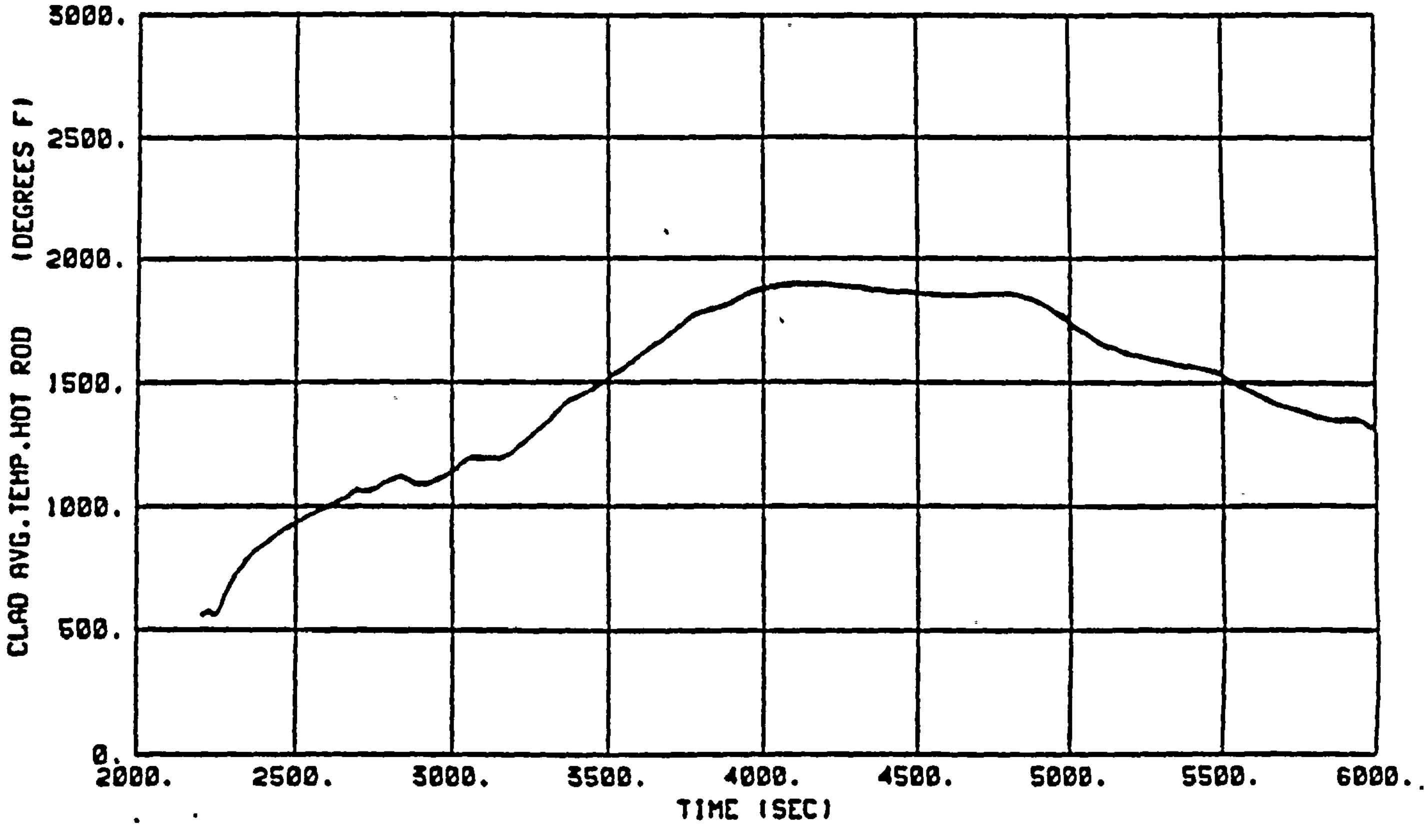

Figure 3.1-32 Hot Spot Clad Temperature (2 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

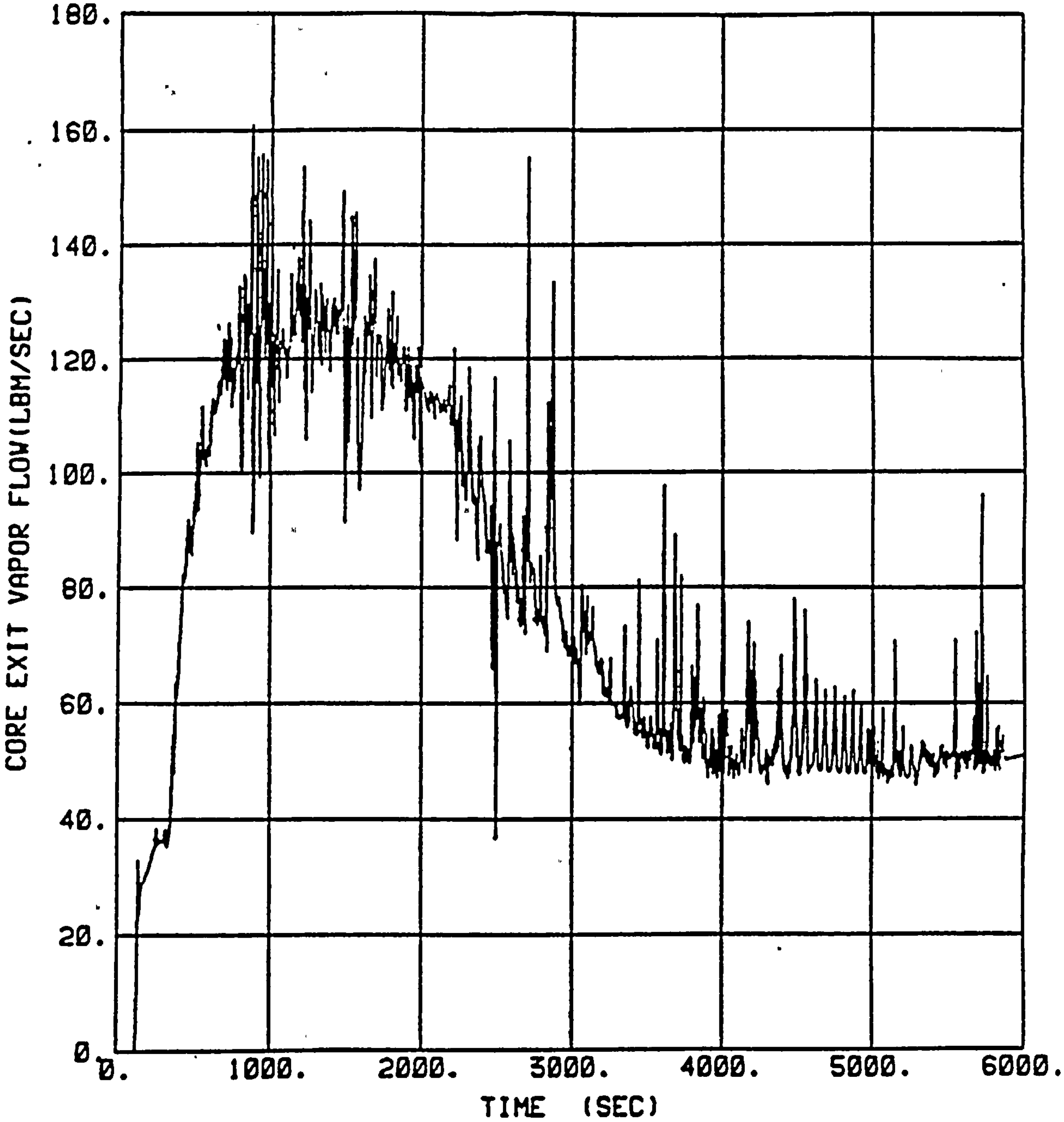

Figure 3.1-33 Core Steam Flowrate (2 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

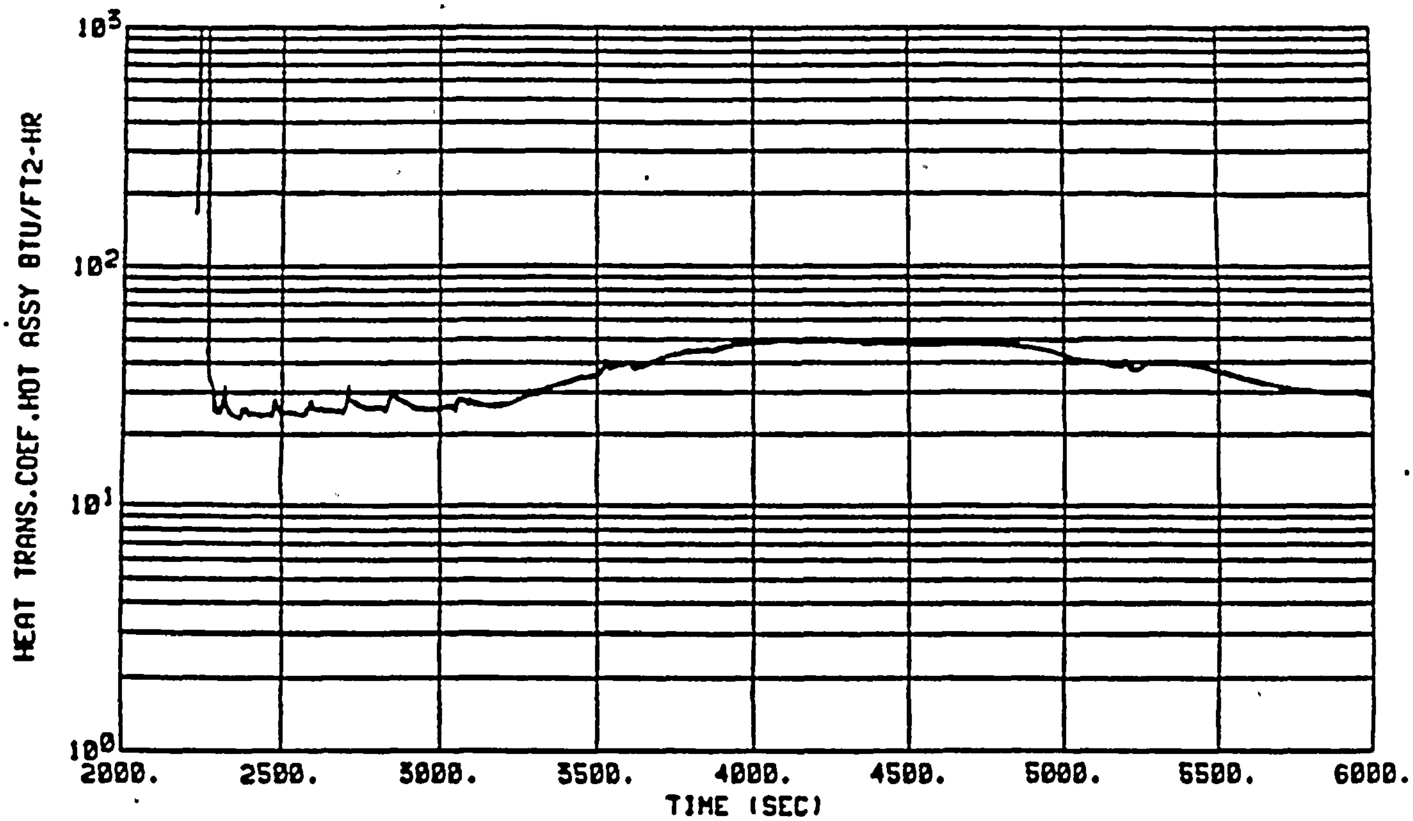

Figure 3.1-34 Hot Spot Heat Transfer Coefficient (2 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

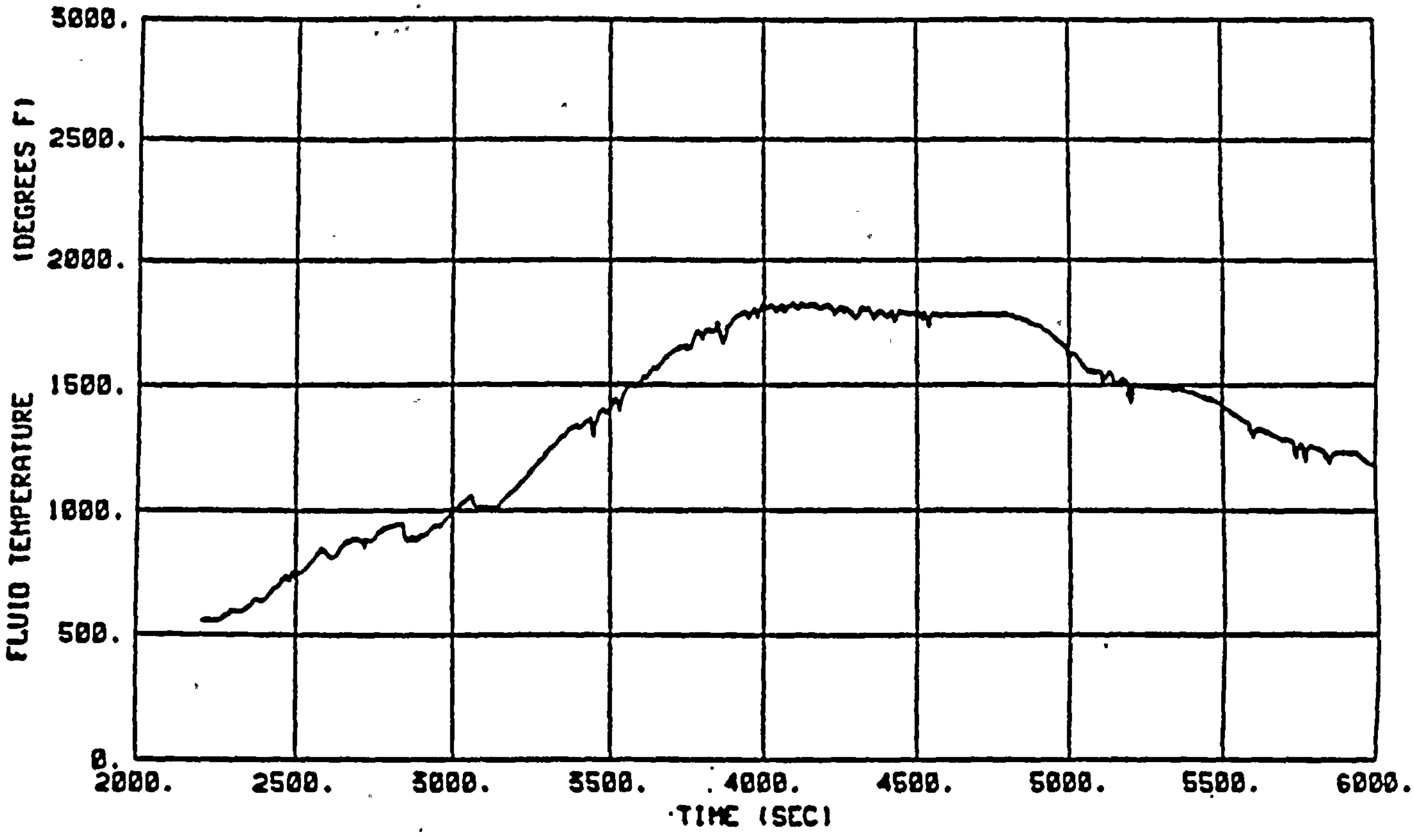

Figure 3.1-35 Hot Spot FLuid Temperature (2 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

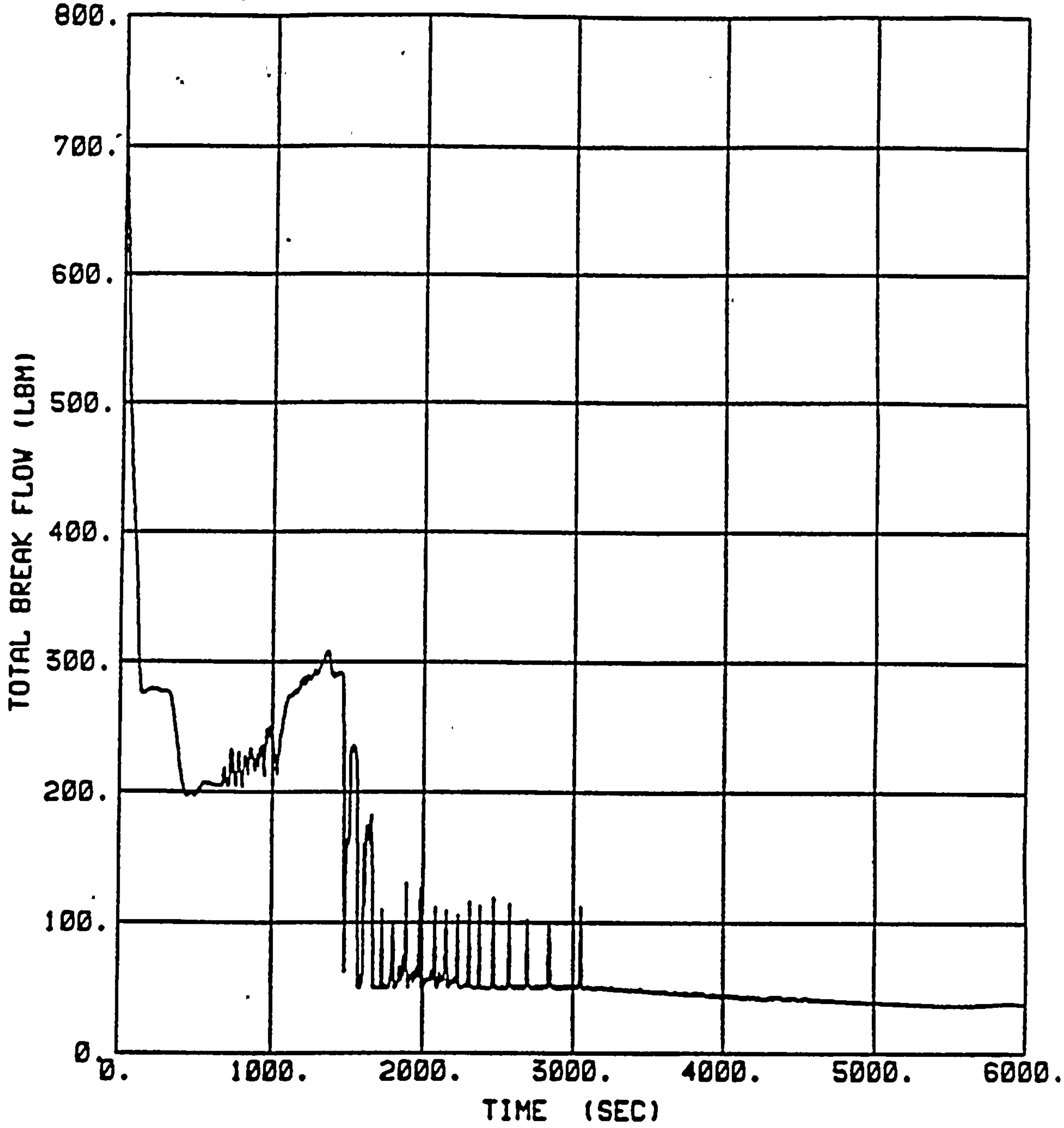

Figure 3.1-36 Total Break Flow (2 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

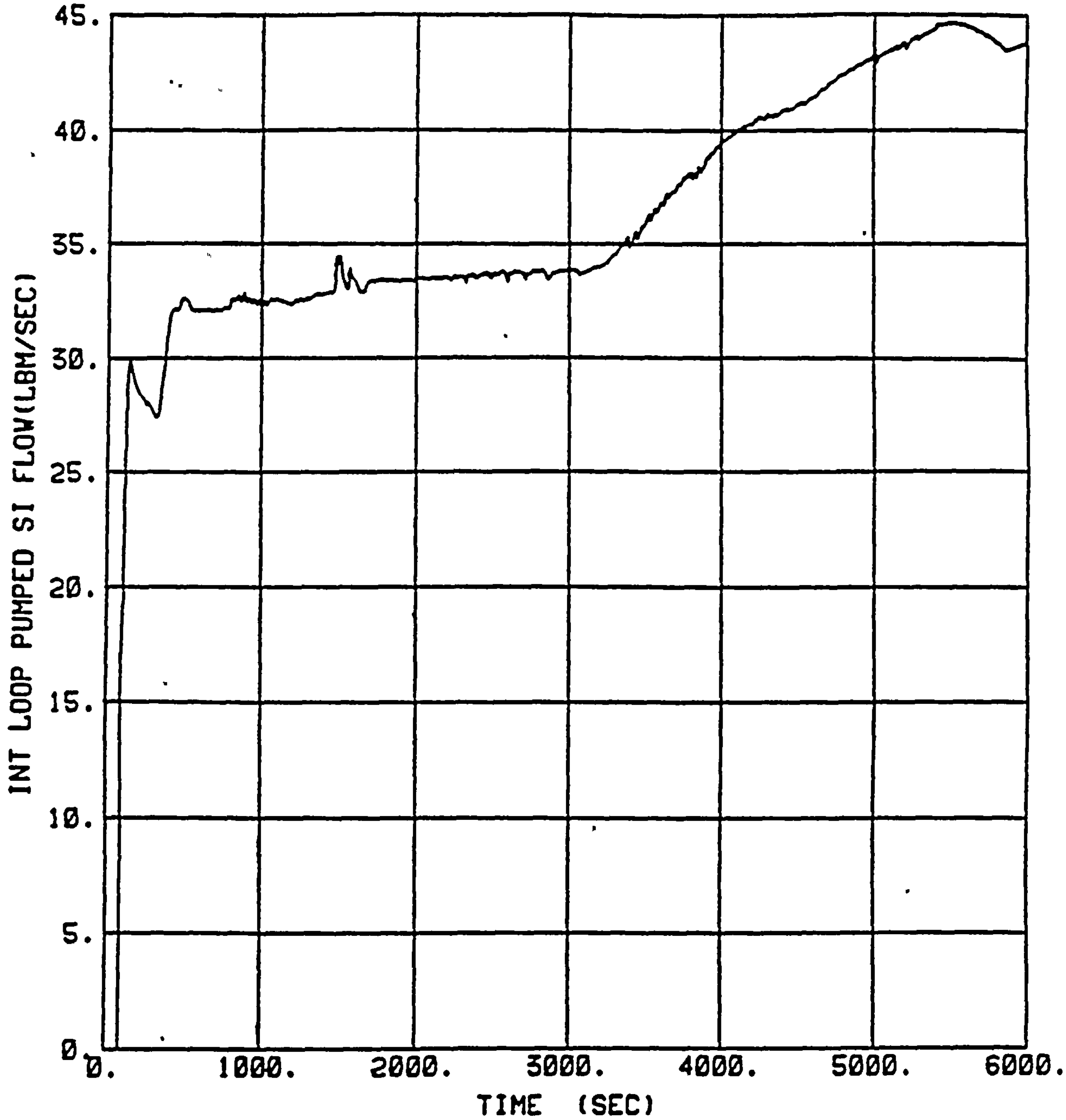

Figure 3.1-37 Intact Loop Pumped SI Flow (2 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

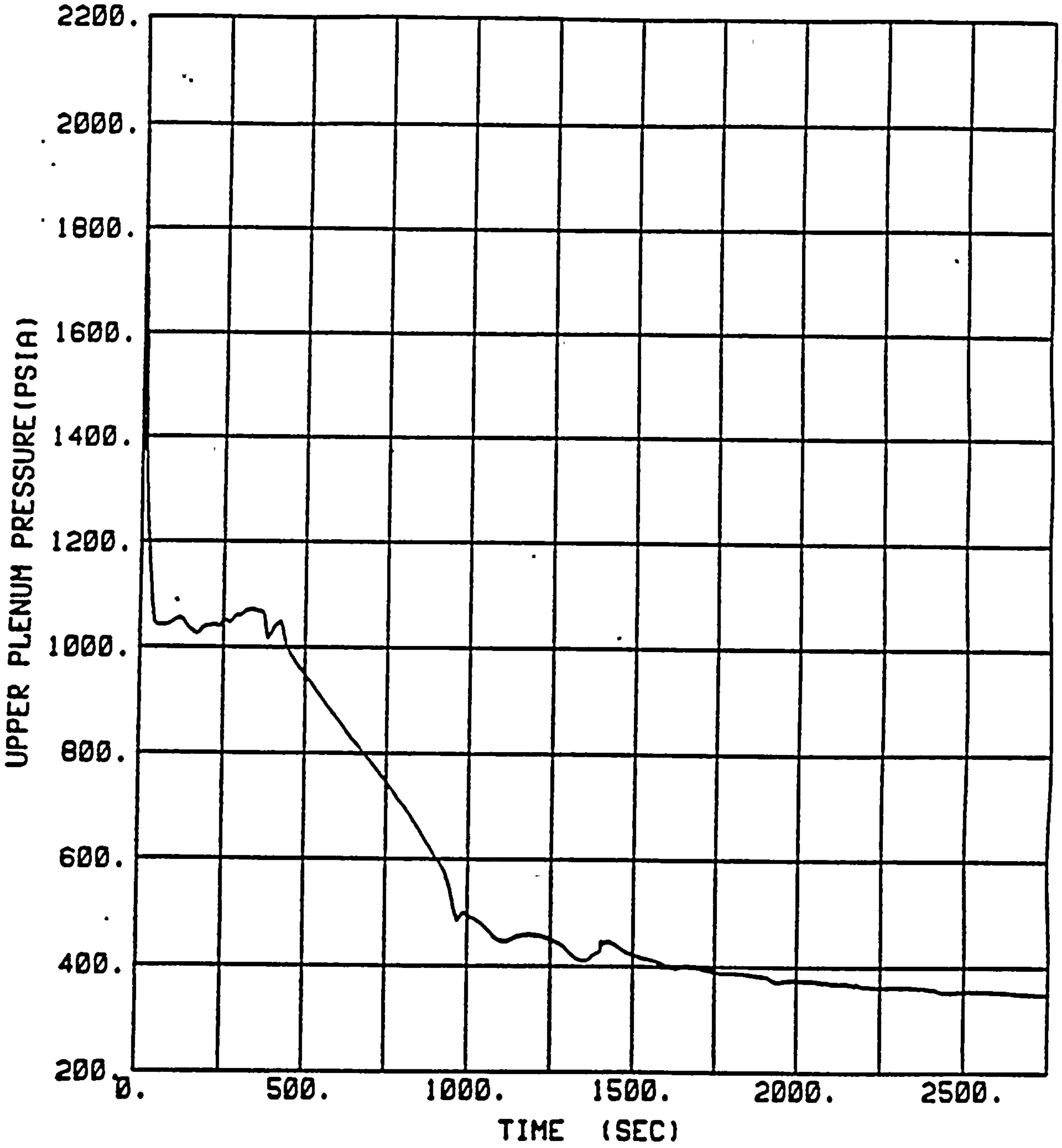

Figure 3.1-38 RCS Pressure (r Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

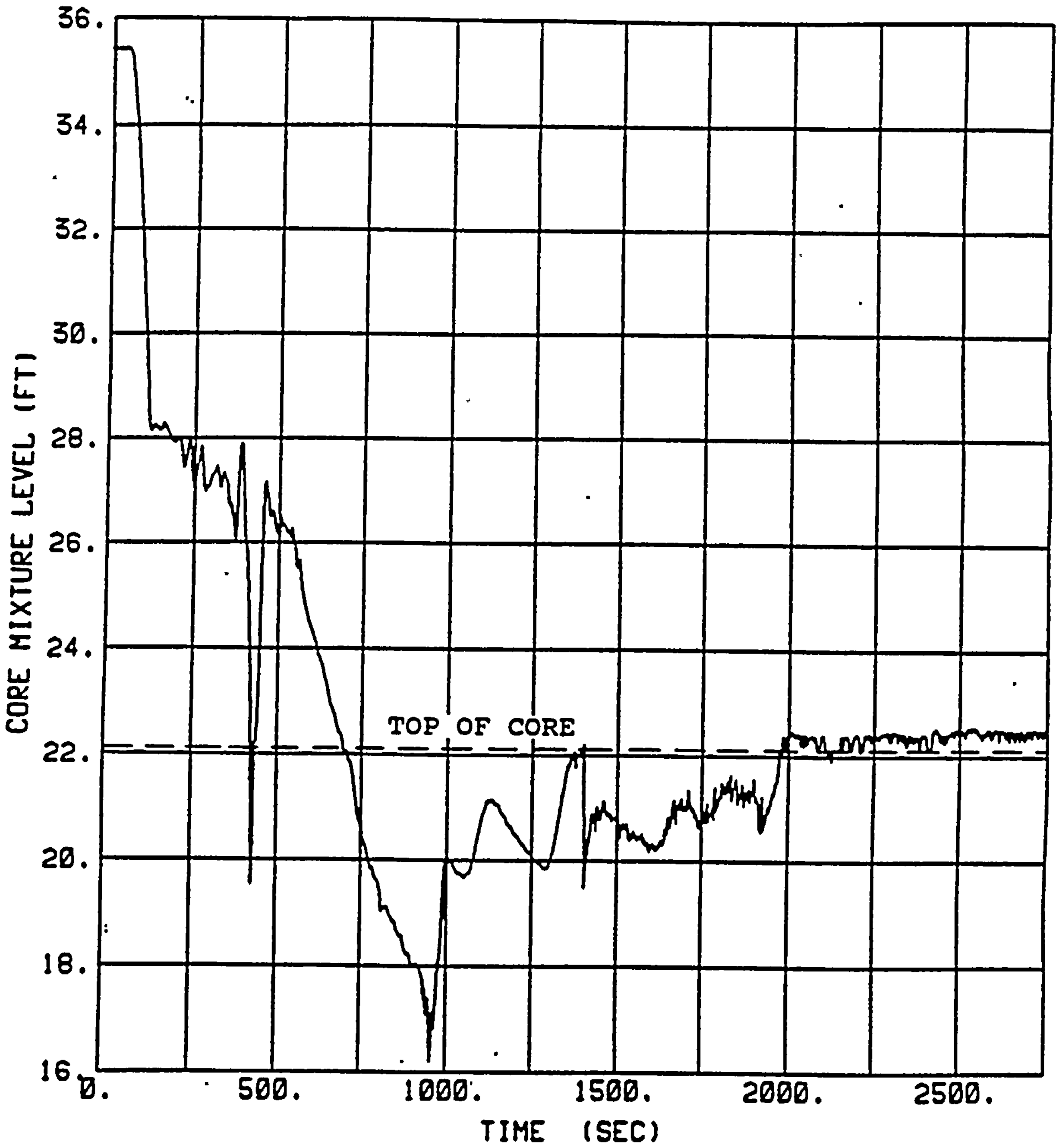

Figure 3.1-39 Core Mixture Height (4 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

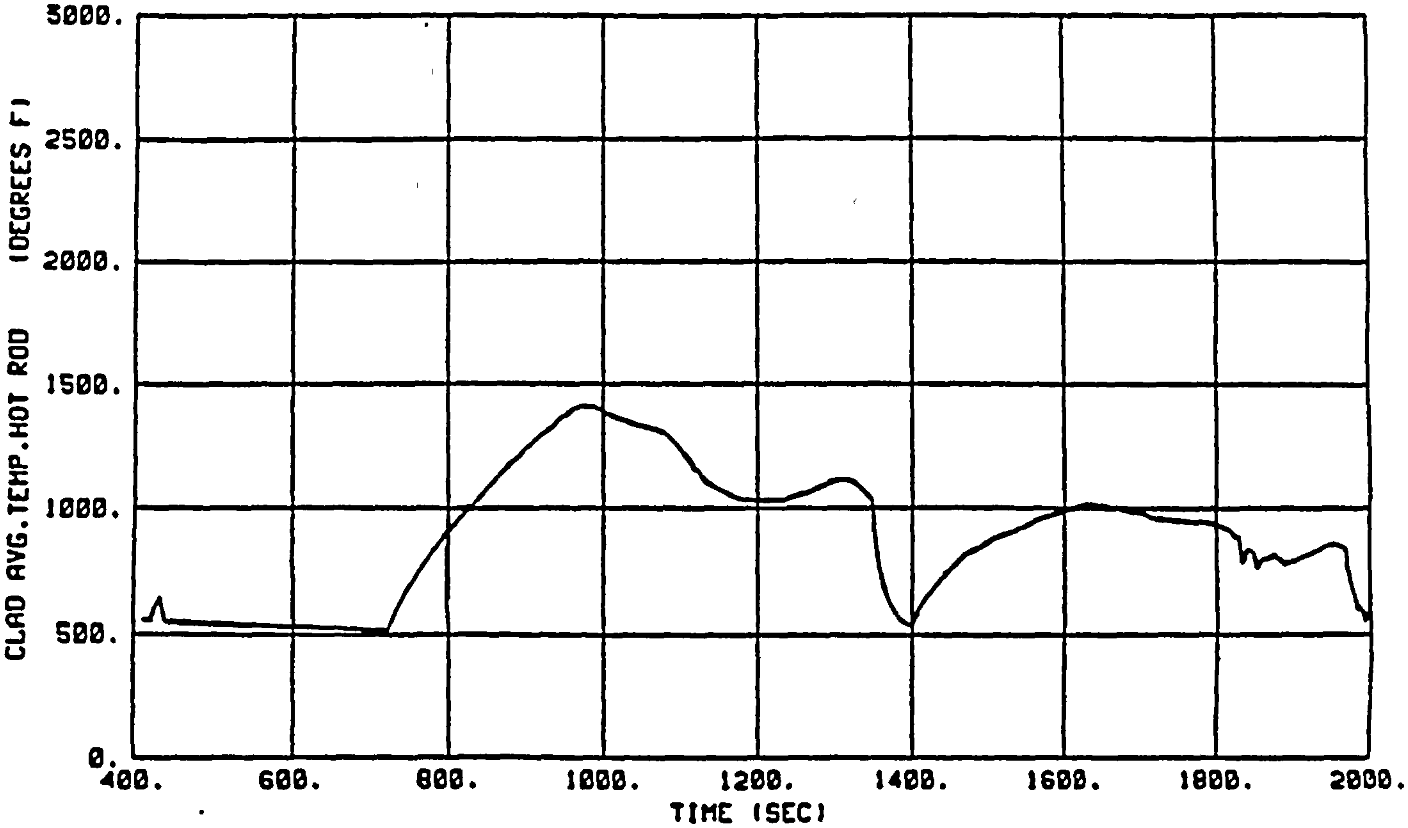

Figure 3.1-40 Hot Spot Clad Temperature (4 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

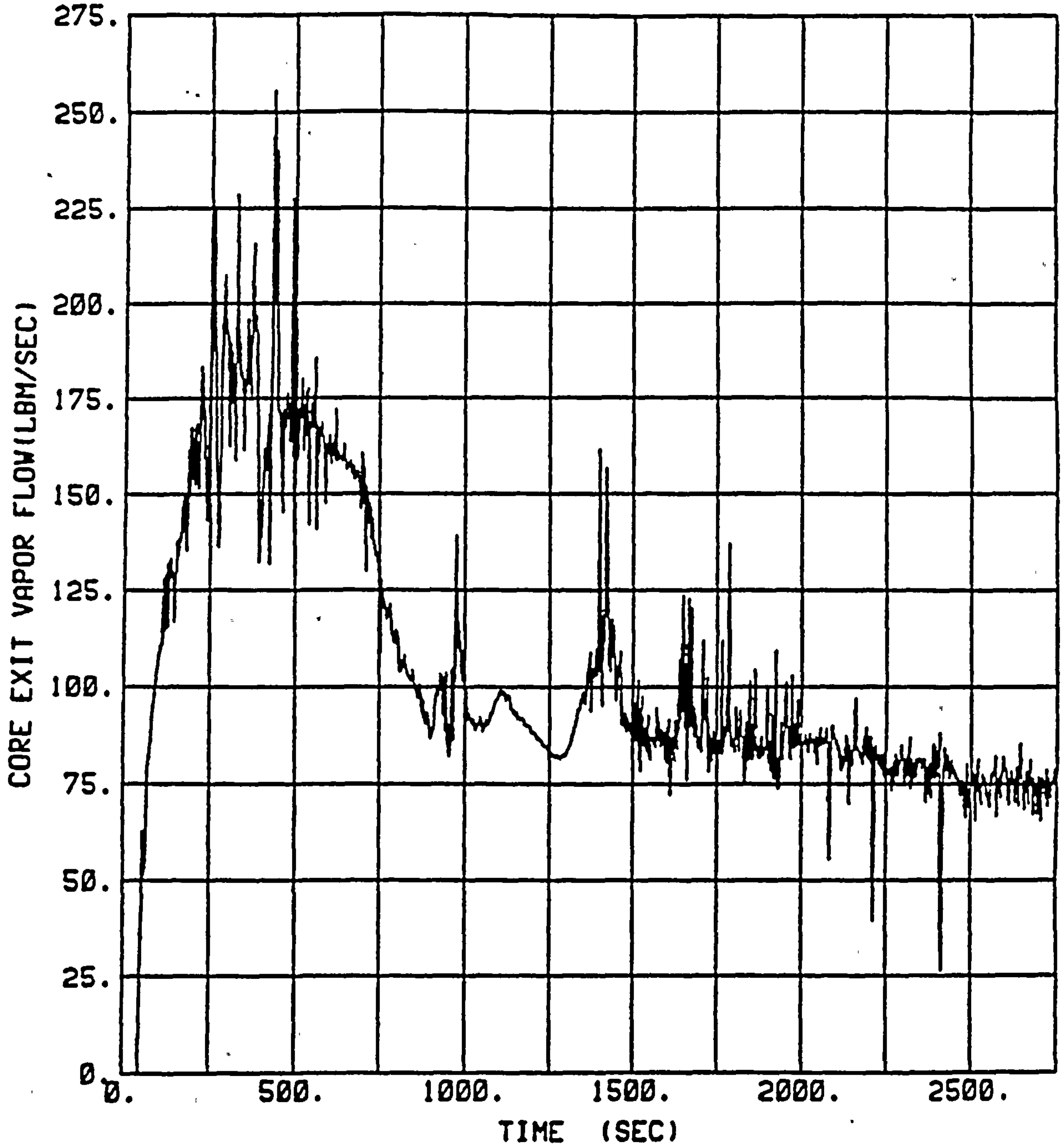

Figure 3.1-41 Core Steam Flowrate (4 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

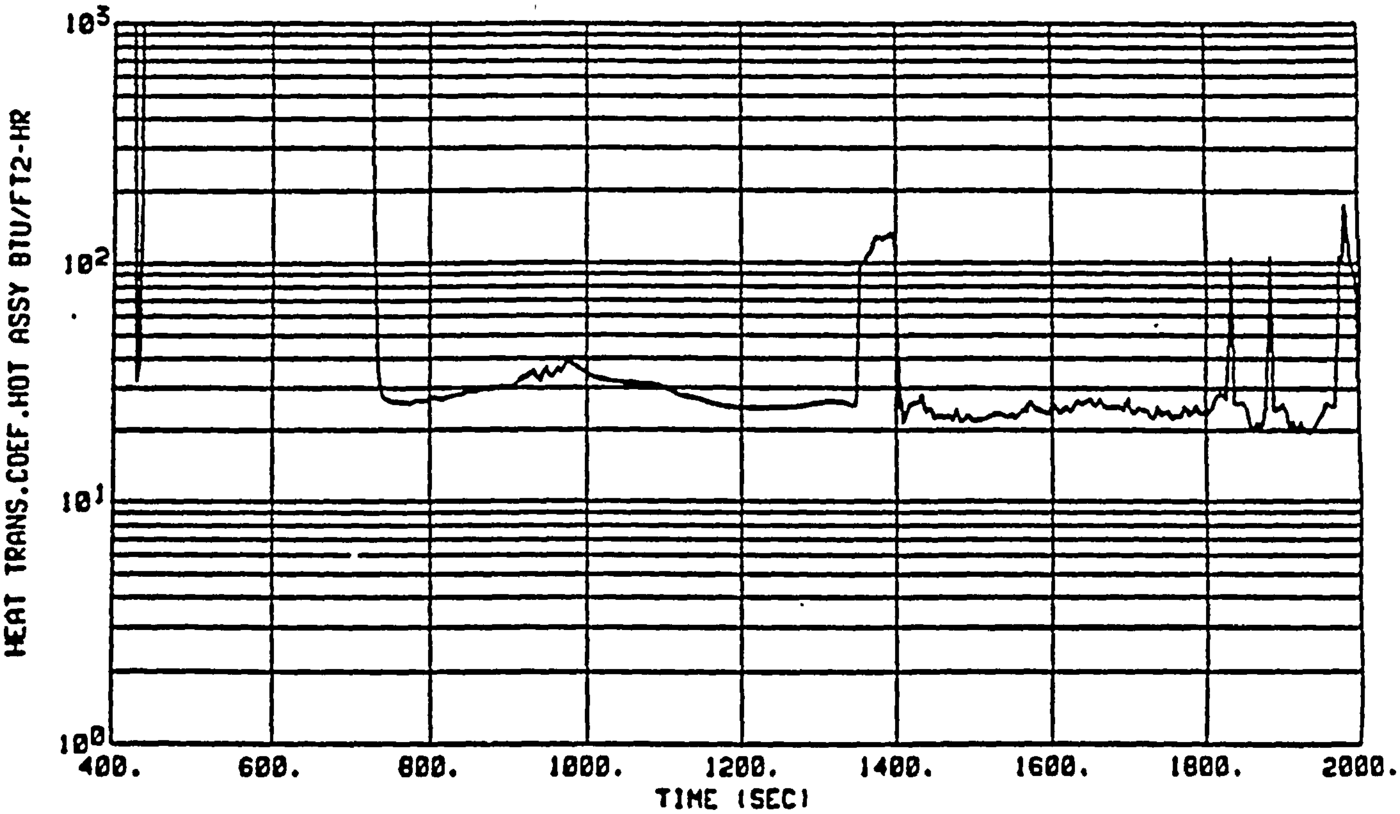

Figure 3.1-42 Hot Spot Heat Transfer Coefficient (4 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

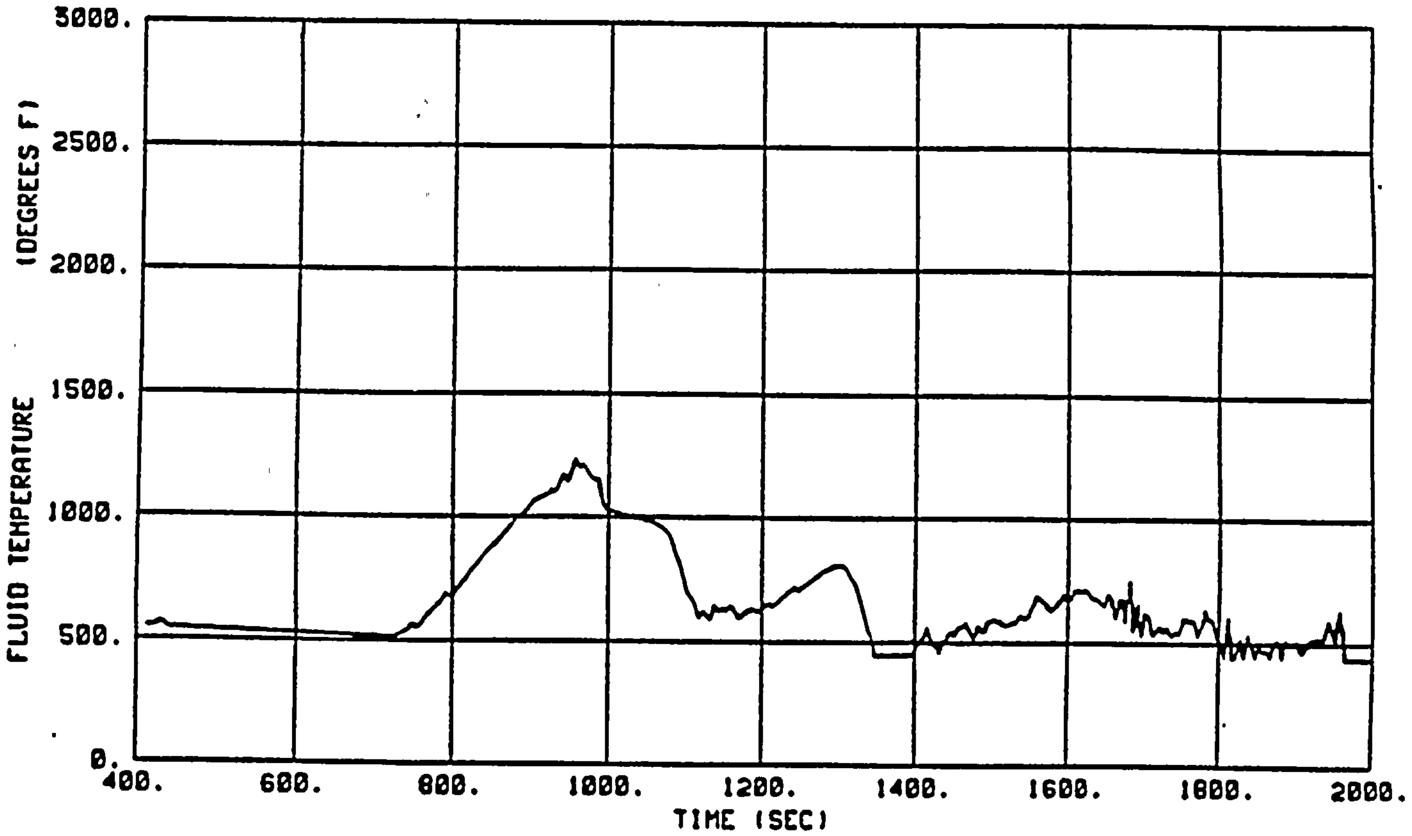

Figure 3.1-43 Hot Spot Fluid Temperature (4 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

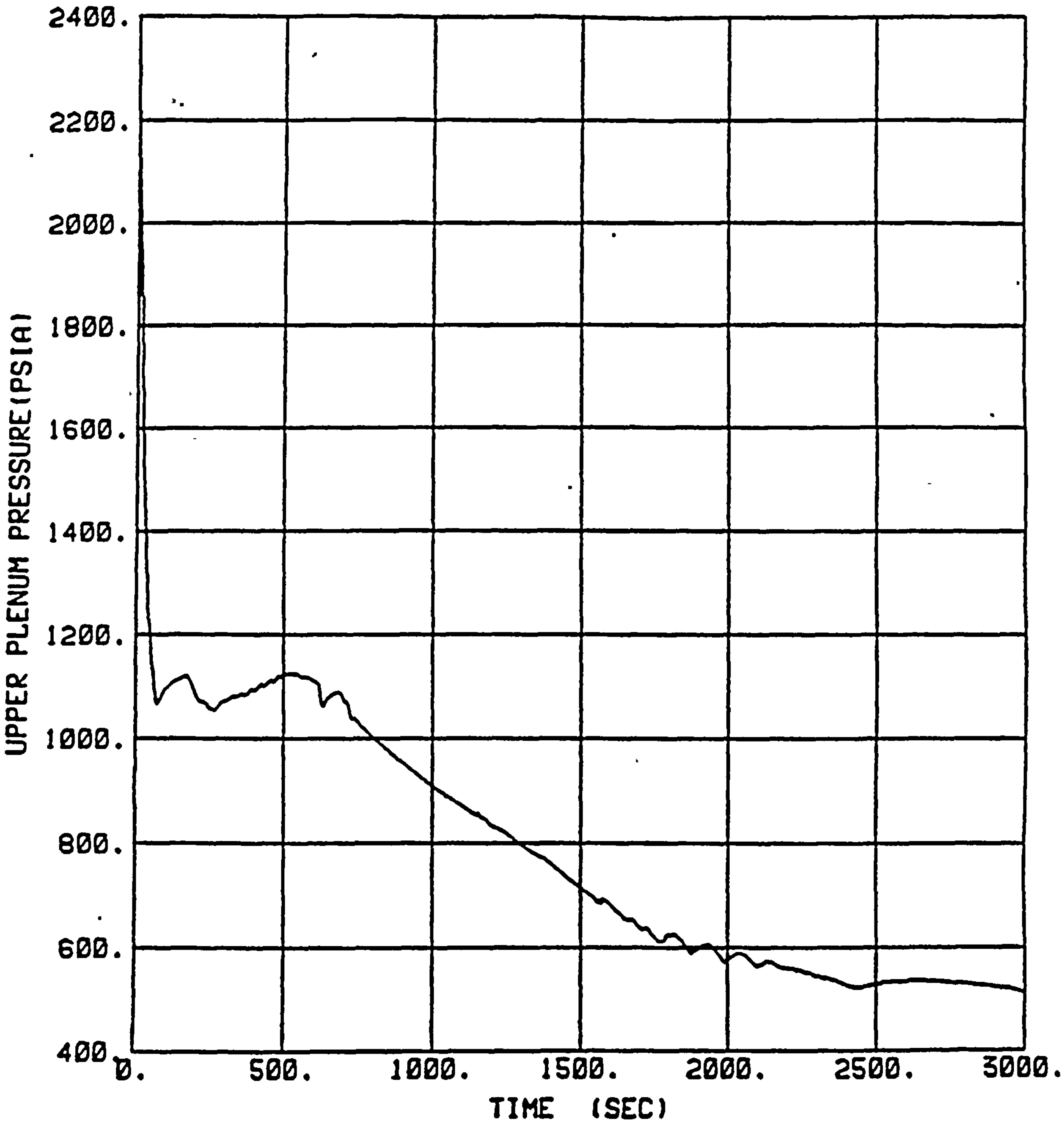

Figure 3.1-44 RCS Pressure (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

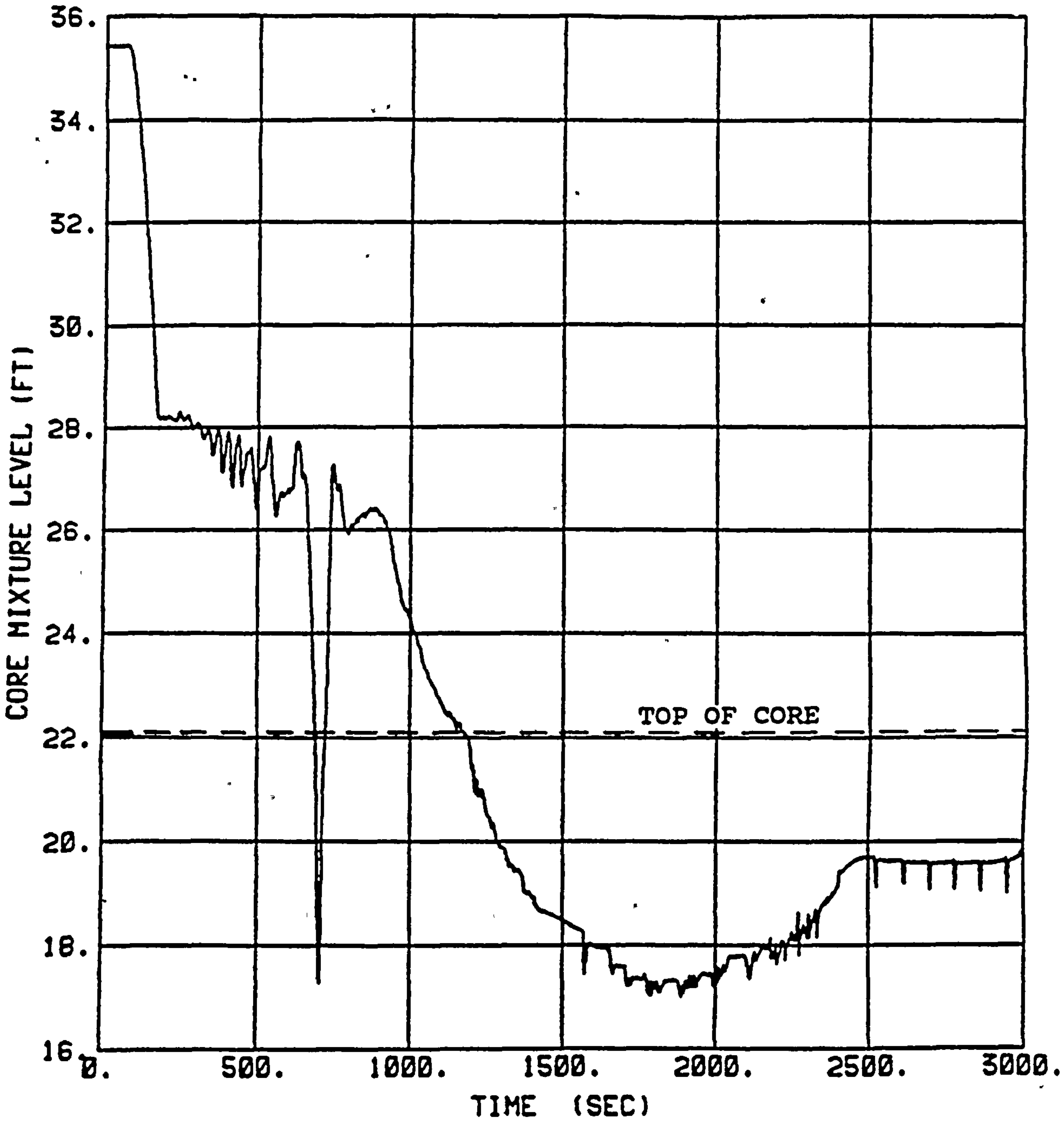

Figure 3.1-45 Core Mixture Height (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

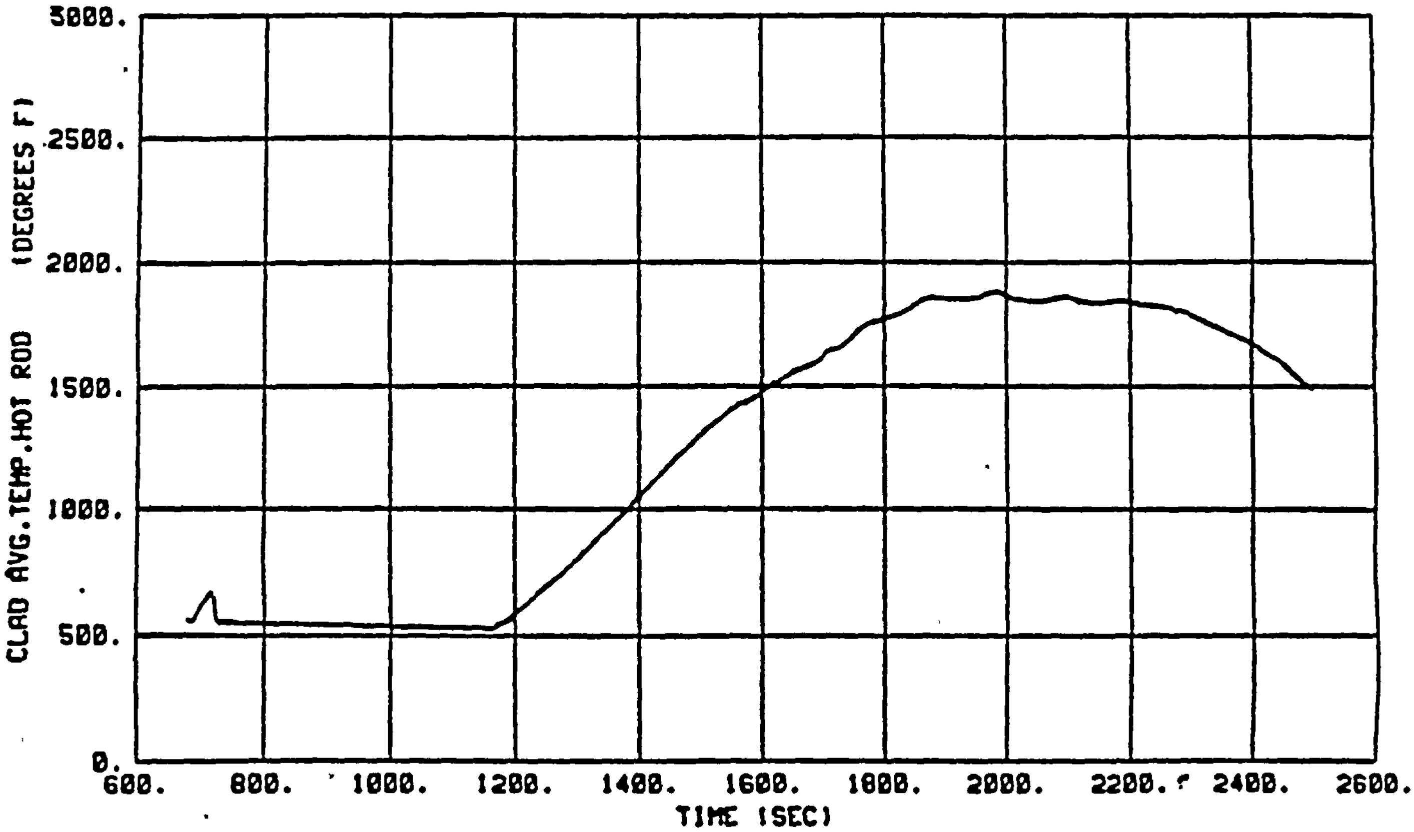

Figure 3.1-46 Hot Spot Clad Temperature (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

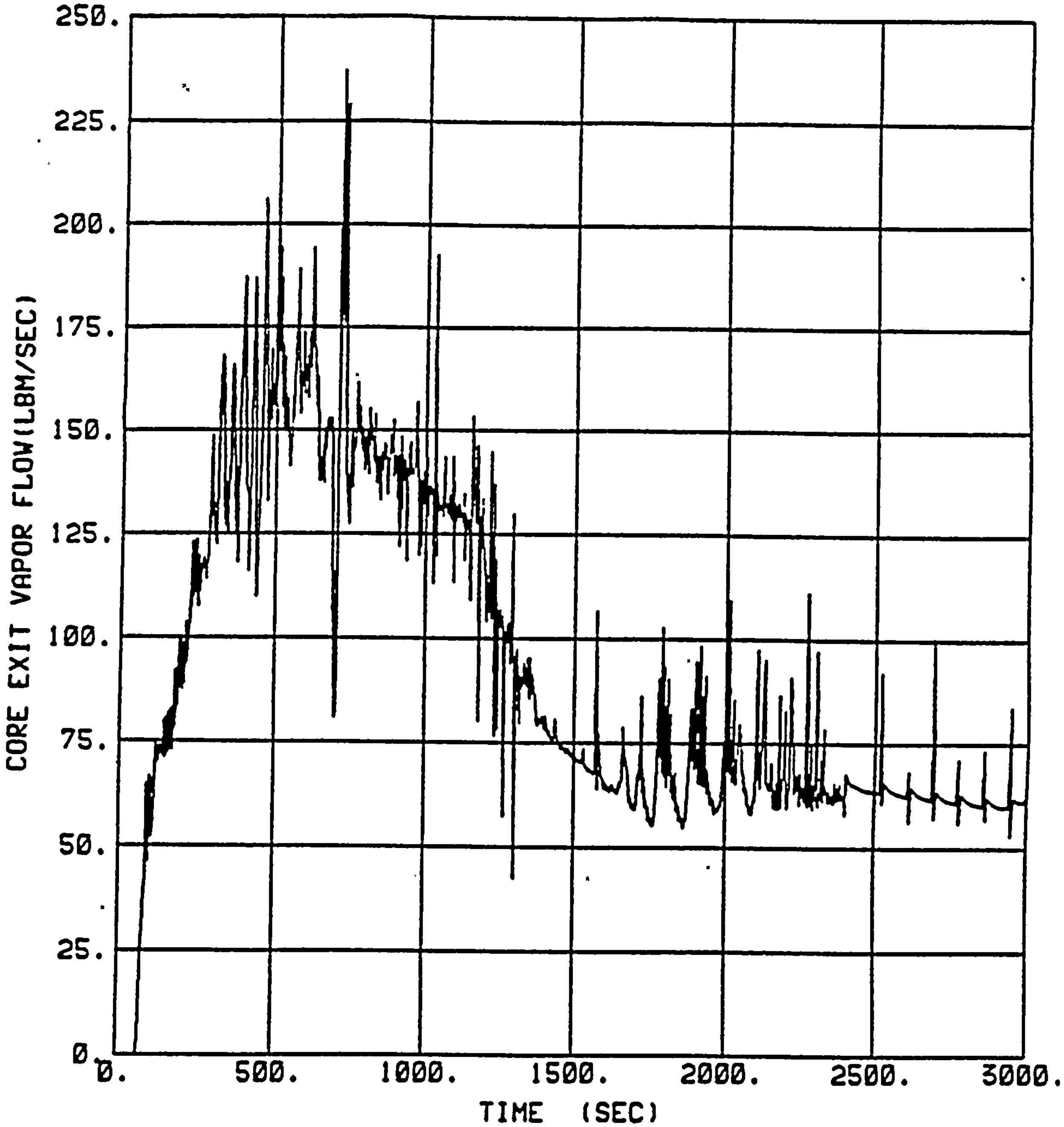

Figure 3.1-47 Core Steam Flow Rate (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

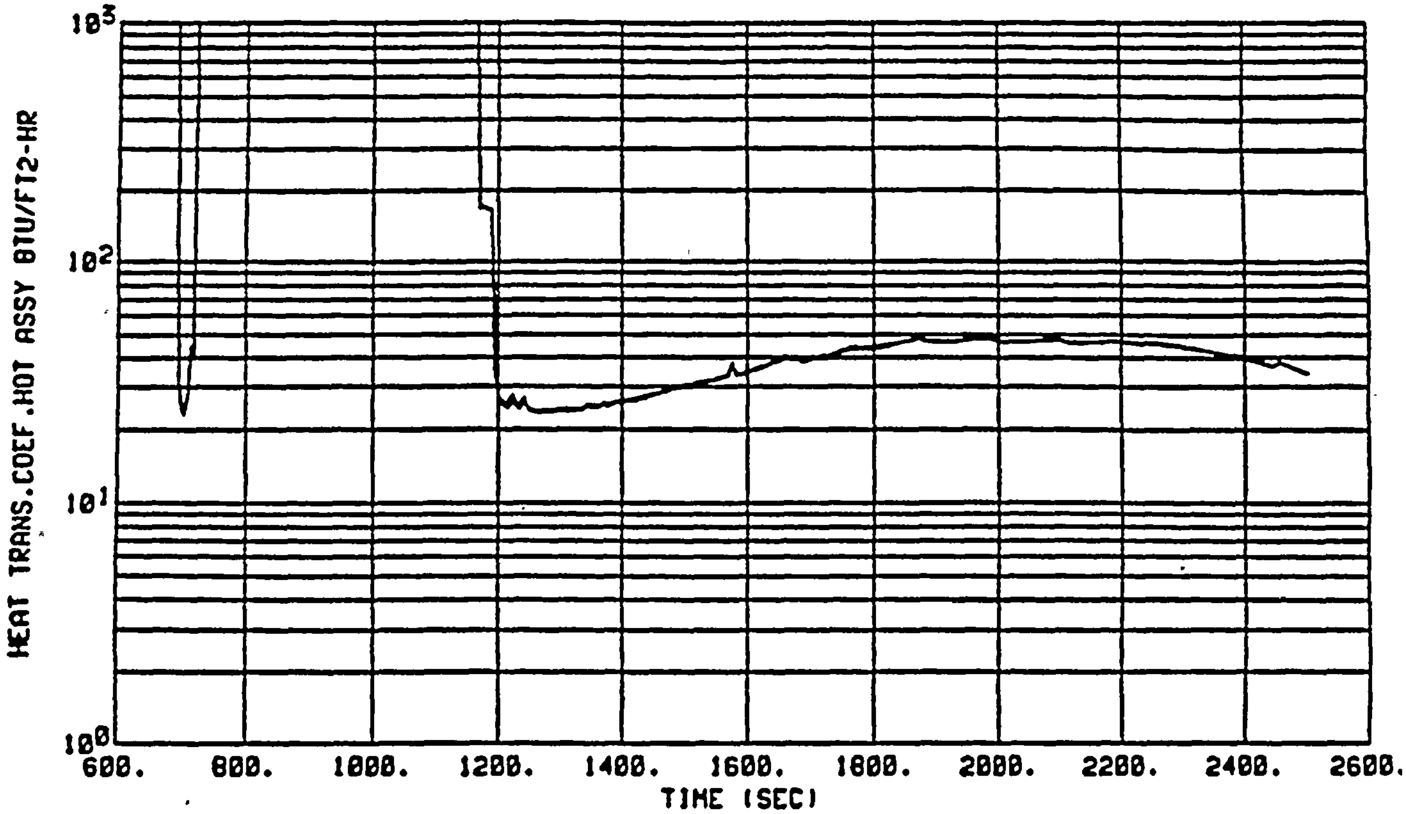

Figure 3.1-48 Hot Spot Heat Transfer Coefficient (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

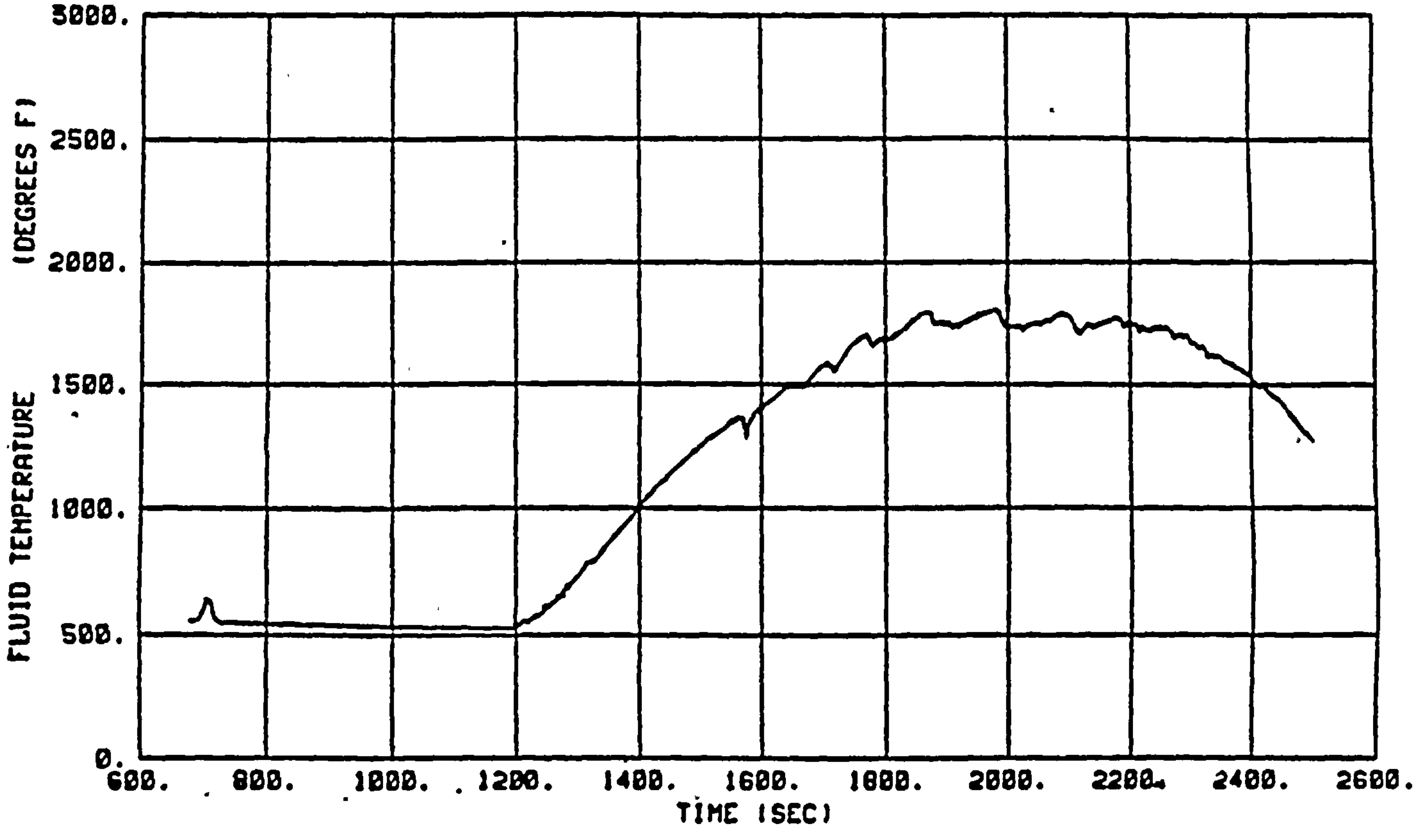

Figure 3.1-49 Hot Spot Fluid Temperature (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

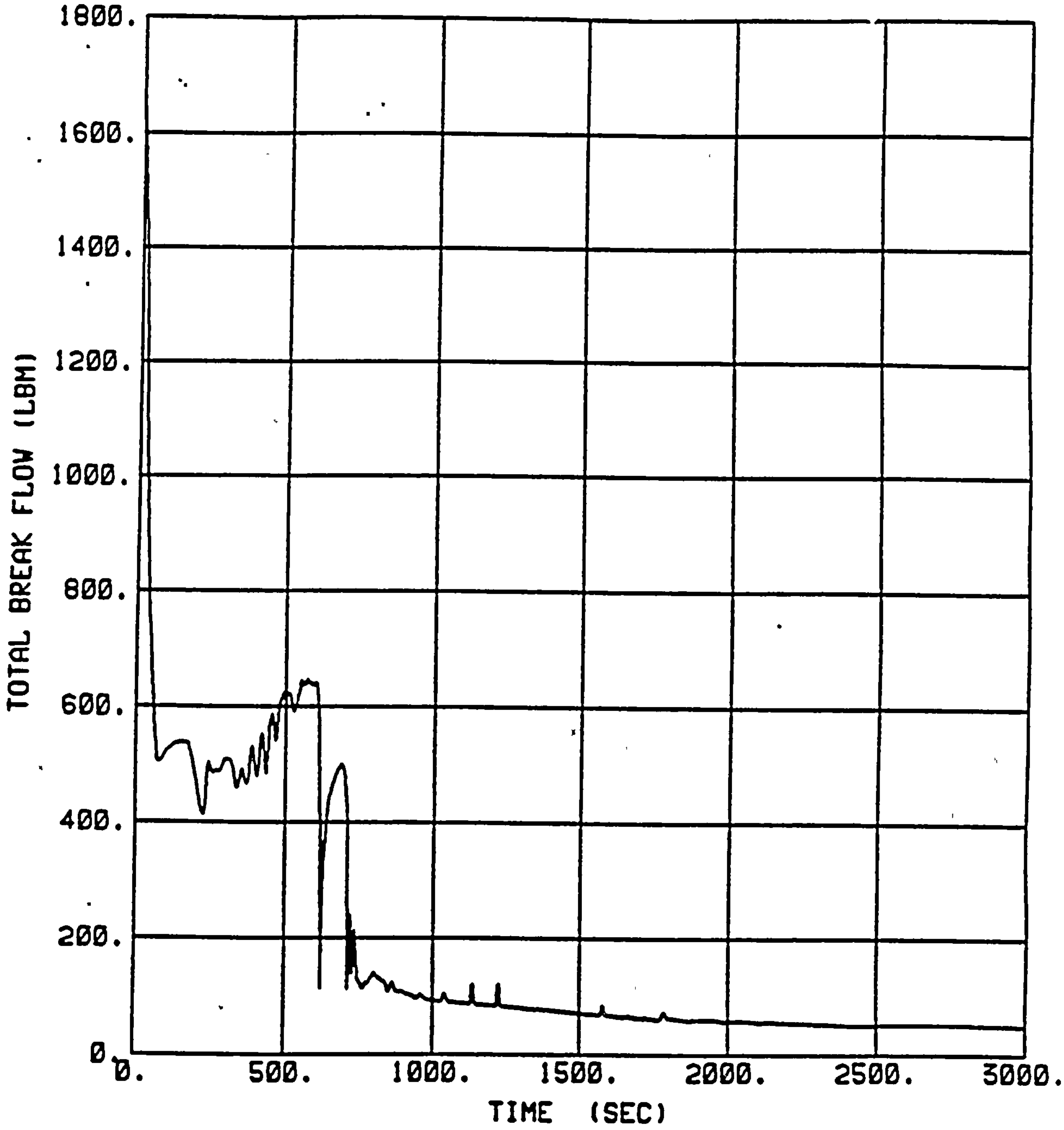

Figure 3.1-50 Total Break Flow (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

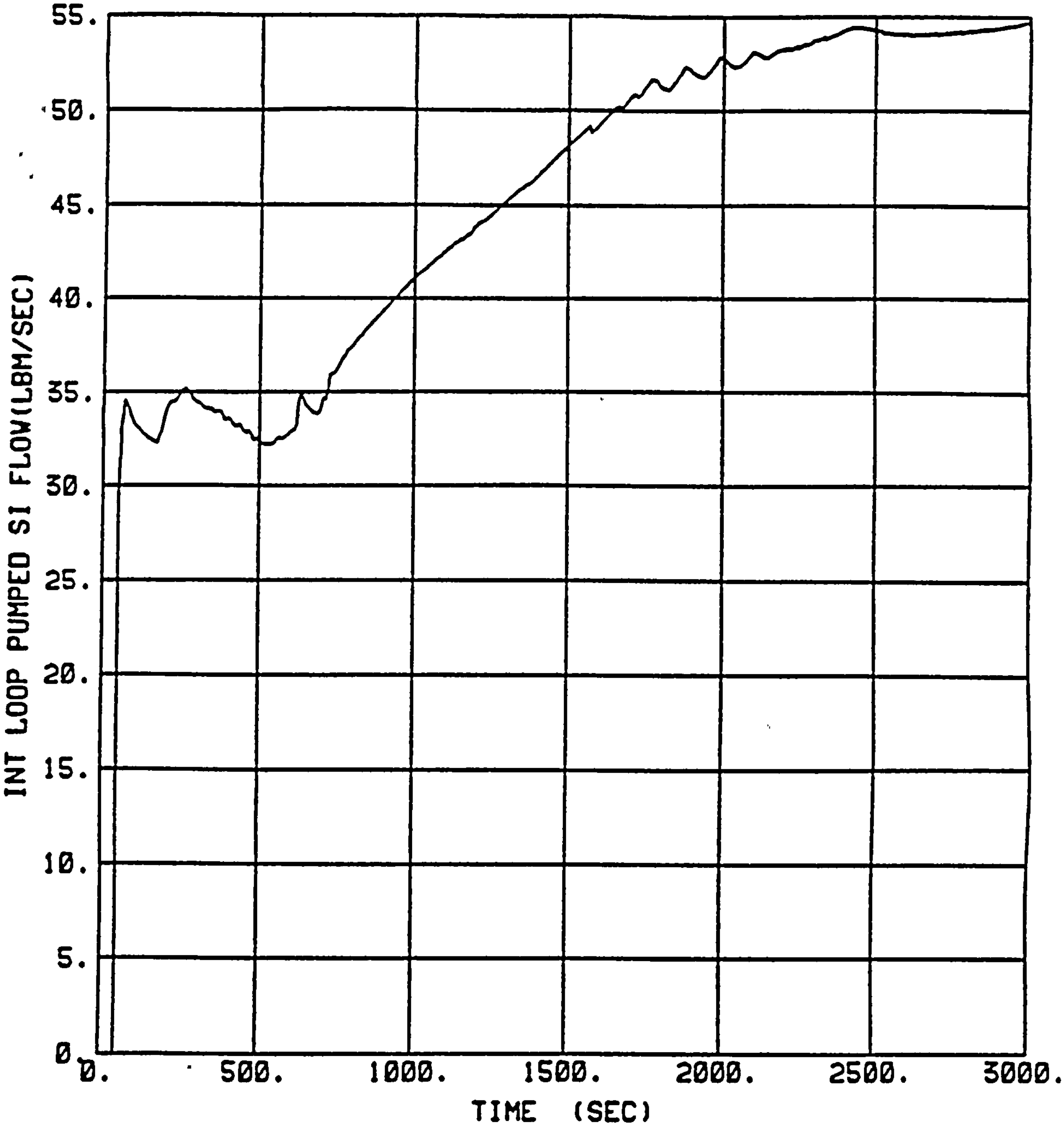

Figure 3.1-51 Intact Loop Pumped SI Flow (3 Inch)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

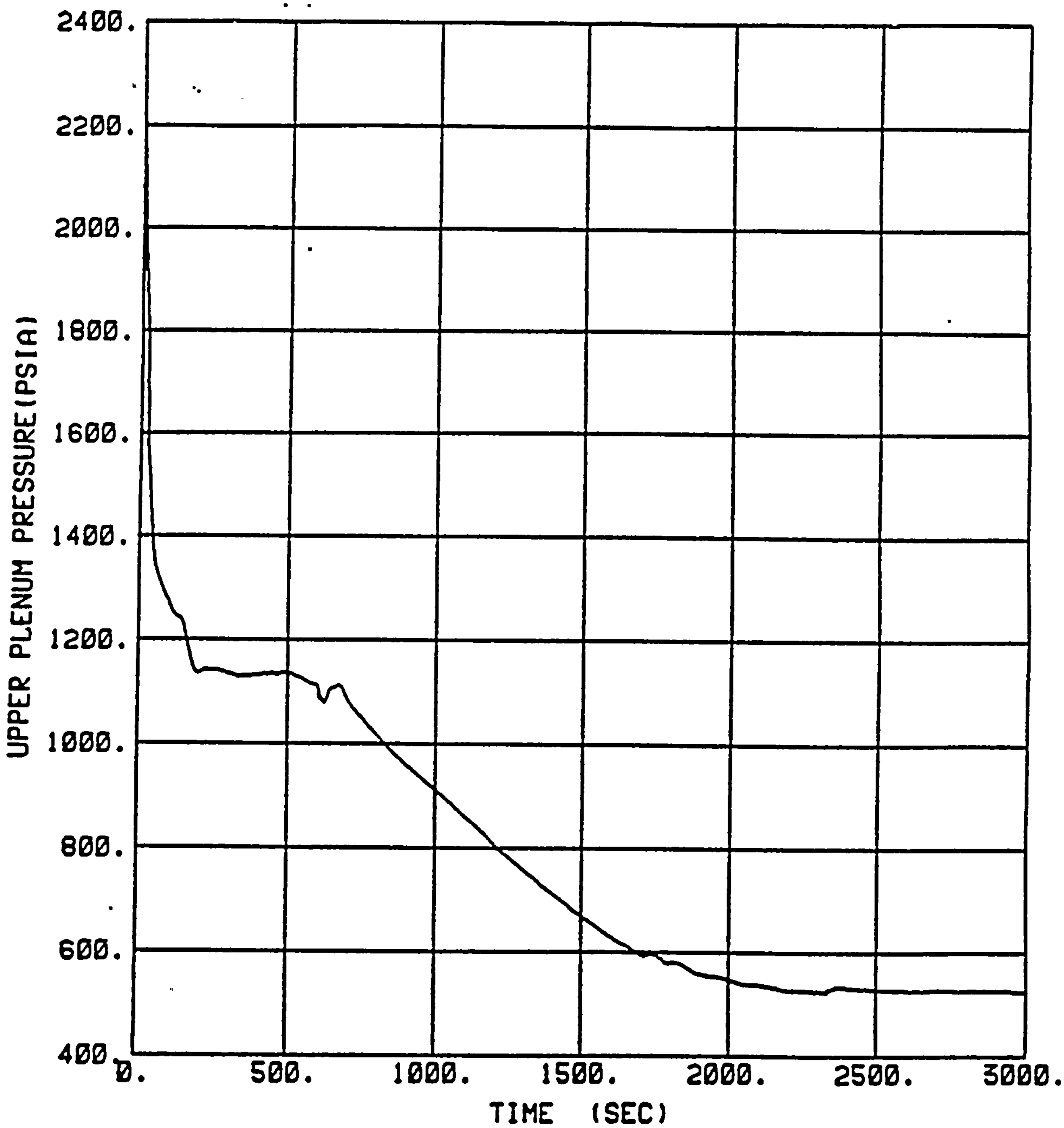

Figure 3.1-52 RCS Pressure (3 Inch)
High Temperature, High Pressure
Donald C. Cook Unit 1

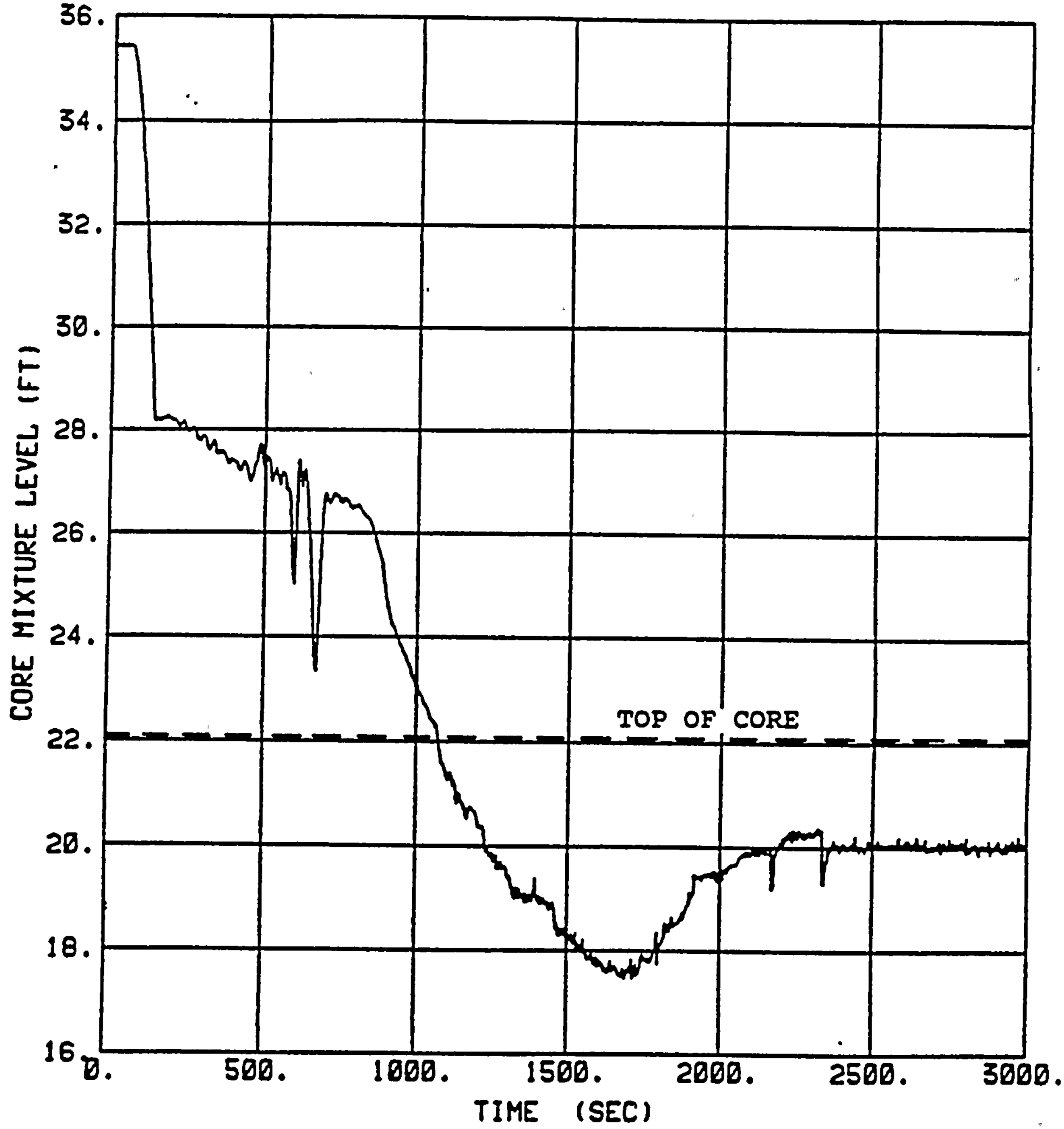

Figure 3.1-53 Core Mixture Height (3 Inch)
High Temperature, High Pressure
Donald C. Cook Unit 1

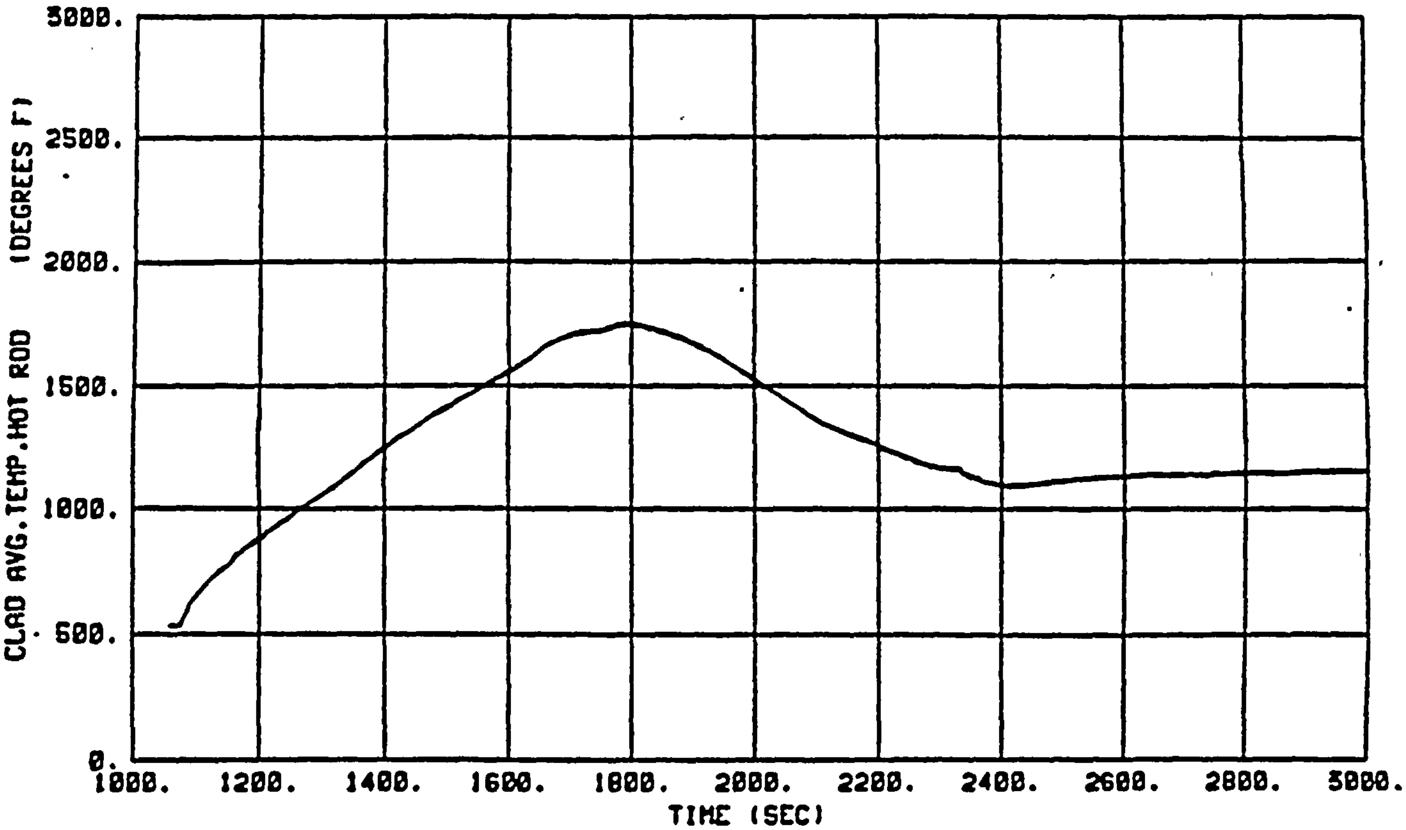

Figure 3.1-54 Hot Spot Clad Temperature (3 Inch)
High Temperature, High Pressure
Donald C. Cook Unit 1

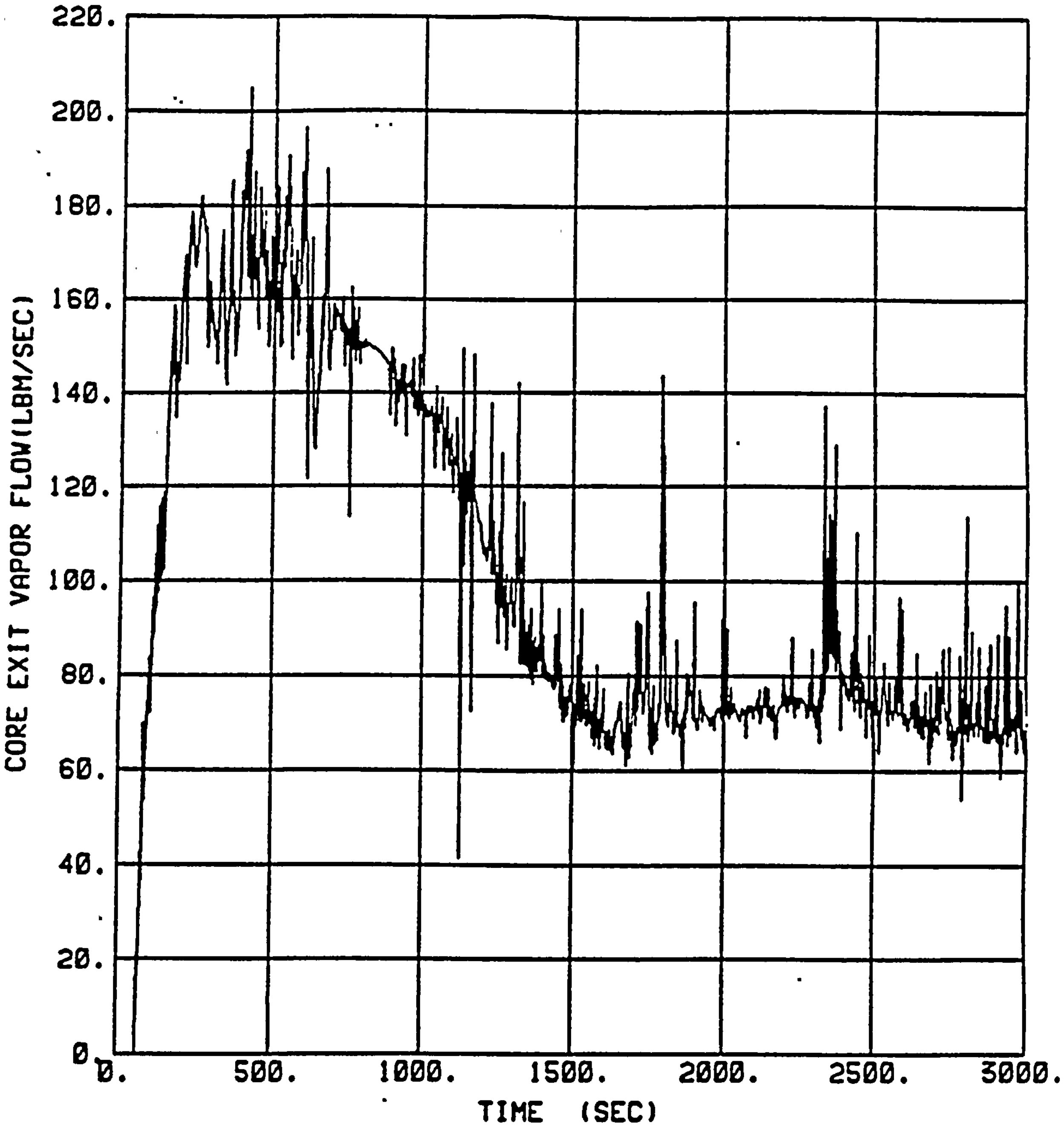

Figure 3.1-55 Core Steam Flowrate (3 Inch)
High Temperature, High Pressure
Donald C. Cook Unit 1

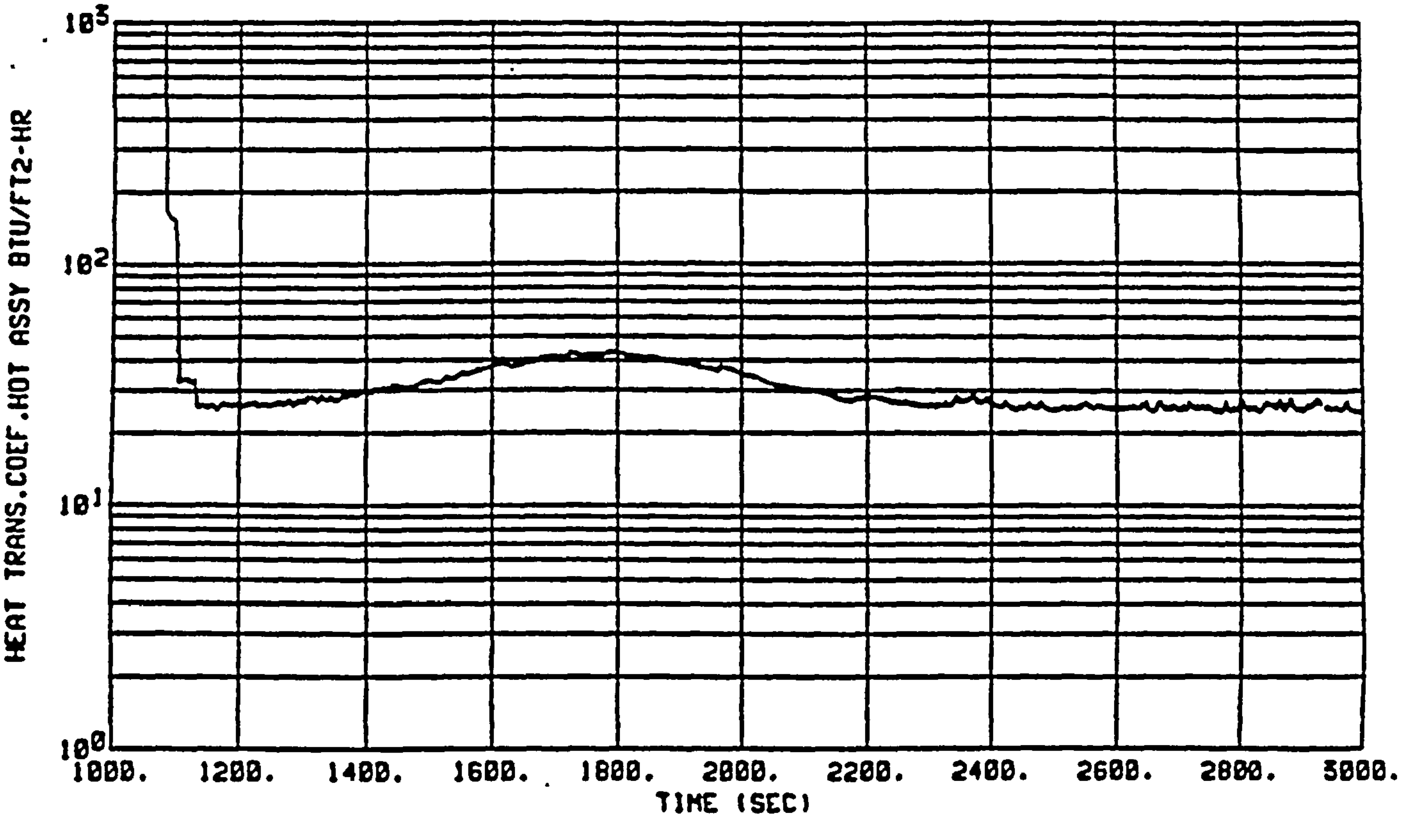

Figure 3.1-56 Hot Spot Heat Transfer Coefficient (3 Inch)
High Temperature, High Pressure
Donald C. Cook Unit 1

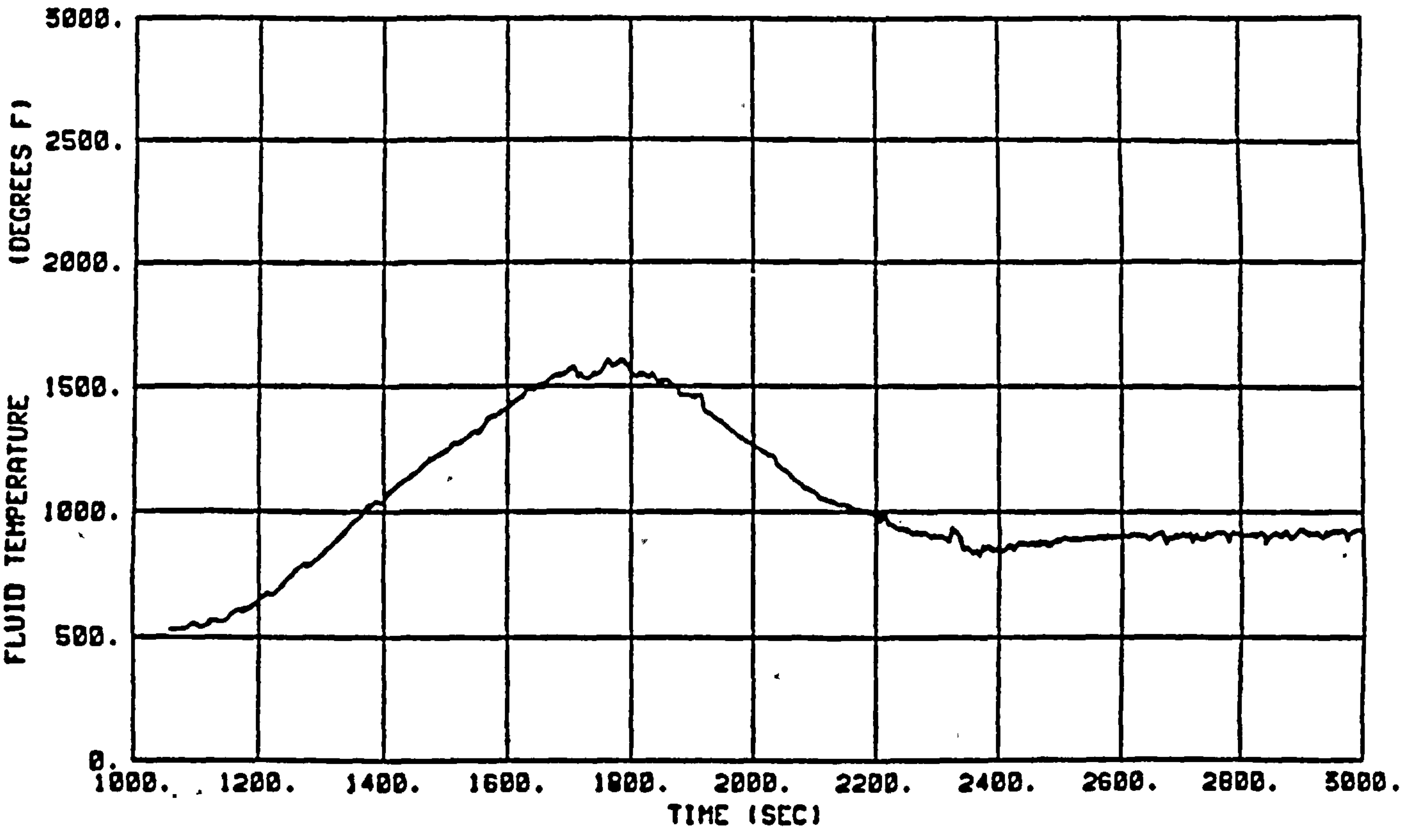

Figure 3.1-57 Hot Spot Fluid Temperature (3 Inch)
High Temperature, High Pressure
Donald C. Cook Unit 1

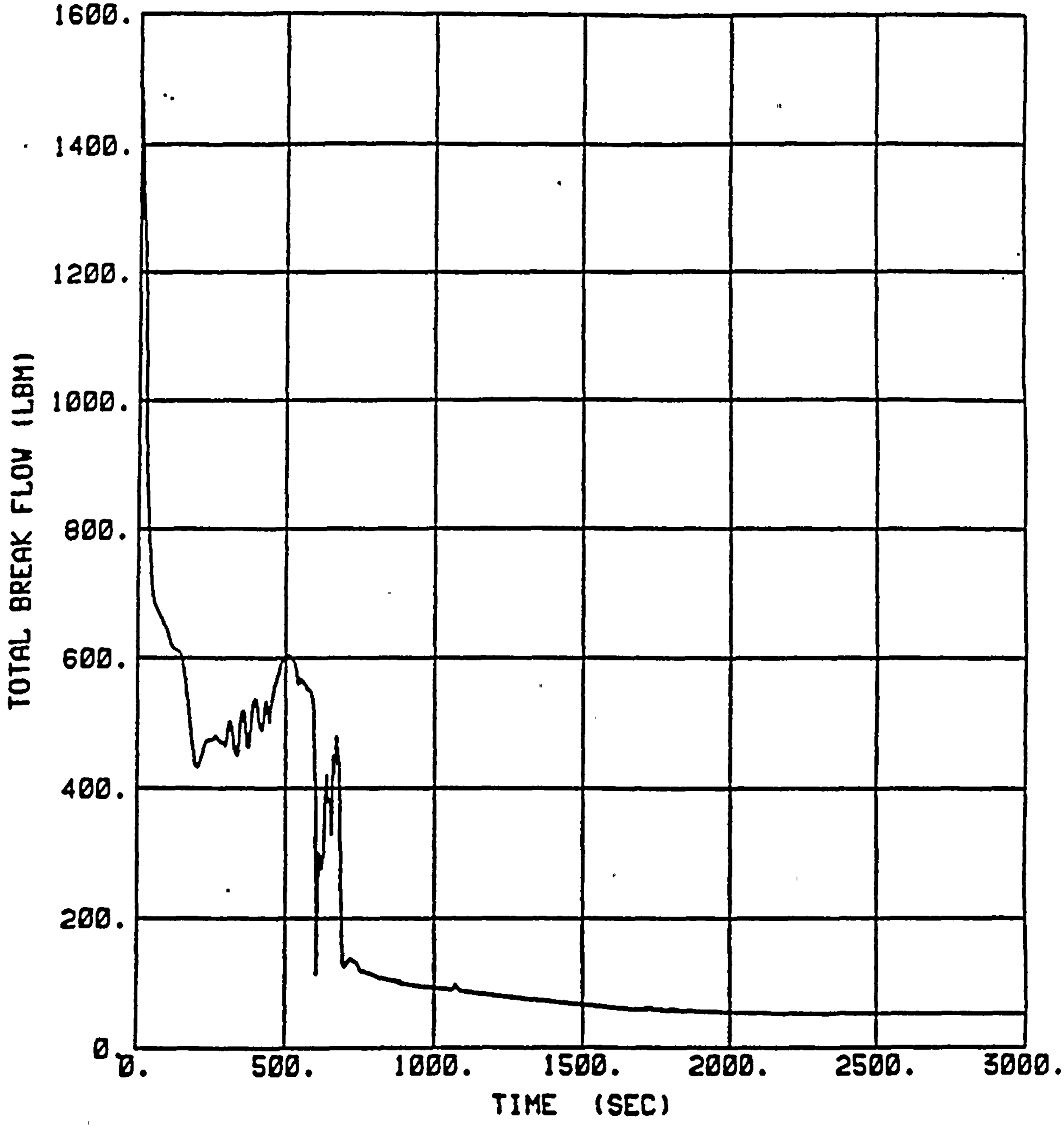

Figure 3.1-58 Total Break Flow (3 Inch)
High Temperature, High Pressure
Donald C. Cook Unit 1

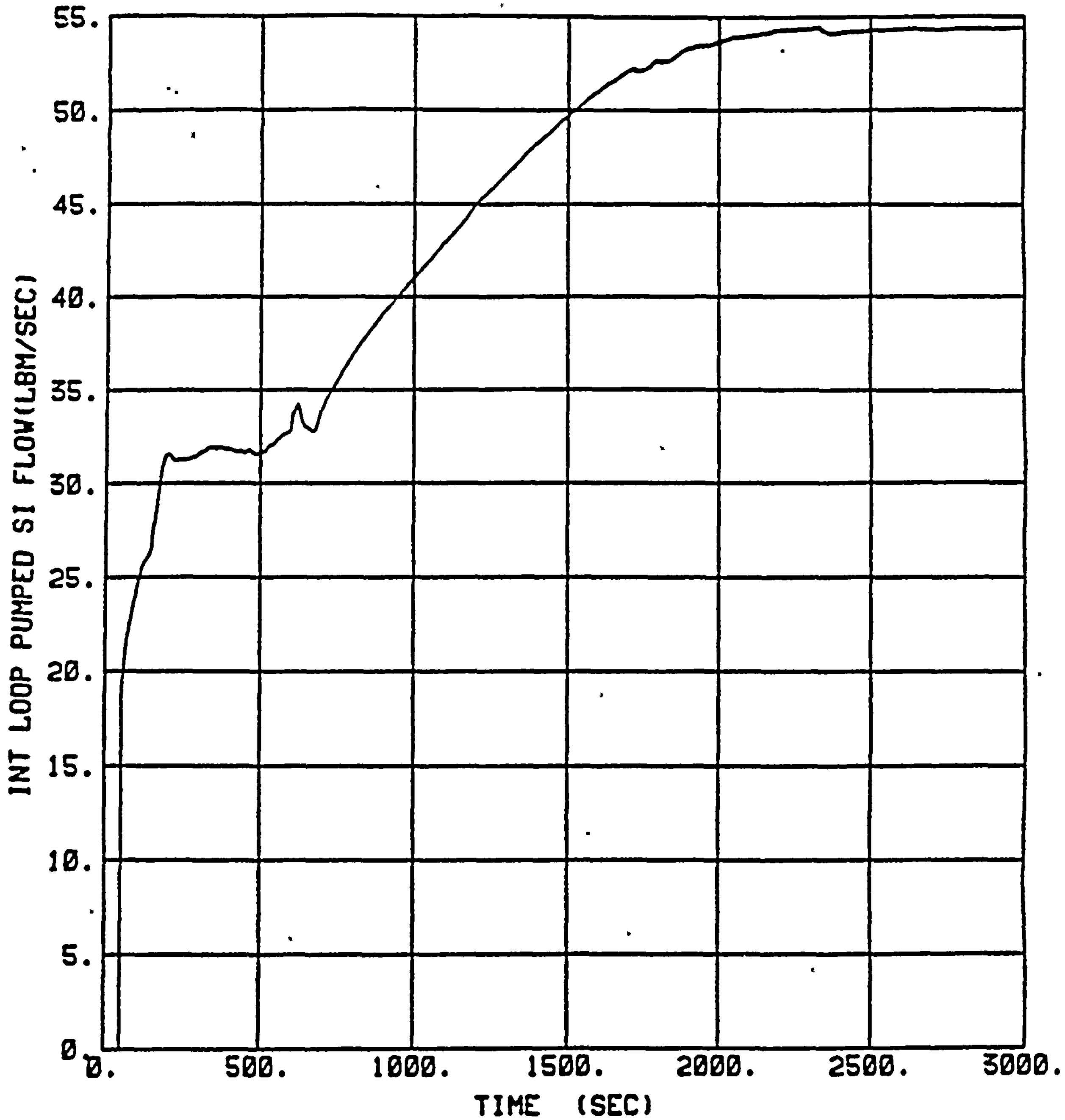

Figure 3.1-59 Intact Loop Pumped SI Flow (3 Inch)
High Temperature, High Pressure
Donald C. Cook Unit 1

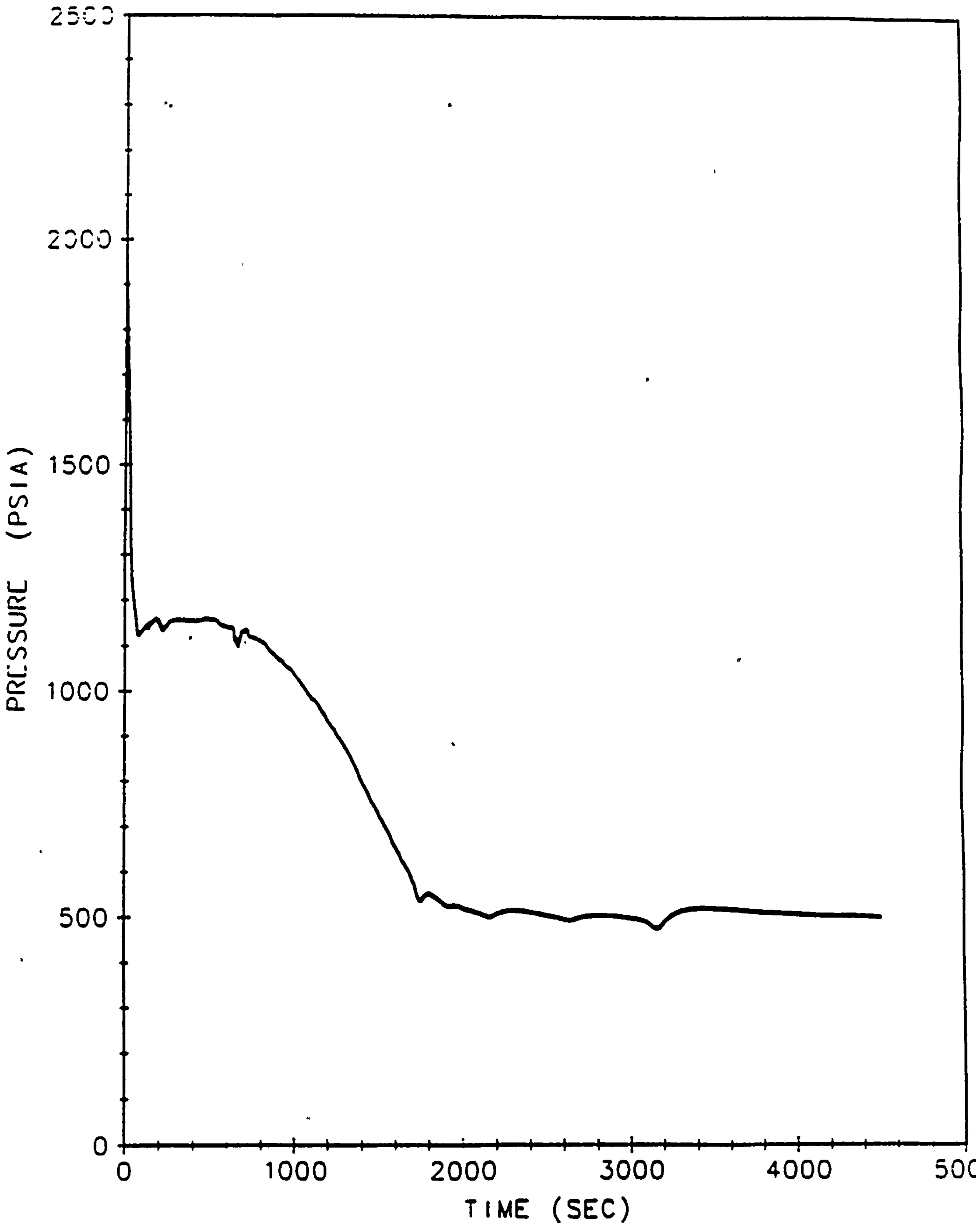

Figure 3.1-60 RCS Pressure (3 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

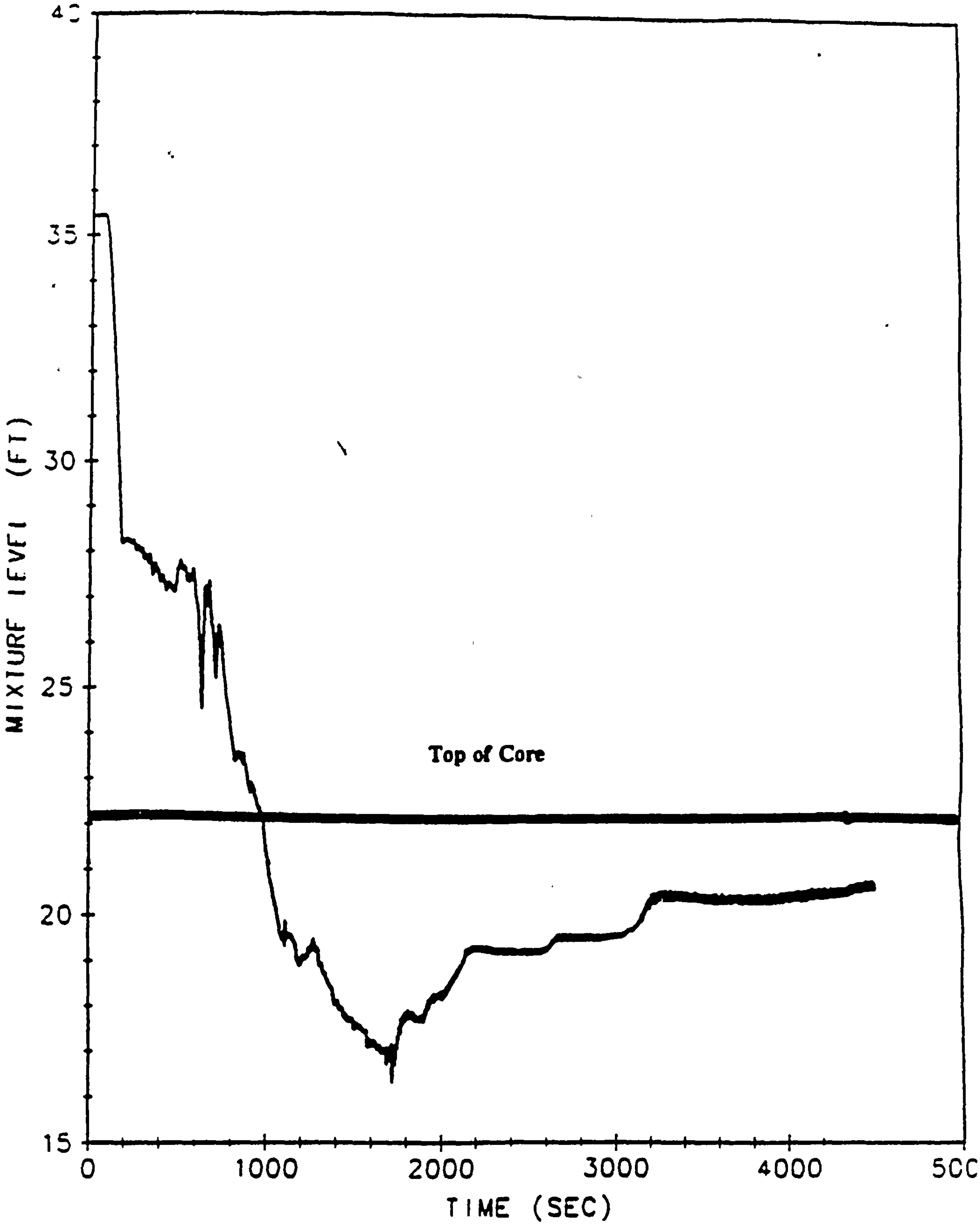

Figure 3.1-61 Core Mixture Level (3 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

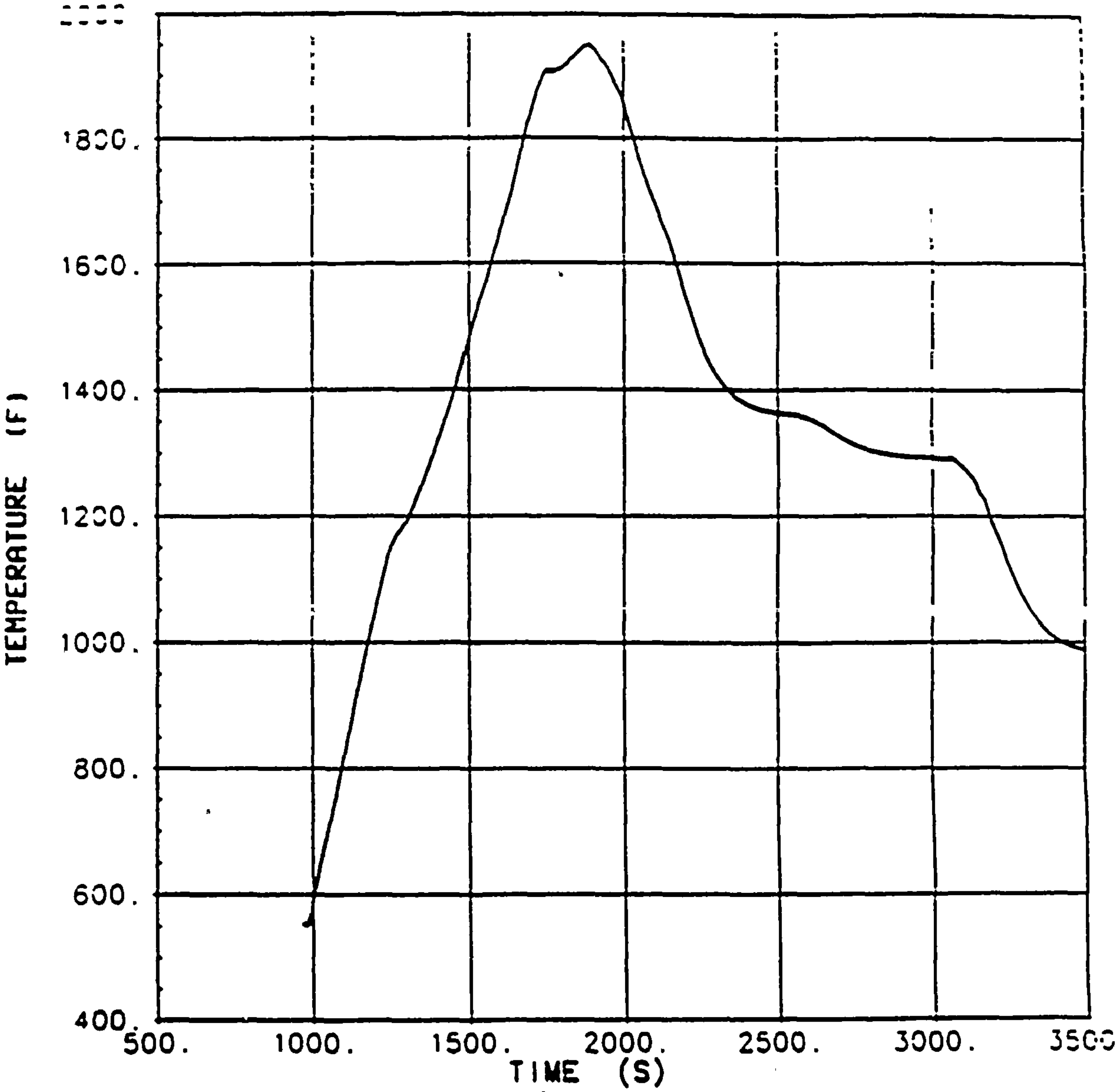

Figure 3.1-62 Peak Clad Temperature (3 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

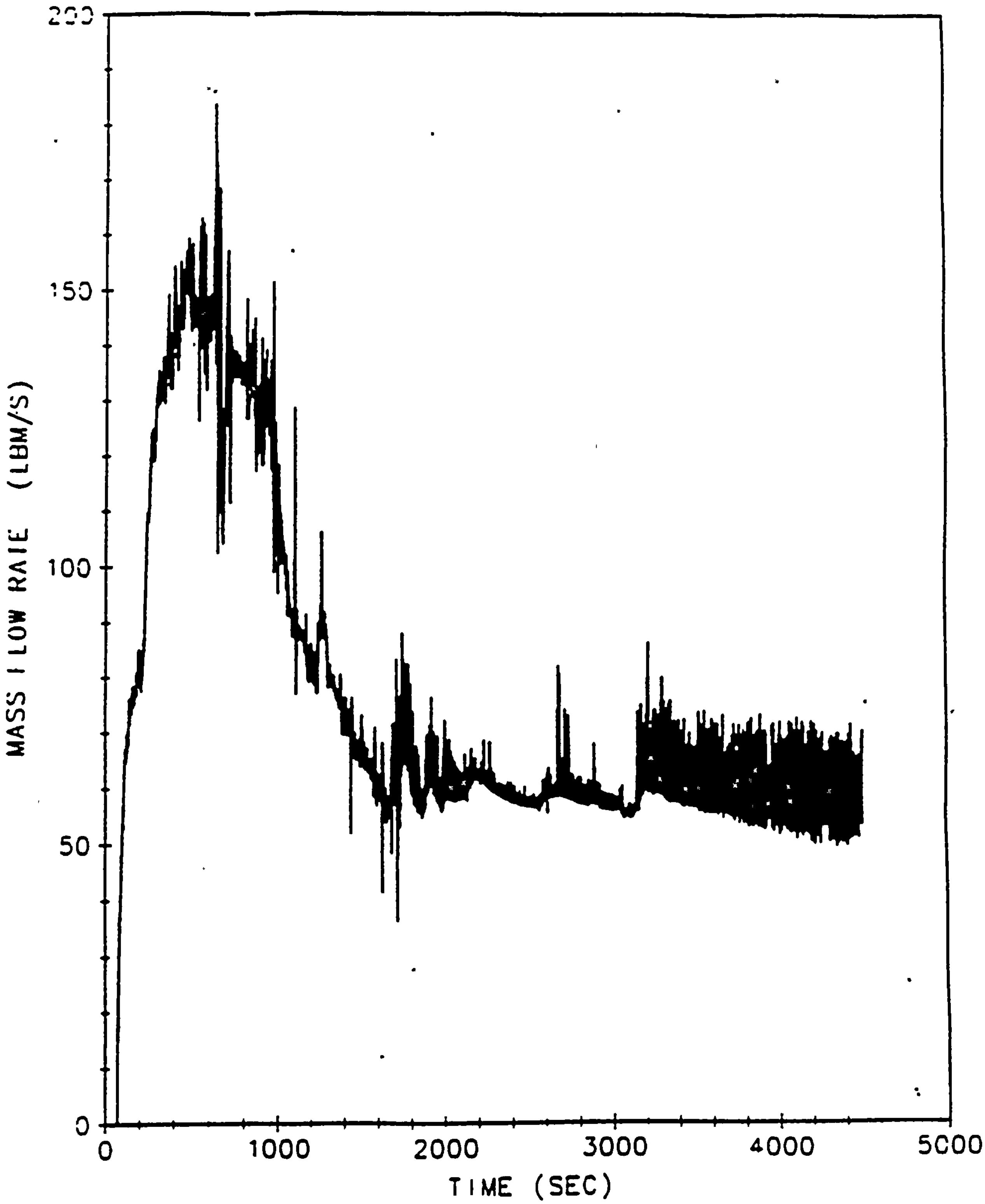

Figure 3.1-63 Core Outlet Steam FLow Rate (3 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

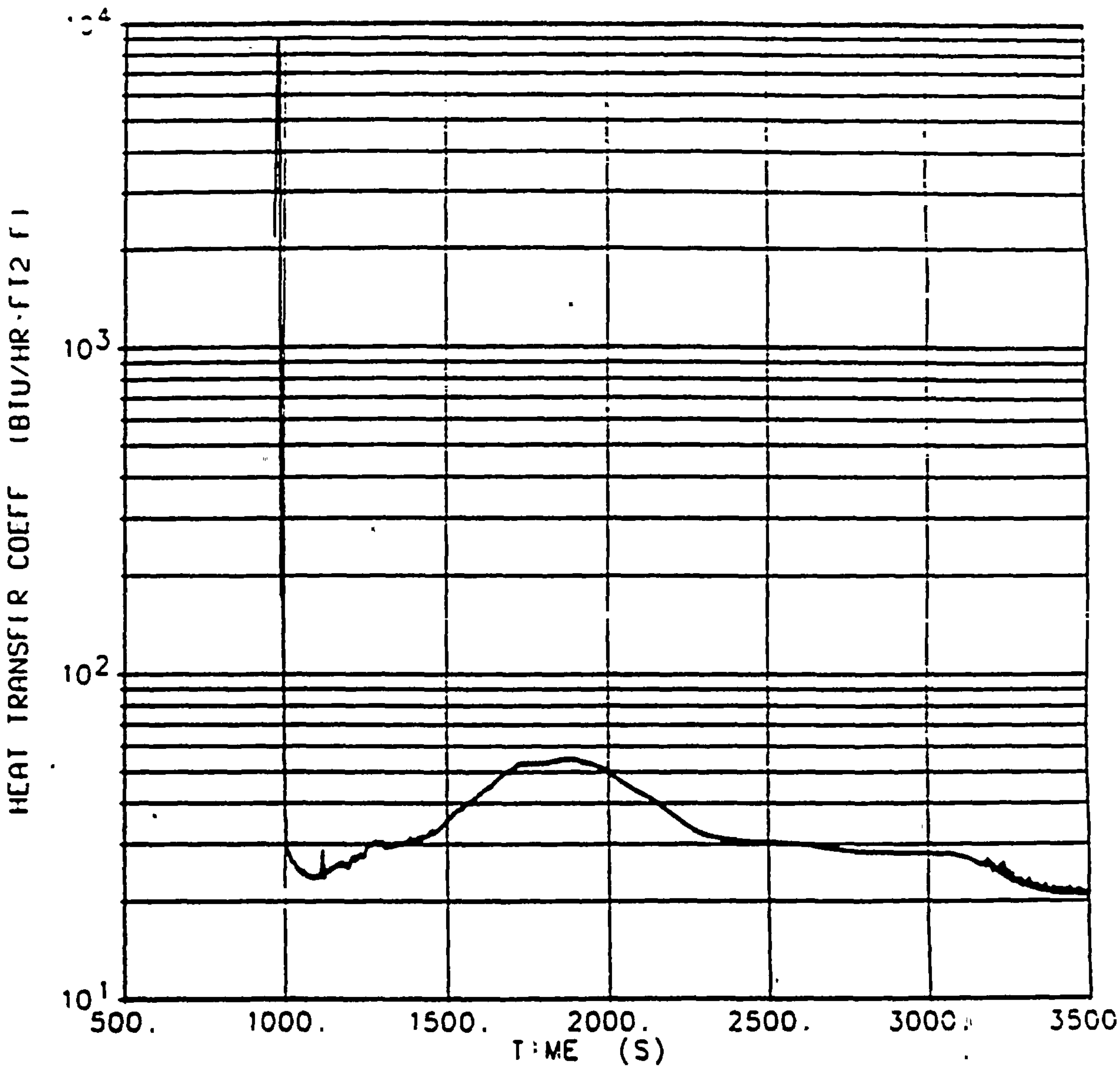

Figure 3.1-64 Hot Spot Rod Surface Heat Transfer Coefficient (3 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

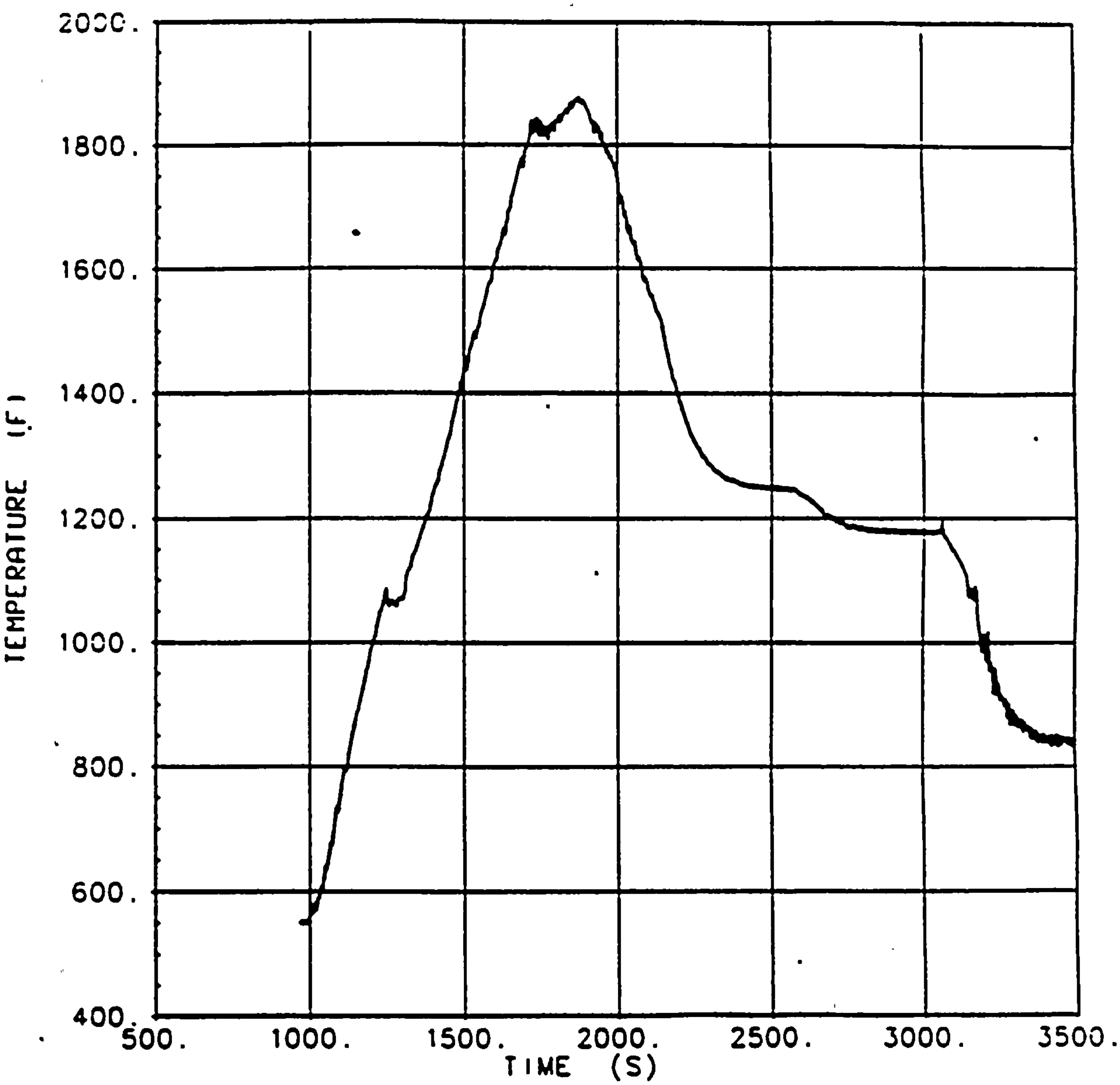

Figure 3.1-65 Hot Spot FLuid Temperature (3 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

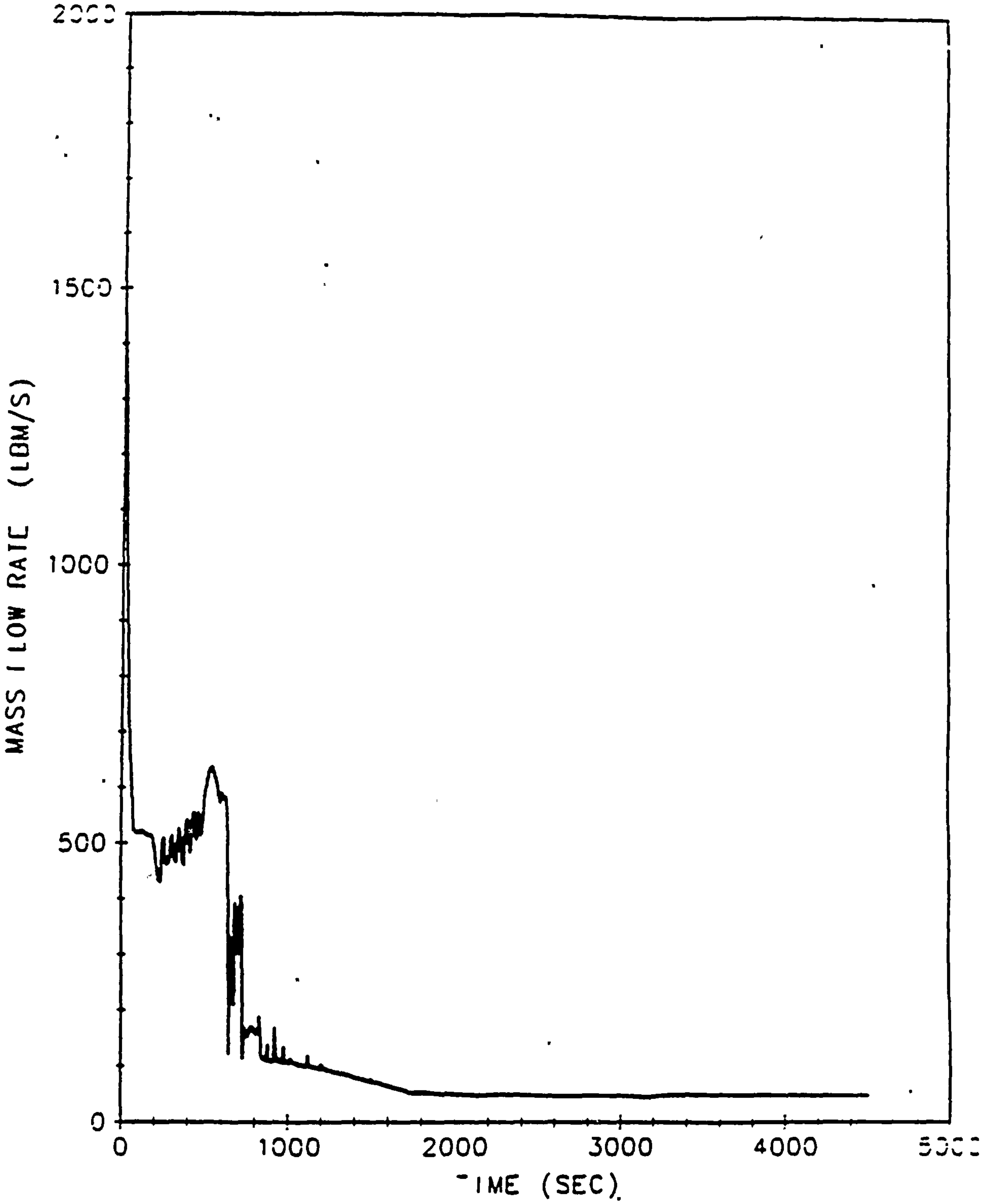

Figure 3.1-66 Cold Leg Break Mass Flow Rate (3 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

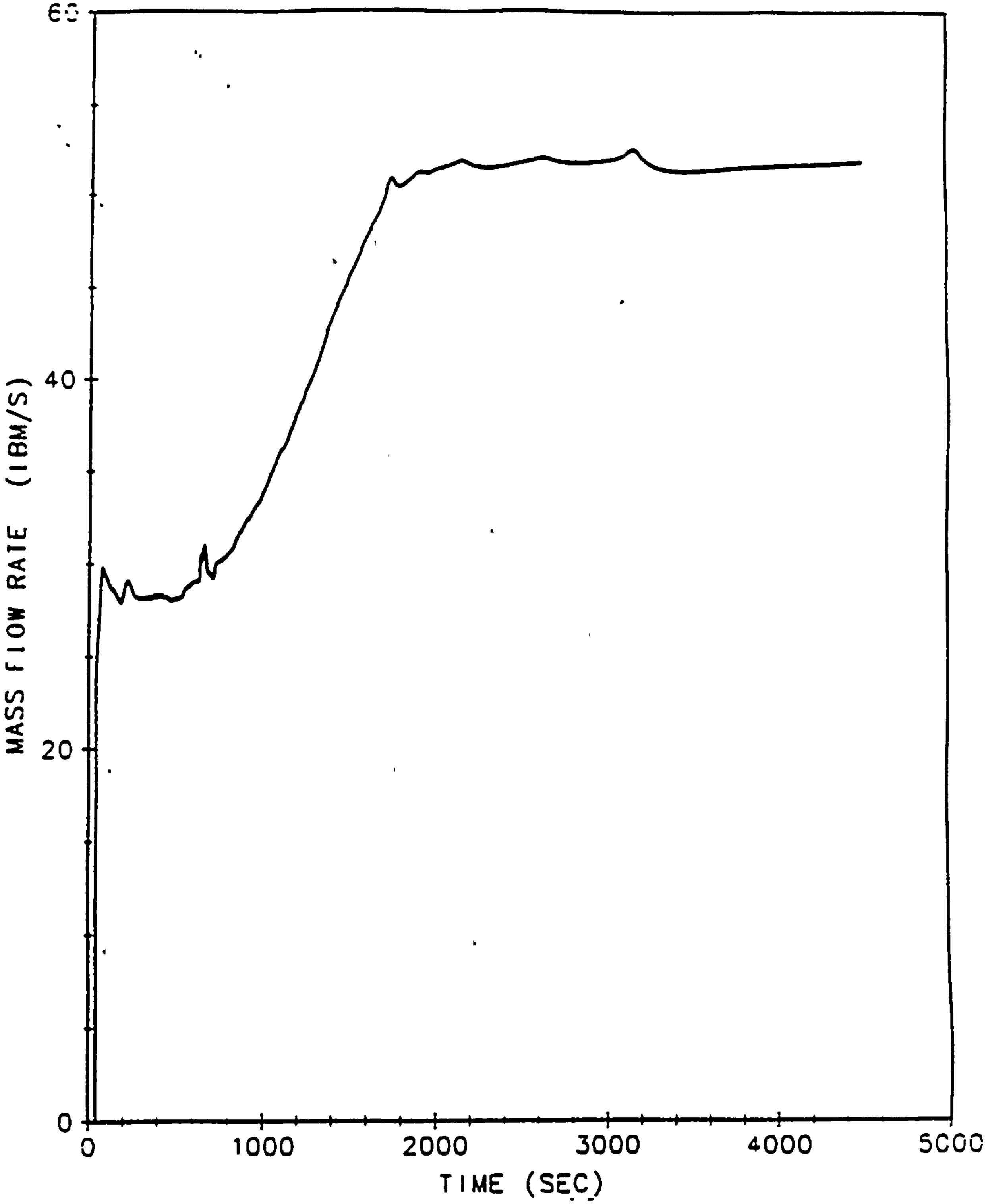

Figure 3.1-67 Safety Injection Mass Flow Rate (3 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

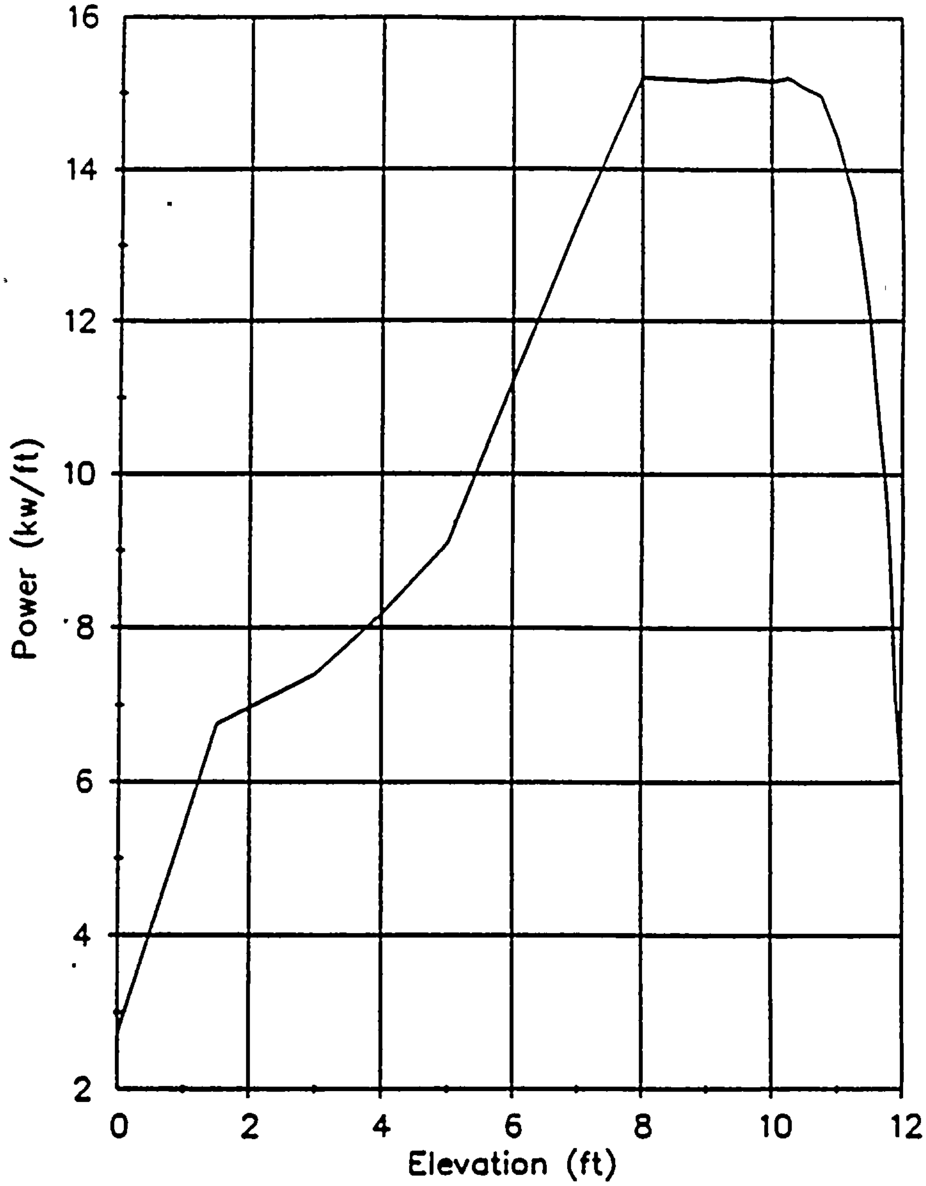

Figure 3.1-68 Hot Rod Power Distribution
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

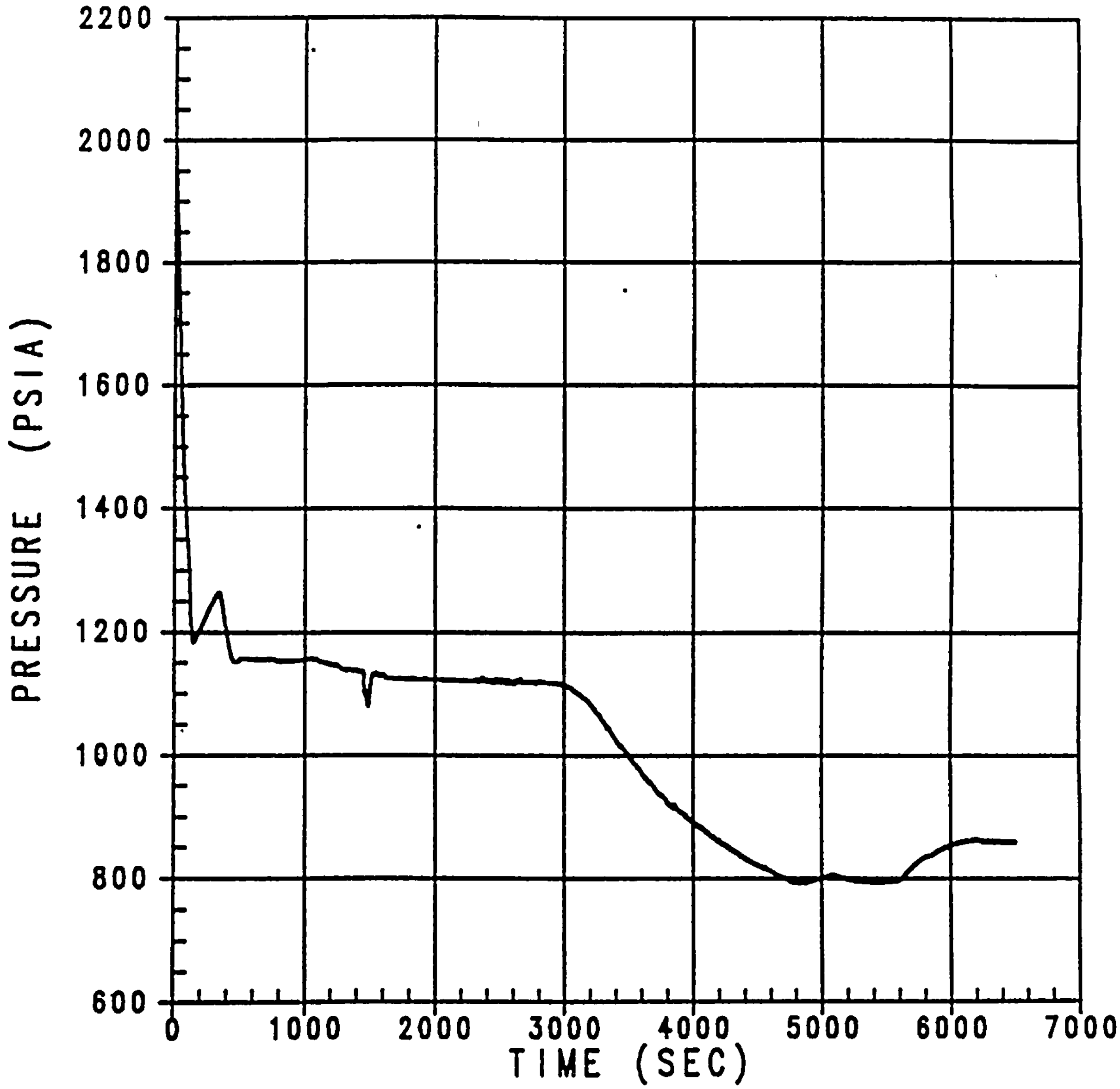

Figure 3.1-69 RCS Pressure (2 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

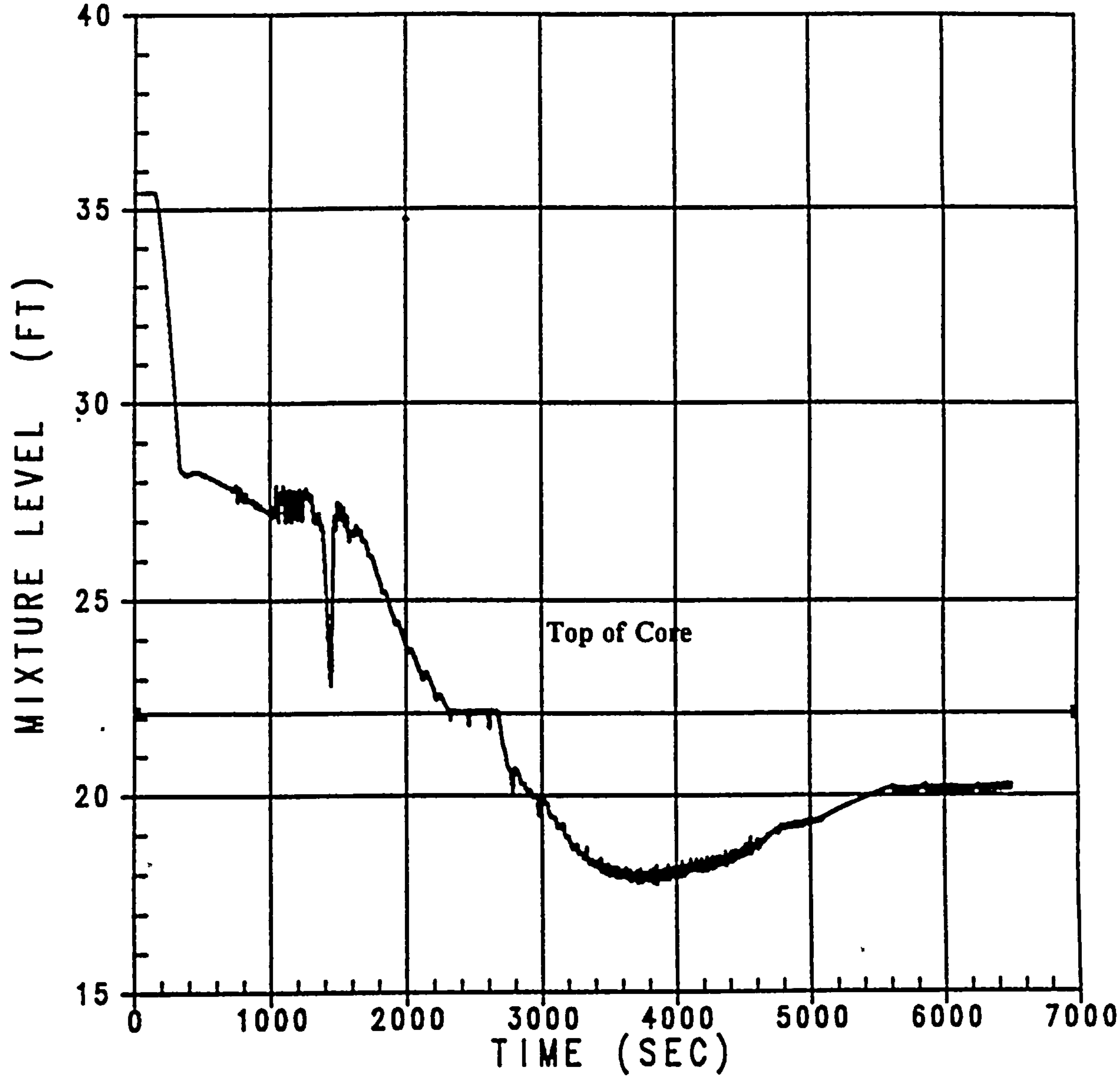

Figure 3.1-70 Core Mixture Level (2 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

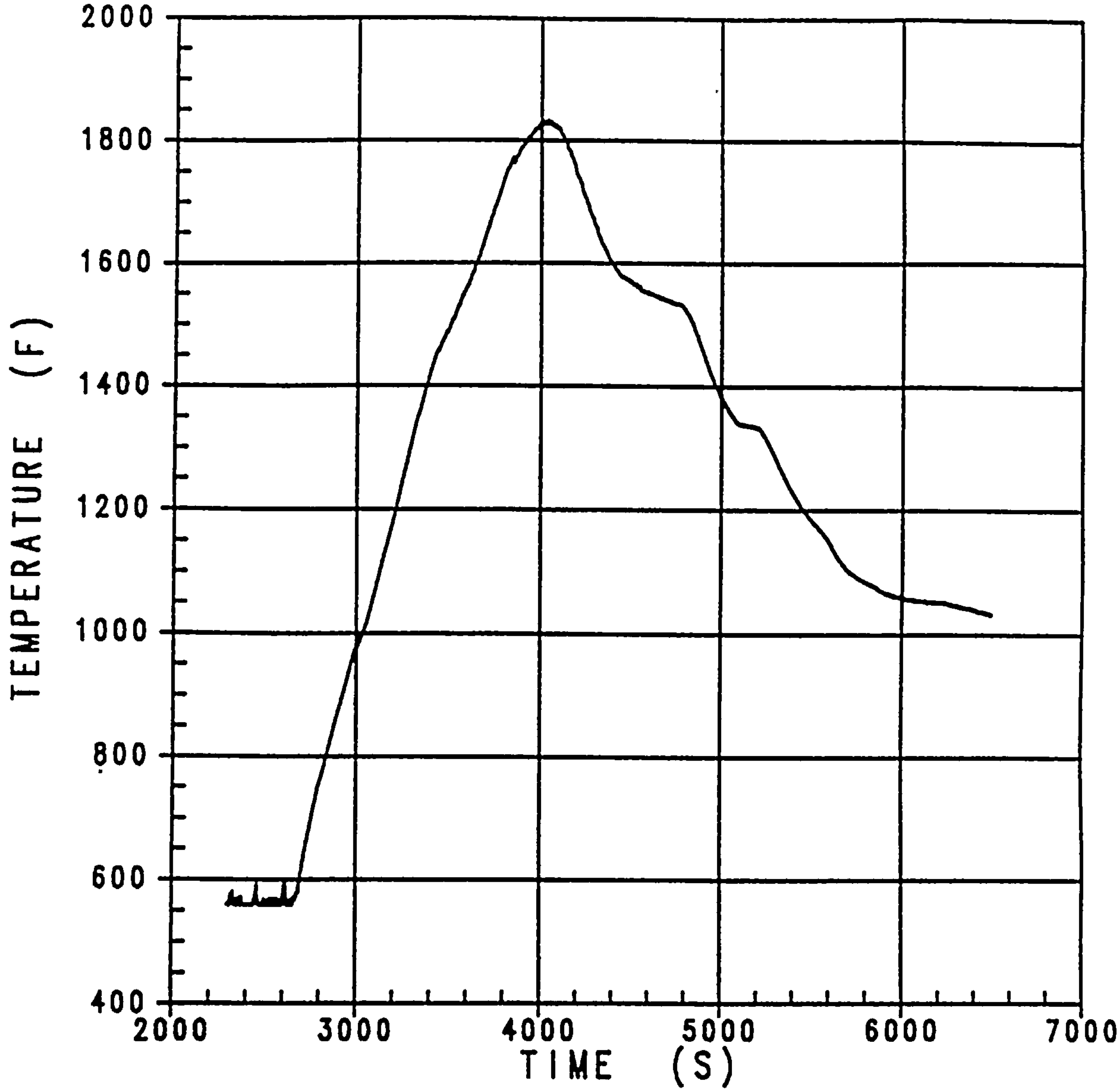

Figure 3.1-71 Peak Clad Temperature (2 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

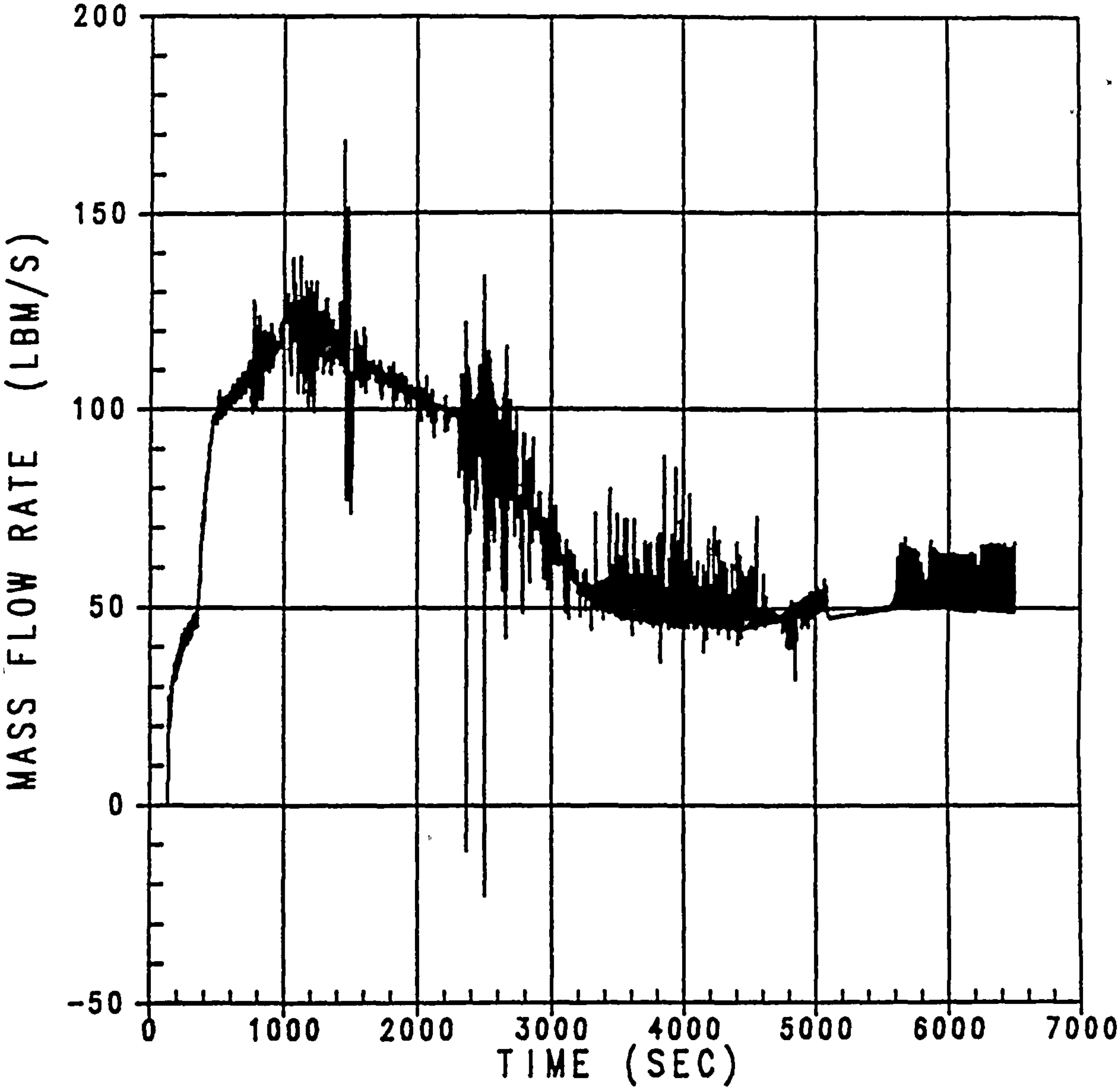

Figure 3.1-72 Core Outlet Steam Flow Rate (2 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

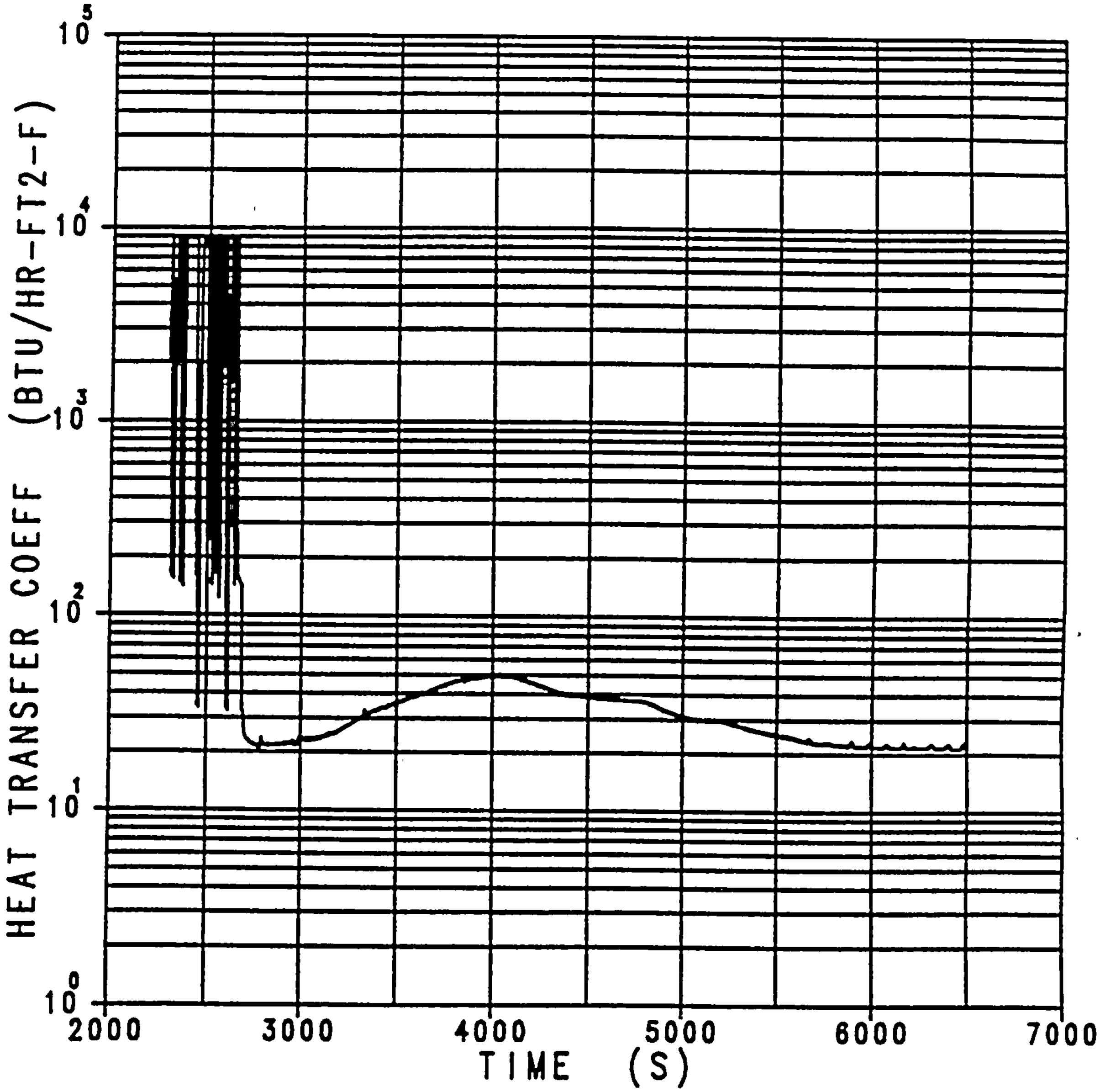

Figure 3.1-73 Hot Spot Rod Surface Heat Transfer Coefficient (2 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

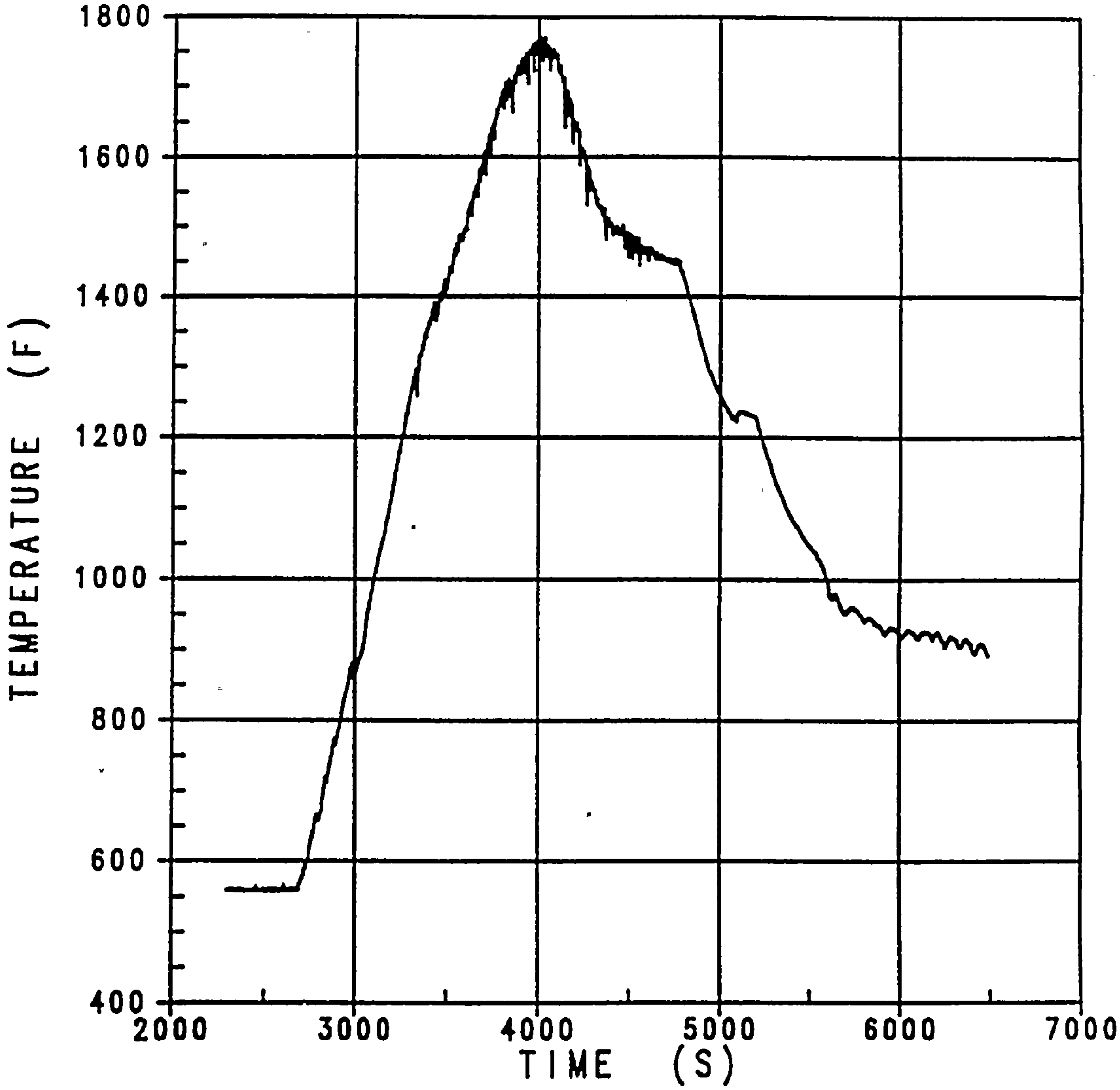

Figure 3.1-74 Hot Spot Fluid Temperature (2 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

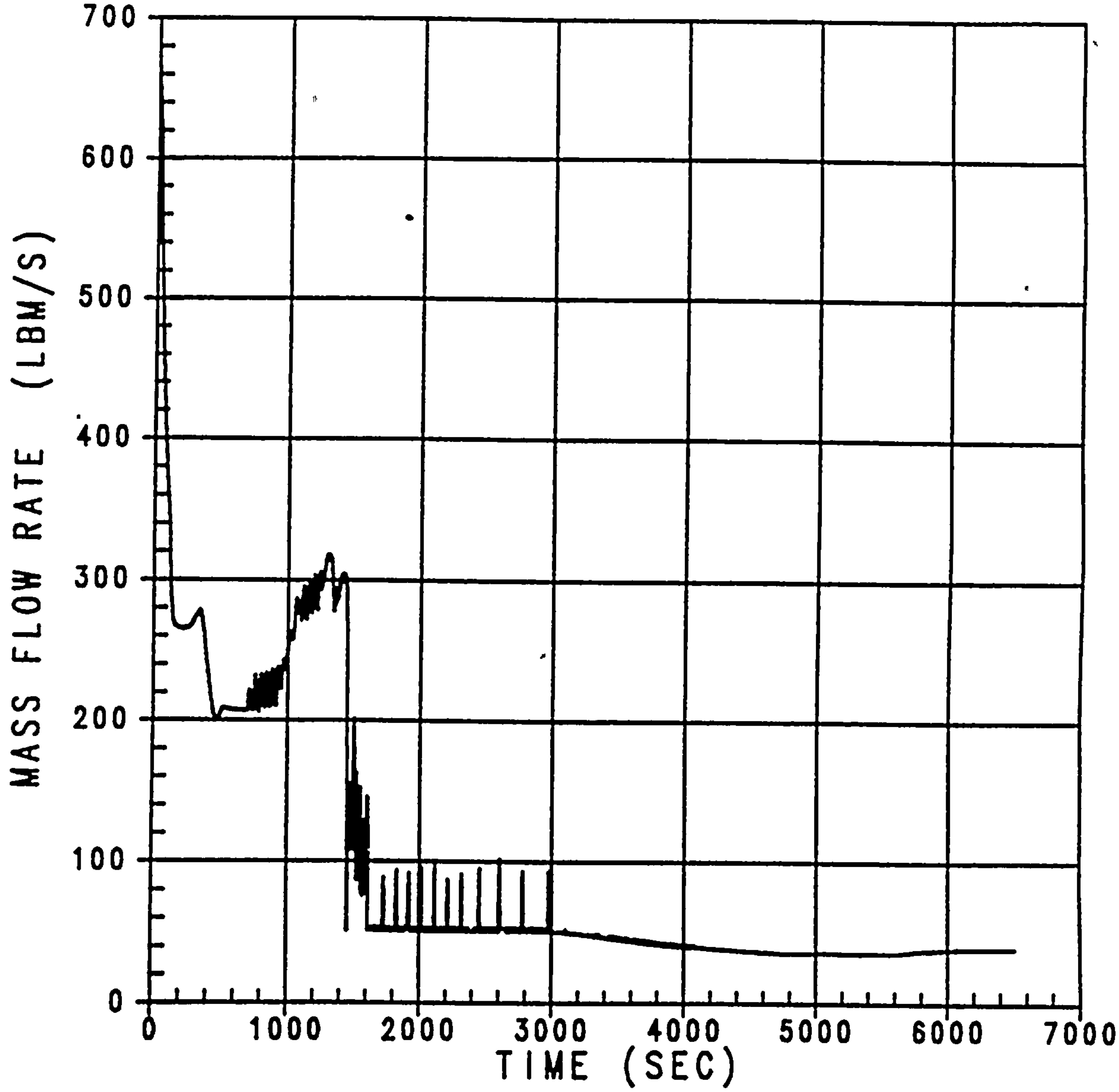

Figure 3.1-75 Cold Leg Break Mass Flow Rate (2 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

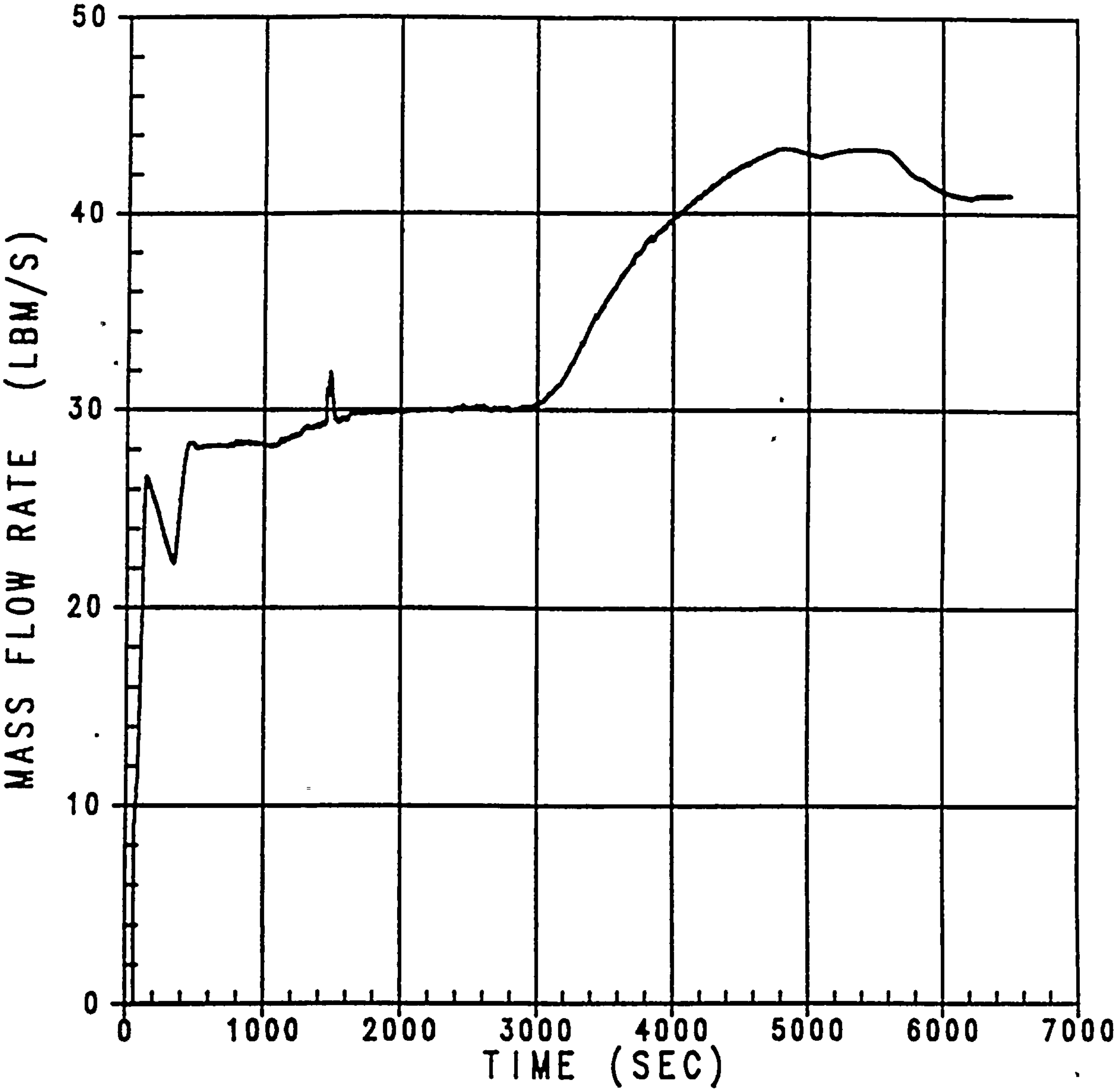

Figure 3.1-76 Safety Injection Mass Flow Rate (2 Inch, 3% MSSV Tolerance)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

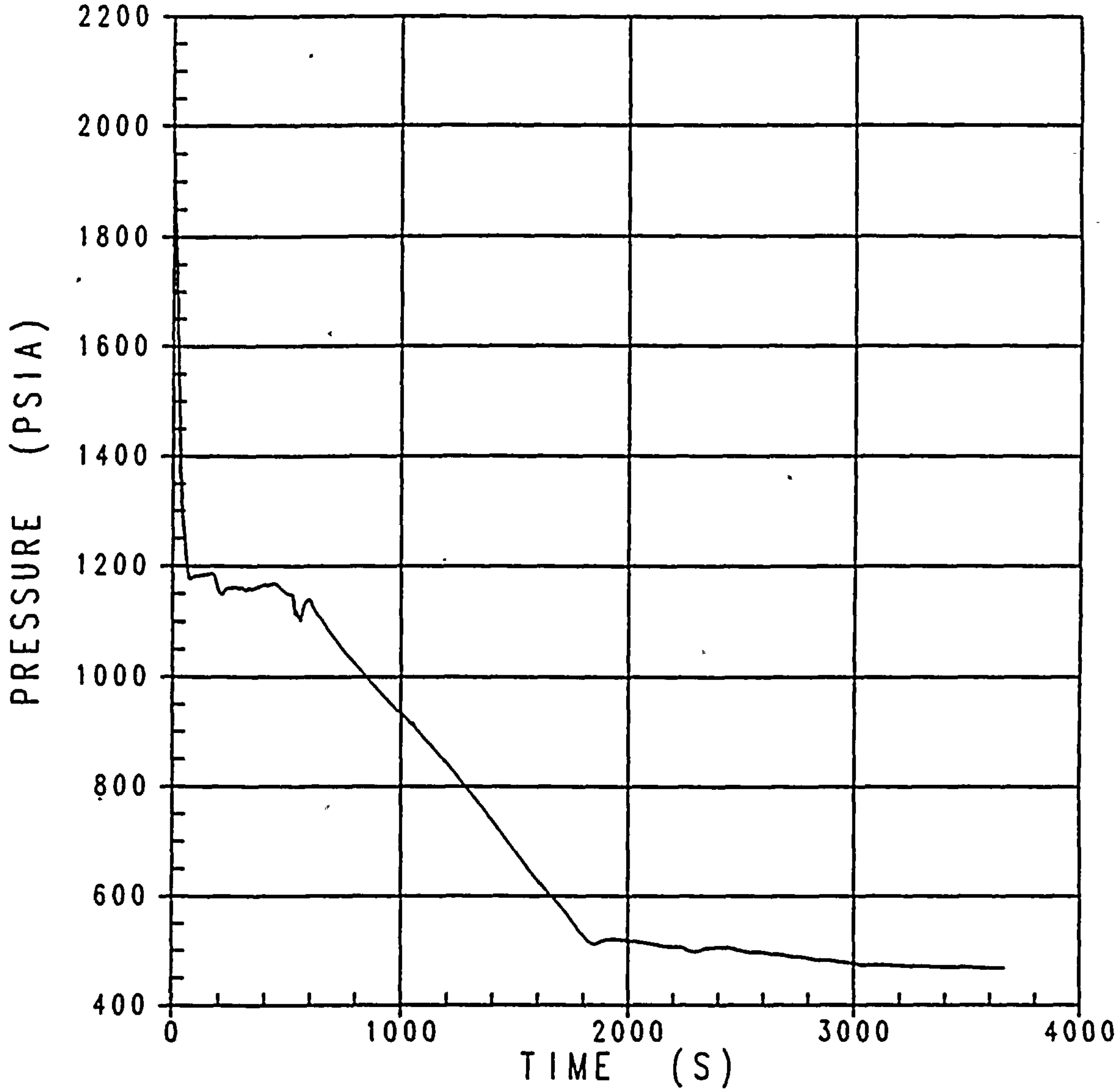

Figure 3.1-77 RCS Pressure (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

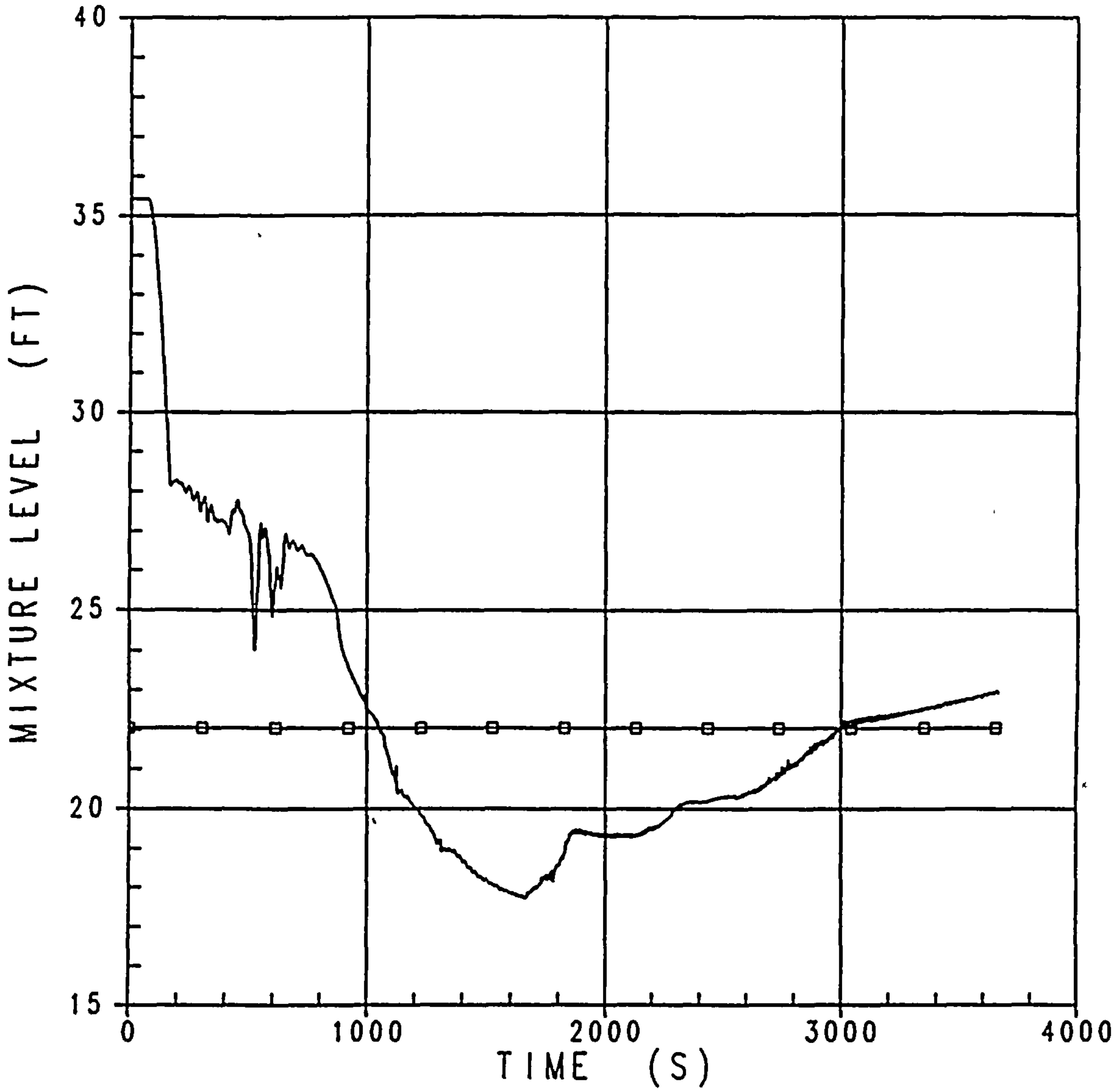

Figure 3.1-78 Core Mixture Level (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

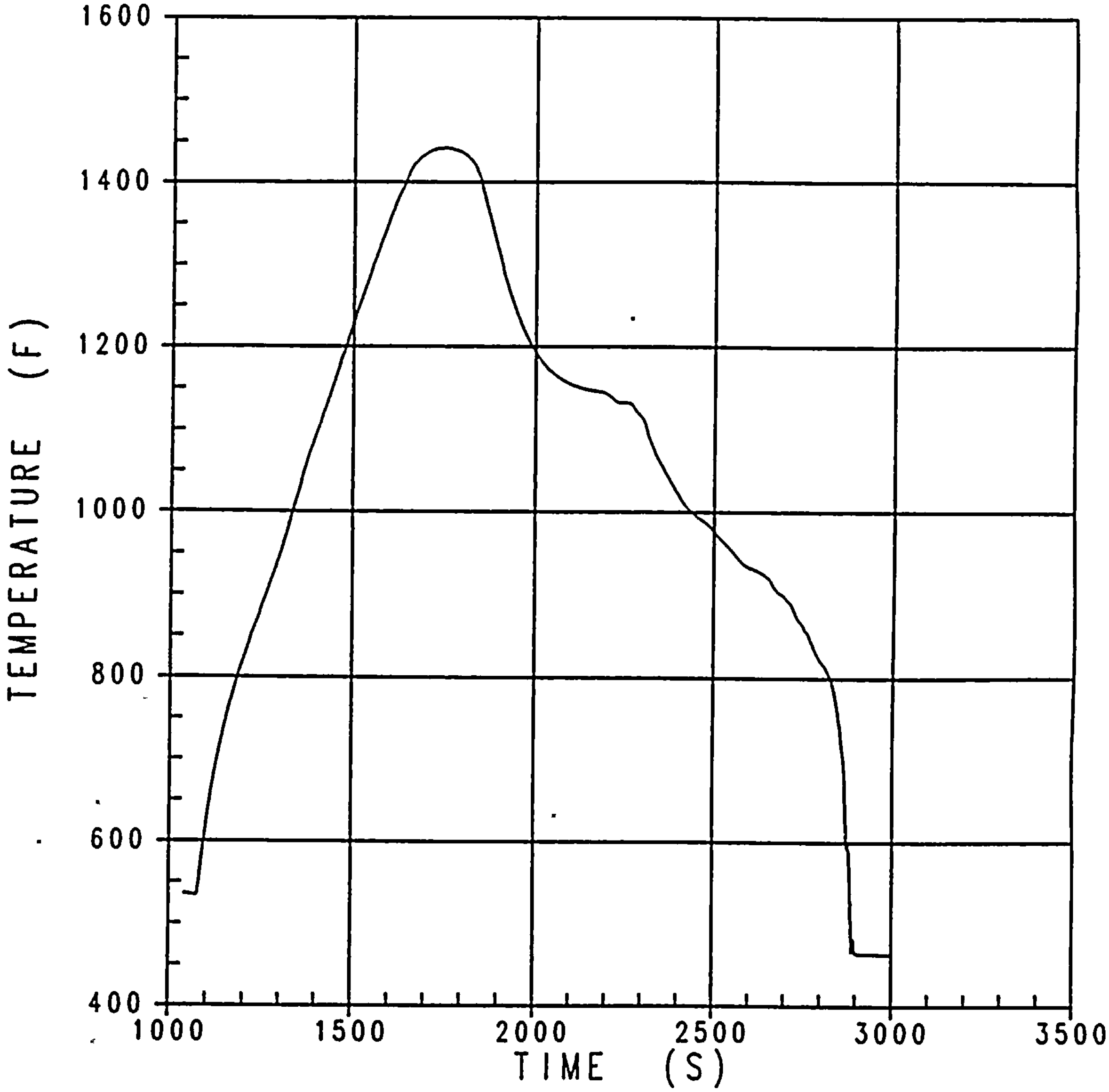

Figure 3.1-79 Hot Spot Clad Temperature (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

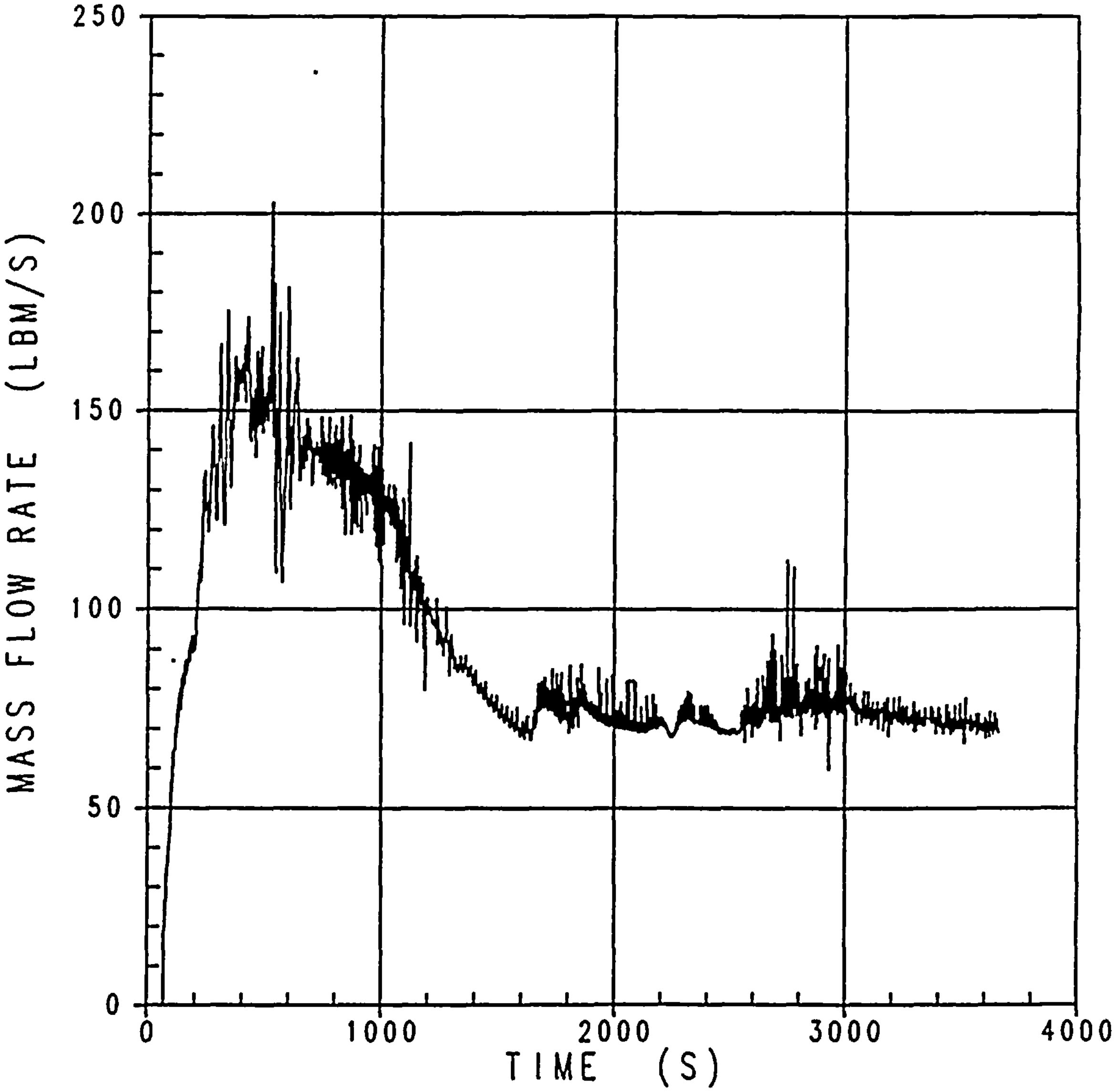

Figure 3.1-80 Core Outlet Steam Flow (3 Inch, 30% SFTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

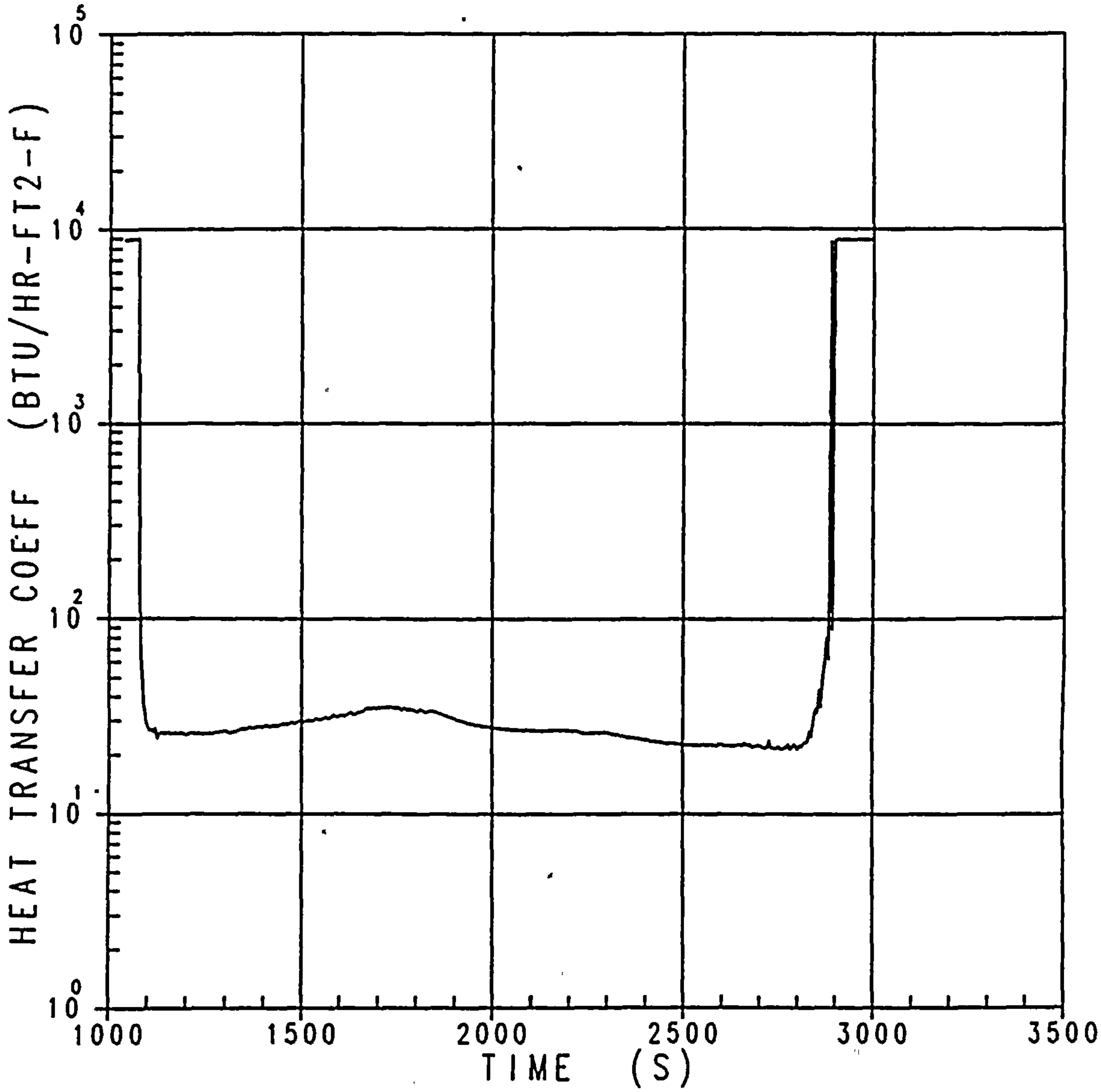

Figure 3.1-81 Hot Spot Rod Surface Heat Transfer Coefficient (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

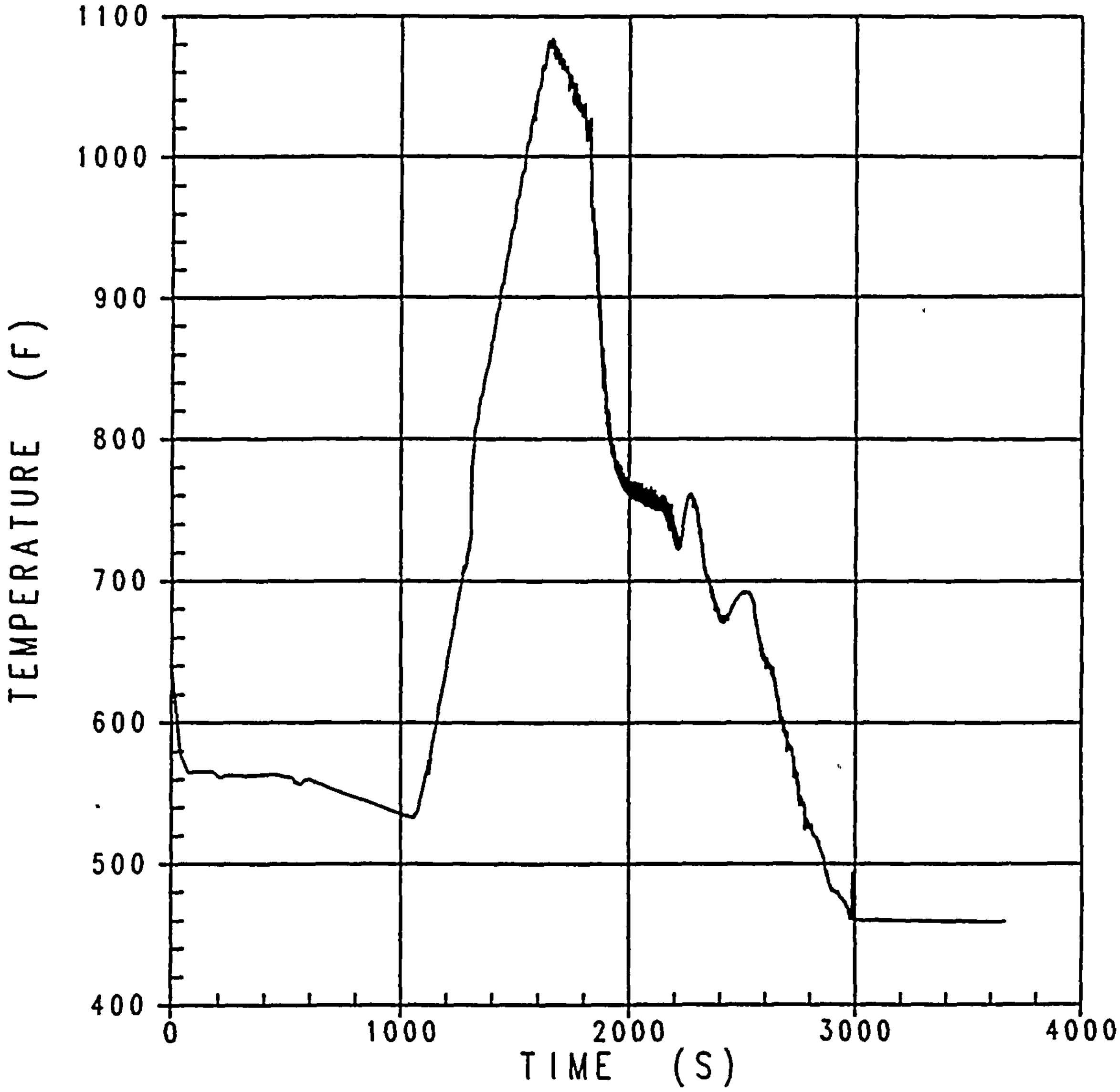

Figure 3.1-82 Hot Spot Fluid Temperature (3 Inch, 3% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

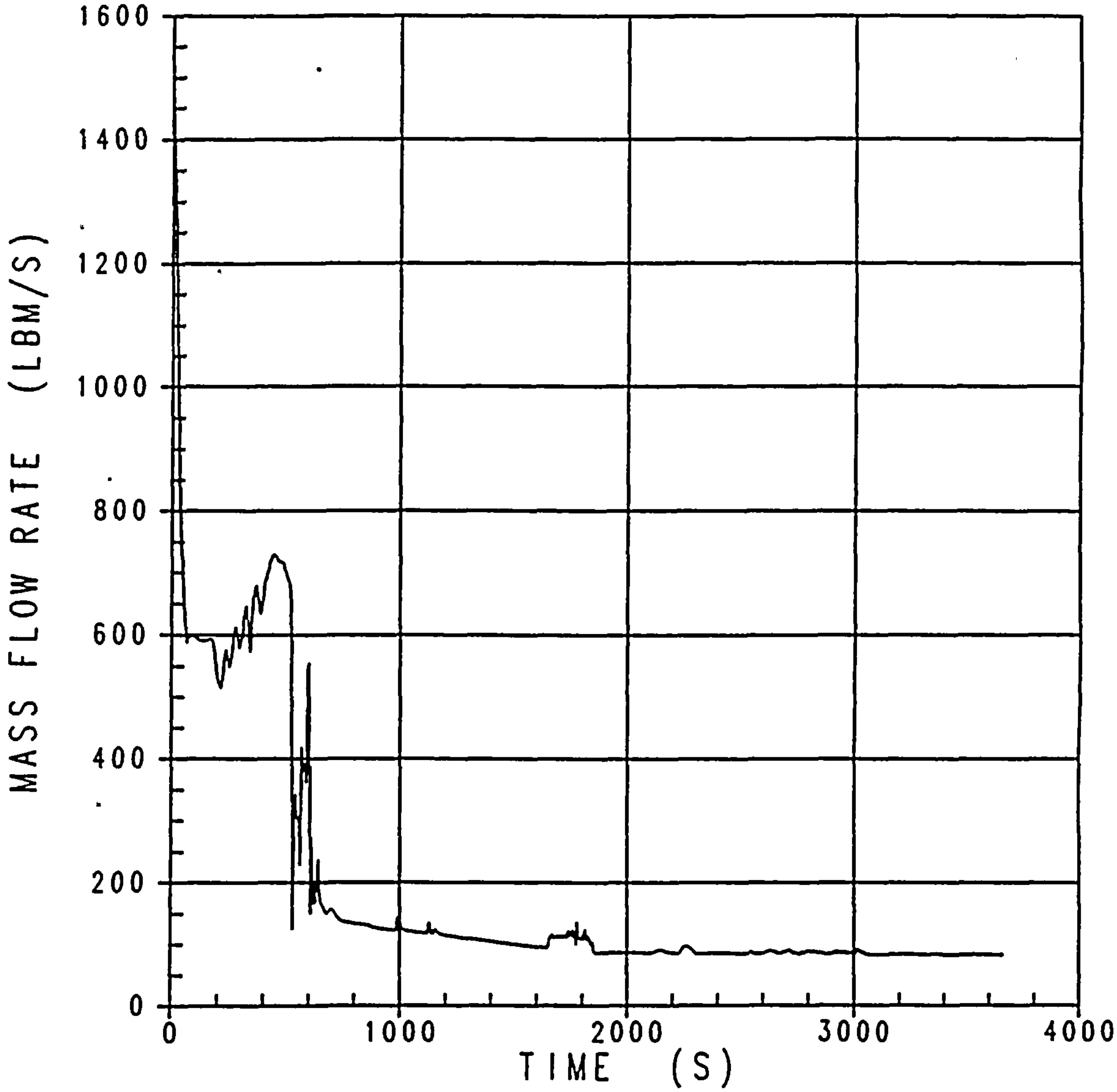

Figure 3.1-83 Cold Leg Break Mass Flow Rate (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

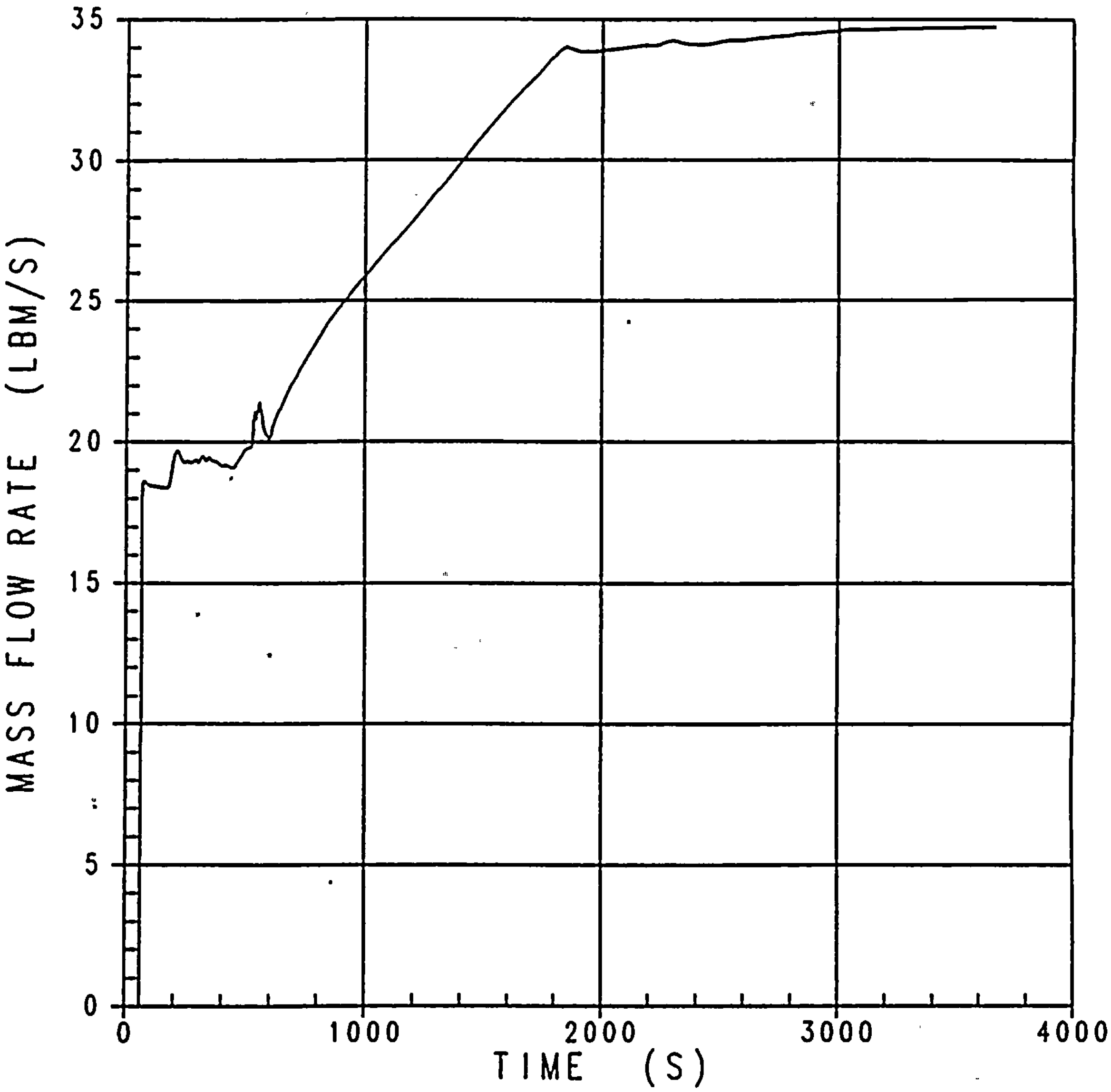

Figure 3.1-84 Broken Loop Safety Injection Mass Flow Rate (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

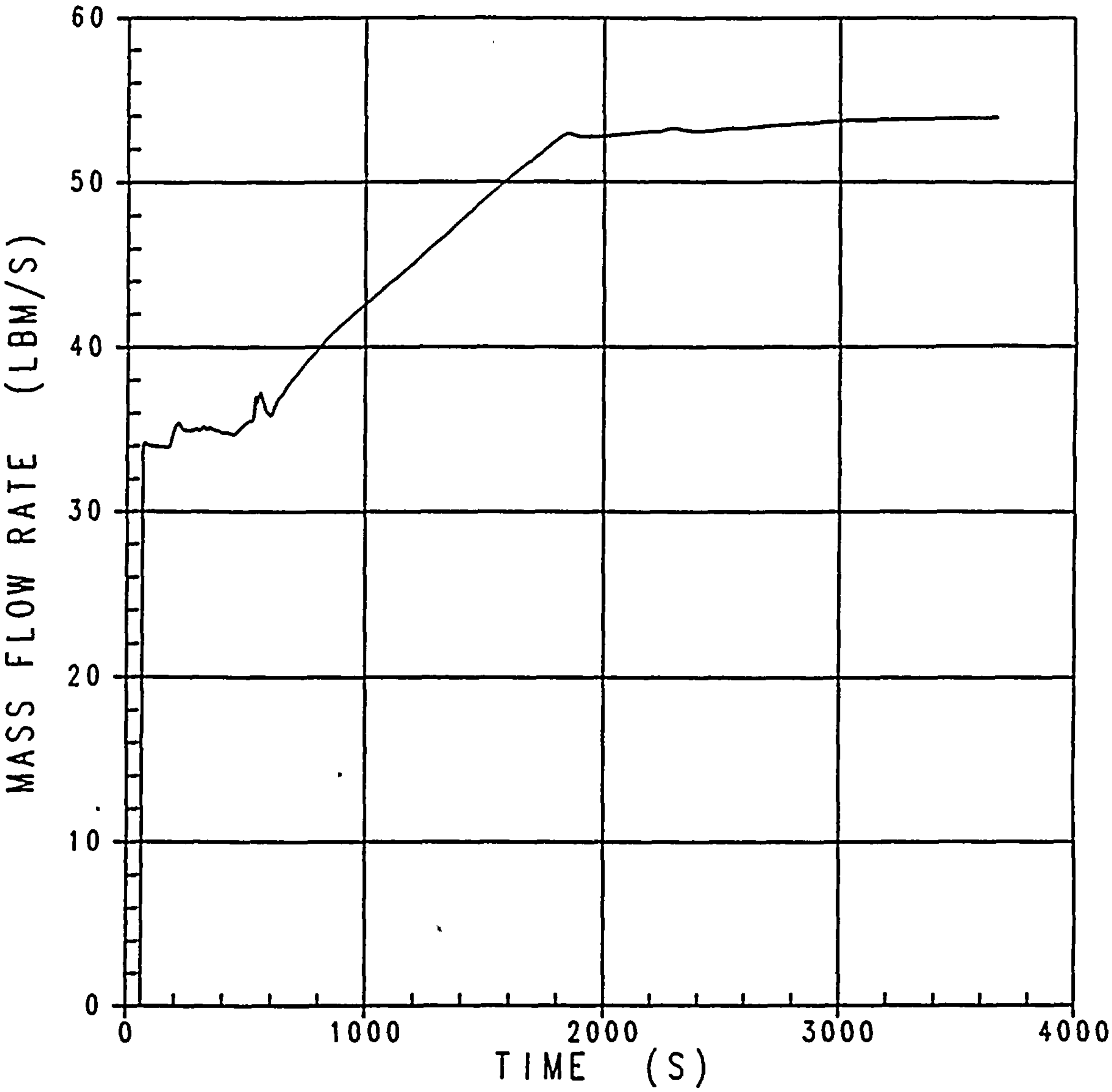

Figure 3.1-85 Lumped Intact Loop SI Mass Flow Rate (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

3.2 LOCA HYDRAULIC FORCES

LOCA hydraulic forces are relatively insensitive to specific steam generator tube plugging levels and the associated changes in thermal design flow, provided the RCS temperatures remain unchanged. The LOCA hydraulic forces analyzed for the Rerating Program are documented in Section 3.2 of WCAP-11902 and used conservative values of 582.3°F and 511.7°F for T_{hot} and T_{cold}, respectively. The LOCA hydraulic forcing functions in Section 3.2 of WCAP 11902 conservatively bound the RCS parameters in Table 2.1-1 of this report, even with consideration of 5% asymmetric flow. WCAP-11902, Section 3.2, remains valid and no changes and/or additions are required.

3.3 NON-LOCA ANALYSES

3.3.1 Introduction

This section evaluates the effects of reduced temperature and pressure operation with a maximum average SGTP level of 30% for Donald C. Cook Nuclear Plant Unit 1 with respect to the non-LOCA safety analyses. The effort performed is to support Unit 1 operation with a core power of 3250 MWt in the range of full-power reactor vessel average temperatures between 553°F and 576.3°F at primary pressure values of 2100 psia or 2250 psia (Cases 1 and 2 of Table 3.3-1, which are identical to cases 2 and 3 of Table 2.1-1).

The current non-LOCA analyses of record for Unit 1 support a rerated core thermal power of 3411 MWt (3425 MWt NSSS) with a full power vessel average temperature between 547°F and 578.7°F at a primary system pressure of 2100 psia or 2250 psia. Cases 3 and 4 of Table 3.3-1 present the range of conditions supported by the current non-LOCA analyses of record. It.is important to note that the current non-LOCA safety analyses of record support the rerating of Donald C. Cook Nuclear Plant Unit 1. However, the unit has never been licensed to operate in accordance with the parameters defined as Cases 3 and 4 of Table 3.3-1.

The Donald C. Cook Nuclear Plant Unit 1 licensing basis, as reported in the UFSAR (Reference 14) includes analyses and evaluations of sixteen non-LOCA events, which are delineated on the next two pages. This licensing-basis has been reviewed to assess the impact associated with the SGTP Program. The following events were re-analyzed as part of the SGTP Program:

Unit 1 UFSAR Section	Accident
14.1.1	Uncontrolled RCCA Bank withdrawal from a Subcritical Condition
14.1.2	Uncontrolled RCCA Bank withdrawal At Power
14.1.3	Rod Cluster Control Assembly Misalignment
14.1.4	RCCA Drop
14.1.6	Loss of Reactor Coolant Flow (Including Locked Rotor)
14.1.8	Loss of External Electrical Load and/or Turbine Trip
14.2.5	Rupture of a Steam Pipe (core response analysis)

14.2.6	Rupture of a CRDM Housing (Rod Ejection)
14.3.4.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside Containment.

The following events/analyses have been evaluated to support the operating conditions associated with the SGTP program:

Unit 1 UFSAR Section	Accident
14.1.5	Chemical and Volume Control System Malfunction
14.1.7	Start-up of an Inactive Loop
14.1.9	Loss of Normal Feedwater
14.1.10	Excessive Heat Removal Due to Feedwater System Malfunctions
14.1.11	Excessive Load Increase Incident
14.1.12	Loss of All AC Power to the Plant Auxiliaries
14.2.8 (Unit 2*)	Rupture of a Main Feedwater Pipe
14.4.11.3	Steamline Break Mass/Energy Release Outside Containment

3.3.2 Non-LOCA Safety Analysis Assumptions Requiring Technical Specification Changes

To enhance operating flexibility for RCS reduced temperature and pressure operation with a maximum average steam generator tube plugging of 30%, certain reactor protection system setpoints were revised. The following requirements were relaxed to enhance operating flexibility as well: EDG start time from ambient conditions; pressurizer code safety valve setpoint tolerance; and shutdown margin for T_{avg} greater than 200°F.

The revised RPS setpoints include the overtemperature ΔT (OTΔT) and the overpower ΔT (OPΔT) reactor trips. The general equations for the OTΔT and OPΔT reactor trip setpoints and the safety analysis limit coefficient values are presented in Table 3.3-3. A detailed

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* The 30% SGTP program also included an evaluation of the Major Rupture of a Feedwater Pipe event (UFSAR Section 14.2.8), which is not part of the Unit 1 licensing basis and is provided for informational purposes only.

discussion of the revised setpoint equations for these reactor trip functions is provided in Section 3.3.2.1.

Discussions of the EDG start time requirement, the pressurizer code safety valve setpoint tolerance adjustment, and shutdown margin relaxation are also presented in the sections that follow. The applicable Technical Specification updates for these revisions/relaxations are provided in Appendix A.

3.3.2.1 Reactor Protection System Trip Setpoints

Revised OTΔT and OPΔT setpoints are based upon new core thermal safety limits, which account for the effects of the RCS parameter changes associated with the increased level of steam generator tube plugging, using the methodology described in Reference 1. These setpoints were revised to increase the available margin between the safety analysis setpoint values and the nominal, or Technical Specification values, such that more ΔT-drift could be accommodated between instrumentation calibrations during the fuel cycle. Presently, the power margin associated with the Rerating Program is being utilized to offset the ΔT-drift that is being experienced during core burnup (i.e., the core power of 3411 MWt is supported by the analyses, but the plant is actually operated with a core full-power value of 3250 MWt). However, since the 30% SGTP parameters do not have this power margin available, there was a need to revise the OTΔT and OPΔT setpoints as part of the SGTP Program.

Figures 3.3-1 through 3.3-4 present the allowable reactor coolant loop average temperature and ΔT conditions as a function of primary coolant pressure, based upon a minimum measured flow (MMF) of 339,100 gpm and a 1.55 chopped cosine axial power distribution. Figure 3.3-1 represents the most limiting 30% SGTP operating configuration (nominal full-power T_{avg} = 576.3°F, nominal pressure = 2100 psia) of the range of conditions described in Table 3.3-1 (Cases 1 and 2) for the calculation of the OTΔT and OPΔT setpoints. The boundaries of operation defined by the OTΔT and OPΔT trips are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions, a trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the limit value (1.40 and 1.42 for typical and thimble cells, respectively; see Table 3.12-3). All points below and to the left of a DNB line for a given pressure have a DNBR greater than the Safety Analysis Limit DNBR value. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable Safety Analysis Limit DNBR at any point.

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high and low RCS pressure (fixed setpoints); overpower and overtemperature ΔT (variable setpoints), and the opening of

the steam generator safety valves, which limit the maximum RCS average temperature. The Safety Analysis Limit DNBR value (1.40 typical and 1.42 thimble), which was used as the DNBR limit for all accidents analyzed with the Revised Thermal Design Procedure (RTDP; Reference 2), is conservative compared to the actual Design Limit DNBR value (1.23 and 1.22 for typical and thimble cells, respectively), required to meet the DNB design basis.

Table 3.3-2 presents the limiting trip setpoints assumed in the accident analyses and the time delay values assumed for each trip function. The difference between the limiting trip point assumed for the analysis and the normal trip point represents an allowance for instrumentation channel error and setpoint error. Nominal trip setpoints are specified in the plant Technical Specifications. During plant start-up tests, it is demonstrated that actual instrument time delays are equal to or less than the assumed values. Additionally, protection system channels are calibrated and instrument response times are determined periodically in accordance with the Technical Specifications.

The 30% SGTP effort assumed that the reference average temperatures (T' and T") used in the OTΔT and OPΔT setpoint equations are scaled to the full-power average RCS temperature each time the cycle average temperature is changed. It is also assumed that the reference pressure (P') in the OTΔT equation is set equal to the appropriate nominal primary system pressure for a particular cycle (either 2100 psia or 2250 psia). These assumptions are key to ensure that the actual plant conditions required to result in an OTΔT and/or OPΔT trip signal to be generated are conservative with respect to assumptions made in the safety analyses. Figures 3.3-1 through 3.3-4 illustrate the OTΔT and OPΔT protection setpoints for the endpoints of the range of full-power vessel average temperatures for the SGTP Program at either 2100 psia or 2250 psia. The calibration of the NIS excore detectors, to compensate for the changes in coolant density each time the cycle operating conditions are changed, is also assumed in the analyses.

The OTΔT and OPΔT reactor trip functions provide primary protection against fuel centerline melting, among other concerns (i.e., DNB and hot-leg boiling). The criterion for no fuel melt is, the uranium dioxide melting temperature shall not be exceeded for at least 95 percent of the limiting fuel rods at a 95 percent confidence level (Reference 1). This criterion is met by limiting the calculated fuel centerline temperature to 4700°F (valid for 60000 MWD/MTU burnup per Reference 16). In many cases, fuel centerline melting can be prevented by limiting gross core thermal power to a prescribed limit (historically 118% of nominal power) independent of axial power distribution. As part of the reload process (via the Reload Safety Analysis Checklist, or RSAC), the peak linear heat generation rate of the core (i.e., peak kw/ft) is determined specifically for fuel centerline melting concerns. Even though the revised OTΔT and OPΔT reactor trip setpoint equations allow the typical gross core average thermal power to slightly exceed the historical value of 118% (Cook Unit 1 analyses indicate that a peak overpower of 119.03% can be achieved with the revised setpoints), fuel centerline melt

concerns are specifically evaluated on a cycle-by-cycle basis as part of the formal reload process to ensure fuel centerline melting does not occur.

Since the revised OTΔT and OPΔT reactor trip setpoint equations allow the typical gross core average thermal power to slightly exceed this 118% value, as noted above, this fact was addressed with respect to the steamline break - core response (SLB-CR) methodology. The full-power steamline break analysis for core response considerations is not in the Donald C. Cook Nuclear Plant licensing basis. Nevertheless, it has been determined that the revised OTΔT and OPΔT setpoint equations, with the OPΔT reference average temperature (T") restricted to values no greater than 563.0°F, provides sufficient assurance that minimum DNBR will be protected during a HFP SLB.

3.3.2.2 Emergency Diesel Generator Start-up Time Relaxation

Those events that must consider a loss of offsite power (i.e., Loss of All AC Power to the Station Auxiliaries, and Steam line Break for core response) have been evaluated with respect to an increase in the EDG start time from 10 seconds to 30 seconds. This start time relaxation of the EDG has been found to be acceptable. This feature also affects the Main Steamline Break Mass/Energy Releases Inside Containment analysis. However, the impact is limited to the containment response portion of the analysis, as the steam mass and energy release calculations are based on the conservative assumption that offsite power is available for the duration of the blowdown.

3.3.2.3 Pressurizer Code Safety Valve Setpoint Tolerance Increase

The following events, which are potentially impacted by an increase in the pressurizer code safety valve setpoint tolerance, have been shown to support an increase from ±1% to ±3% setpoint tolerance: Loss of External Electrical Load; Loss of Normal Feedwater; Loss of All AC Power to the Station Auxiliaries; and Locked Rotor/Shaft Break events. Thus, a setpoint tolerance of ±3% for the pressurizer code safety valves is acceptable.

3.3.2.4 Shutdown Margin Relaxation

All of the current Donald C. Cook Nuclear Plant Unit 1 licensing-basis analyses (i.e., the analyses supporting the Rerating Program) that model shutdown margin (SDM) assume 1.3%Δk/k, except for the Steamline Break for Core Response (SLB-CR) event. The re-analysis of the SLB-CR event for the SGTP Program was performed with a SDM assumption of 1.3%Δk/k. As such, all of the non-LOCA safety analyses that model SDM support the reduced SDM value of 1.3%Δk/k.

3.3.2.5 Steamline Break Protection System Modification

The coincidence logic currently required for safety injection initiation and steamline isolation on high steam flow and low steam pressure or low-low T_{avg} for Unit 1 is to be modified to match that installed at Unit 2. This logic is part of the steamline break protection system. A detailed description of both of the steamline break protection systems currently installed in each of the units is presented in Section 3.5.4. The proposed Unit 1 modification, which will result in the two units having identical steamline break protection systems, consists of replacing SI actuation on high steam flow coincident with low steam pressure, or high steam flow coincident with low-low T_{avg}, with SI actuation on low steam pressure only. The proposed Unit 1 modification also replaces steamline isolation on high steam flow coincident with low steam pressure with SLI on low steam pressure only. The coincidence requirement for high steam flow with low steam pressure of the current Unit 1 design increases the likelihood that safeguards initiation might be delayed compared to the proposed Unit 1 modified design, where only a low steam pressure signal is required. In the case where the coincidence logic prohibits safety injection and steamline isolation on high steam flow with low steam pressure, one of the other signals must be received before the safeguards are initiated. As such, the currently installed Unit 1 steamline break protection system design assumed in the Steamline Break Mass/Energy Releases Outside Containment calculations (Section 3.3.4.7), the Rupture of a Steam Pipe analysis (Section 3.3.5.6), and the Steamline Break Mass/Energy Releases Inside Containment analysis (Section 3.5.4) bounds the proposed modifications to the Unit 1 steamline break protection system, as the requirement to satisfy the coincidence, discussed above, can result in SI and/or steamline isolation later in a transient than had SI and/or SLI actuated by low steam pressure alone. A delay in the initiation of safeguards is conservative for all three of the previously listed events. Also, an evaluation regarding the Major Rupture of a Feedwater Pipe event for Unit 1 has been performed (Section 3.3.4.8), which specifically addresses the proposed Unit 1 steamline break protection system modifications. It should be noted that the deleted SI actuation function, i.e., high steam flow coincident with low-low T_{avg}, is not modeled in any of the non-LOCA safety analyses.

3.3.3 Methodology

The Unit 1 non-LOCA safety evaluation for the SGTP Program was performed using current Westinghouse methodology and computer codes. The following four sub-sections discuss: the Initial Conditions assumed, which reflects the change from the Improved Thermal Design Procedure (ITDP) (utilized for the Rerating Program of Unit 1) to the Revised Thermal Design Procedure (RTDP) for most of the events that are DNB limited; the Computer Codes Utilized; and the 5% RCS Flow Asymmetry.

3.3.3.1 Initial Conditions

For most accidents which are DNB limited, nominal values of initial conditions and minimum measured flow (339,100 gpm) are assumed. The allowances on reactor power, RCS temperature and pressure are determined on a statistical basis and are included in the limit DNBR as described in WCAP-11397 (Reference 2). This procedure is known as the "Revised Thermal Design Procedure".

For accidents that are not DNB limited or in which RTDP is not employed the initial conditions are obtained by adding the maximum steady-state errors to rated values. The following steady-state errors are considered:

a.	Core Power	+2% calorimetric error allowance
b.	Average RCS Temperature	±4.1°F controller deadband and measurement error allowance; also a +1.0°F bias for cold-leg streaming
c.	Pressurizer Pressure	±67 psi steady-state fluctuations and measurement error allowance (see paragraph below)
d.	Reactor Flow	Thermal Design Flow (332,800 gpm)

It should be noted that the pressurizer pressure uncertainty includes an allowance of 23 psi for "readability," which is only applicable for DNB considerations. However, the 67 psi uncertainty was conservatively applied to all non-LOCA analyses for simplicity. Thus, there is an additional 23 psi of pressure margin that can be realized, if necessary, for non-DNB events.

Table 3.3-4 summarizes initial conditions and computer codes used in the accident analysis, and shows which accidents employed a DNB analysis using the RTDP.

3.3.3.2 Computer Codes Utilized

Summaries of the principal computer codes used in the transient analyses are given below. The codes used in the analysis of each transient have been listed in Table 3.3-4.

FACTRAN

FACTRAN calculates the transient temperature distribution in a cross-section of a metal clad UO_2 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, and 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 a sophisticated fuel-to-clad gap heat transfer calculation.

C. The necessary calculations to handle post-departure from nucleate boiling transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the materials.

FACTRAN is further discussed in Reference 3.

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, 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 homogenous, 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, overpower ΔT, overtemperature ΔT, 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 minimum value of DNBR as calculated for typical or thimble cell.

LOFTRAN is further discussed in Reference 4.

TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which was patterned after steady-state codes presently 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 temperatures, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided; e.g., channel wise 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 5.

THINC IV

The THINC IV computer program, as approved by the NRC, is used to determine coolant density, mass velocity, enthalpy, vapor void, static pressure, and DNBR distributions along parallel flow channels within a reactor core under all expected operating conditions. The THINC IV code is described in detail in Reference 6.

3.3.3.3 5% RCS Flow Asymmetry

A 5% RCS flow asymmetry is supported by the non-LOCA safety analyses. Specifically, a reduction of RCS flow in one loop up to 5% below the nominal average per loop flow rate is acceptable, as long as the total minimum measured RCS flow is equal to or greater than 339,100 gpm. Should more than one loop be below the 84,775 gpm/loop flow rate, the sum of the loop flow shortfalls can be no greater than 5% of one loop.

The non-LOCA events that potentially are sensitive to asymmetric RCS flow include: Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition (RWFS), Partial Loss of Forced Reactor Coolant Flow (PLOF), Reactor Coolant Pump Locked Rotor/Shaft Break (Locked Rotor), Loss of Normal Feedwater, Excessive Heat Removal Due to Feedwater System Malfunctions, Loss of All AC Power to the Station Auxiliaries, Steamline Break for Core Response, Rod Ejection at zero power conditions (HZP Rod Ejection), and Rupture of a Main Feedwater Pipe. The balance of the non-LOCA events are not sensitive to RCS flow asymmetry.

The following events explicitly accounted for the effects of asymmetric RCS flow as part of the SGTP Program analyses: RWFS, PLOF, Locked Rotor, and HZP Rod Ejection. Specifically, the PLOF and Locked Rotor analyses model the fault to occur in the loop with the highest flow (i.e., 5% above the nominal per loop minimum measured flow). Thus, the low flow reactor trip (conservatively assumed as a percentage of <u>nominal</u> flow as opposed to a percentage of

normalized flow) is delayed as much as possible, also the largest overall flow reduction is obtained with this model. For the RWFS and Rod Ejection analyses, a RCS flow corresponding to two out of four reactor coolant pumps in service (Mode 3 flow) is assumed. A conservative flow fraction is used, which bounds a worst case 5% flow asymmetry scenario in Mode 3 where the loops that would be providing the most flow are out of service. The safety analysis criteria for all of the aforementioned reanalyses continued to be met after explicitly accounting for the asymmetric RCS flow.

The remainder of the events that are potentially sensitive to asymmetric RCS flow, but did not explicitly account for the effects in the specific analysis, have been evaluated and were found to be able to accommodate a RCS flow asymmetry of 5%. Thus, it can be concluded that 5% RCS flow asymmetry is supported (either directly or indirectly) by the Cook Nuclear Plant Unit 1 non-LOCA safety analyses and evaluations.

3.3.4 Non-LOCA Safety Evaluation: Transients Evaluated

The sections that follow contain the detailed descriptions of the impact of the SGTP Program on the applicable non-LOCA transients. This first grouping of transients are those which could be evaluated and the second grouping (Section 3.3.5) are transients which required re-analysis. In all cases the appropriate UFSAR acceptance criteria are satisfied.

3.3.4.1 Chemical and Volume Control System Malfunction

Reactivity can be added to the core by feeding primary grade water into the RCS via the reactor makeup portion of the CVCS. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the RCS. The CVCS is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which provides the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the Primary Water Makeup Control Valve supplies water to the RCS which can dilute the reactor coolant. Inadvertent dilution can be readily terminated by closing this valve. In order for makeup water to be added to the RCS, at least one charging pump must also be running in addition to the primary water pumps.

The rate of addition of unborated water makeup to the RCS is limited by the capacity of the primary water pumps. The maximum addition rate in this case is 225 gpm with both primary water pumps running. The 225 gpm reactor makeup water delivery rate is based on a pressure drop calculation comparing the pump curves with the system resistance curve. This is the maximum delivery based on the unit piping layout. Normally, only one primary water supply pump is operating while the other is on standby.

The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flow rates of boric acid and primary grade water on the control board..

In order to dilute, two separate operations are required. First, the operator must switch from the automatic makeup mode to the dilute mode; second, the start button must be depressed. Omitting either step would prevent dilution. This makes the possibility of inadvertent dilution very remote.

Information on the status 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 CVCS. 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.

To cover the phases of the plant operation and to account for the reduction in the volume of the RCS due to the increase in the level of SGTP up to 30%, boron dilution during startup and power operation were examined. Included in the evaluation was the effect of the difference in the density of unborated makeup water and the density of the reactor coolant. The evaluation is to show that, from initiation of the event, sufficient time is available to allow the operator to determine the cause of the addition and take corrective action before excessive shutdown margin is lost.

Results and Conclusions

Because of the steps involved in the dilution process, an erroneous dilution is considered highly unlikely. Nevertheless, if it does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and take corrective action before excessive shutdown margin is lost for the phases of the plant operation (start-up and at-power) affected by the SGTP Program.

3.3.4.2 Startup of an Inactive Loop

In accordance with Technical Specification 3/4.4.1, Cook Nuclear Plant Unit 1 operation during Modes 1 and 2 with less than four reactor coolant loops is not permitted. Since three loop operation during Modes 1 and 2 is prohibited, the Startup of an Inactive Loop event does not have to be considered as part of the 30% SGTP Program.

3.3.4.3 Loss of Normal Feedwater

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat

generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The reactor trip on low-low water level in any steam generator provides the necessary protection against a loss of normal feedwater.

The auxiliary feedwater system is started automatically. The turbine driven auxiliary feedwater pump utilizes steam from the secondary system and exhausts to the atmosphere. The motor driven auxiliary feedwater pumps are supplied by power from the diesel generators if a loss of offsite power occurs. The pumps take suction directly from the condensate storage tank for delivery to the steam generators.

An evaluation of the system transient has been performed to show that following a loss of normal feedwater, with initial plant conditions consistent with those defined in the SGTP Program, the auxiliary feedwater system is capable of returning the plant to a safe condition by removing the stored and residual heat, thus preventing either overpressurization of the RCS or uncovery of the core.

The results of the evaluation demonstrate that the Loss of Normal Feedwater event can support the 30% SGTP conditions. The limiting peak pressurizer level occurs under low T_{avg} (553°F) conditions. This is consistent with the current Loss of Normal Feedwater analysis of record performed as part of the Rerating Program. The conclusions presented in the Donald C. Cook Nuclear Plant Unit 1 UFSAR (Reference 14) remain applicable for 30% SGTP conditions, since the Loss of Normal Feedwater analysis under rerated conditions (i.e., Cases 3 and 4 of Table 3.3-1) yield more severe results than those obtained from the sensitivity cases investigated for the SGTP Program. This is due to the benefits from the power level reduction (3411 MWt → 3250 MWt) and the increase in the lower bound T_{avg} (547°F → 553°F) more than offsets the heat removal penalties caused by the increase in SGTP level (15% → 30%) and the thermal design flow reduction (354,000 gpm → 332,800 gpm).

3.3.4.4 Excessive Heat Removal due to Feedwater System Malfunctions

Reductions in feedwater temperature or additions of excessive feedwater are means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The Overpower-Overtemperature Protection (high neutron flux, overpower ΔT, and overtemperature ΔT trips) prevents any power increase which could lead to DNBR less than minimum allowable value in the event that the steam generator High-High Level Protection has not been actuated.

Excessive feedwater flow may be caused by full opening of a feedwater control valve due to a Feedwater Control System malfunction or an operator error. At power conditions, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity.

The excessive heat removal due to Feedwater System Malfunction events are examined primarily to demonstrate core protection. For the SGTP Program, an evaluation of the system transient has been performed to show that acceptable consequences will occur in the event of an excessive feedwater addition, due to control system malfunction or operator error which allows one or more feedwater control valve(s) to open fully. This evaluation considered both at power and zero power scenarios with the reactor being operated under both automatic and manual rod control conditions.

A feedwater malfunction event as described above results in an increase in the rate at which heat is removed from the reactor coolant. An increase in the level of tube plugging in the steam generators results in a reduction in the heat transfer characteristics between the primary coolant and the steam system. Thus, a less severe cooldown would be experienced for this event under the 30% SGTP conditions. However, the RCS flow reduction due to the larger number of tubes being plugged is a DNB penalty for the at power events. Furthermore, the reduction in core power from the rerated value of 3411 MWt to 3250 MWt provides a DNB benefit.

The evaluation performed for the SGTP Program conservatively ignored the benefit associated with the reduced ability of the excessive feedwater flow to cool the primary coolant. The evaluation conservatively minimized the benefit associated with the rated thermal power reduction and conservatively maximized the penalty due to the RCS flow reduction with respect to the power and flow values assumed in the analyses of record. The parameters assumed in the analyses of record for the feedwater malfunction events are consistent with those presented as Cases 3 and 4 of Table 3.3-1.

The evaluation concluded that both the at power and zero power feedwater malfunction transients can support 30% SGTP conditions. The reactivity insertion rate assumed in the current UFSAR analysis (120 pcm/sec) for the zero power event continues to be conservative for the 30% SGTP conditions. It should be noted that the revised OTΔT and OPΔT setpoint equations do not impact the feedwater malfunction events, due to the fact that the analyses do not take credit for the protection offered by these trip functions. The conclusions presented in the Donald C. Cook Nuclear Plant Unit 1 UFSAR for the Excessive Heat Removal Due to Feedwater System Malfunctions (UFSAR Section 14.1.10) remain applicable.

3.3.4.5 Excessive Increase in Secondary Steam Flow

An excessive load increase incident is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10% step load increase and a 5% per minute ramp load increase in the range of 15 to 100% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control.

During power operation, steam dump to the condenser is controlled by reactor coolant condition signals, i.e., high reactor coolant temperature indicates a need for steam dump. A single controller malfunction does not cause steam dump; an interlock is provided which blocks the opening of the valves unless a large turbine load decrease or turbine trip has occurred.

Protection against an excessive load increase accident is provided by the following reactor protection system signals:

- Overpower ΔT
- Overtemperature ΔT
- Power range high neutron flux
- Low pressurizer pressure

An excessive increase in steam load results in an increase in the rate at which heat is removed from the reactor coolant. An increase in the level of tube plugging in the steam generators results in a reduction in the heat transfer characteristics between the primary coolant and the steam system. Thus, a less severe cooldown would be experienced for this event under the 30% SGTP conditions. However, the RCS flow reduction due to the large number of tubes being plugged is a DNB penalty for this event. Conversely, the reduction in the core power from the rerated value of 3411 MWt to 3250 MWt provides a DNB benefit.

The evaluation performed for the SGTP Program conservatively ignored the benefit associated with the reduced ability of the excessive steam flow to cool the primary coolant. The evaluation conservatively minimized the benefit associated with the rated thermal power reduction and conservatively maximized the penalty due to the RCS flow reduction with respect to the power and flow values assumed in the analyses of record. The parameters assumed in the analyses of record for the Excessive Load increase Incident are consistent with those presented as Cases 3 and 4 of Table 3.3-1.

The evaluation concluded that the Excessive Load Increase Incident can support the 30% SGTP conditions. The revised OTΔT and OPΔT setpoint equations do not impact this event, as the current analysis of record resulted in the plant reaching a stabilized condition at the higher power level, i.e., no reactor trip occurred for this event. The conclusions presented in the Donald C. Cook Nuclear Plant Unit 1 UFSAR for the Excessive Load Increase Incident (UFSAR Section 14.1.11) remain applicable.

3.3.4.6 Loss of All AC Power to the Station Auxiliaries

The loss of all AC power to the station auxiliaries event, as with the loss of normal feedwater incident, is a limiting transient with respect to pressurizer overfill. The decrease in primary to secondary heat transfer ability, due to the increase in SGTP, aggravates the heatup portion of the transient, and increases the potential for filling the pressurizer. As such, the loss of all AC power to the station auxiliaries is evaluated for the SGTP Program.

A complete loss of all (non-emergency) AC power (e.g. offsite power) may result in the loss of all power to the plant auxiliaries, i.e., the RCPs, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accomplished by a turbine generator trip at the station, or by a loss of the onsite AC distribution system.

This transient is analyzed to show the adequacy of the heat removal capability of the auxiliary feedwater system. The transient is more severe than the loss of load event analyzed because in this case the decrease in heat removal by the secondary system is accomplished by a flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip due to: (1) turbine trip; (2) upon reaching one of the trip setpoints in the primary and secondary systems as a result of the flow coastdown and decrease in secondary heat removal; or (3) due to loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

Following a loss of power with turbine and reactor trips, the sequence described below will occur:

A. Plant vital instruments are supplied from emergency DC power sources.

B. As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for steam dump. If the steam flow rate through the power relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

C. As the no load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot standby condition.

D. The standby diesel generators, started on loss of voltage on the plant emergency busses, begin to supply plant vital loads.

The motor driven auxiliary feedwater pumps are supplied power by the diesels and the turbine-driven pump utilizes steam from the main steam system. Both type pumps are designed to supply rated flow within 80 seconds of the initiating signal even if a loss of all non-emergency AC power occurs simultaneously with loss of normal feedwater. The turbine exhausts the used steam to the atmosphere. The auxiliary feedwater pumps take suction from the condensate storage tank for delivery to the steam generators.

Following the RCP coastdown caused by the loss of AC power, the natural circulation capability of the RCS will remove decay heat from the core, aided by auxiliary feedwater in the secondary system. The results of an evaluation is presented here to show that the natural circulation flow in the RCS, following a loss of all AC power to the station auxiliaries with initial plant conditions consistent with those defined in the SGTP Program is sufficient to remove decay heat from the core.

The results of the evaluation demonstrate that the Loss of All AC Power to the Station Auxiliaries (LOOP) event can support the 30% SGTP conditions. The limiting peak pressurizer level occurs under low T_{avg} (553°F) conditions. This is consistent with the current LOOP analysis of record, which was performed for the Rerating Program. The conclusions presented in the Donald C. Cook Nuclear Plant Unit 1 UFSAR (Reference 14) remain applicable for the 30% SGTP conditions, since the LOOP analysis under rerated conditions (i.e, Cases 3 and 4 of Table 3.3-1) yield more severe results than those obtained from the sensitivity cases investigated for the SGTP Program. This is due to the benefits from the power level reduction (3411 MWT → 3250 MWt) and the increase in the lower bound T_{avg} (547°F → 553°F) more than offsets the heat removal penalties caused by the increase in SGTP level (15% → 30%), the additional delay in AFW delivery (60 seconds → 80 seconds) due to the relaxed EDG start time delay (10 seconds → 30 seconds), and the thermal design flow reduction (354,000 gpm → 332,800 gpm).

3.3.4.7 Steamline Break Mass/Energy Releases Outside Containment

The existing mass and energy (M/E) releases following a steamline break (SLB) outside containment were performed to support the range of conditions possible for the Rerating Program of Unit 1 (Cases 3 and 4 of Table 3.3-1), as well as to position Unit 2 for a potential future uprating (i.e., 3600 MWt NSSS). Thus, the M/E releases are based upon a rated

thermal power of 3600 MWt. The core reactivity parameters were chosen to conservatively maximize the reactivity feedback effects of the cooldown resulting from a blowdown from either Donald C. Cook Nuclear Plant unit. The changes associated with the SGTP Program for Unit 1, i.e., RCS flow reduction, reduced primary-to-secondary heat transfer capability, and reduction in the rated thermal power, are less limiting parameters relative to the assumptions currently made for the M/E release calculations following a SLB outside containment.

Furthermore, the adjustment in the K_4 safety analysis value of the OPΔT setpoint equation (discussed in Section 3.3.2.1) does not impact the SLB M/E Release Outside Containment analysis, which is the only non-LOCA safety analysis that relies on this trip function for primary protection. This is because a conservatively larger K_4 value of 1.18 was assumed in the SLB M/E Release Outside Containment analysis. The revised safety analysis K_4 value for Unit 1 is 1.172. The increase in the EDG start time delay from 10 seconds to 30 seconds has no effect on this analysis as well, since it is conservative to maintain offsite power such that reactor coolant pump operation is maintained (which aids in maximizing the steam releases). Therefore, it can be concluded that the current licensing basis outside containment SLB M/E releases (UFSAR Section 14.4.11.3) continue to bound Unit 1 operation as defined by the SGTP Program.

It is key to notice that the existing outside containment SLB M/E release analysis-of-record became part of the Cook Nuclear Plant Unit 1 licensing basis (and Unit 2 for that matter) following the approval of the Boron Injection Tank (BIT) Removal submittal (Reference 18). The existing SLB M/E Release Outside Containment analysis assumed:

a. End-of-life shutdown margin of 1.3 %Δk/k at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position.

b. Minimum capability for the injection of boric acid solution corresponding to the most restrictive single failure in the safety injection system. The ECCS consists of the following systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, 3) the high head safety injection system, and 4) the charging system. Only the charging system and the passive accumulators are modeled for the steamline break accident analysis for M/E releases outside containment. Centrifugal Charging pump head degradation of 10% was assumed.

c. Coincidence logic required for SI and SLI consistent with the current Unit 1 steamline break protection system. A proposed modification will change the logic associated with this system. However, as discussed in Section 3.3.2.5, the current analysis, which assumes the current Unit 1 steamline break protection system, bounds the proposed modifications to the Unit 1 steamline break protection system.

3.3.4.8 Major Rupture of a Feedwater Pipe

The feedline break event is currently presented in the Unit 1 UFSAR (Section 14.2.8) for "informational purposes only," as this event is not part of the Unit 1 licensing-basis. However, Cook Unit 2 does have the feedline break event in its licensing scope. The SGTP Program for Unit 1 includes an evaluation to demonstrate that the response of Unit 1 to a feedline break is bounded by the existing Unit 2 feedline break analysis. A key stipulation for this evaluation is that the Unit 1 steamline break protection logic, which is currently classified as the "Old" system, must be modified to match the "Hybrid" steamline break protection logic that is in place at Cook Nuclear Plant Unit 2.

A detailed evaluation was performed, which included sensitivity cases using the LOFTRAN code. The evaluation specifically assessed the plant parameter changes associated with the SGTP Program (Cases 1 and 2 of Table 3.3-1) relative to the Unit 2 parameters corresponding to rerated conditions (i.e., 3600 MWt NSSS). Sensitivity cases investigated the effects of increasing the pressurizer safety valve setpoint tolerance from ±1% to ±3% and increasing the EDG start time delay from 10 seconds to 30 seconds. The sensitivity cases also included the effects associated with the setpoint tolerance increasing from ±1% to ±3%, which has been previously evaluated (Reference 19).

The evaluation concluded that the results presented in the Donald C. Cook Nuclear Plant Unit 2 UFSAR for the Major Rupture of Main Feedwater Pipe event (UFSAR Section 14.2.8) are applicable to Unit 1, provided that the steamline break protection logic installed at Unit 1 is modified to match that installed in Unit 2. Furthermore, the evaluation concluded that an increase in the EDG start time from 10 seconds to 30 seconds; an increase in the pressurizer safety valve setpoint tolerance from ±1% to ±3%, as well as the inclusion of the 1.0°F bias to account for the cold-leg streaming phenomenon can be accommodated.

3.3.5 Non-LOCA Safety Evaluation: Transients Analyzed

The subsections that follow contain the details of the accidents re-analyzed to support 30% SGTP operation of Unit 1. In all cases, the applicable UFSAR acceptance criteria are satisfied.

3.3.5.1 Uncontrolled RCCA Withdrawal From A Subcritical Condition

The uncontrolled RCCA withdrawal from a subcritical condition event is analyzed to determine the impact of the reduced RCS flow as a result of the increased steam generator tube plugging level of 30%. This event is analyzed to demonstrate core protection. Although the no-load temperature does not change for the SGTP Program, the reduction in nominal RCS flow is non-conservative with respect to the DNB transient.

An RCCA withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of RCCA banks resulting in a power excursion. While the occurrence of a transient of this type is highly unlikely, such a transient could be caused by a malfunction of the Reactor Control or Control Rod Drive Systems. This could occur with the reactor either subcritical or at power. The "at power" case is discussed in Section 3.3.5.2.

Reactivity is added at a prescribed and controlled rate in bringing the reactor from a shutdown condition to a low power level during startup by RCCA bank withdrawal. Although the initial startup procedure uses the method of boron dilution, the normal startup is with RCCA bank withdrawal. RCCA bank motion can cause much faster changes in reactivity than can be made by changing boron concentration.

The control rod drive mechanisms are wired into preselected banks, and these bank configurations are not altered during the core life. The RCCA's are therefore physically prevented from being withdrawn in other than their respective banks. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. The RCCA drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed by assuming the simultaneous withdrawal of the combination of the two banks of the maximum combined worth at maximum speed.

Should a continuous control rod assembly withdrawal be initiated, the transient will be terminated by the following reactor trip functions.

1. Source range neutron flux level trip - actuated when either of two source range channels indicates a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed when either intermediate range flux channel indicates a flux level above the source range cutoff level. It is

automatically reinstated when both intermediate range channels indicate a flux level below the source range cutoff level.

2. Intermediate range neutron flux level trip - actuated when either of two intermediate range channels indicates a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed when two of the four power range channel are reading above approximately 10 percent of full power flux and is automatically reinstated when three of the four power range channels indicate a flux level below this value.

3. Power range neutron flux level trip (low setting) - actuated when two out of the four power channels indicate a flux level above approximately 25 percent of full power flux. This trip function may be manually bypassed when two of the four power range channels indicate a flux level above approximately 10 percent of full power flux and is automatically reinstated when three of the four channels indicate a flux level below this value.

4. Power range neutron flux level trip (high setting) - actuated when two out of the four power range channels indicate a flux level above a preset setpoint. This trip function is always active.

In addition, control rod stops on high intermediate range flux level and high power range flux level serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast power rise terminated by the reactivity feedback effect of the negative fuel temperature coefficient. This self-limitation of the initial power burst results from a fast negative fuel temperature feedback (Doppler effect) and is of prime importance during a startup incident since it limits the power to a tolerable level prior to protective action. After the initial power burst, the neutron flux is momentarily reduced and then, if the incident is not terminated by a reactor trip, the neutron flux increases again, but at a much slower rate.

Termination of the startup incident by the previously discussed protection channels prevents core damage. In addition, the reactor trip from pressurizer high pressure serves as a backup to terminate the incident before an overpressure condition could occur.

Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first an average core nuclear power transient calculation, then an average core heat transfer calculation, and finally the departure from nucleate boiling ratio (DNBR)

calculation. The average core nuclear calculation is performed using spatial neutron kinetics methods (TWINKLE) to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. The average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN. The average heat flux is next used in THINC IV for transient DNBR calculations.

Analysis of this transient incorporates the neutron kinetics, including six delayed neutron groups and the core thermal and hydraulic equations. In addition to the neutron flux response, the average fuel, clad and water temperature, and also the heat flux response, are computed.

In order to give conservative results for a startup incident, the following additional assumptions are made concerning the initial reactor conditions:

1. Since the magnitude of the neutron flux peak reached during the initial part of the transient, for any given rate of reactivity insertion, is strongly dependent on the Doppler power reactivity coefficient, a conservatively low value (i.e., small in absolute value) is used for the startup incident (-0.9×10^{-4} Δk/% power).

2. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the neutron flux response time constant. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator temperature reactivity coefficient. Although during normal operation (100% rated power) the moderator coefficient will not be positive at any time in core life, a highly conservative value has been used in the analysis to yield the maximum peak core heat flux. The analysis is based on a moderator coefficient which was at least +5 pcm/°F at the zero power nominal average temperature, and which became less positive for higher temperatures. This was necessary since the TWINKLE computer code used in the analysis is a diffusion theory code rather than a point kinetics approximation and the moderator temperature feedback cannot be artificially held constant with temperature.

3. The reactor is assumed to be at hot zero power (547°F). This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel to water heat transfer, a larger fuel thermal capacity, and a less negative (smaller absolute magnitude) Doppler coefficient. The less negative Doppler coefficient reduces the Doppler feedback effect thereby increasing the neutron flux peak. The high neutron flux peak combined with a high fuel thermal capacity and larger thermal conductivity

yields a larger peak heat flux. Initial multiplication factor (k_o) is assumed to be closely approaching 1.0 since this results in the maximum neutron flux peak.

4. Two reactor coolant pumps are assumed to be in operation. This is conservative with respect to the DNB transient.

5. The most adverse combination of instrumentation and setpoint errors, as well as delays for trip signal actuation and control rod assembly release, are taken into account. A 10% increase has been assumed for the power range flux trip setpoint raising it from the nominal value of 25% to a value of 35% in addition to taking no credit for the source and intermediate range protection. Reference to Figure 3.3-5, however, shows that the rise in nuclear flux is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition to the above, the rate of negative reactivity insertion corresponding to the trip action is based on the assumption that the highest worth control rod assembly is stuck in its fully withdrawn position.

The accident is analyzed using the Standard Thermal Design Procedure with the initial conditions listed in Table 3.3-4. The analysis was performed for a reactivity insertion rate of 75 pcm*/sec. This reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute).

* 1 pcm = 10^{-5} $\Delta k/k$

Results and Conclusions

The nuclear power, heat flux, fuel average temperature, and clad temperature versus time for a 75 pcm/sec insertion rate are shown in Figures 3.3-5 and 3.3-6. This insertion rate, coupled with the 30% SGTP conditions, yields a minimum DNBR which remains above the limit value. For the Rod Withdrawal from subcritical event, the core axial power distribution is severely peaked to the bottom of the core. The W-3 DNB correlation is used to evaluate DNBR in the span between the lower non-mixing vane grid and the first mixing vane grid. The WRB-1 correlation remains applicable for the rest of the fuel assembly. For all regions of the core the DNB design bases are met.

3.3.5.2 Uncontrolled Control Rod Assembly Bank Withdrawal At Power

An uncontrolled Rod Cluster Control Assembly (RCCA) withdrawal at power results in an increase in core heat flux. Since the heat extraction from the steam generator lags behind the power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, to minimize the possibility of breaching the cladding, the Reactor Protection System is designed to terminate any such transient before the DNBR falls below the limit value.

The automatic features of the Reactor Protection System which minimize adverse effects to the core in an RCCA Bank Withdrawal incident at power include the following:

1. Nuclear power range instrumentation actuates a reactor trip on high neutron flux if two out of four channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two out of four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power distribution, coolant average temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two out of four ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with coolant average temperature so that the allowable fuel power rating is not exceeded.

4. A high pressure reactor trip, actuated from any two out of four pressure channels, is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

5. A high pressurizer water level reactor trip, actuated from any two out of three level channels, is set at a fixed point.

In addition to the above listed reactor trips, there are the following RCCA Withdrawal blocks.

a. High neutron flux (one out of four)
b. Overpower ΔT (two out of four)
c. Overtemperature ΔT (two out of four)

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of Reactor Coolant System conditions is illustrated in Figures 3.3-1 through 3.3-4. These figures represent the allowable conditions of reactor

coolant loop average temperature and power with the design power distribution in a two-dimensional plot.

The purpose of this analysis is to demonstrate the manner in which the above protective systems function for various reactivity insertion rates from different initial conditions. Reactivity insertion rates and initial conditions govern which protective function occurs first.

Method of Analysis

This transient is analyzed by the LOFTRAN code. The core limits as illustrated in Figure 3.3-1 through 3.3-4 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

The analysis is performed to bound the conditions of high and low average temperature with high and low RCS pressures for Unit 1.

This accident is analyzed with the RTDP described in Reference 2. Plant characteristics and initial conditions are listed in Table 3.3-4. For an uncontrolled rod withdrawal at power accident, the following conservative assumptions are made:

A. Nominal values are assumed for the initial reactor power, pressure, and RCS temperatures. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 2.

B. Reactivity coefficients - two cases are analyzed:

   1. Minimum Reactivity Feedback. A +5 pcm/°F moderator temperature coefficient of reactivity and a least negative Doppler only power coefficient (see Table 3.3-4) are assumed.

   2. Maximum Reactivity Feedback. A conservatively large negative moderator temperature coefficient and a most negative Doppler only power coefficient (See Table 3.3-4) are assumed.

C. 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, while the delays for the trip signal actuation are assumed at their maximum values.

D. The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

E. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combinations of the two control banks having the maximum combined worth at maximum speed.

Results

Figures 3.3-7 through 3.3-9 show the transient response for a rapid RCCA bank 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 T_{avg} and pressure result and margin to DNB is maintained.

The transient response for a slow RCCA bank withdrawal from full power is shown in Figures 3.3-10 through 3.3-12. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA bank withdrawal. Again, the minimum DNBR is greater than the limit value.

Figure 3.3-13 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 functions provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT functions. The minimum DNBR is always greater than the limit value.

Figures 3.3-14 and 3.3-15 show the minimum DNBR as a function of reactivity insertion rate for RCCA bank withdrawal incidents starting at 60 and 10 percent power respectively. 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 neither case does the DNBR fall below the limit value.

Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than the limit value for all fuel types. Also, the pressurizer does not fill.

3.3.5.3 Rod Cluster Control Assembly Misalignment

The rod cluster control assembly misalignment events are primarily examined to demonstrate core protection. Although the reduction in rated thermal power is a benefit for the DNB evaluation, the reduction in RCS flow is non-conservative with respect to the DNB transient. As such, the rod cluster control assembly misalignment events are analyzed to determine the impact of the SGTP Program.

Rod cluster control assembly misalignment accidents include:

A. A dropped RCCA
B. A dropped RCCA bank
C. Statically misaligned RCCA

Each RCCA has a position indicator channel which displays position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by rod bottom light. Group demand position is also indicated.

RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the secondary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the RCCAs of a rod group are driven in parallel, any single malfunction which would cause rod withdrawal would affect a minimum of one group. Mechanical malfunctions are in the direction of insertion, or immobility.

A dropped RCCA or RCCA bank is detected by:

a. Sudden drop in the core power level as seen by the nuclear instrumentation system;

b. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;

c. Rod at bottom signal;

d. Rod position deviation monitor;

e. Rod position indication.

Misaligned RCCA are detected by:

a. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;

b. Rod position deviation monitor;

c. Rod position indicators.

The resolution of the rod position indicator channel is ±5 percent (±12 steps). Deviation of any assembly from its group by twice this distance will not cause power distributions worse than the design limits. The rod position deviation monitor alerts the operator to rod deviation before it can exceed ten percent of span (± 24 steps). If the rod position deviation monitor is not operable, the operator is required to take action as required by the Technical Specifications.

Method of Analysis

A. One or more dropped RCCAs from the same group.

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Nominal values for initial reactor power, temperature, and RCS pressure are assumed to bound the operation of Unit 1 with 30% SGTP. The initial conditions are presented in Table 3.3-4. Uncertainties for initial conditions are included in the limit DNBR.

Statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the THINC IV code. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Reference 9. Note that operation with automatic rod control is assumed for the analysis. Also note that the analysis does not take credit for the negative flux rate reactor trip.

B. Statically Misaligned RCCA

Steady state power distributions are analyzed using the methodology described in Reference 9. The peaking factors are then used as input to the THINC IV code to calculate the DNBR.

Results

A. One or more Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. The core is not adversely affected during this period, since power is decreasing rapidly. Following plant stabilization, normal rod retrieval or shutdown procedures are followed. The operator may manually retrieve the RCCA by following approved operating procedures.

Power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control-system interaction is monotonic, thus removing power overshoot as a concern, and establishing the automatic rod control mode of operation as the limiting case.

For a dropped RCCA event in the automatic rod control mode, the Rod Control System detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figures 3.3-16 and 3.3-17 show a typical transient response to a dropped RCCA (or RCCAs) in automatic control. Uncertainties in the initial condition are included in the DNB evaluation as described in Reference 9. In all cases, the minimum DNBR remains above the limit value.

B. Dropped RCCA bank

A dropped RCCA bank typically results in a reactivity insertion greater than 500 pcm. The core is not adversely affected during the insertion period, since power is decreasing rapidly. The transient will proceed as described in "A" above; however, the return to power will be less due to the greater worth of an entire bank. Following plant stabilization, normal rod retrieval or shutdown procedures are followed to further cool down the plant.

C. Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted, or where bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value.

The insertion limits in the Technical Specifications may vary from time to time depending on a number of limiting criteria. It is preferable, therefore, to analyze the misaligned RCCA case at full power for a position of the control bank as deeply inserted as the criteria on minimum DNBR and power peaking factor will allow. The full power insertion limits on control bank D must then be chosen to be above that position and will usually be dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

With bank D inserted to its full insertion limit and one RCCA fully withdrawn, DNBR does not fall below the limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values (as given in Table 3.3-4) but with the increased radial peaking factor associated with the misaligned RCCA.

DNB calculations have not been performed specifically for RCCAs missing from other banks; however, power shape calculations have been done as required for RCCA ejection analysis. Inspection of the power shapes shows that the DNB and peak kw/ft situation is less severe than the bank D case discussed above assuming insertion limits on the other banks equivalent to a bank D insertion limit.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values, (as given in Table 3.3-4) but with the increased radial peaking factor associated with the misaligned RCCA.

DNB does not occur for the RCCA misalignment incident and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which would cause fuel melting.

Following the identification of a RCCA group misalignment condition by the operator, the operator is required to take action as required by the plant Technical Specifications and operating instructions.

3.3.5.4 Loss of Reactor Coolant Flow (Including Locked Rotor Analysis)

A loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature which is magnified by a positive MTC. This increase could result in DNB with subsequent adverse effects to the fuel if the reactor were not tripped promptly. The trip systems available to mitigate the consequence of this accident are discussed in the UFSAR.

Simultaneous loss of electrical power to all reactor coolant pumps at full power is the most severe credible loss of flow condition. For this condition reactor trip together with flow sustained by the inertia of the coolant and rotating pump parts will be sufficient to prevent RCS overpressurization and the DNB ratio from exceeding the limit values.

The decrease in reactor coolant system flow rate events are primarily examined to demonstrate core protection. The reduction in RCS flow, as a result of the increase in steam generator tube plugging to 30%, is non-conservative with respect to the DNB transient. As such, analyses are presented to discuss the impact of this change.

Method of Analysis

The following loss of flow cases are analyzed:

1. Loss of four pumps from nominal full power conditions with four loops operating.

2. Loss of one pump from nominal full power conditions with four loops operating.

The normal power supplies for the pumps are four buses connected to the generator. Each bus supplies power to one pump. When a generator trip occurs, the pumps are automatically transferred to a bus supplied from external power lines, and the pumps will continue to supply coolant flow to the core. The simultaneous loss of power to all reactor coolant pumps is a highly unlikely event. Since each pump is on a separate bus, a single bus fault would not result in the loss of more than one pump.

A full plant simulation is used in the analysis to compute the core average and hot spot heat flux transient responses, including flow coastdown, temperature, reactivity and control rod insertion effects.

These data are then used in a detailed thermal-hydraulic computation to compute the margin to DNB using the RTDP. This computation solves the continuity, momentum and energy equations of fluid flow together with the WRB-1 DNB correlation.

The analyses are performed to bound the conditions of the SGTP Program. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 2. Nominal values are assumed for the initial reactor power, pressure, and RCS temperatures. The initial conditions used are listed in Table 3.3-4.

This transient is analyzed by three digital computer codes. First the LOFTRAN code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN.

Finally, the THINC IV code is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble cell for each type of fuel.

Results

Figures 3.3-18 through 3.3-20 show the transient response for the loss of power to all RCPs with four loops in operation. The reactor is assumed to be tripped on undervoltage signal. Figure 3.3-20 shows the DNBR to be always greater than the limit value for the most limiting fuel assembly cell.

Figures 3.3-21 through 3.3-23 show the transient response for the loss of one RCP with four loop operation. The reactor is assumed to be tripped on low flow signal. Figure 3.3.-23 shows the DNB to be always greater than the limit value for the most limiting fuel assembly cell.

The sequence of events following each of these transients is included in Table 3.3-5..

Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not significantly reduced. Thus, the average fuel and clad temperature do not increase significantly above their respective initial values.

Conclusions

The analysis shows that the DNBR will not decrease below the limit value at any time during the transient. Thus, no fuel adverse effects or clad rupture is predicted, and all applicable acceptance criteria are met.

Locked Rotor Accident

A transient analysis has been performed for the instantaneous seizure of a reactor coolant pump rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the steam generator causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer causes a pressure increase which in turn actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in a sequence dependent on the rate of insurge and pressure increase. The power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect as well as the pressure-reducing effect of the spray are not included in this analysis.

The locked rotor event is examined to determine the DNB transient and to demonstrate that the peak RCS pressure and peak clad temperature remain below the limit values. The reduction in RCS flow, due to the increase in the SGTP level, is non-conservative with respect to the DNB evaluation. As such, the locked rotor event was re-analyzed.

Method of Analysis

Two digital-computer codes are used to analyze this transient. The LOFTRAN code is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN code, using the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes the use of a film boiling heat transfer coefficient.

The analysis is performed to bound the conditions associated with the SGTP Program. As in the previous UFSAR analysis, the analysis assumes offsite power is available following the reactor trip and turbine trip.

Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion begins 1 second after the flow in the affected loop reaches 87 percent of nominal flow. No

credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip.

Although these operations are expected to occur and would result in a lower peak RCS pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are assumed to initially open at 2575 psia and achieve rated flow at 2580 psia. This analysis assumed an initial pressurizer pressure of 2317 psia. Table 3.3-4 presents the initial conditions assumed for the peak pressure transient.

Evaluation of the Peak Clad Temperature

For this accident, DNB is assumed to occur in the core; therefore an evaluation of the consequences with respect to fuel rod thermal transients is performed. The assumption of rods going into DNB as a conservative initial condition is made in order to determine the clad temperature and zirconium water reaction. This analysis assumed an initial pressurizer pressure of 2100 psia. Results obtained from analysis of this hot spot condition represent the upper limit with respect to clad temperature and zirconium water reaction. In the evaluation, the rod power at the hot spot is assumed to be 2.5 times the average rod power (i.e., $F_Q = 2.5$) at the initial core power level. Table 3.3-4 presents the initial conditions assumed for the peak clad temperature transient.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN code using the Bishop-Sandberg-Tong film boiling correlation (Reference 13). The fluid properties are evaluated at film temperatures (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

For the peak clad temperature analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to clad temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during

the transient, the gap coefficient was assumed to increase from a steady state value consistent with initial fuel temperature to 10,000 BTU/hr-ft^2-°F at the initiation of the transient. Thus the large amount of energy stored in the fuel because of the small initial value is released to the clad at the initiation of the transient.

Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). In order to take this phenomenon into account, the following correlation, which defines the rate of the zirconium-steam reaction, was introduced into the models (Reference 10).

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp\left(-\frac{45,500}{1.986T}\right)$$

where:

w = amount reacted, mg/cm^2
t = time, seconds
T = temperature, K
The reaction heat is 1510 cal/g

Evaluation of Rods-in-DNB

An evaluation is made to determine what percentage, if any, of rods are expected to be in DNB during the transient. For this evaluation, the predicted core conditions are used as input to a THINC IV calculation of the minimum DNBR during the transient. Results of the THINC IV evaluation are then used to determine the percentage of fuel rods which experience DNB. Table 3.3-4 presents the initial conditions assumed for the rods-in-DNB evaluation.

Results

The transient results for the locked rotor accident are shown in Figures 3.3-24 through 3.3-26. The peak RCS pressure (2641 psia) reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits (this peak pressure is also below 110% of the design pressure). The pressure response shown in Figure 3.3-25 is the response at the point in the Reactor Coolant System having the maximum pressure. Also, the peak clad surface temperature (1934°F, shown in Figure 3.3-26) is considerably less than 2700°F. The sequence of events is included in Table 3.3-5.

For the most limiting fuel assembly, less than 7% of the rods reach a DNBR value less than the limit value for the 30% SGTP conditions.

Conclusions

A. Since the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted conditions stress limits, the integrity of the primary coolant system is not endangered.

B. Since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F (the temperature at which clad embrittlement may be expected), the core will remain in place and intact with no loss of core cooling capability.

3.3.5.5 Loss of External Electrical Load

The complete loss of steam load from full power is examined primarily to show the adequacy of the pressure relieving devices and also to demonstrate core protection. The reduction in RCS flow, as result of increasing the level of SGTP, is non-conservative with respect to the DNB behavior. Primary protection for this event is provided by the high pressurizer pressure, OTΔT, high pressurizer water level, and low-low steam generator water level reactor trips.

The loss of external electrical load may result from an abnormal variation in network frequency or other adverse network operating conditions. It may also result from a trip of the turbine generator or in an unlikely opening of the main breaker from the generator which fails to cause a turbine trip but causes a rapid large NSSS load reduction by the action of the turbine control.

Method of Analysis

The loss of load transients are analyzed by employing the detailed digital computer program LOFTRAN. The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

An analysis is performed to bound the conditions of the SGTP Program. Nominal values are assumed for the initial reactor power, temperature, and pressure. This accident is analyzed with the RTDP. Plant characteristics and initial conditions are listed in Table 3.3-4.

Major assumptions are summarized below:

A. Initial Operating Conditions - nominal conditions for reactor power, pressure, and RCS temperatures are assumed for statistical DNB analyses.

B. Moderator and Doppler Coefficients of Reactivity - the loss of load is analyzed with both maximum and minimum reactivity feedback. The maximum feedback cases assume a large negative moderator temperature coefficient and the most negative Doppler power coefficient. The minimum feedback cases assume a positive moderator temperature coefficient and the least negative Doppler coefficients.

C. Reactor Control - from the standpoint of the maximum pressures attained it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

D. Pressurizer Spray and Power-Operated Relief Valves - two cases for both the minimum and maximum moderator feedback cases are analyzed:

1. Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available.

2. No credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable.

E. Steam Release - no credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through the safety valves limits the secondary steam pressure.

F. Feedwater Flow - main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur; however, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

G. Reactor trip is actuated by the first Reactor Protection System trip setpoint reached. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, high pressurizer water level, and low-low steam generator water level.

Results

The transient responses for a loss of load from full power operation are shown for four cases: minimum and maximum reactivity feedback, with and without pressure control (Figures 3.3-27 through 3.3-46).

Figures 3.3-27 through 3.3-31 show the transient responses for the loss of load with minimum reactivity feedback assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the overtemperature ΔT trip signal.

The minimum DNBR remains well above the limit value. The pressurizer relief and safety valves prevent overpressurization of the primary system. The steam generator safety valves

prevent overpressurization of the secondary system, maintaining pressure below 110 percent of design value.

Figures 3.3-32 through 3.3-36 show the responses for the total loss of steam load with maximum reactivity feedback. All other plant parameters are the same as the above. The DNBR increases throughout the transient and never drops below its initial value. Pressurizer relief valves and steam generator safety valves prevent overpressurization in primary and secondary systems, respectively. The reactor is tripped by the low-low steam generator water level signal. The pressurizer safety valves are not actuated for this case.

In the event that feedwater flow is not terminated at the time of turbine trip for this case, flow would continue under automatic control with the reactor at a reduced power. The operator would take action to terminate the transient and bring the plant to a stabilized condition. If no action were taken by the operator the reduced power operation would continue until the condenser hotwell was emptied. A low-low steam generator water level reactor trip would be generated along with auxiliary feedwater initiation signals. Auxiliary feedwater would then be used to remove decay heat with the results less severe than those presented in Section 3.3.4.3, Loss of Normal Feedwater Flow.

The loss of load accident was also studied assuming the plant to be initially operating at full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figures 3.3-37 through 3.3-41 show the transient responses with minimum reactivity feedback. The neutron flux remains essentially constant at full power until the reactor is tripped. The DNBR never goes below its initial value throughout the transient. In this case the pressurizer safety valves are actuated, and maintain system pressure below 110 percent of the design value.

Figures 3.3-42 through 3.3-46 show the transient responses with maximum reactivity feedback with the other assumptions being the same as in the preceding case. Again, the DNBR increases throughout the transient and the pressurizer safety valves are actuated to limit primary pressure.

The sequence of events following each of these transients is included in Table 3.3-6.

<u>Conclusions</u>

Results of the analyses show that the plant design is such that a loss of load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits. The integrity of the core is maintained by

operation of the reactor protection system, i.e., the DNBR will be maintained above the limit value.

Thus the conclusions presented is the UFSAR remain valid for 30% SGTP.

3.3.5.6 Rupture of a Steam Pipe

Although the no-load temperature does not change for the SGTP Program, and a reduction in the heat transfer capability, due to the increased number of plugged steam generator tubes, would result in a less severe cooldown, the impact of the RCS flow reduction needs to be addressed for the steamline break accident. The reanalysis also assumed a reduction in the available shutdown margin from 1.60 to 1.30% Δk/k at no-load conditions. An evaluation has been performed for those cases that model a coincident loss of offsite power in order to address the increase in the EDG start time from 10 to 30 seconds.

This analysis was performed assuming the coincidence logic required for SI and SLI consistent with the current Unit 1 steamline break protection system. A proposed modification to the Unit 1 steamline break protection system will change this logic. However, this analysis bounds the proposed modifications to the Unit 1 steamline break protection system, as discussed in Section 3.3.2.5.

A rupture of a steam pipe results in an uncontrolled steam release from a steam generator. The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the causes a reduction in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient (MTC), the cooldown results in a reduction of core shutdown margin. If the most reactive RCCA is assumed stuck in its fully withdrawn position, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential concern mainly because of the high hot channel factors which exist when the most reactive assembly is assumed stuck in its fully withdrawn position. The core is ultimately shut down by boric acid delivered by the ECCS.

The analysis of a steam pipe rupture is performed to demonstrate that:

> Assuming a stuck assembly, with or without offsite power, and assuming a single failure in the engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact.
>
> Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

Method of Analysis

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

A. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN code has been used.

B. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer code, THINC IV, has been used to determine if DNB occurs for the core conditions computed in item A above.

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

A. End-of-life shutdown margin (1.30% $\Delta k/k$) at no load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position: Operation of the control rod banks during core burnup is restricted in such a way (to not violate the rod insertion limits presented in the Technical Specifications) that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

B. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position: The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus temperature at 1050 psia corresponding to the negative moderator temperature coefficient used is shown in Figure 3.3-47. The Doppler power feedback assumed for this analysis is presented in Figure 3.3-48.

The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculation. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod. To verify the conservatism of this method, the reactivity as well as the power distribution was checked for the limiting conditions for the cases analyzed. This core analysis considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and non-uniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity

calculated including the above local effects for the statepoints. These results verify conservatism; i.e., underprediction of negative reactivity feedback from power generation.

C. Minimum capability for injection of boric acid (2400 ppm) solution corresponding to the most restrictive single failure in the safety injection system. The ECCS consists of the following systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, 3) the high head safety injection system, and 4) the charging system. Only the charging system and the passive accumulators are modeled for the steam line break accident analysis. Centrifugal Charging pump head degradation of 10% was assumed.

The modeling of the safety injection system in LOFTRAN is described in Reference 4. Figure 3.3-49 presents the safety injection flow rates as a function of RCS pressure assumed in the analysis. The flow corresponds to that delivered by one charging pump delivering its full flow to the cold leg header. No credit has been taken for the low concentration borated water, which must be swept from the lines downstream of the RWST prior to the delivery of boric acid to the reactor coolant loops. For this analysis, a boron concentration of 0 ppm for the boron injection tank is assumed. It should be noted that this analysis also considers the operation of the Centrifugal Charging Pump Minimum Flow Isolation Valves. These valves are assumed to close following the receipt of a SI signal and reopen when RCS pressure rises above 2000 psig. The SI flow rates assumed in the steamline break analysis, graphically shown in Figure 3.3-49, correspond to Centrifugal Charging Pump Minimum Flow Isolation Valves being in the closed position.

For the cases where offsite power is assumed, the sequence of events in the safety injection system is the following. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the charging pump starts. In 27 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed and to draw suction from the RWST. The volume containing the low concentration borated water is swept into core before the 2400 ppm borated water reaches the core. This delay, described above, is inherently included in the modeling.

In cases where offsite power is not available, a 30 second delay is assumed to start the EDGs and to commence loading the necessary safety injection equipment onto them.

D. Design value of the steam generator heat transfer coefficient including allowance for fouling factor.

E. Four combinations of break sizes and initial plant conditions have been considered in determining the core power transient which can result from large area pipe breaks.

a. Complete severance of a pipe downstream of the steam flow restrictor with the plant initially at no load conditions and all reactor coolant pumps running.

b. Complete severance of a pipe inside the containment at the outlet of the steam generator with the same plant conditions as above.

c. Case (a) above with loss of off-site power simultaneous with the generation of the Safety Injection Signal (loss of AC power results in coolant pump coastdown).

d. Case (b) above with the loss of off-site power simultaneous with the Safety Injection Signal.

A fifth case, in which the spurious opening of a steam dump, relief, or safety valve occurs, was considered. An evaluation concluded that the DNBR remains above the limit value for this case.

e. A break equivalent to a steam flow of 247 lbs per second at 1100 psi from one steam generator with off-site power available.

F. Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow, and, thus, are different for each case studied.

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

reaches a trip point. Following a trip at power the RCS contains more stored energy than at no-load, the average coolant temperature is higher than at no-load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no-load condition at time zero.

In addition, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the RCS cooldown are more severe than steam line breaks occurring at power.

G. In computing the steam flow during a steam line break, the Moody Curve (Reference 11) for fl/D = O is used.

H. The fast acting steamline isolation valves are assumed to close in less than 11 seconds from receipt of actuation signal. The 11 second closure time of the isolation valves is based upon the actuating signal being generated by the steam flow in two steam lines - high coincident with steam line pressure - low functions. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the isolation valves fails to close.

<u>Results</u>

The limiting case for Cases a through e was shown to be the double-ended rupture located upstream of the flow restrictor with offsite power available. Table 3.3-7 lists the limiting statepoint for this worst case. The results presented are conservative indication of the events which would occur assuming a steam line rupture since it is postulated that all of the conditions described above occur simultaneously.

Figures 3.3-50 through 3.3-53 show the RCS transients and core heat flux following a main steam line rupture (complete severance of a pipe) upstream of the flow restrictor at initial no-load condition. The sequence of events for this transient is presented in Table 3.3-8.

Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by high differential pressure between any steamline and the remaining steamlines or by high steam flow signals in coincidence with either low-low RCS temperature or low steam line

pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast acting isolation valves in the steam lines by high containment pressure signals or high steam flow coincident with low steamline pressure or low-low T_{avg}. Even with the failure of one valve, release is limited to approximately 13 seconds for the other steam generators while the one generator blows down. The steam line stop valves are assumed to be fully closed in less than 11 seconds from receipt of a closure signal (steam flow in two steam lines - high coincident with steam line pressure-low).

As shown in Figure 3.3-53, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 2400 ppm enters the RCS. A peak core power less than the nominal full power value is attained.

The calculation assumes the boric acid is mixed with, and diluted by the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and in the safety injection system. The variation of mass flow rate in the RCS due to water density changes is included in the calculation as is the variation of flow rate in the safety injection system due to changes in the RCS pressure. The safety injection system flow calculation includes the line losses in the system as well as the pump head curve. Note that since the RCS pressure (Figure 3.3-51) drops below 2015 psia and never repressurizes above that value, the automatic operation to open the Centrifugal Charging Pump Minimum Flow Isolation Valves would not occur during this event. Therefore, there would be not reduction in SI flow below that assumed in the safety analysis.

The assumed steam release for an accidental depressurization of the main steam system (Case e) is the maximum capacity of any single steam dump, relief, or safety valve. Safety injection is initiated automatically by low pressurizer pressure. Operation of one centrifugal charging pump is assumed. Boron solution at 2400 ppm enters the RCS providing sufficient negative reactivity to prevent core damage. The transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements or the energy stored in the other steam generators. Since the transient occurs over a period of about 5 minutes, the neglected stored energy is likely for this event to have a significant effect in slowing the cooldown. The DNB transient is bounded by the limiting case for a steamline rupture.

The DNB analysis for the limiting case (double-ended rupture located upstream of the flow restrictor) showed that the minimum DNBR remained above the limit value.

<u>Conclusions</u>

The analysis has shown that the criteria stated earlier are satisfied.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis, in fact, shows that no DNB occurs for the rupture (including an accidental depressurization of the main steam system) assuming the most reactive RCCA stuck in its fully withdrawn position.

3.3.5.7 Rupture of Control Rod Drive Mechanism Housing (RCCA Ejection)

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a RCCA and drive shaft. The consequence of this mechanical failure, in addition to being a small break loss-of-coolant accident, is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. This event has been analyzed as part of the 30% SGTP Program to address the reduction in RCS flow due to the increase in the SGTP level.

If an RCCA ejection accident were to occur, a fuel rod thermal transient which could cause DNB may occur together with limited fuel damage. The amount of fuel damage that can result from such an accident will be governed mainly by the worth of the ejected RCCA and the power distribution attained with the remaining control rod pattern. The transient is limited by the Doppler reactivity effects of the increase in the fuel temperature and is terminated by reactor trip actuated by neutron flux signals, before conditions are reached that can result in damage to the reactor coolant pressure boundary, or significant disturbances in the core, its support structures or other reactor pressure vessel internals which would impair the capability to cool the core.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast flux increase terminated by the reactivity feedback effect of the Doppler coefficient. This self limitation of the power burst is of primary importance since it limits the power to a tolerable level during the delay time for protective action. Should a RCCA Ejection accident occur, the following automatic features of the RPS are available to terminate the transient.

a. The source-range high neutron flux reactor trip is actuated when either of the independent source-range channels indicates a neutron flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed when either intermediate-range flux channel indicates a flux level above a specified level. It is automatically reinstated when both intermediate-range channels indicate a flux level below a specified level.

b. The intermediate-range high neutron flux reactor trip is actuated when either of two independent intermediate-range channels indicates a flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed when two of the four power-range channels give readings above

approximately 10% of full power and is automatically reinstated when three of the four channels indicate a power below this value.

c. The power-range high neutron flux reactor trip (low setting) is actuated when two-out-of-four power-range channels indicate a power level above approximately 25% of full power. This trip function may be manually bypassed when two of the four power-range channels indicate a power level above approximately 10% of full power and is automatically reinstated when three of the four channels indicate a power level below this value.

d. The power-range high neutron flux reactor trip (high setting) is actuated when two-out-of -four power-range channels indicate a power level above a preset setpoint (typically 109% of full power). This trip function is always active.

e. The high nuclear flux rate reactor trip is calculated when the positive rate of change of neutron flux on two-out-of-four nuclear power-range channels indicates a rate above the preset setpoint. This trip function is always active.

Due to the extremely low probability of a RCCA Ejection accident, this event is classified as an ANS Condition IV event (Limiting Fault). The ultimate acceptance criteria for this event is that any consequential damage to either the core or the RCS must not prevent long-term cooling, and that any offsite dose consequences must be within the guidelines of 10 CFR 100. To demonstrate compliance with these requirements, it is sufficient to show that the RCS pressure boundary remains intact, and that no fuel dispersal into the coolant, gross lattice distortions, or severe shock waves will occur in the core. Therefore, the limiting criteria is described in Reference 12.and summarized below:

A. Average fuel pellet enthalpy at hot spot below 225 cal/g for unirradiated fuel and 200 cal/g for irradiated fuel.

B. Average clad temperature at the hot spot below the temperature at which clad embrittlement may be expected (3000°F).

C. Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits.

D. Fuel melting will be limited to less than ten percent 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion A above.

The analysis performed is to bound the parameters associated with the increased SGTP level of 30%.

Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient. A detailed discussion of the method of analysis can be found in Reference 12.

Average Core Analysis

The spatial kinetics computer code, TWINKLE, is used for the average core transient analysis. This code solves the two group neutron diffusion theory kinetic equation in one, two or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multi-region, transient fuel-clad-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in Section 3.3.3.2.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spot before and after ejection are coincident. This is very conservative since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel and clad transient heat transfer computer code, FACTRAN. This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A conservative radial power distribution is used within the fuel rod.

FACTRAN uses the Jens-Lottes or Dittus-Boelter correlation (References 8 and 15, respectively) to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation (see Reference 13) to determine the film boiling coefficient after DNB. The Bishop-Sandberg-Tong correlation is conservatively used assuming zero bulk fluid quality. The DNB ratio is not calculated, instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady-state temperature distribution to agree with the fuel heat transfer design codes. Further description of FACTRAN appears in Section 3.3.3.2.

A detailed three-dimensional calculation of a worst case scenario (Reference 12) demonstrates an upper limit to the number of rods-in-DNB for the RCCA Ejection accident as 10%. Since the severity of the Cook Nuclear Plant Unit 1 analysis does not exceed this worst case analysis, the maximum number of rods in DNB following a RCCA Ejection will be less than 10%, although neither the number of rods in DNB nor the minimum DNBR value is explicitly calculated in the Cook Nuclear Plant Unit 1 analysis. The most limiting break size resulting from a RCCA Ejection will not be sufficient to uncover the core or cause DNB at any later time. Since the maximum number of fuel rods experiencing DNB is limited to 10%, the fission product release will not exceed that associated with the guidelines of 10 CFR 100.

System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore by calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC IV calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in the LOFTRAN computer code. This code calculates the pressure transient taking into account fluid transport in the RCS and heat transfer to the steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 3.3-9 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using either three dimensional static methods or by a synthesis method employing one dimensional and two dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation to provide worst case results.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties, including an allowance for nuclear power peaking due to densification.

Power distribution before and after ejection for a worst case can be found in Reference 12. During plant startup physics testing, ejected rod worths and power distributions are measured in the zero and full power configurations and compared to values used in the analysis. Experience has shown that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single channel feedbacks, correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one dimensional (axial) spatial kinetics method is employed, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to three dimensional analysis (Reference 12).

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning of life and end of life are adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative

compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results. The resulting moderator temperature coefficient is at least +5 pcm/°F at the appropriate zero or full power nominal average temperature, and becomes less positive for higher temperatures. This is necessary since the TWINKLE computer code utilized in the analyses is a diffusion-theory code rather than a point-kinetics approximation and the moderator temperature feedback cannot be artificially held constant with temperature.

The Doppler reactivity defect is determined as a function of power level using a one dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler weighting factor will increase under accident conditions, as discussed above.

Delayed Neutron Fraction, β_{eff}

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.70% at beginning of life and 0.50% at end of life. The accident is sensitive to β_{eff} if the ejected rod worth is equal to or greater than β_{eff} as in zero power transients. In order to allow for future cycles, pessimistic estimates of β_{eff} of 0.50% at beginning of a cycle and 0.40% at end of a cycle were used in the analysis.

Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 3.3-9 and includes the effect of one stuck RCCA adjacent to the ejected rod. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip points is reached before significant shutdown reactivity is inserted into the core. This is particularly important conservatism for hot full power accidents.

The minimum design shutdown margin available for this plant at HZP may be reached only at end of life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, an adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations have shown that the effect of two stuck RCCAs (one of which is the worst ejected rod) is to reduce the shutdown by about an additional 1% Δk/k. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to HZP.

Depressurization calculations have been performed assuming the maximum possible size break (2.75 inch diameter) located in the reactor pressure vessel head. The results show a rapid pressure drop and a decrease in system water mass due to the break. The ECCS is actuated on low pressurizer pressure within one minute after the break. The RCS pressure continues to drop and reaches saturation (1100 to 1300 psi depending on the system temperature) in about two to three minutes. Due to the large thermal inertia of primary and

secondary system, there has been no significant decrease in the RCS temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2% Δk/k due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than 10 minutes after the break. The addition of borated safety injection flow (supplied from the RWST) starting one minute after the break is much more than sufficient to ensure that the core remains subcritical during the cooldown.

Reactor Protection

Reactor protection for a rod ejection is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip although the analysis modeled the high neutron flux trip (high and low setting) only. These protection functions are part of the reactor trip system. No single failure of the reactor trip system will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

Results

Table 3.3-9 summarizes the results. Cases are presented for both beginning and end of life at zero and full power.

A. Beginning of Cycle, Full Power

Control Bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.15% Δk/k and 6.8 respectively. The peak clad average temperature was 2299°F. The peak spot fuel center temperature reached melting, conservatively assumed at 4900°F. However, melting was restricted to less than 10% of the pellet.

B. Beginning of Cycle, Zero Power

For this condition, Control Bank D was assumed to be fully inserted and banks B and C were at their insertion limits. The worst ejected rod is located in Control Bank D and has a worth of 0.65% Δk/k and a hot channel factor of 12.0. The peak clad average temperature reached 2130°F, the fuel center temperature was 3120°F.

C. End of Cycle, Full Power

Control Bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.19%

$\Delta k/k$ and 7.1 respectively. This resulted in a peak clad average temperature of 2245°F. The peak hot spot fuel center temperature reached melting at 4800°F. However, melting was restricted to less than 10% of the pellet.

D. End of Cycle, Zero Power

The ejected rod worth and hot channel factor for this case were obtained assuming Control Bank D to be fully inserted and banks B and C at their insertion limits. The results were 0.75% $\Delta k/k$ and 19.0 respectively. The peak clad average and fuel center temperatures were 2322°F and 3258°F. The Doppler weighting factor for this case is significantly higher than for the other cases due to the very large transient hot channel factor.

For all the cases analyzed, average fuel pellet enthalpy at the hot spot remains below 200 cal/g.

The nuclear power and hot spot fuel and clad temperature transients for two cases (end of life zero power and end of life full power) are presented in Figures 3.3-54 through 3.3-57.

The ejection of an RCCA constitutes a break in the RCS, located in the reactor pressure vessel head. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other LOCA to recover from the event.

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits (Reference 12). Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for Cook Nuclear Plant Unit 1 will not result in an excessive pressure rise or further adverse effects to the RCS.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are under-moderated, and bowing will tend to increase the under-moderation at the hot spot. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat

from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

Conclusions

Even on a pessimistic basis, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no likelihood of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no likelihood of further consequence to the RCS. The analyses have demonstrated the fission product release as a result of fuel rods entering DNB is limited to less than 10% of the fuel rods in the core.

3.3.5.8 Steamline Break Mass/Energy Releases Inside Containment

The non-LOCA discussion regarding the reanalysis of the steamline break mass and energy releases inside containment can be found in Sections 3.5.4 and 3.5.5. It should be noted that the changes associated with the SGTP Program for Unit 1, i.e., RCS flow reduction, reduced primary-to-secondary heat transfer capability, and reduction in the rated thermal power, are less limiting parameters relative to the assumptions currently made for the M/E release calculations following a SLB inside containment. The parameter changes associated with the SGTP program do not warrant reanalysis of this event. However, evaluations are currently in place (References 7 and 17) to address several non-conservative assumptions in the analysis. A reanalysis effort was undertaken for the steamline break mass and energy releases inside containment as part of the SGTP Program, such that the Reference 7 and 17 evaluations will no longer be required.

3.3.6 Conclusions of the Non-LOCA Safety Evaluation

The non-LOCA safety analyses and evaluations presented in this section support the operation of Donald C. Cook Nuclear Plant Unit 1 with SGTP, as described in Table 3.3-1 (Cases 1 and 2).

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TABLE 3.3-1
DONALD C. COOK NUCLEAR PLANT UNIT 1 NSSS PERFORMANCE PARAMETERS
USED IN NON-LOCA SAFETY ANALYSES

Parameter	Case 1	Case 2	Case 3	Case 4
	(30% SGTP Program)		(Rerating Program) [1]	
NSSS Power, MWt	3262	3262	3425	3425
Core Power, MWt	3250	3250	3413	3413
RCS Flow, gpm/loop [2]	83200	83200	88500	88500
Minimum Measured Flow, total gpm [3]	339,100	339,100	366,400 [4]	366,400 [4]
RCS Temperature, °F				
Core Outlet	589.7	611.9	583.6	614.0
Vessel Outlet	586.8	609.1	580.7	611.2
Core Average	555.8	579.4	549.7	581.8
Vessel Average	553.0	576.3	547.0	578.7
Vessel/Core Inlet	519.2	543.5	513.3	546.2
Steam Generator Outlet	518.9	543.2	513.1	546.0
Zero Load	547.0	547.0	547.0	547.0
RCS Pressure, psia	2250 or 2100	2250 or 2100	2250 or 2100	2250 or 2100
Steam Pressure, psia	595	749	603	820
Steam Flow (10^6 lb/hr total)	14.12	14.17	14.98	15.07
Feedwater Temp., °F	434.8	434.8	442.	442.
SG Tube Plugging, %	30	30	10	10

(1) Cook Unit 1 is not licensed to operate at the rerated conditions specified by Cases 3 and 4 with 30% steam generator tube plugging (SGTP) levels. However, several events that were previously performed using these conditions were subsequently evaluated to support the 30% SGTP program. Hence, the rerated conditions are also specified in this table for completeness.

(2) RCS Flow (Thermal Design Flow) - The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based upon this flow.

(3) Minimum Measured Flow - The flow specified in the Technical Specifications which must be confirmed or exceeded by the flow measurements obtained during plant startup and is the flow used in reactor core DNB analyses for plants applying the Revised Thermal Design Procedure. MMF based upon a 1.9% flow measurement uncertainty. Analyses also bound a MMF of 341,100 gpm which reflects a 2.5% flow measurement uncertainty.

(4) A MMF of 366,400 gpm was assumed in the Rerating Program analyses. A safety evaluation was performed to support a reduction of MMF to 361,600 gpm (SECL-90-280).

TABLE 3.3-2
TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN
NON-LOCA ACCIDENT ANALYSIS[c]

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delay (Seconds)
Power range high neutron flux, high setting	118 percent	0.5
Power range high neutron flux, low setting	35 percent	0.5
Overtemperature ΔT	Variable, see Figure 3.3-1 through 3.3-4 and Table 3.3-3	8.0[a]
Overpower ΔT	Variable, see Figure 3.3-1 through 3.3-4 and Table 3.3-3	8.0[d]
High pressurizer pressure	2420 psig	2.0
Low pressurizer pressure	1825 psig	2.0
High pressurizer water level	100% NRS	2.0
Low reactor coolant flow (From loop flow detectors)	87 percent loop flow	1.0
Undervoltage trip	b	1.5
Low-low steam generator level	0.0 percent of narrow range level span	2.0
High steam generator level Turbine Trip	82 percent of narrow range level span	2.5
Feedwater Isolation		11.0

[a] Total time delay (including RTD bypass loop fluid transport delay effect, bypass loop piping thermal capacity, RTD time response, and trip circuit, channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall. The time delay assumed in the analysis supports the 6 second response time of the RTD time response, trip circuit delays, and channel electronics delay presented in the Technical Specifications.

[b] No explicit value assumed in the analysis. Undervoltage trip setpoint assumed reached at initiation of analysis.

[c] The control rod scram time to dashpot is 2.4 seconds.

[d] Overpower ΔT reactor trip was assumed in the steamline break mass/energy release outside containment calculations.

TABLE 3.3-3
OTΔT AND OPΔT SETPOINT EQUATION AND SAFETY ANALYSIS LIMIT COEFFICIENT VALUES

Overtemperature ΔT equation:

$$OT\Delta T \leq \Delta T_o \; [K_1 - K_2 \; [\frac{1+\tau_1 s}{1+\tau_2 s}] \; (T - T') + (P - P') - f_1(\Delta I)$$

where,

K_1 = 1.35
K_2 = 0.023
τ_1 = 22 seconds
τ_2 = 4 seconds
s = Laplace transform operator
T' = 553.0 to 576.3°F
K_3 = 0.0011
P' = 2100 or 2250 psia
$f_1(\Delta I)$: Dead-band: from -37 to +3%ΔI
Positive Wing: 2.34%/%ΔI for each percent ΔI > +3%ΔI
Negative Wing: 0.33%/%ΔI for each percent ΔI < -37%ΔI

Overpower ΔT equation:

$$OP\Delta T \leq \Delta T_o \; [K_4 - K_5 \; [\frac{\tau_3 s}{1+\tau_3 s}] \; T - K_6 \; (T - T'') - f_2(\Delta I)$$

where,

K_4 = 1.172
K_5 = 0.0177; this gain is not modeled in the non-LOCA safety analyses
τ_3 = 10 seconds
s = Laplace transform operator
T" = 553.0 to 563.0°F
K_6 = 0.0015
P' = 2100 or 2250 psia
$f_2(\Delta I)$: = 0

TABLE 3.3-4
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Moderator Density (ΔK/gm/cc)	Reactivity Coefficients Assumed: Doppler	DNB Correlation	Revised Thermal Design Procedure	Reactor Initial NSSS Thermal Power Output (MWt)	Vessel Vessel Coolant Flow (GPM)	Average Temperature (°F)	Pressurizer Pressure (PSIA)
Uncontrolled Rod Cluster Assembly Bank Withdrawal from a Subcritical Condition	TWINKLE FACTRAN THINC IV	Refer to Section 3.3.5.1		Min (1)	W-3/WRB-1 See Section 3.3.4.3	No	0	146,432	547	2033
Uncontrolled Rod Cluster Assembly Bank Withdrawal At Power (2)	LOFTRAN	+5	.54	Min and Max (3)	WRB-1	Yes	3270 1962 327	339,100	576.3 564.58 549.93	2100
Rod Cluster Control Assembly Misalignment	LOFTRAN THINC IV	NA*	NA	NA	WRB-1	Yes	3270	339,100	576.3	2100
Uncontrolled Boron Dilution	NA	NA	NA	NA	NA	NA	3425 0	NA	NA	NA
Loss of Forced Reactor Coolant Flow	LOFTRAN FACTRAN THINC IV	+5	NA	Max	WRB-1	Yes	3270	339,100	576.3	2100
Locked Rotor (Peak Pressure)	LOFTRAN	+5	NA	Max	NA	NA	3335	332,800	581.4	2317
Locked Rotor (Peak Clad Temp)	LOFTRAN FACTRAN	+5	NA	Max	NA	NA	3335	332,800	581.4	2033
Locked Rotor (Rods-in-DNB)	LOFTRAN FACTRAN THINC IV	+5	NA	Max	WRB-1	Yes	3270	339,100	576.3	2100

* NA - Not Applicable
(1) Minimum Doppler power defect (pcm/%power) = -9.55 + 0.035Q where Q is in % power.
(2) Multiple power levels, Tavg, and reactivity feedback cases were examined.
(3) Maximum Doppler power defect (pcm/% power) = -19.4 + 0.065Q.
(4) Minimum and Maximum reactivity feedback cases were examined.

TABLE 3.3-4 (continued)
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Moderator Density (ΔK/gm/cc)	Reactivity Coefficients Assumed: Doppler	DNB Correlation	Revised Thermal Design Procedure	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
Loss of Electrical Load and/or Turbine Trip (4)	LOFTRAN	+5	.54	Max and Min	WRB-1	Yes	3262	339,100	576.3	2100
Loss of Normal Feedwater (5)	LOFTRAN	+5	NA	Max	NA	NA	3494	354,000	551.5	2285
Excessive Heat Removal Due to Feedwater System Malfunction (5)	LOFTRAN	NA	.54	Min	WRB-1	Yes	3425 0	366,400	578.7 547	2100
Excess Load Increase Incident (5)	LOFTRAN	NA	0 and .54	Max and Min	WRB-1	Yes	3425	366,400	578.7	2100
Loss of Offsite Power to the Station Auxiliaries (5)	LOFTRAN	+5	NA	Max	NA	NA	3494	354,000	542.5	2285
Rupture of a Steam Pipe	LOFTRAN THINC IV	See Figure 3.3-47	NA	See Figure 3.3-48	W-3	NA	0	332,800	547	2100
Rupture of a Control Rod Drive Mechanism Housing	TWINKLE FACTRAN	See Section 3.3.5.7	NA	Min	NA	NA	3335 0	332,800 146,432	581.4 547	2033

* NA - Not Applicable
(1) Minimum Doppler power defect (pcm/%power) = -9.55 + 0.035Q where Q is in % power.
(2) Multiple power levels, Tavg, and reactivity feedback cases were examined.
(3) Maximum Doppler power defect (pcm/% power) = -19.4 + 0.065Q.
(4) Minimum and Maximum reactivity feedback cases were examined.
(5) Values presented correspond to the respective rerating analysis. Subsequent evaluations support the 30% SGTP parameters given as Cases 1 and 2 of Table 3.3-1.

TABLE 3.3-5
SEQUENCE OF EVENTS FOR LOSS OF
FLOW AND LOCKED ROTOR ACCIDENTS

Accident	Event	Time (sec.)
Complete Loss of Flow	All pumps lose power and begin coasting down, undervoltage trip signal generated	0.0
	Rods begin to drop	1.50
	Minimum DNBR occurs	3.40
Partial Loss of Flow	One operating pump loses power and begins coasting down	0.0
	Low reactor coolant flow trip setpoint reached in faulted loop	1.74
	Rods begin to drop	2.74
	Minimum DNBR occurs	3.90
Locked Rotor	One pump rotor seizes	0.0
	Low reactor coolant flow trip setpoint reached in faulted loop	0.04
	Rods begin to drop	1.04
	Maximum percentage of rods in DNB predicted	2.6
	Maximum RCS pressure occurs	3.20
	Maximum clad temperature occurs	3.49

TABLE 3.3-6
SEQUENCE OF EVENTS FOR LOSS OF
EXTERNAL ELECTRICAL LOAD

Case	Event	Time (sec.)
Minimum Feedback with Pressure Control	Loss of external electrical load	0.0
	OTΔT trip setpoint reached	14.2
	Peak RCS pressure occurs	15.5
	Rods begin to drop	16.2
	Minimum DNBR occurs	18.0
Maximum Feedback with Pressure Control	Loss of external electrical load	0.0
	Minimum DNBR occurs	0.0
	Peak RCS pressure occurs	10.0
	Low-low steam generator level trip setpoint reached	68.1
	Rods begin to drop	70.1
Minimum Feedback without Pressure Control	Loss of external electrical load	0.0
	Minimum DNBR occurs	0.0
	High pressurizer pressure trip setpoint reached	8.4
	Rods begin to drop	10.4
	Peak RCS pressure occurs	12.0
Maximum Feedback without Pressure Control	Loss of external electrical load	0.0
	Minimum DNBR occurs	0.0
	High pressurizer pressure trip setpoint reached	8.9
	Rods begin to drop	10.9
	Peak RCS pressure occurs	12.5

TABLE 3.3-7
LIMITING STEAMLINE BREAK STATEPOINT
DOUBLE ENDED RUPTURE INSIDE CONTAINMENT
WITH OFFSITE POWER AVAILABLE

Time sec	Pressure psia	Heat Flux Fraction	Inlet Cold °F	Temp. Hot °F	Flow Fraction	Boron PPM	Reactivity Percent	Density gm/cc
180.2	601.93	.228	336.6	463.3	1.0	7.13	.001	.849

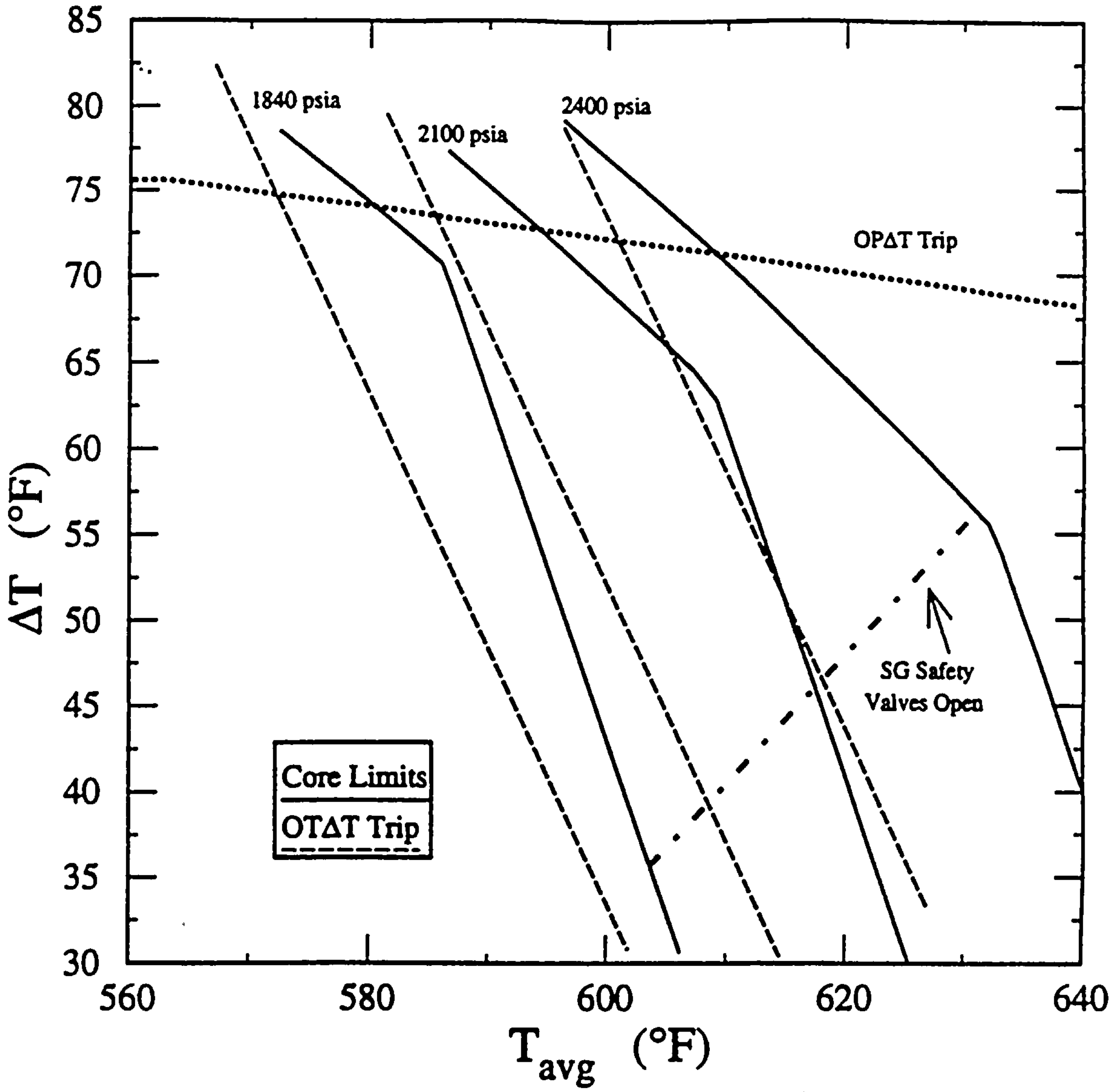
TABLE 3.3-8
TIME SEQUENCE OF EVENTS

DOUBLE ENDED RUPTURE INSIDE CONTAINMENT
WITH OFFSITE POWER AVAILABLE

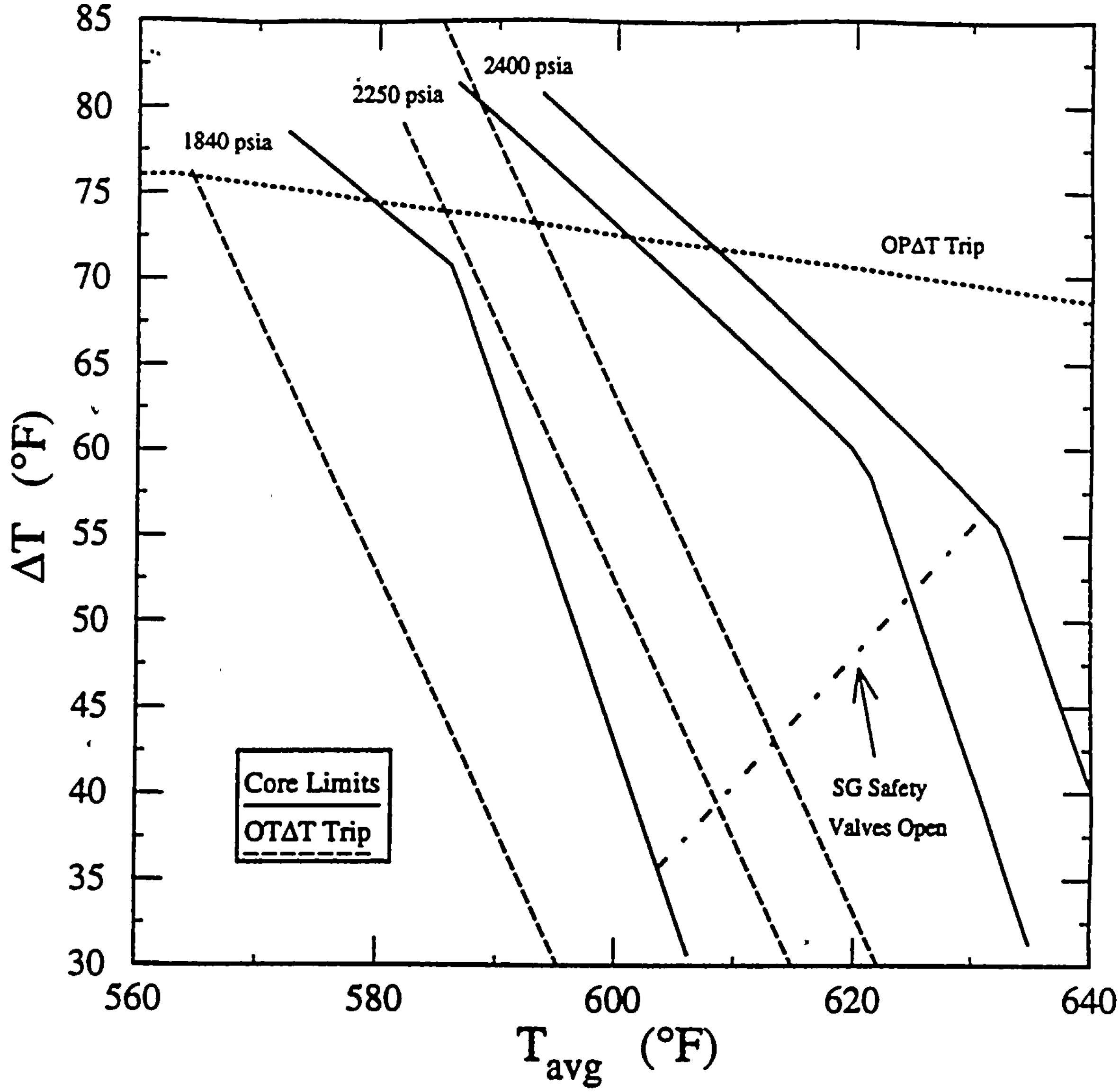
Event	Time (sec)
Steam line rupture occurs	0.00
Low steam line pressure coincident with high steam flow in two steam lines reached	2.06
Feedwater Isolation (All loops)	10.06
Criticality attained	12.40
Steamline Isolation (Loops 2, 3 and 4)	13.06
Pressurizer empties	13.20
SI flow starts	29.06
Boron from SI reaches the core	39.80
Peak heat flux attained	179.2
Core becomes subcritical	180.0

TABLE 3.3-9
PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Time in Life	HZP Beginning	HFP Beginning	HZP End	HFP End
Power Level (%)	0	102	0	102
Ejected Rod Worth (%Δk)	0.65	0.15	0.75	0.19
Delayed Neutron Fraction	0.0050	0.0050	0.0040	0.0040
Feedback Reactivity Weighting	2.071	1.30	2.755	1.30
Trip Reactivity (%Δk)	2.	4.	2.	4.
F_q Before Rod Ejection	2.50	2.50	2.50	2.50
F_q After Rod Ejection	12.	6.8	19.	7.1
Number of Operational Pumps	2.	4.	2.	4.
Maximum Fuel Pellet Average Temperature (°F)	2764	4056	2963	3969
Maximum Fuel Center Temperature (°F)	3120	4968	3258	4872
Maximum Clad Average Temperature (°F)	2130	2299	2322	2245
Maximum Fuel Stored Energy (cal/gm)	112.7	177.3	122.2	172.7
Fuel Melt in Hot Pellet, %	0	<10	0	<10



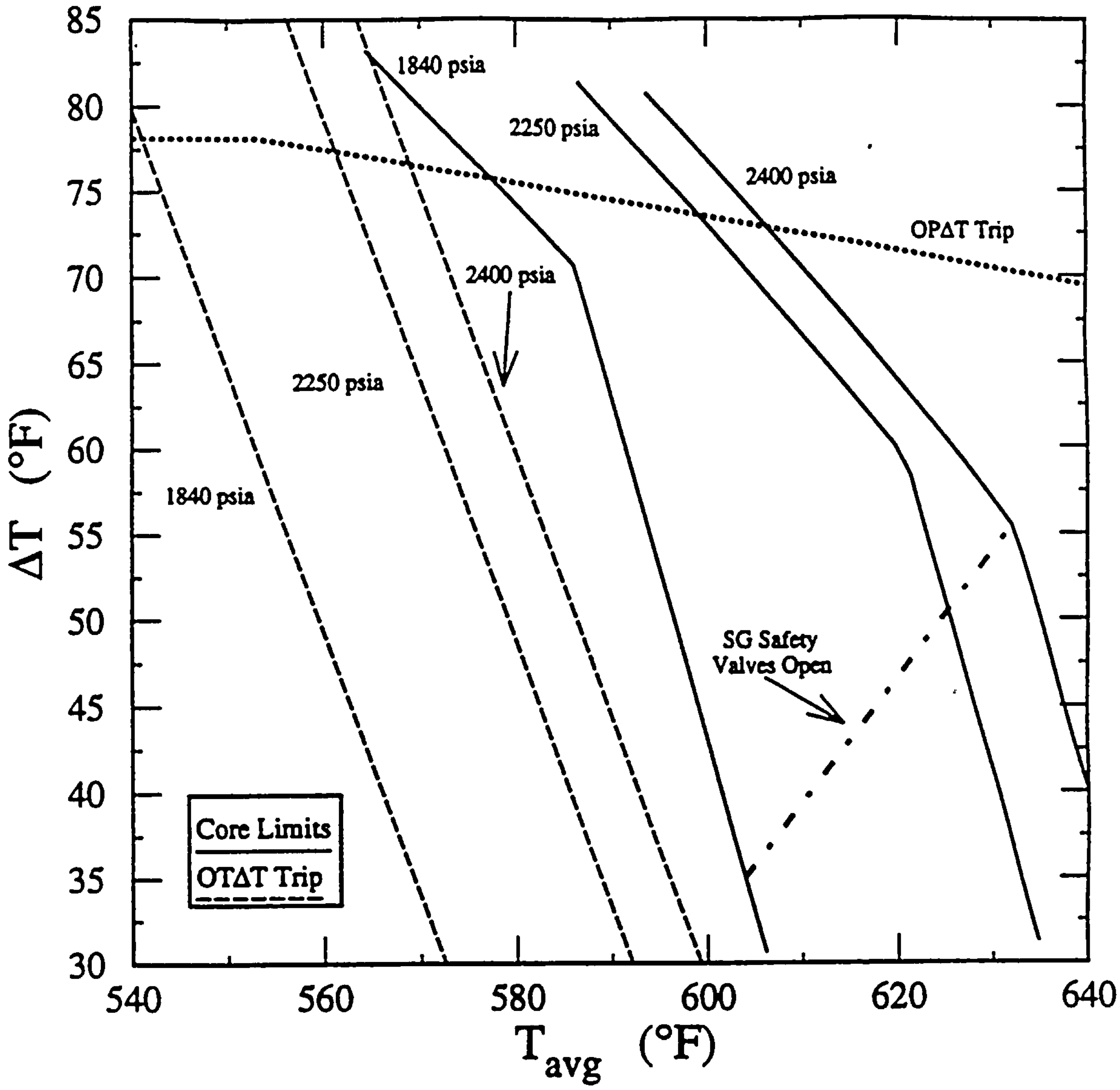
DONALD C. COOK **NUCLEAR PLANT** **UNIT 1**
FIGURE 3.3-1 Illustration of Overtemperature and Overpower ΔT Protection Nominal Tavg = 576.3°F Nominal Pressure = 2100 psia



DONALD C. COOK
NUCLEAR PLANT
UNIT 1

FIGURE 3.3-2

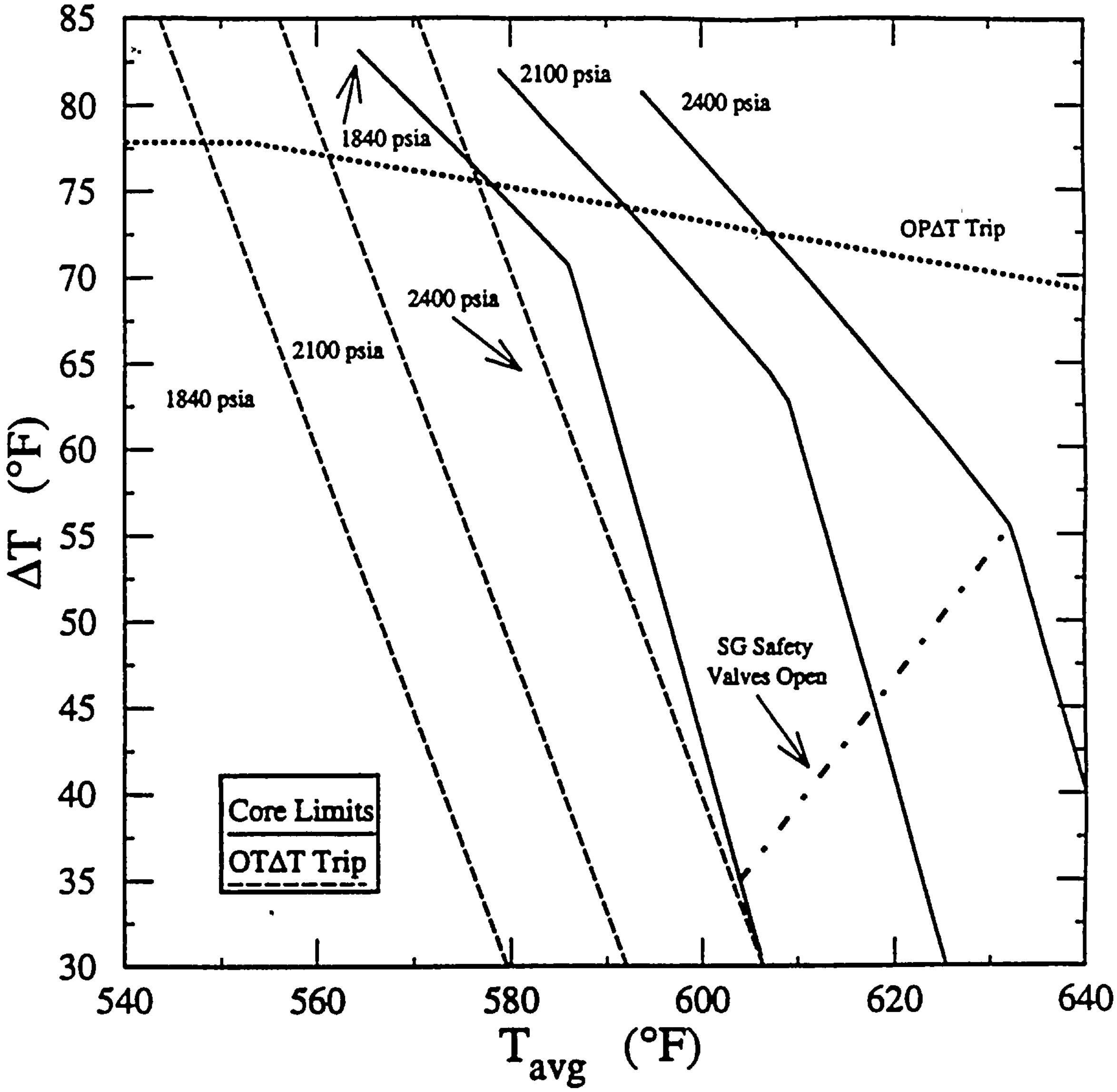
Illustration of Overtemperature and Overpower ΔT Protection
Nominal Tavg = 576.3°F
Nominal Pressure = 2250 psia



DONALD C. COOK
NUCLEAR PLANT
UNIT 1

FIGURE 3.3-3

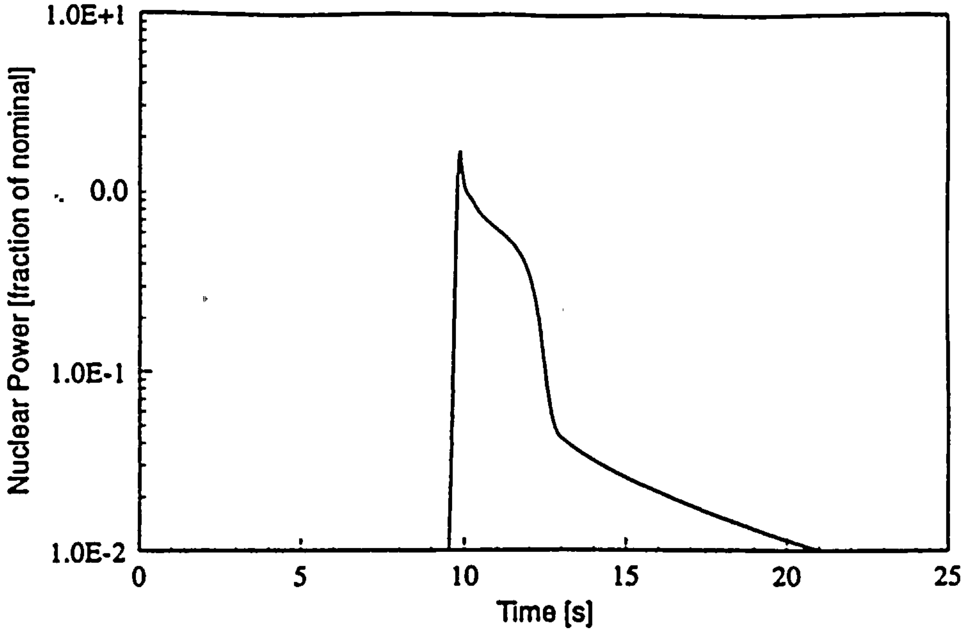
Illustration of Overtemperature and Overpower ΔT Protection
Nominal Tavg = 553.0°F
Nominal Pressure = 2250 psia

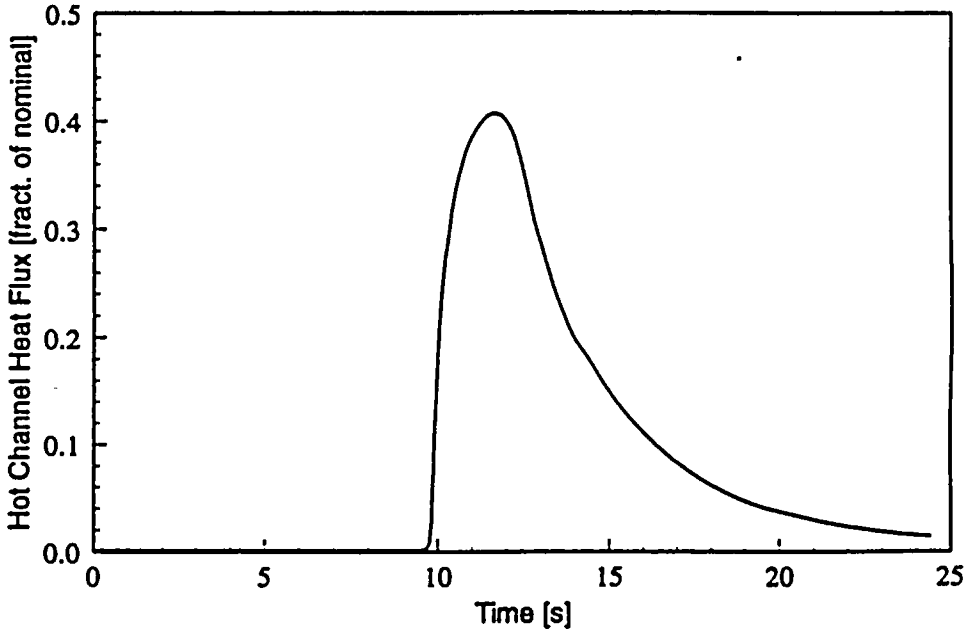


DONALD C. COOK NUCLEAR PLANT UNIT 1

FIGURE 3.3-4

Illustration of Overtemperature and Overpower ΔT Protection
Nominal Tavg = 553.0°F
Nominal Pressure = 2100 psia

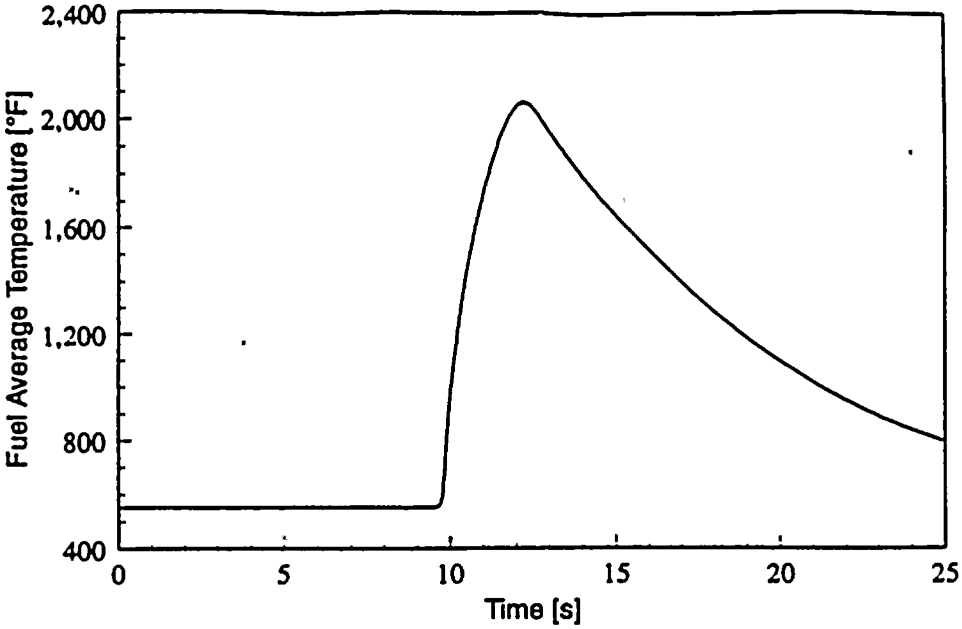




DONALD C. COOK
NUCLEAR PLANT
UNIT 1

FIGURE 3.3-5

Nuclear Power and Hot Channel Heat Flux vs. Time For The Rod Withdrawal From Subcritical Event



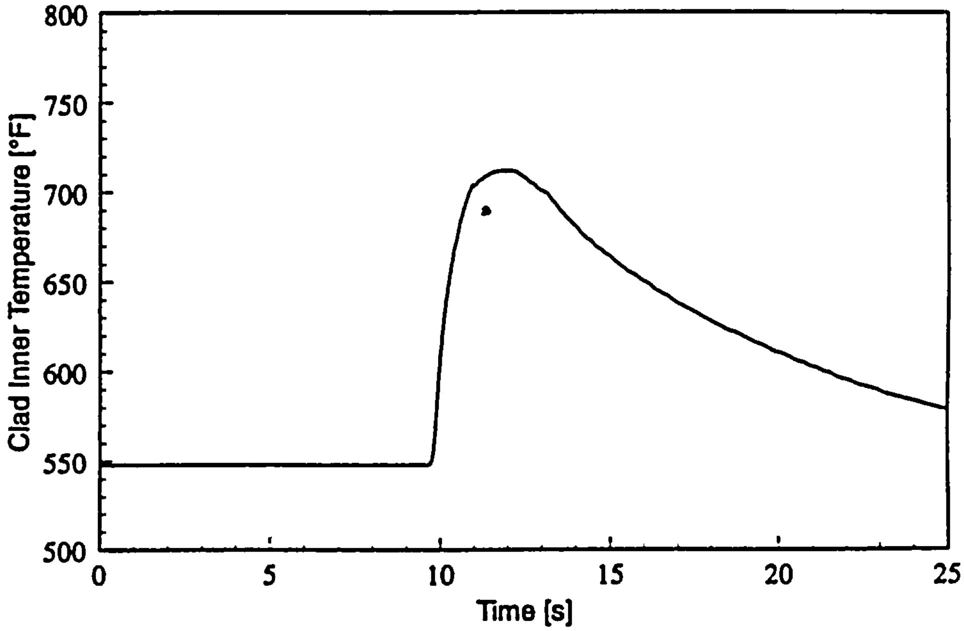


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FIGURE 3.3-6

Fuel Average and Clad Temperature vs. Time For The Rod Withdrawal From Subcritical Event

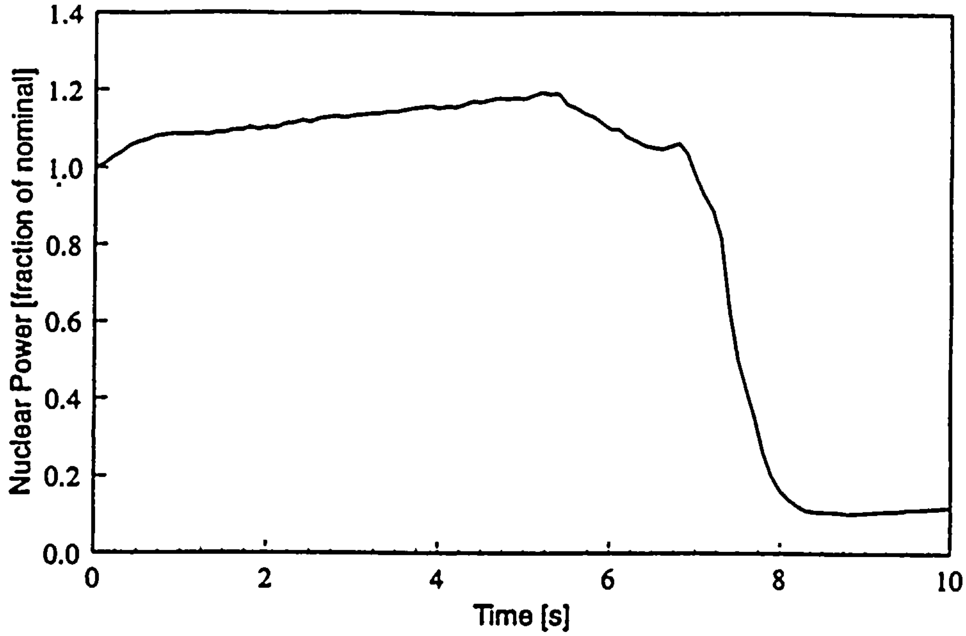

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FIGURE 3.3-7

Nuclear Power vs.Time For The RCCA Withdrawal
At Power Event, Full Power, 80 PCM/Sec. Insertion Rate
Maximum Reactivity Feedback

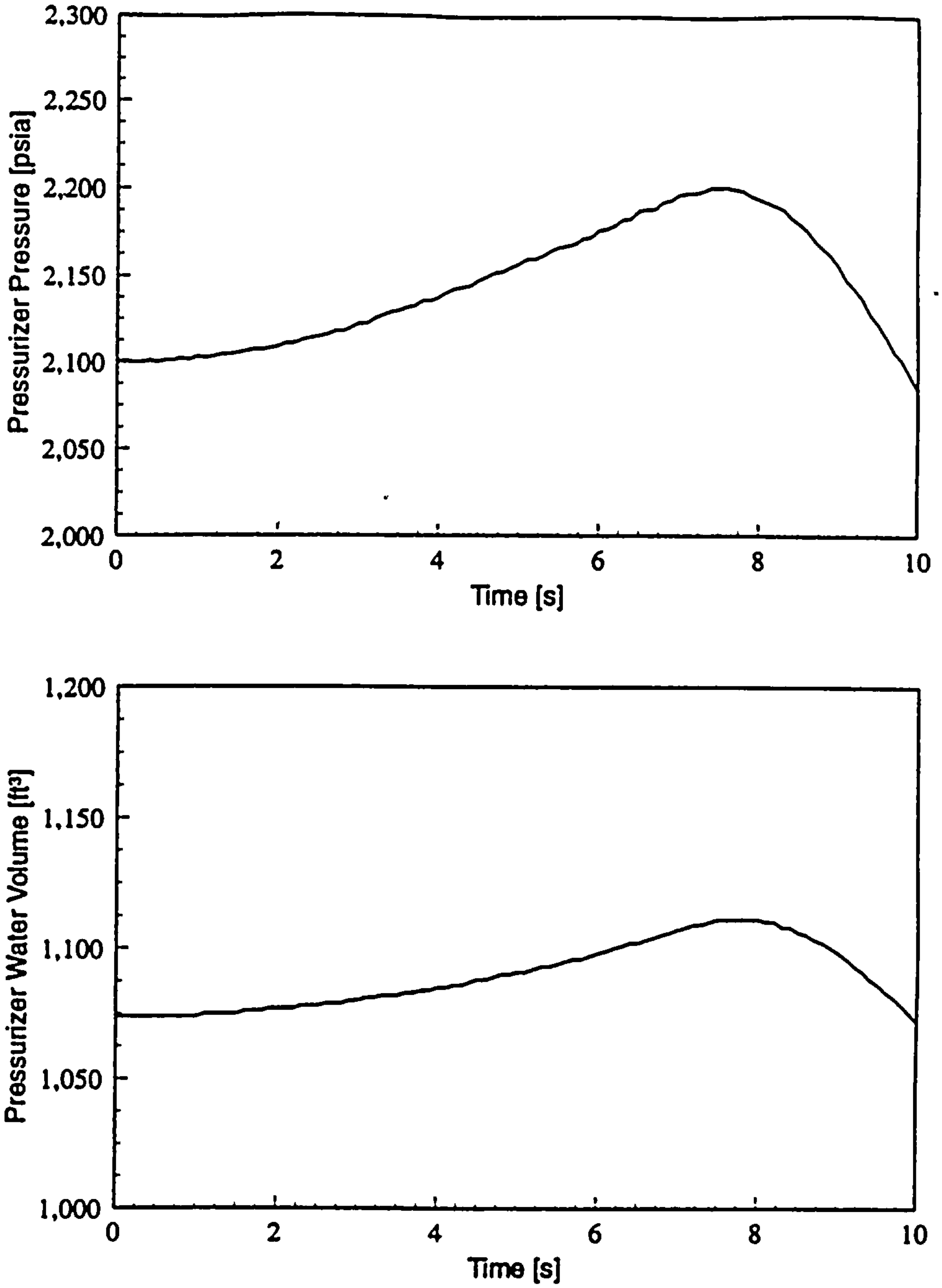

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FIGURE 3.3-8
Pressurizer Pressure and Pressurizer Water Volume vs.Time
For The RCCA Withdrawal At Power Event,
Full Power, 80 PCM/Sec. Insertion Rate
Maximum Reactivity Feedback

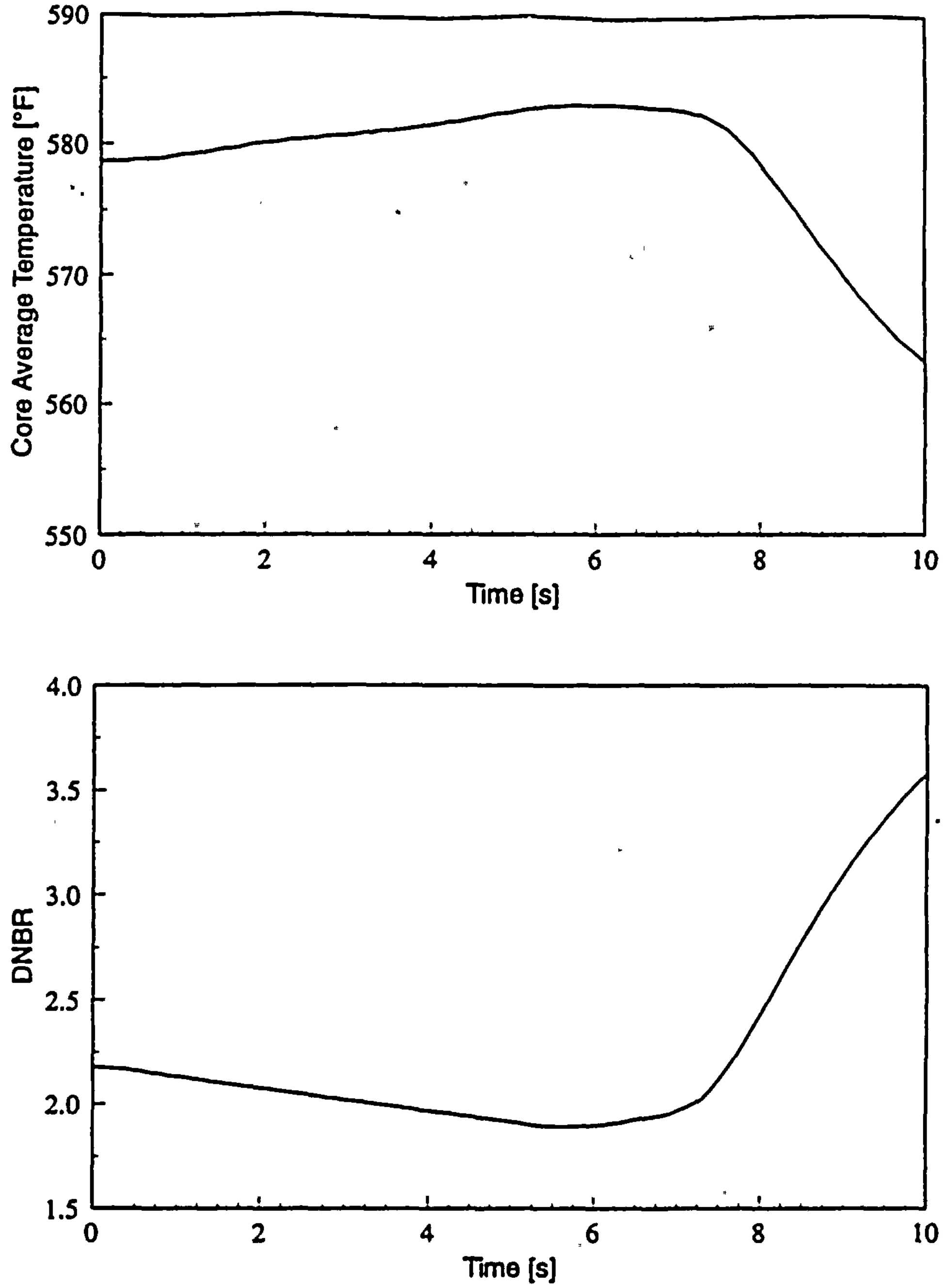

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FIGURE 3.3-9
Core Average Temperature and DNBR vs.Time
For The RCCA Withdrawal At Power Event,
Full Power, 80 PCM/Sec. Insertion Rate
Maximum Reactivity Feedback

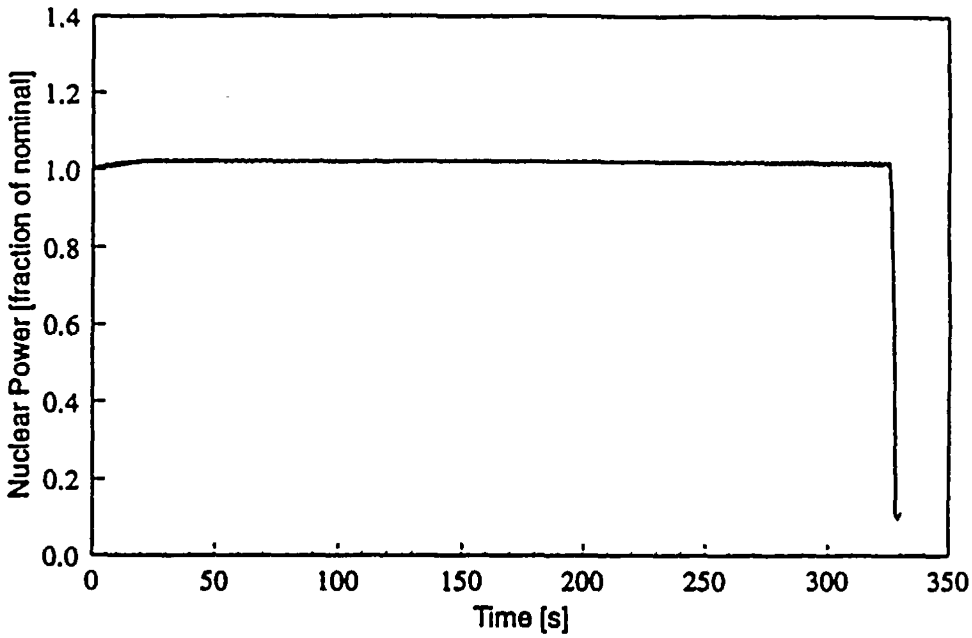

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FIGURE 3.3-10

Nuclear Power vs.Time For The RCCA Withdrawal
At Power Event, Full Power, 4 PCM/Sec. Insertion Rate
Maximum Reactivity Feedback

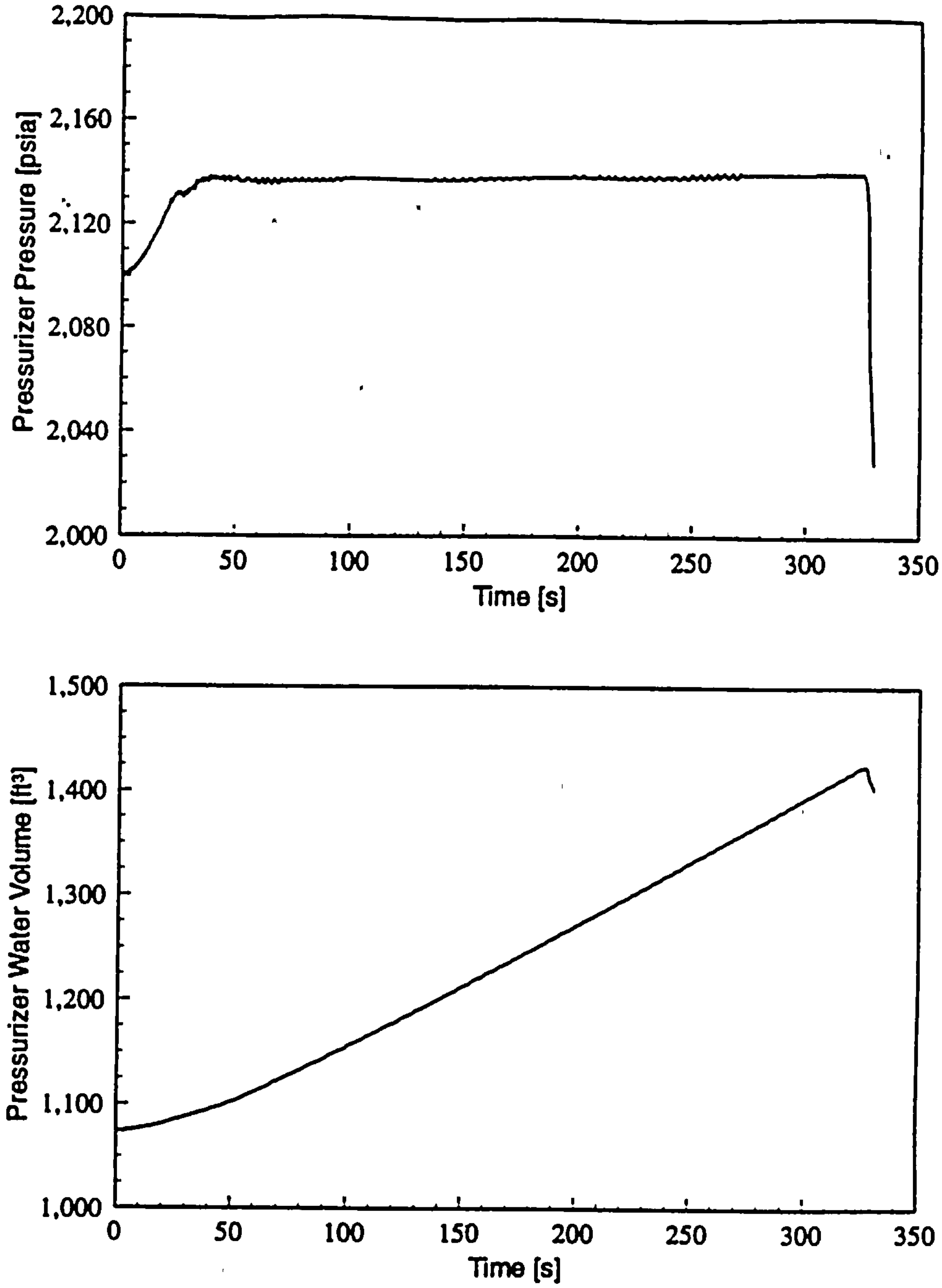

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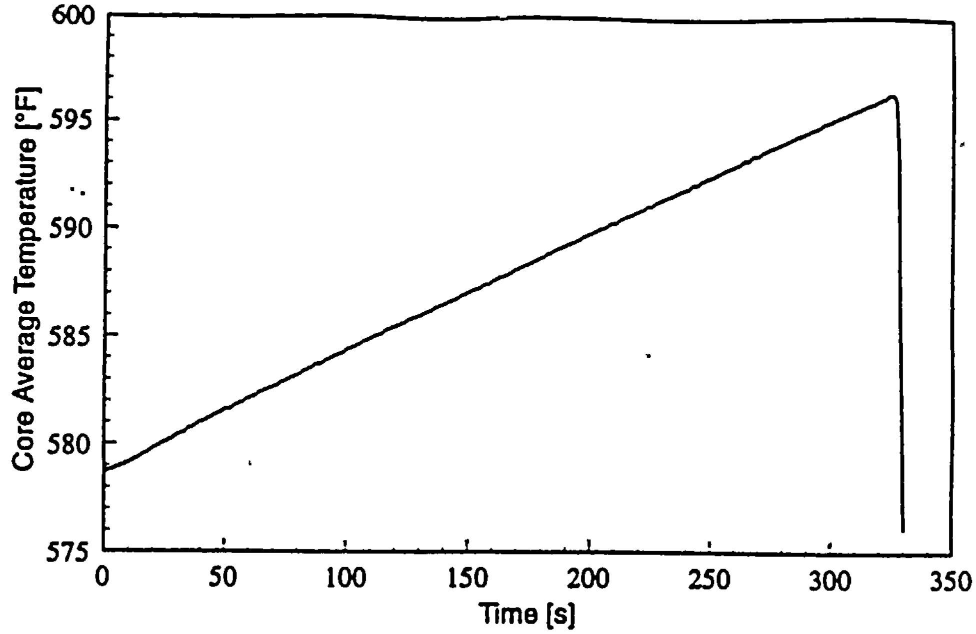
FIGURE 3.3-11
Pressurizer Pressure and Pressurizer Water Volume vs.Time For The RCCA Withdrawal At Power Event, Full Power, 4 PCM/Sec. Insertion Rate Maximum Reactivity Feedback



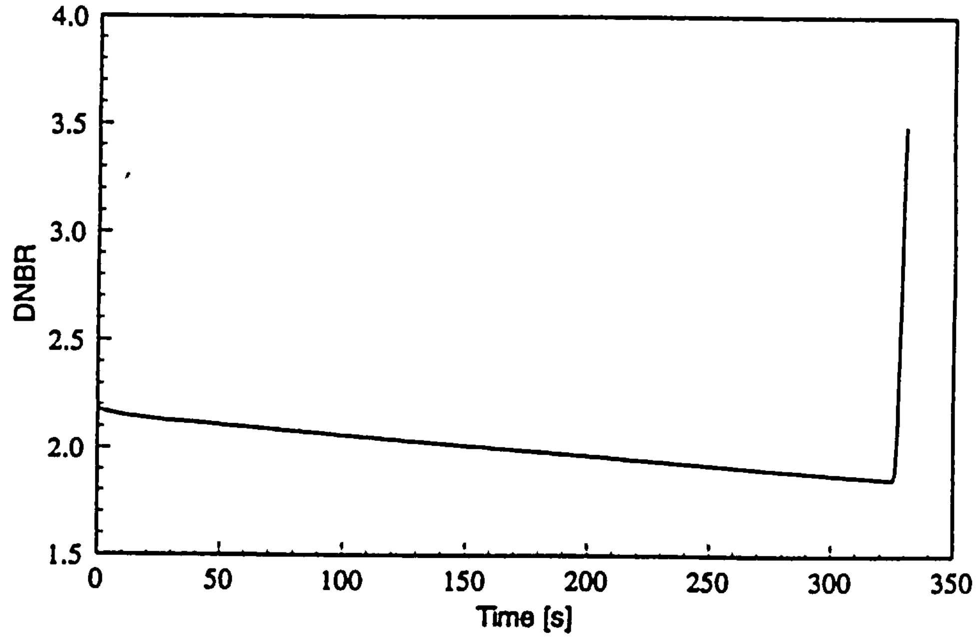


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FIGURE 3.3-12
Core Average Temperature and DNBR vs.Time
For The RCCA Withdrawal At Power Event,
Full Power, 4 PCM/Sec. Insertion Rate
Maximum Reactivity Feedback

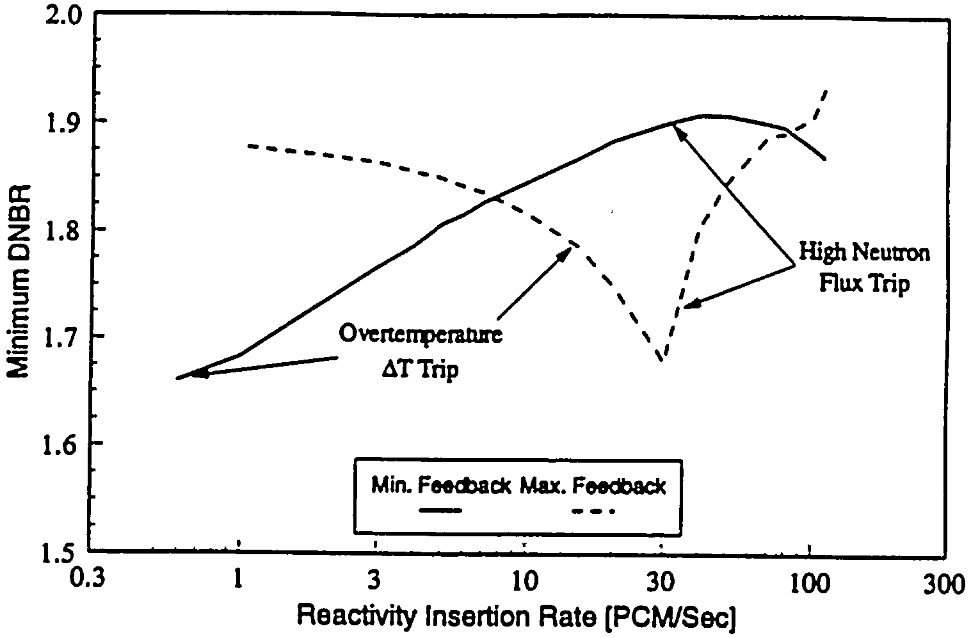

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FIGURE 3.3-13

Minimum DNBR vs. Reactivity Insertion Rate
For The RCCA Withdrawal At Power Event, 100% Power

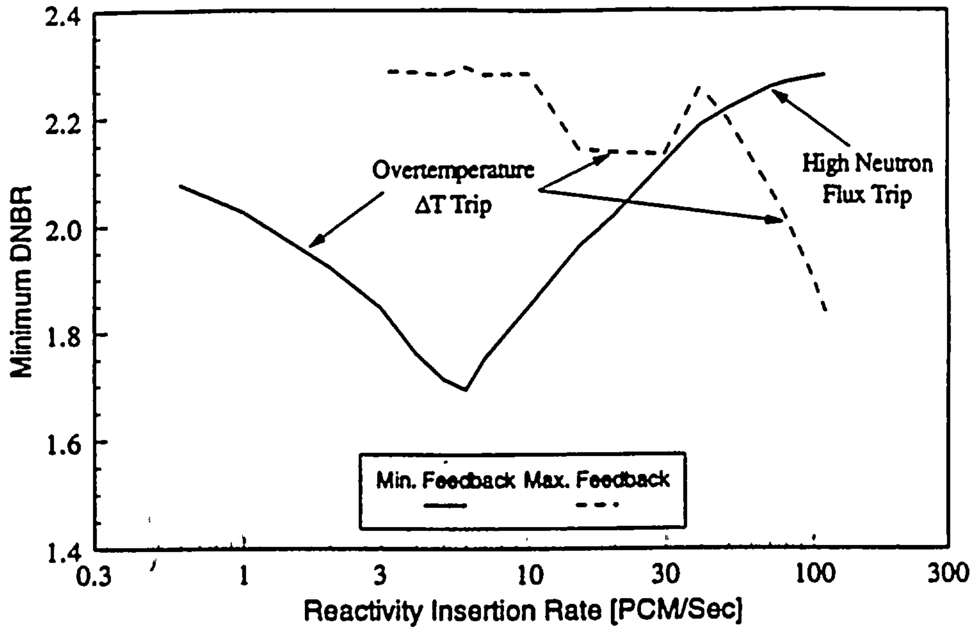

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FIGURE 3.3-14

Minimum DNBR vs. Reactivity Insertion Rate
For The RCCA Withdrawal At Power Event, 60% Power

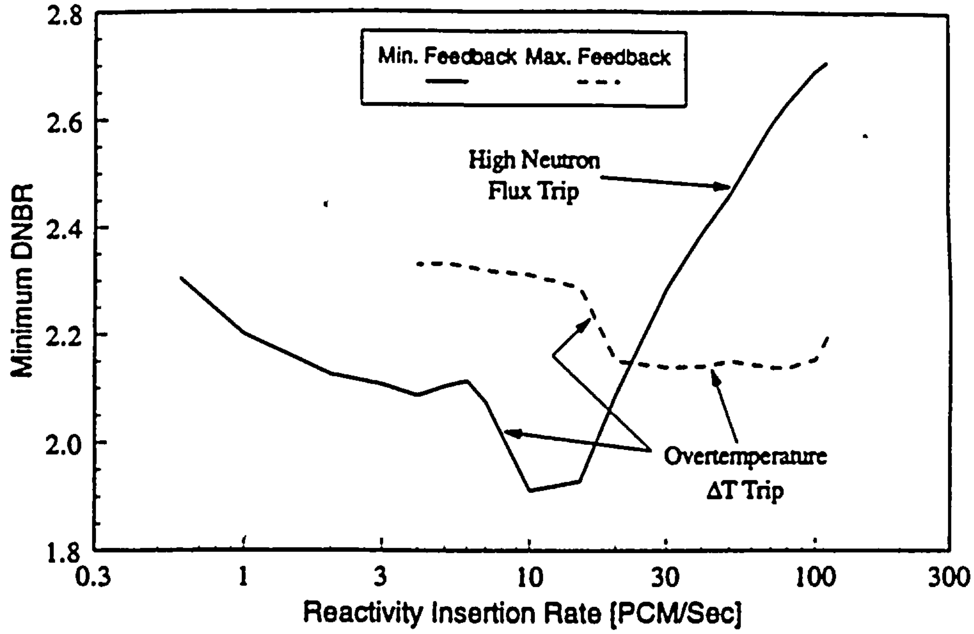

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FIGURE 3.3-15

Minimum DNBR vs. Reactivity Insertion Rate
For The RCCA Withdrawal At Power Event, 10% Power

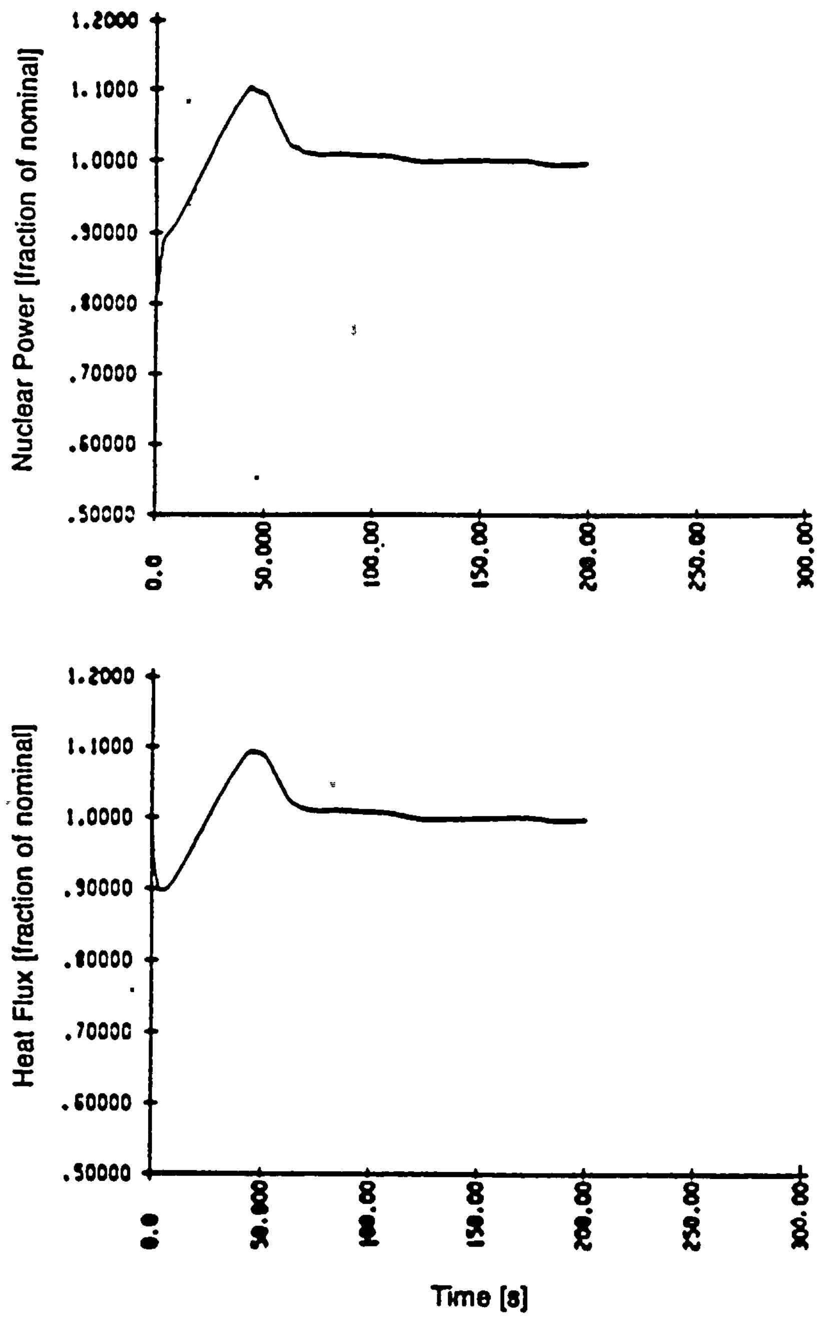

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FIGURE 3.3-16

Nuclear Power and Core Heat Flux vs. Time for a Typical Response to a Dropped RCCA(s) in Automatic Control

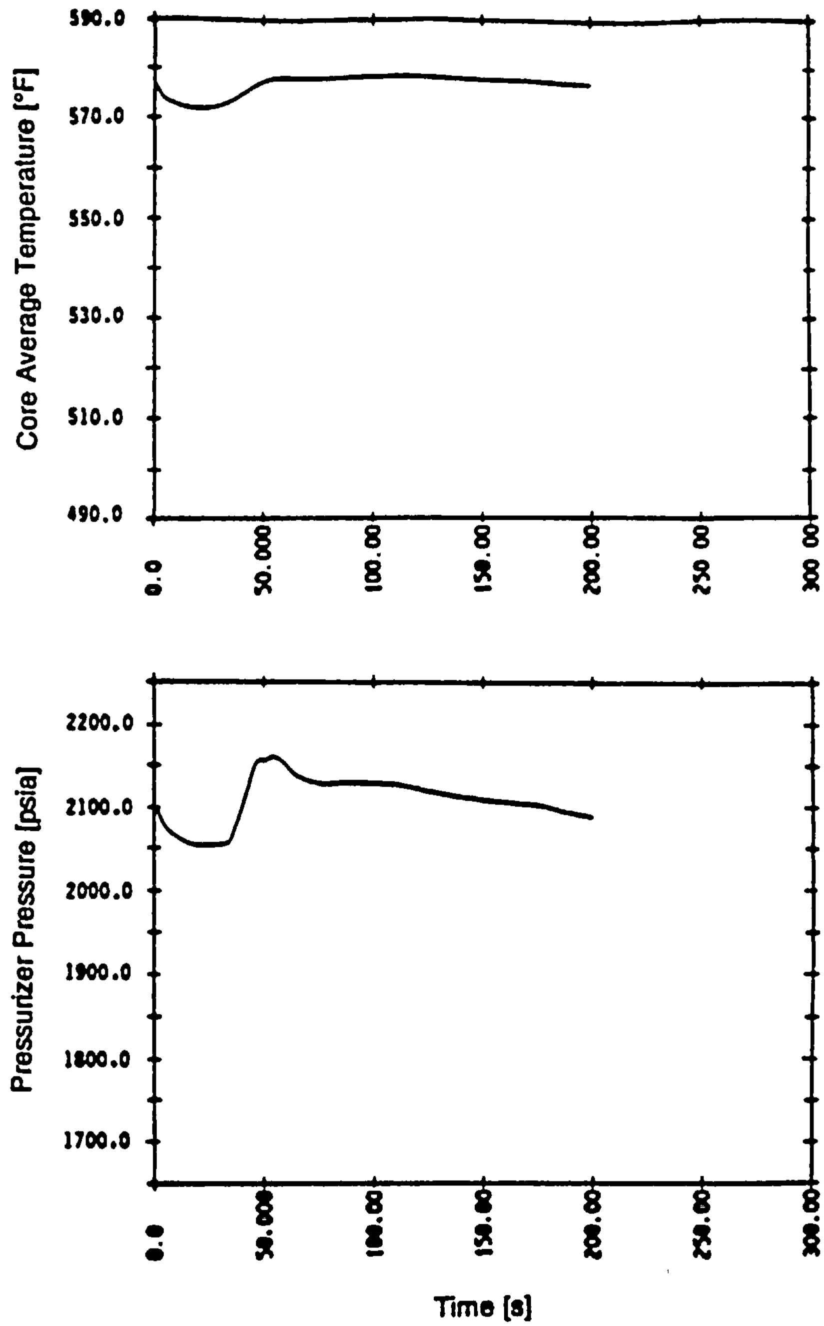

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FIGURE 3.3-17

Average Coolant Tempertaure and Pressurizer Pressure vs. Time for a Typical Response to a Dropped RCCA(s) in Automatic Control

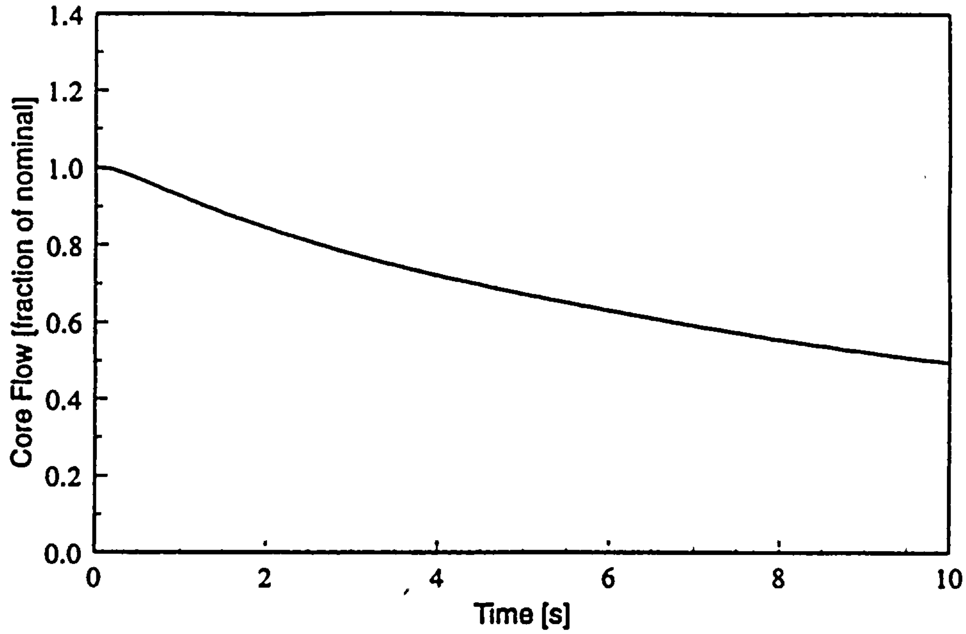

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FIGURE 3.3-18

Total Core Flow vs. Time for
The Complete Loss Of Flow Event

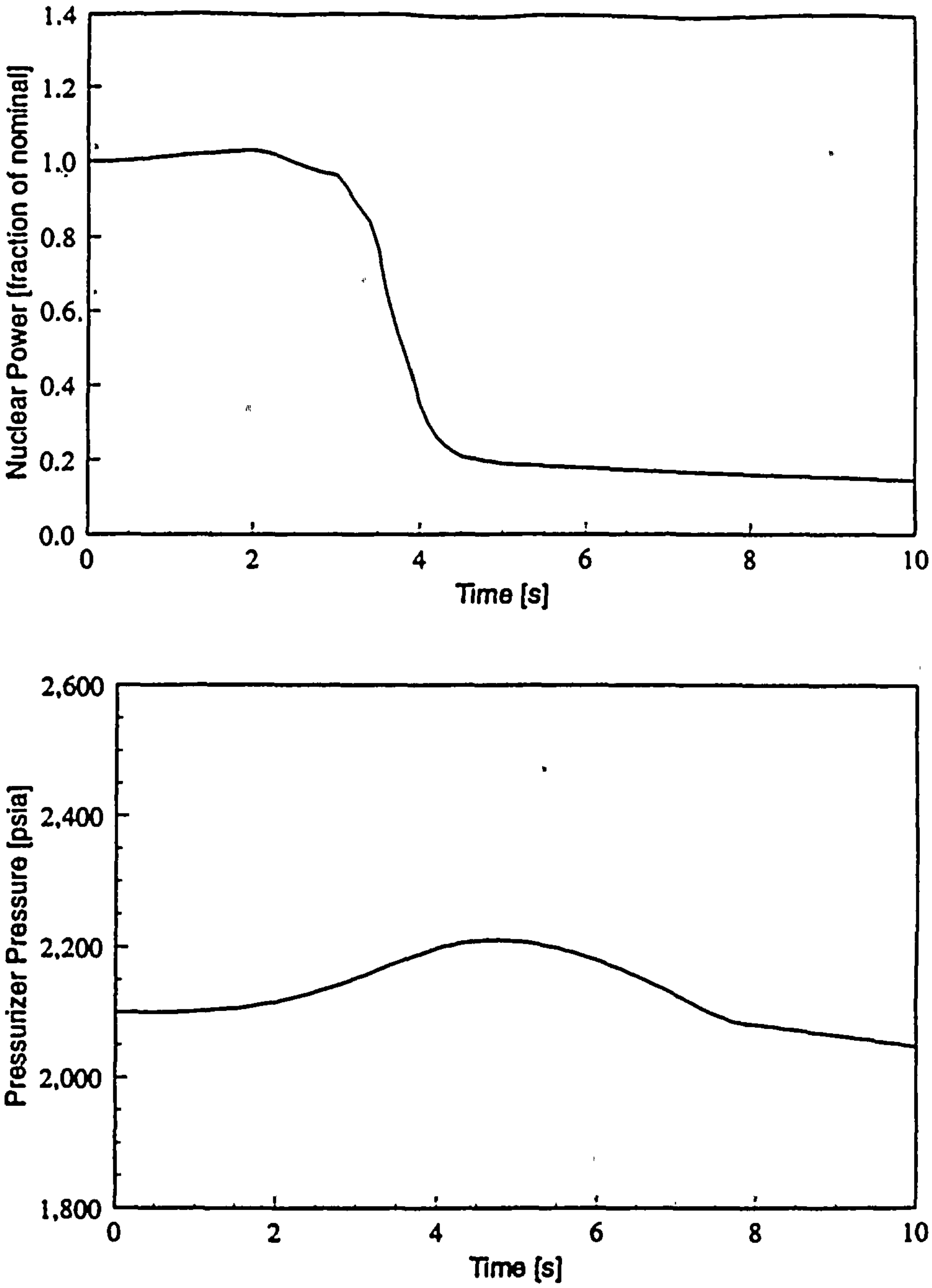

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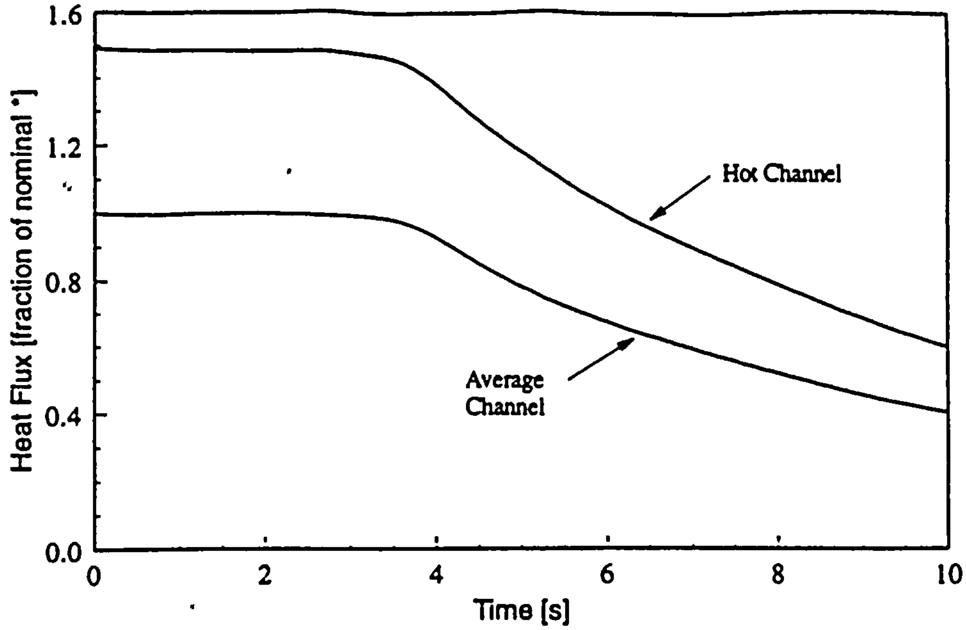
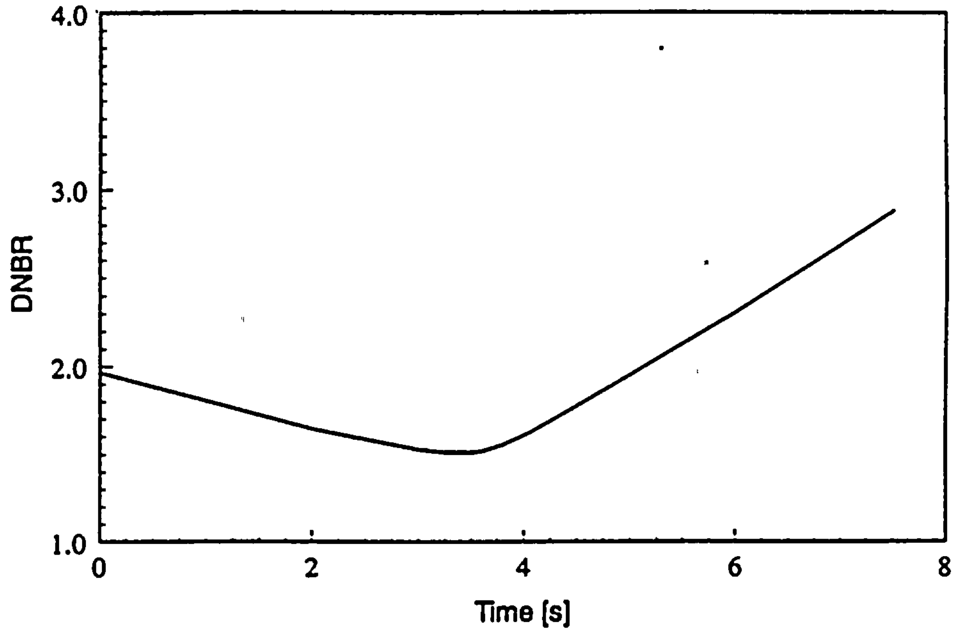
FIGURE 3.3-19

Nuclear Power and Pressurizer Pressure vs. Time for The Complete Loss Of Flow Event





* Heat fluxes are shown as a fraction of the nominal average channel heat flux

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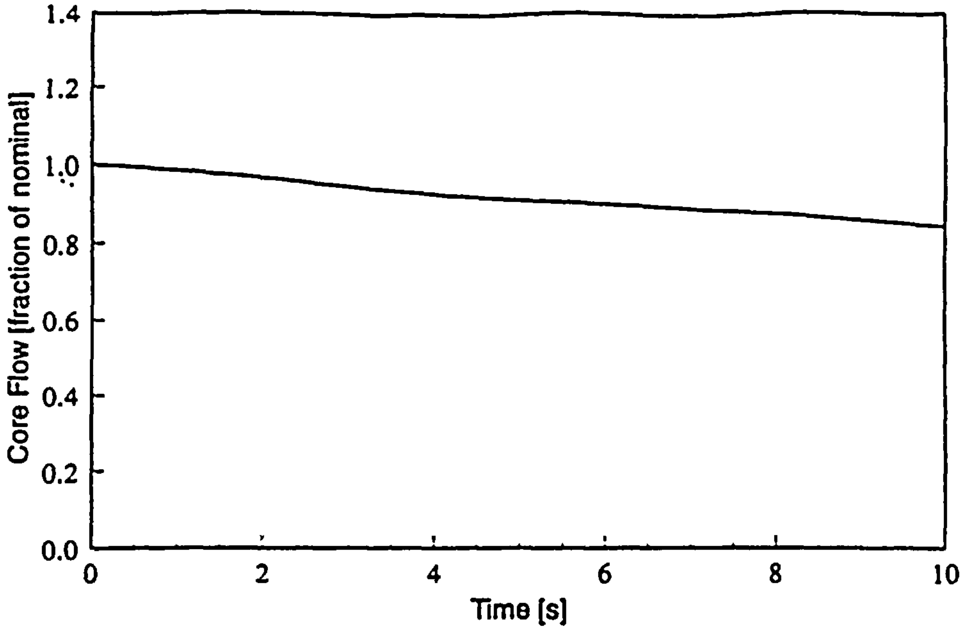
FIGURE 3.3-20

Average and Hot Channel Heat Fluxes and DNBR vs. Time for the Complete Loss Of Flow Event



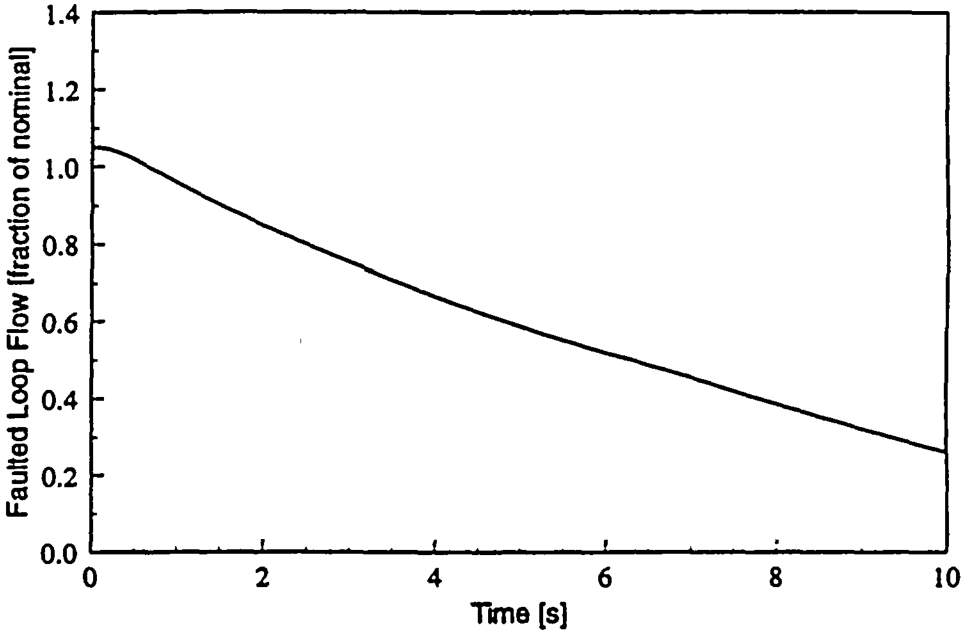


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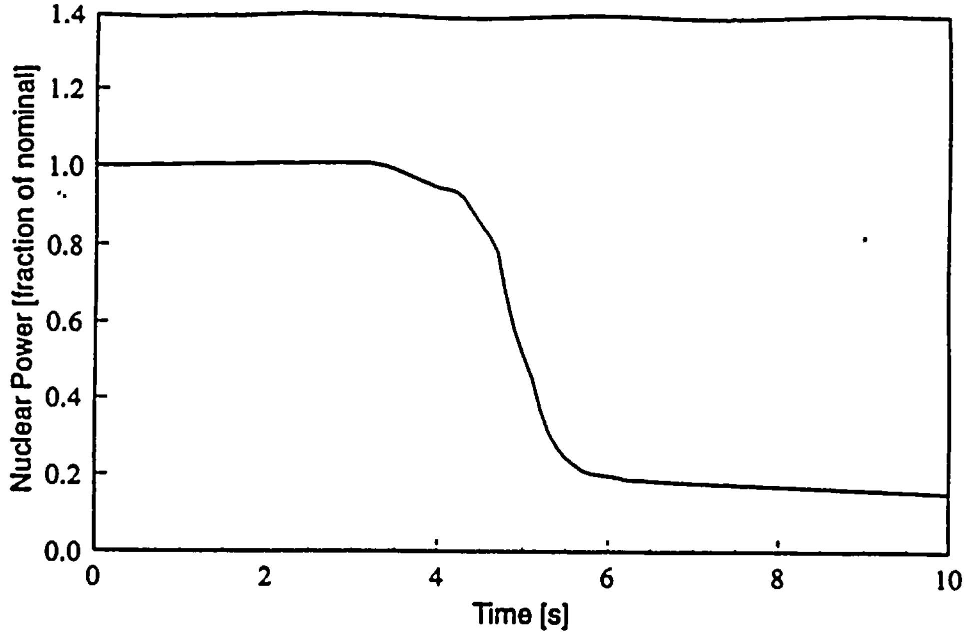
FIGURE 3.3-21

Total Core Flow and Faulted Loop Flow vs. Time for The Partial Loss Of Flow Event



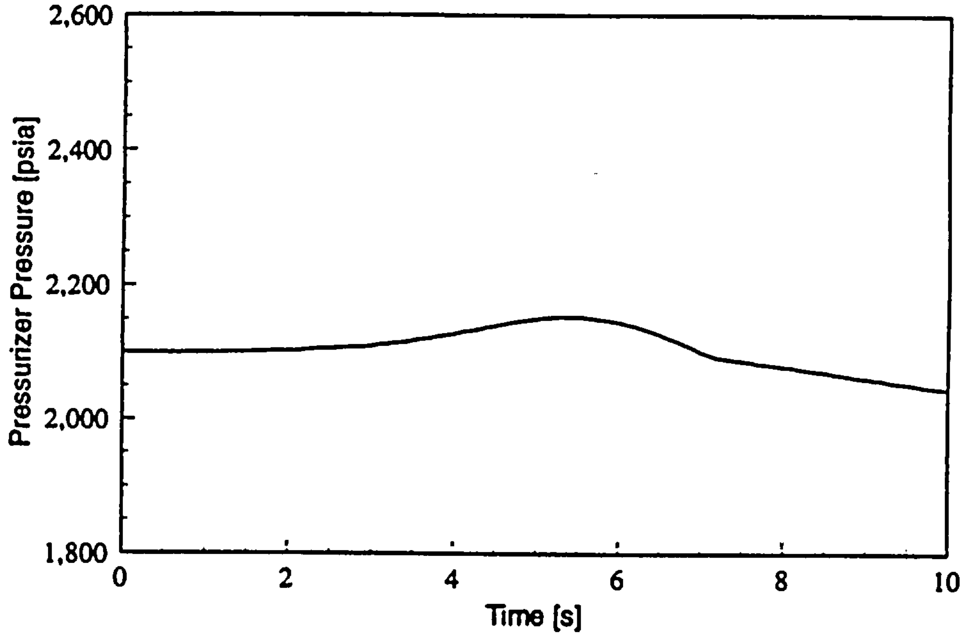


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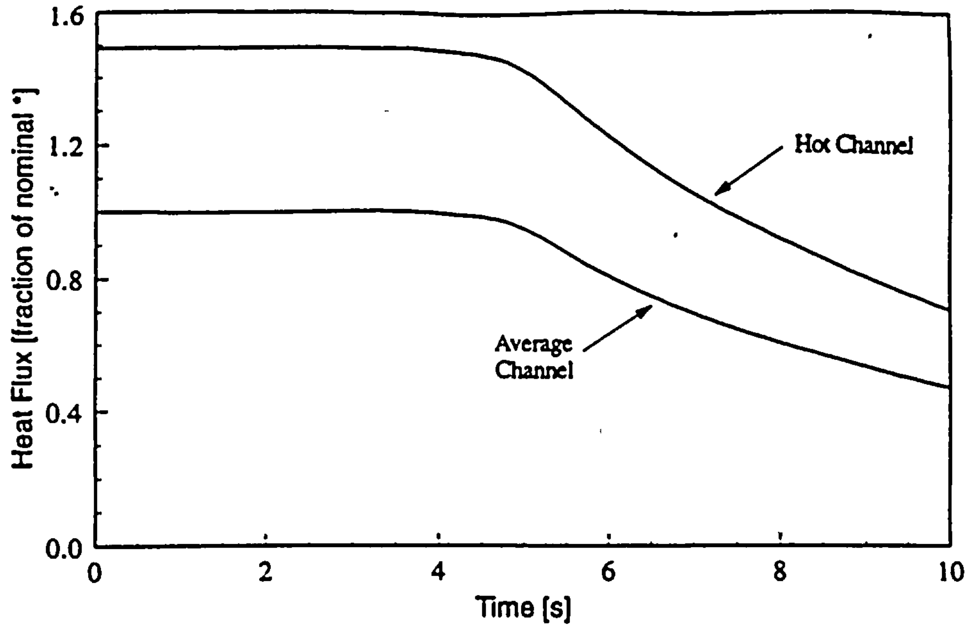
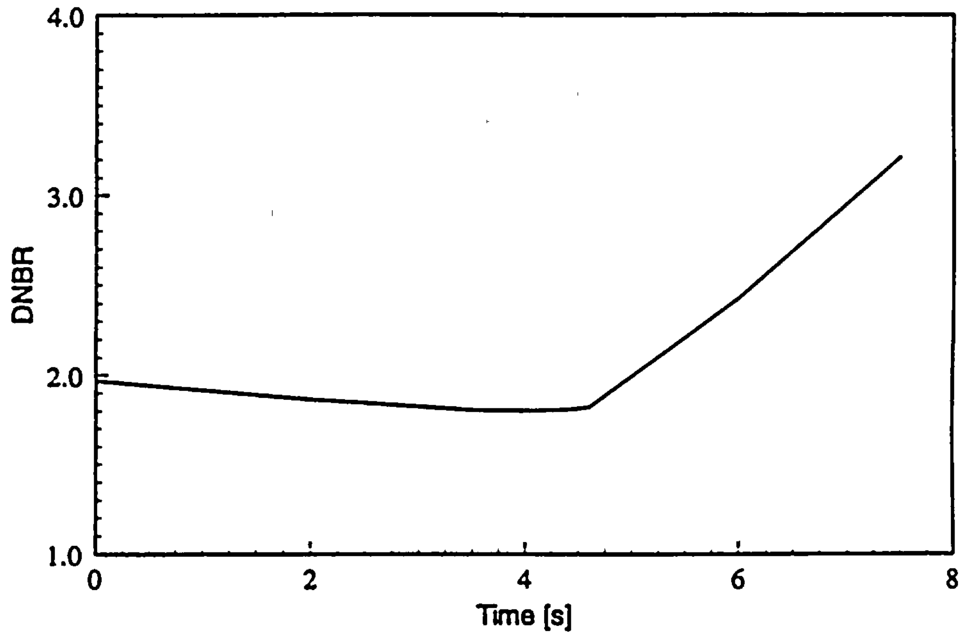
FIGURE 3.3-22 Nuclear Power and Pressurizer Pressure vs. Time for The Partial Loss Of Flow Event





* Heat fluxes are shown as a fraction of the nominal average channel heat flux

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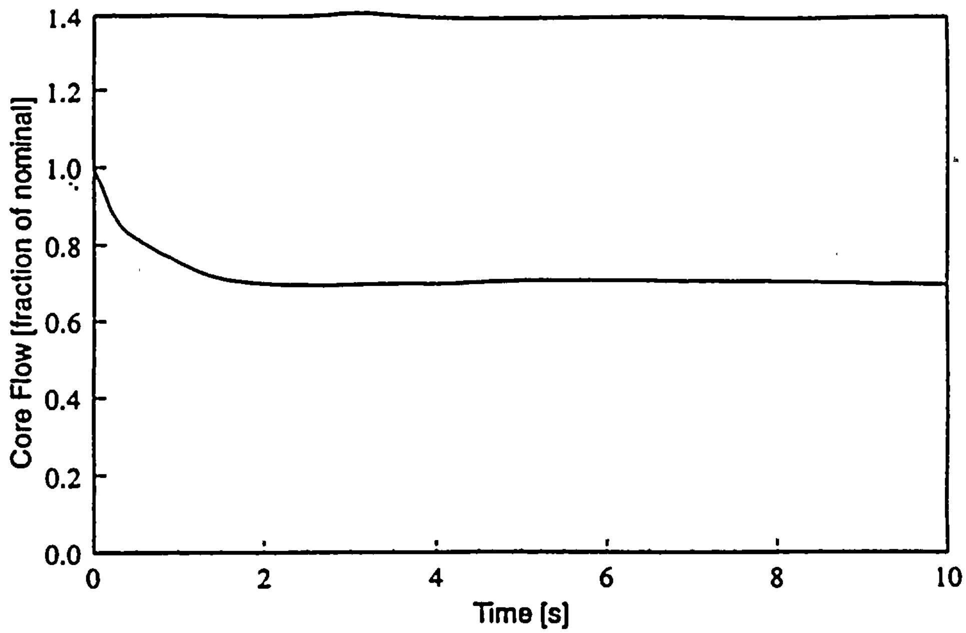
FIGURE 3.3-23

Average and Hot Channel Heat Fluxes and DNBR vs. Time for the Partial Loss Of Flow Event



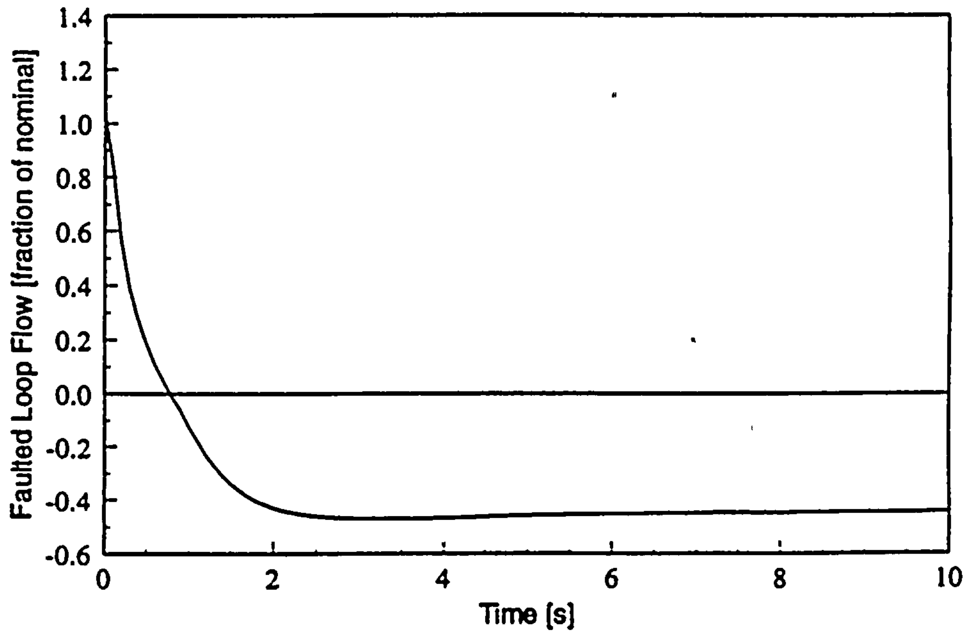


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FIGURE 3.3-24 Total Core Flow and Faulted Loop Flow vs. Time For The Locked Rotor Event

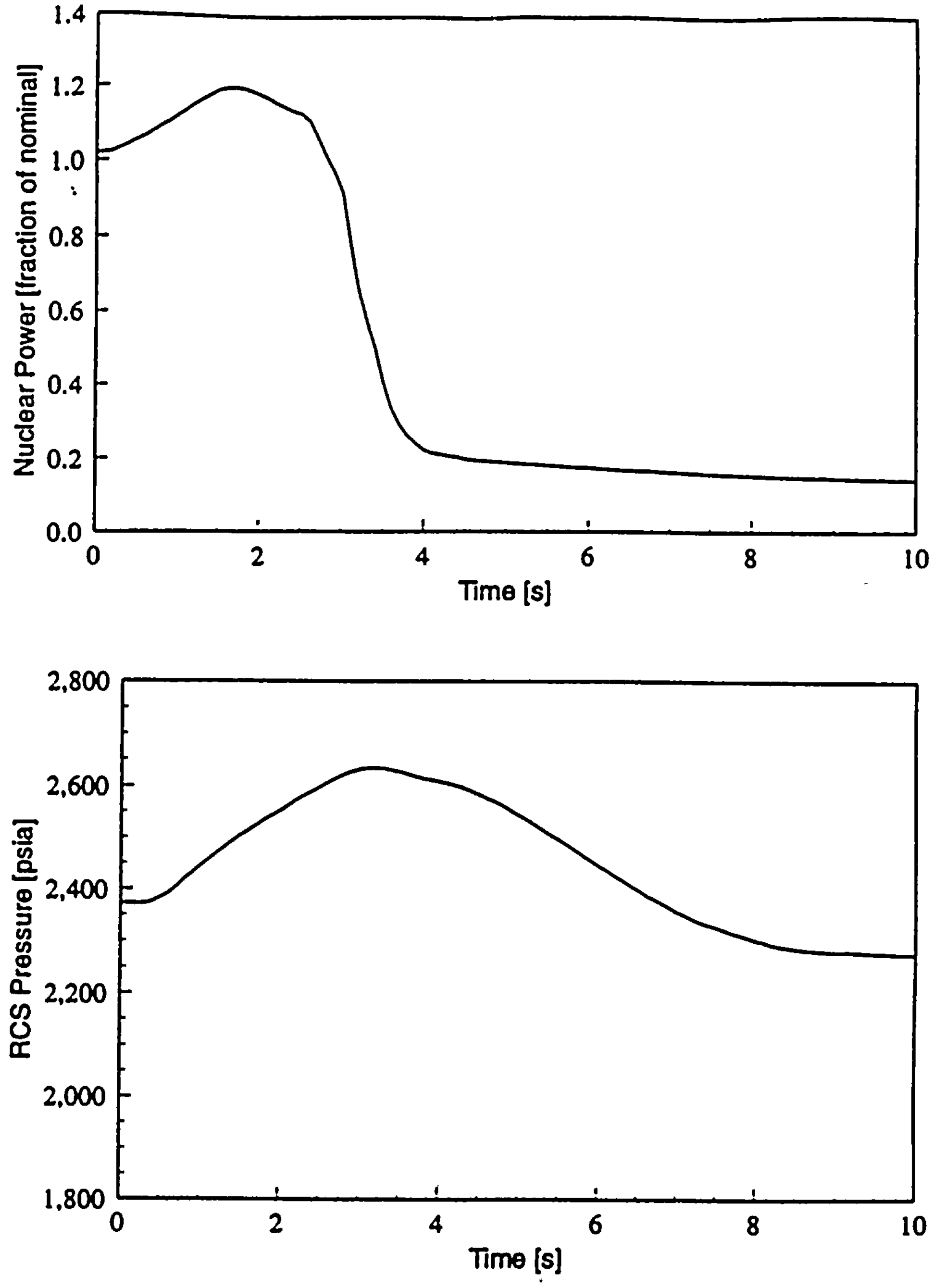

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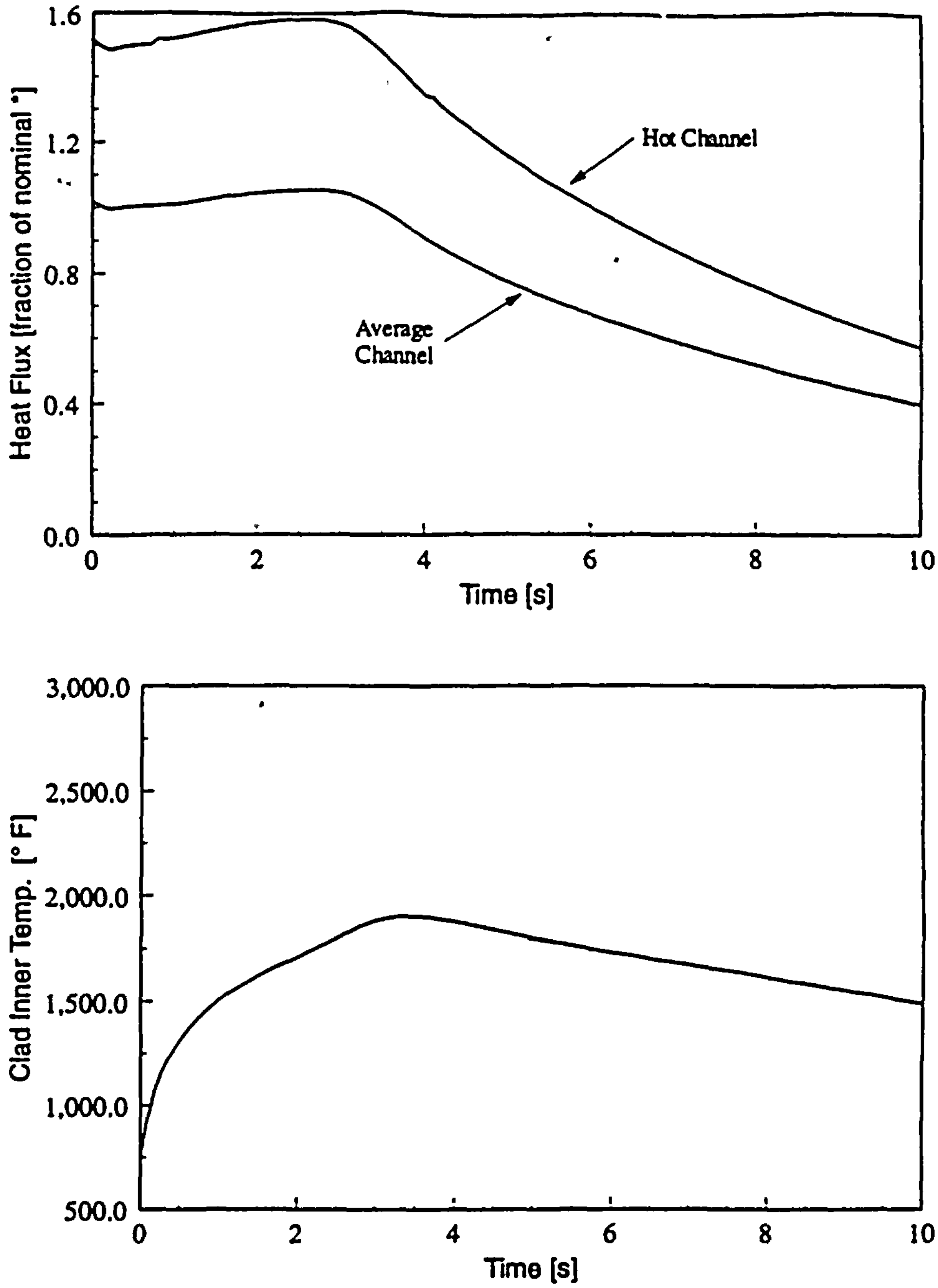
FIGURE 3.3-25

Nuclear Power and RCS Pressure vs. Time
For The Locked Rotor Event



* Heat fluxes are shown as a fraction of the nominal average channel heat flux

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FIGURE 3.3-26
Average and Hot Channel Heat Fluxes vs. Time and Clad Inner Temperature vs. Time For The Locked Rotor Event

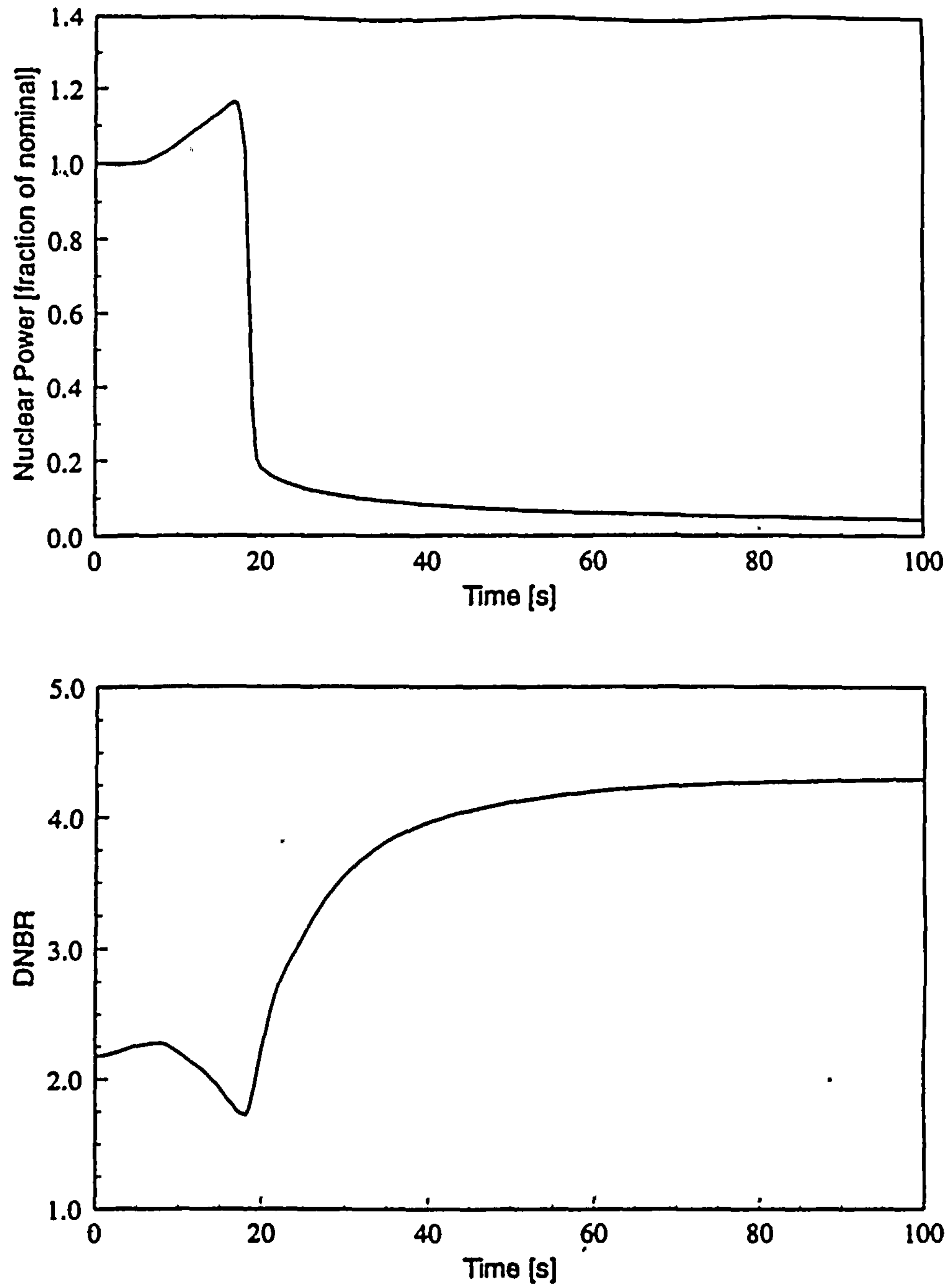

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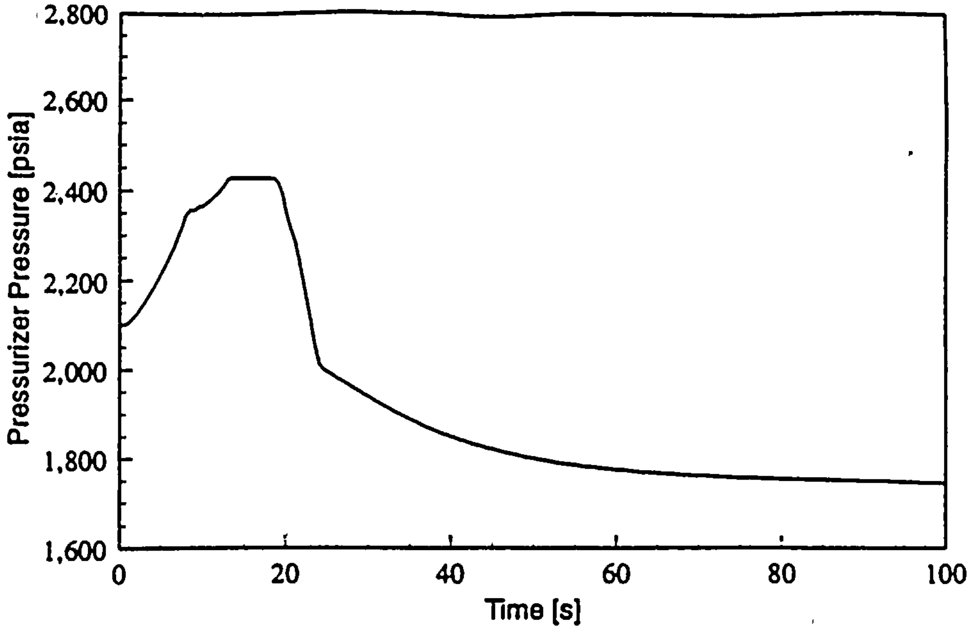
FIGURE 3.3-27

Nuclear Power and DNBR vs.Time For Loss of Load, Minimum Reactivity Feedback With Pressurizer Spray and PORVs



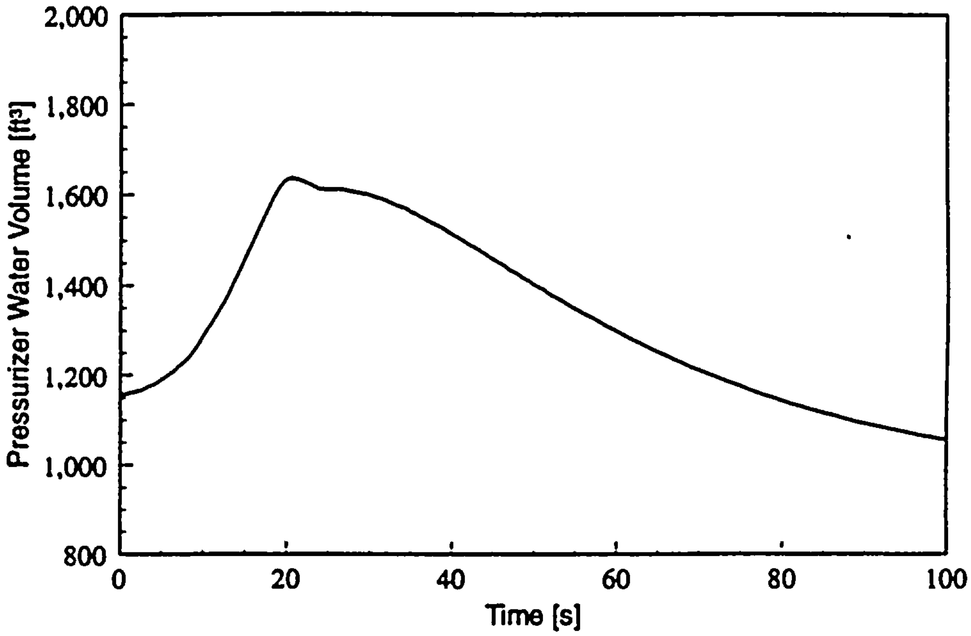


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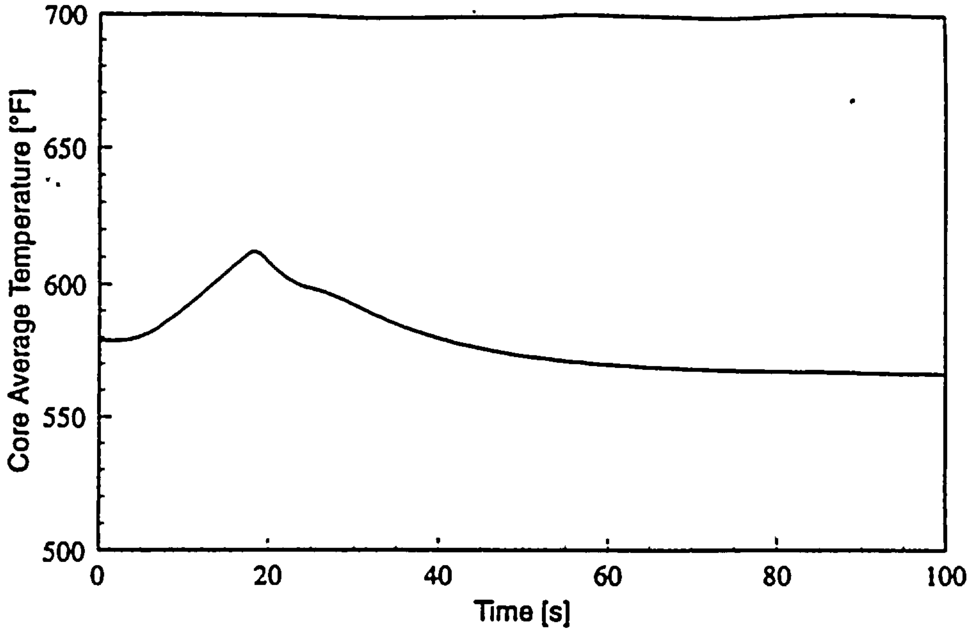
FIGURE 3.3-28

Pressurizer Pressure and Pressurizer Water Volume vs. Time
For Loss of Load, Minimum Reactivity Feedback
With Pressurizer Spray and PORVs



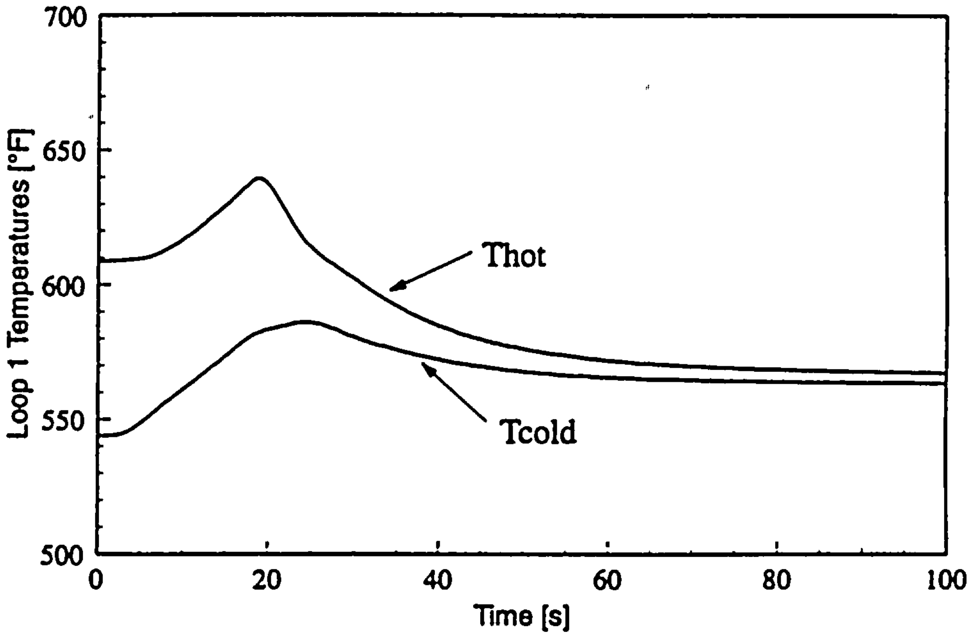


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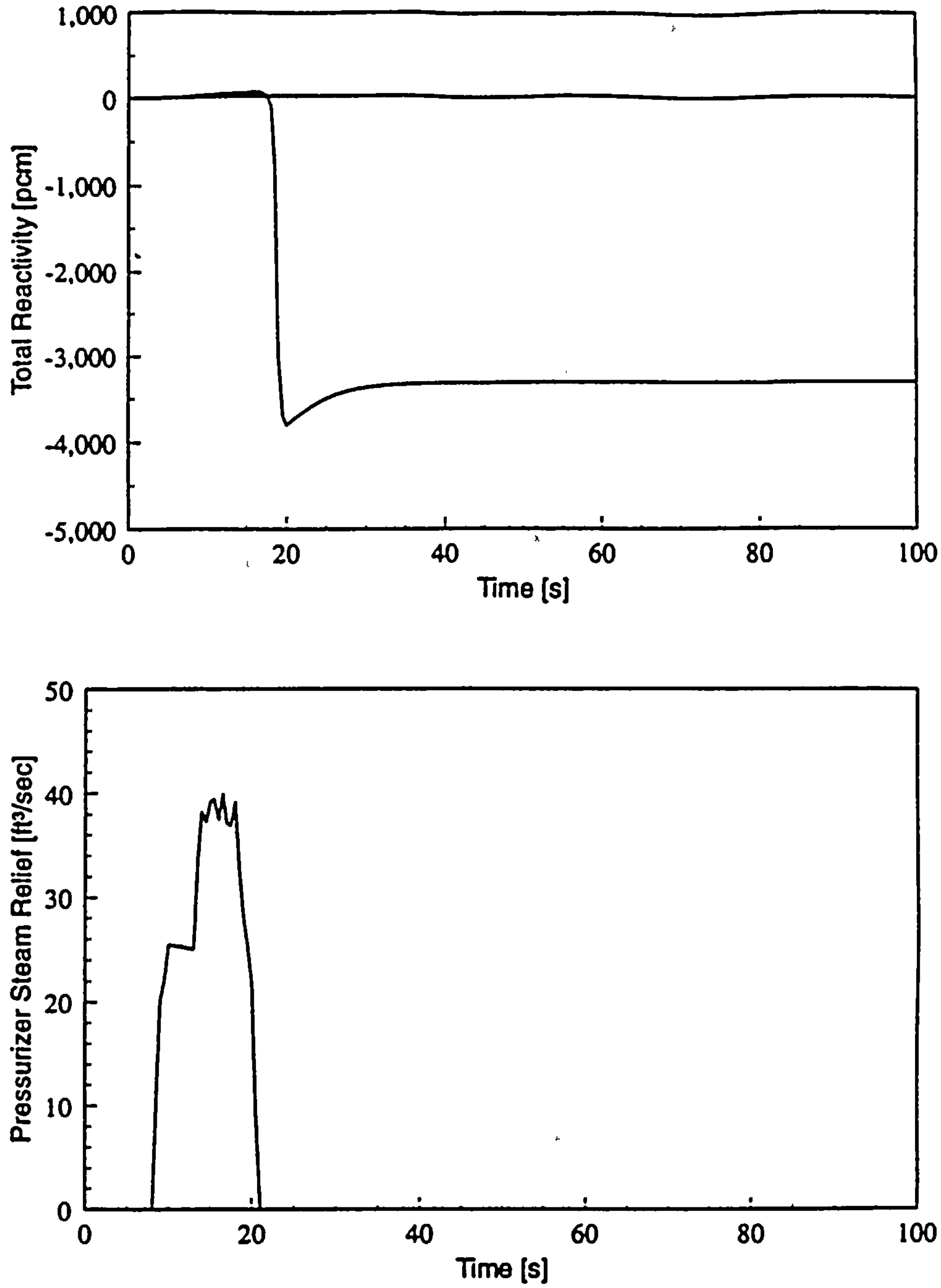
FIGURE 3.3-29

Core Average and Loop 1 Temperatures vs. Time For Loss of Load, Minimum Reactivity Feedback With Pressurizer Spray and PORVs



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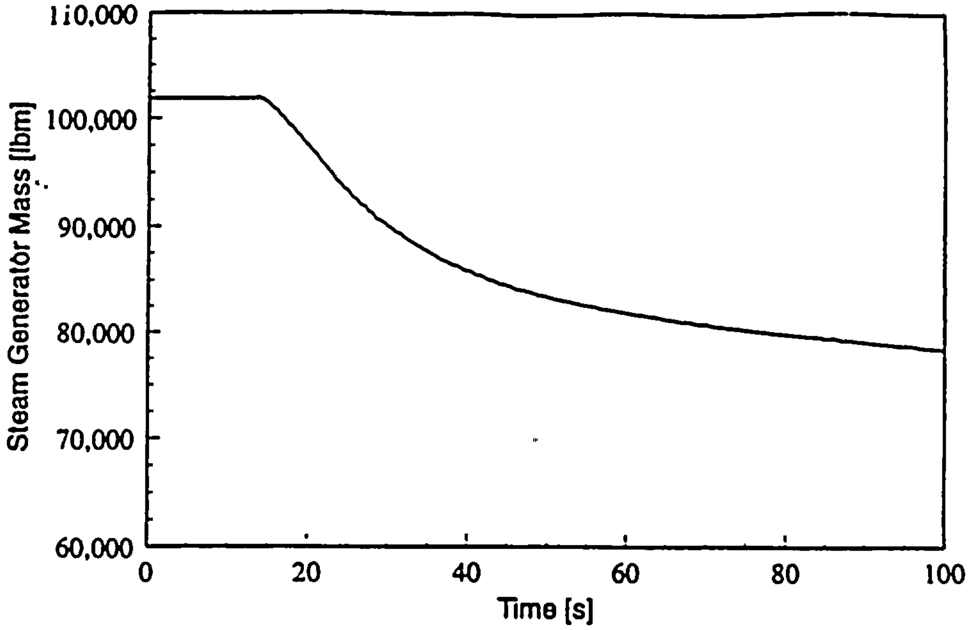
UNIT 1

FIGURE 3.3-30

Total Reactivity and Pressurizer Steam Relief vs.Time
For Loss of Load, Minimum Reactivity Feedback
With Pressurizer Spray and PORVs



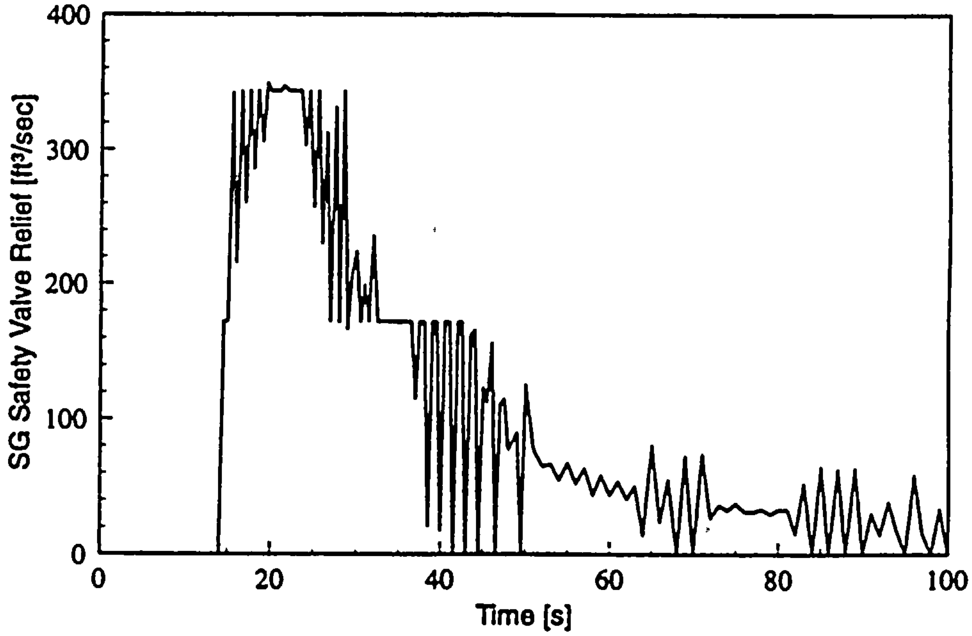


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FIGURE 3.3-31 Steam Generator Mass and Safety Valve Relief vs.Time For Loss of Load, Minimum Reactivity Feedback With Pressurizer Spray and PORVs

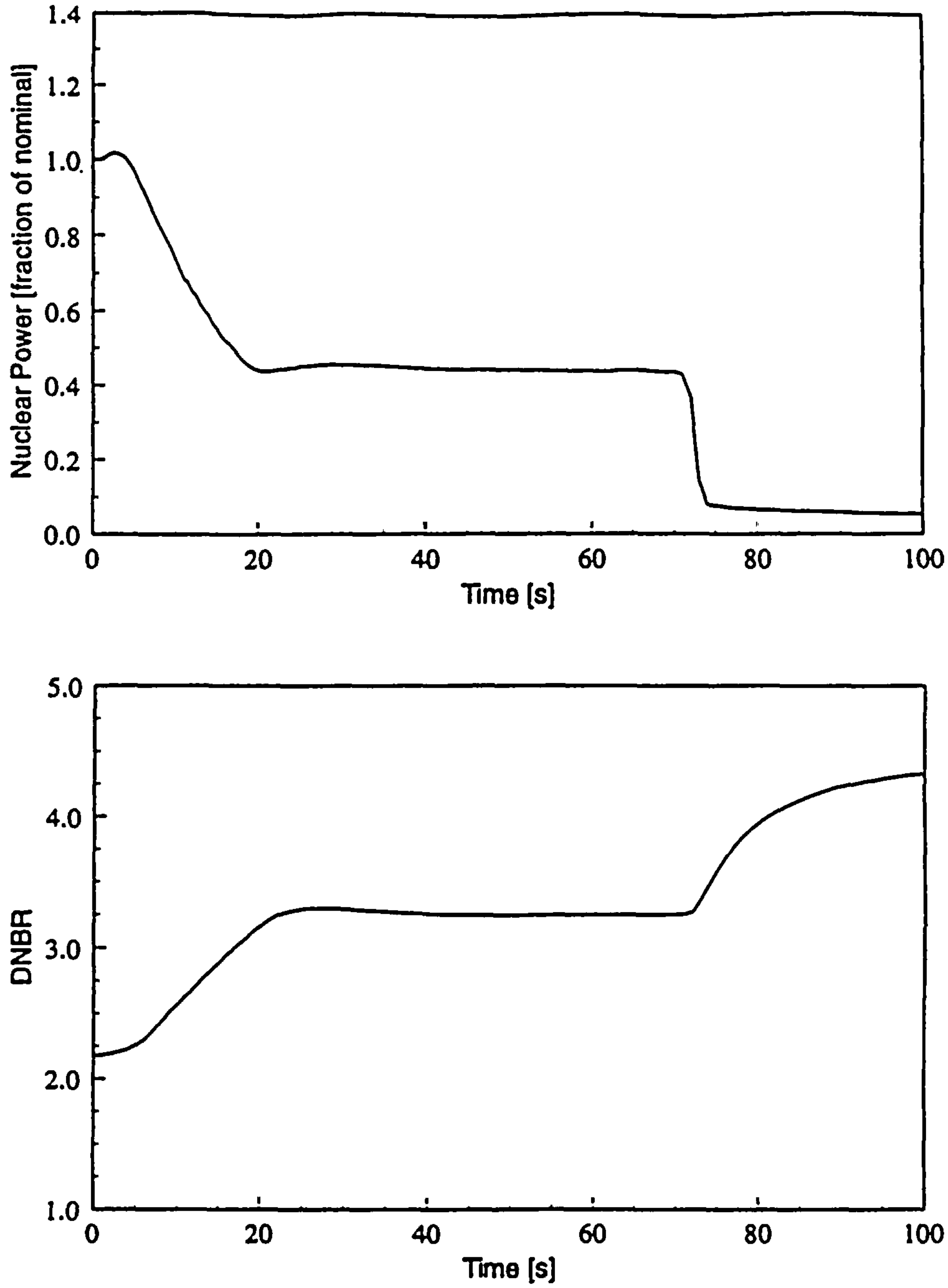

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FIGURE 3.3-32

Nuclear Power and DNBR vs.Time For
Loss of Load, Maximum Reactivity Feedback
With Pressurizer Spray and PORVs

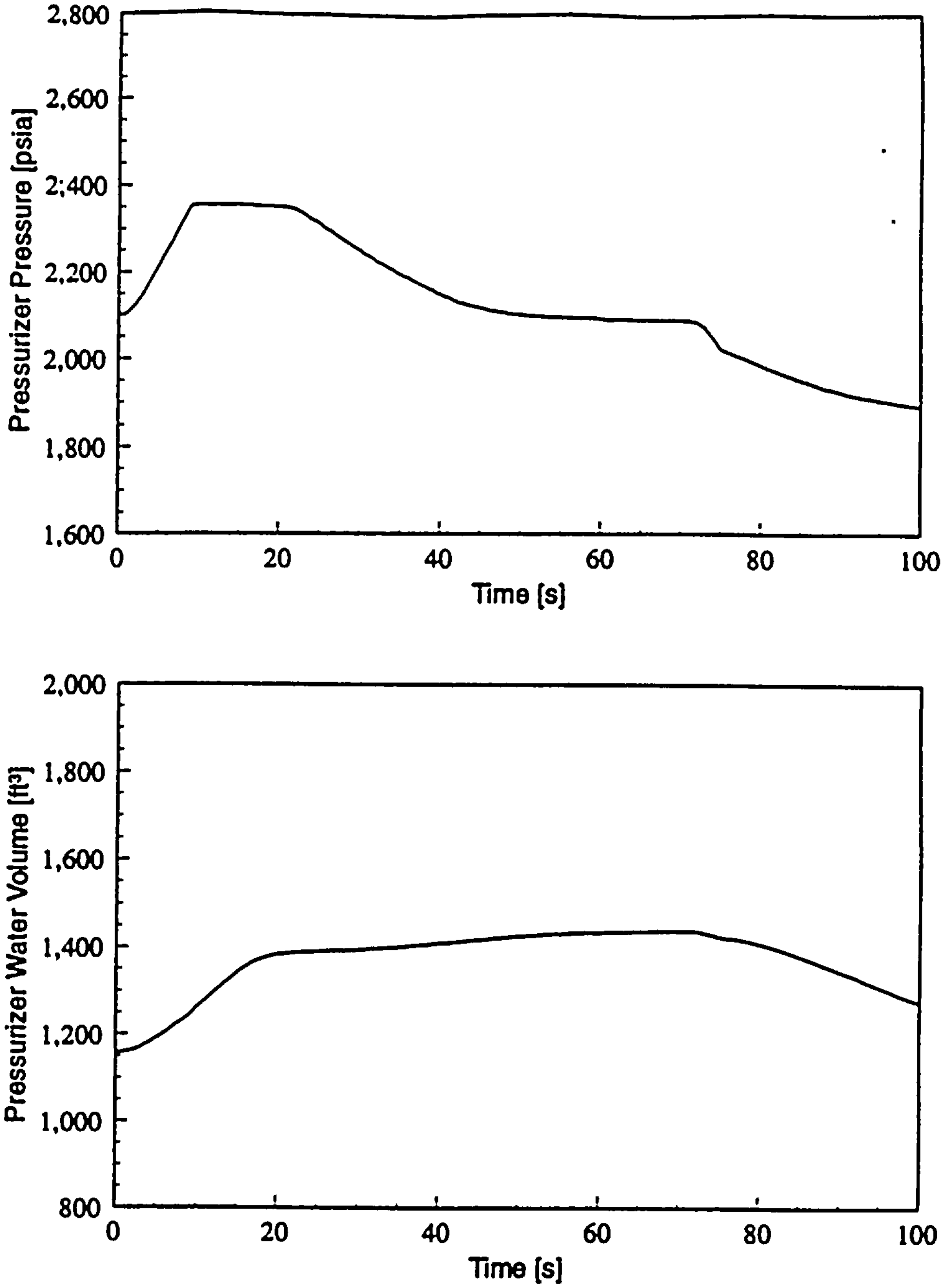

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FIGURE 3.3-33

Pressurizer Pressure and Pressurizer Water Volume vs.Time
For Loss of Load, Maximum Reactivity Feedback
With Pressurizer Spray and PORVs

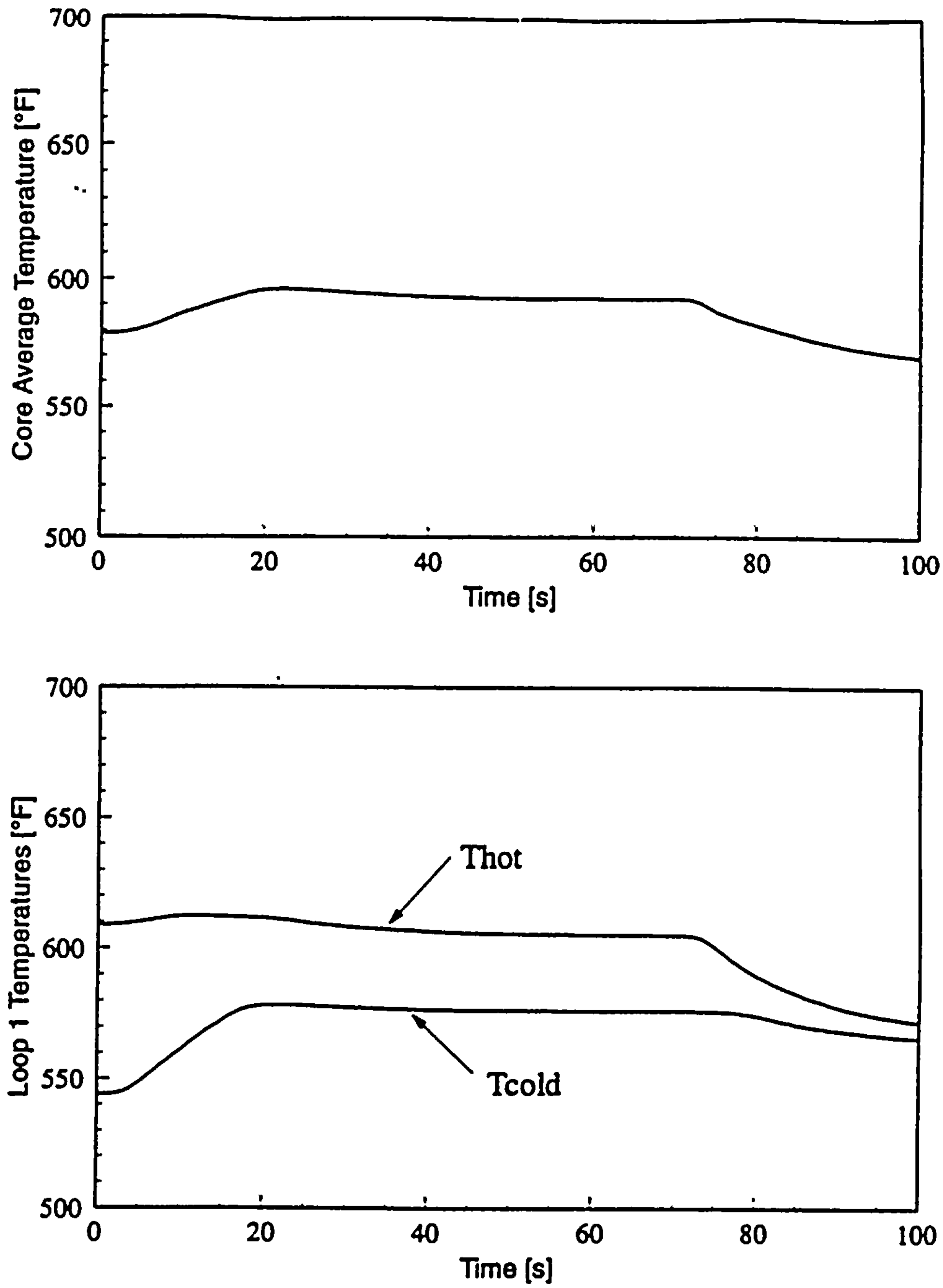

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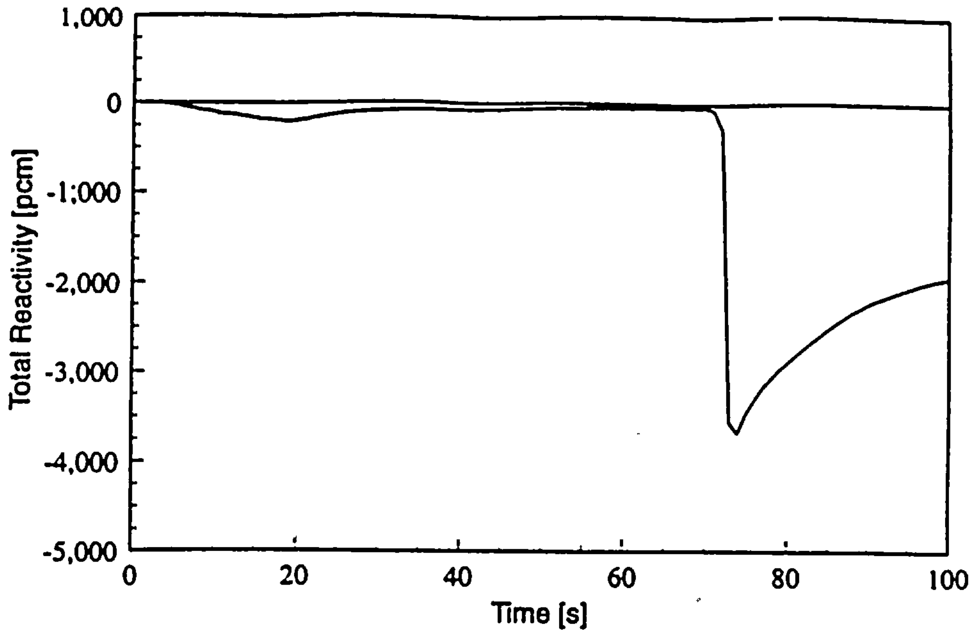
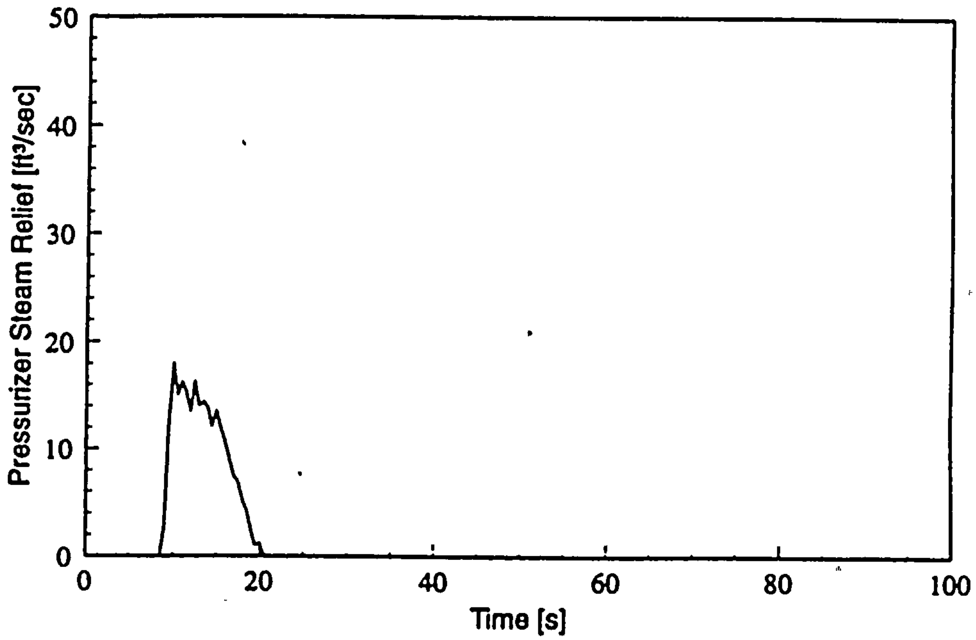
FIGURE 3.3-34

Core Average and Loop 1 Temperatures vs. Time For Loss of Load, Maximum Reactivity Feedback With Pressurizer Spray and PORVs





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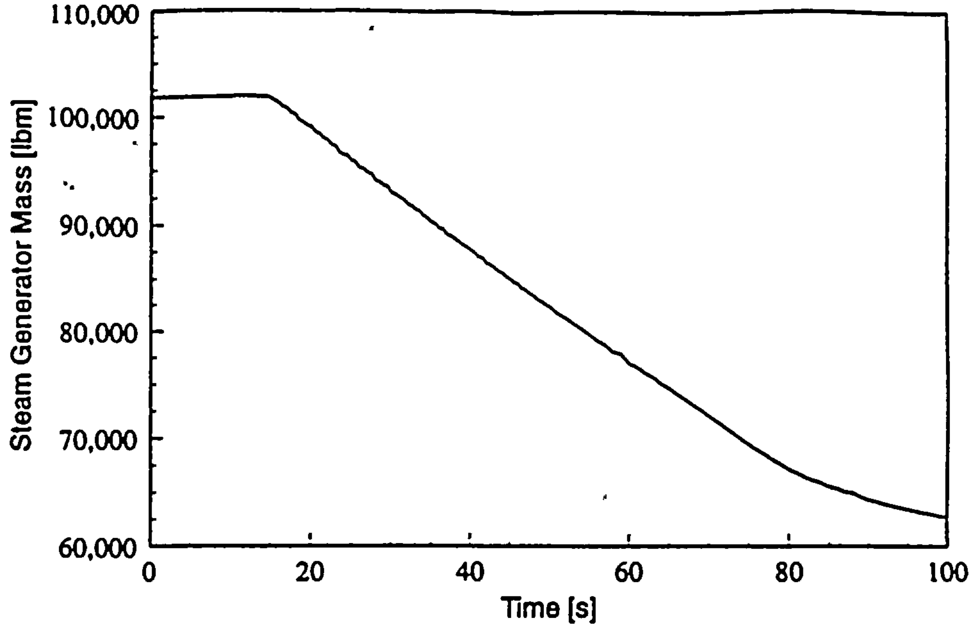
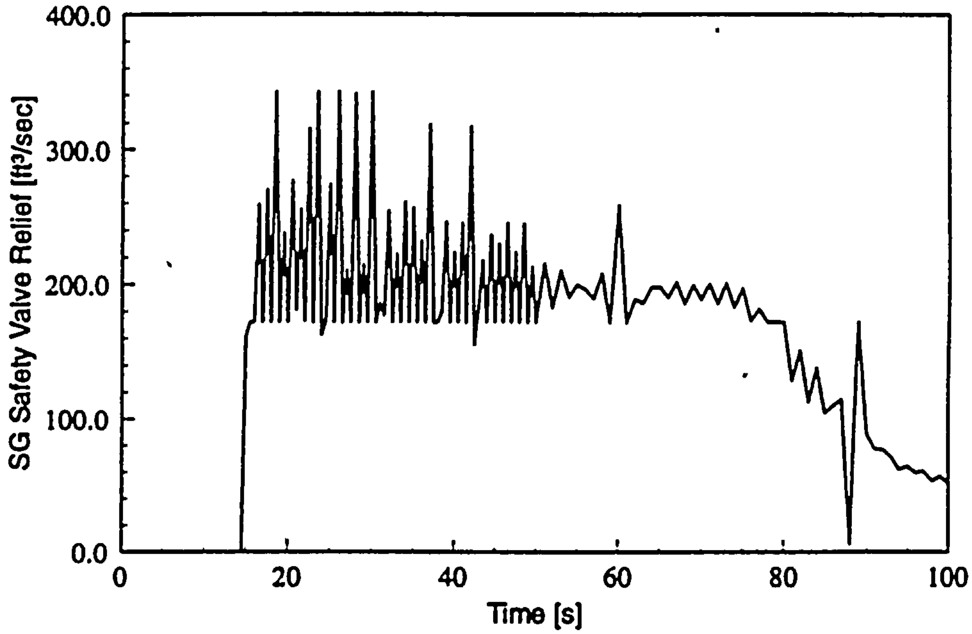
UNIT 1

FIGURE 3.3-35

Total Reactivity and Pressurizer Steam Relief vs.Time
For Loss of Load, Maximum Reactivity Feedback
With Pressurizer Spray and PORVs





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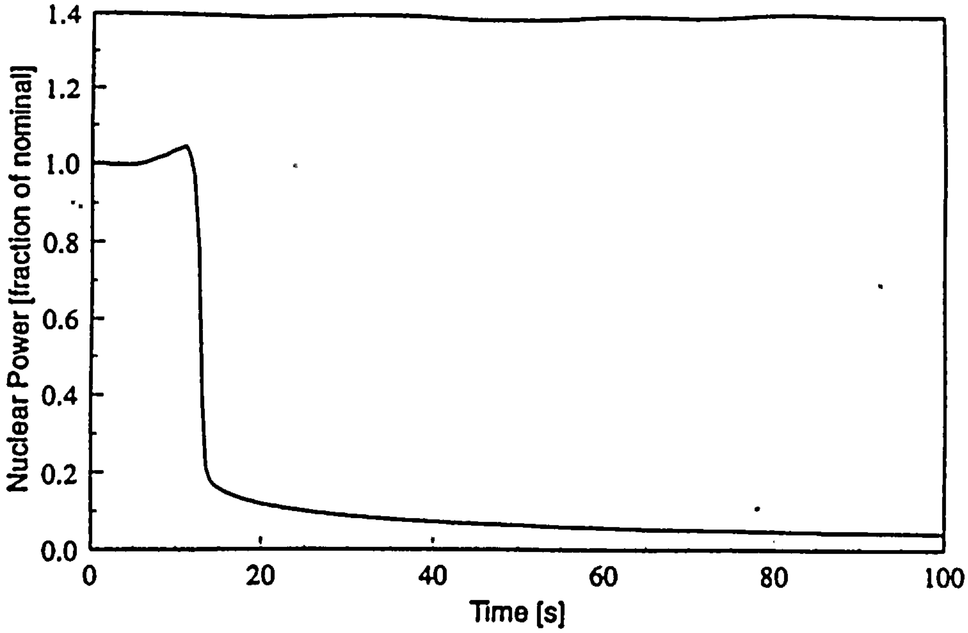
UNIT 1

FIGURE 3.3-36

Steam Generator Mass and Safety Valve Relief vs.Time
For Loss of Load, Maximum Reactivity Feedback
With Pressurizer Spray and PORVs



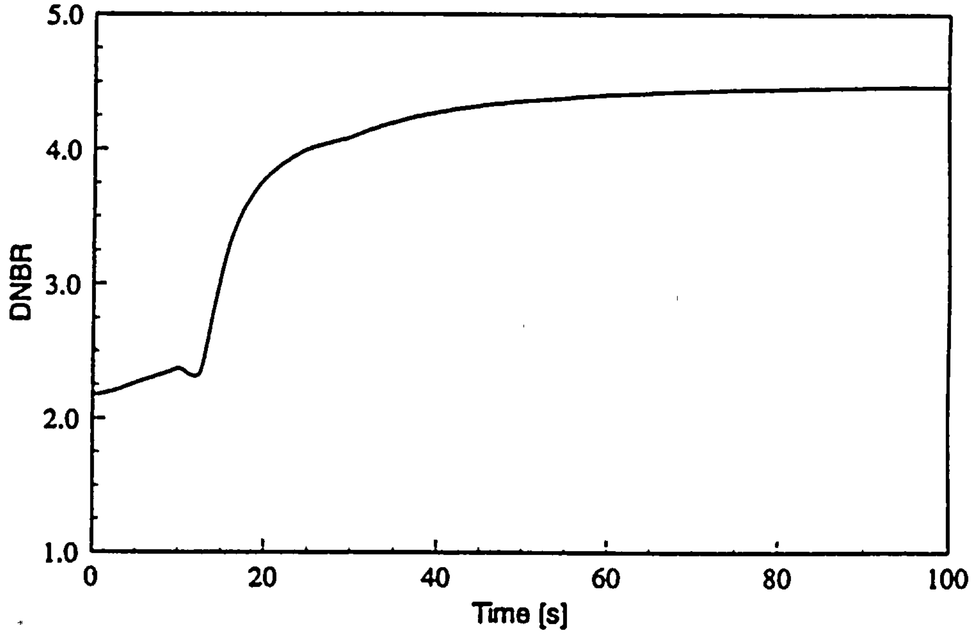


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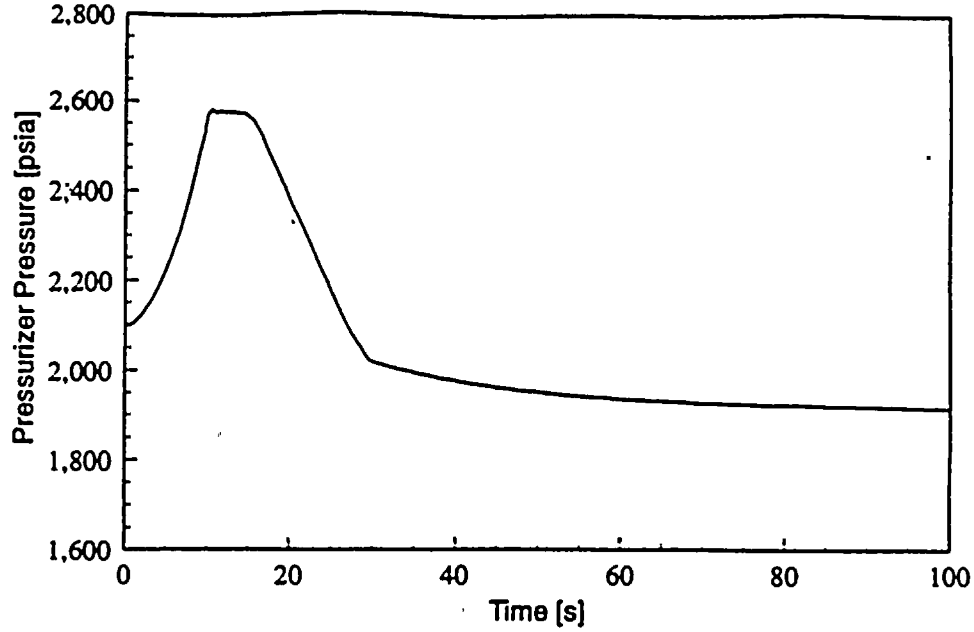
FIGURE 3.3-37

Nuclear Power and DNBR vs.Time For Loss of Load, Minimum Reactivity Feedback Without Pressurizer Spray and PORVs



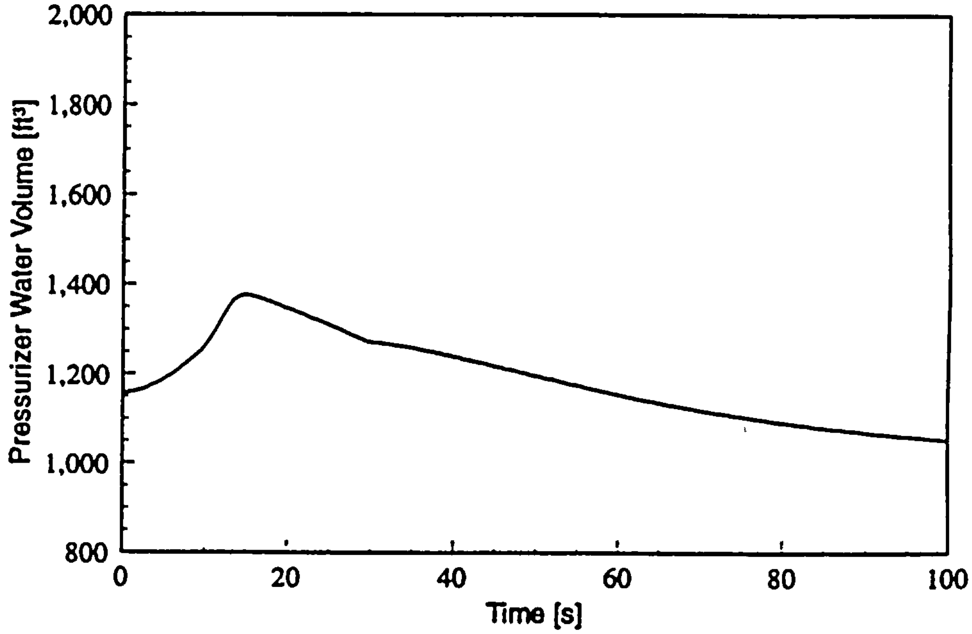


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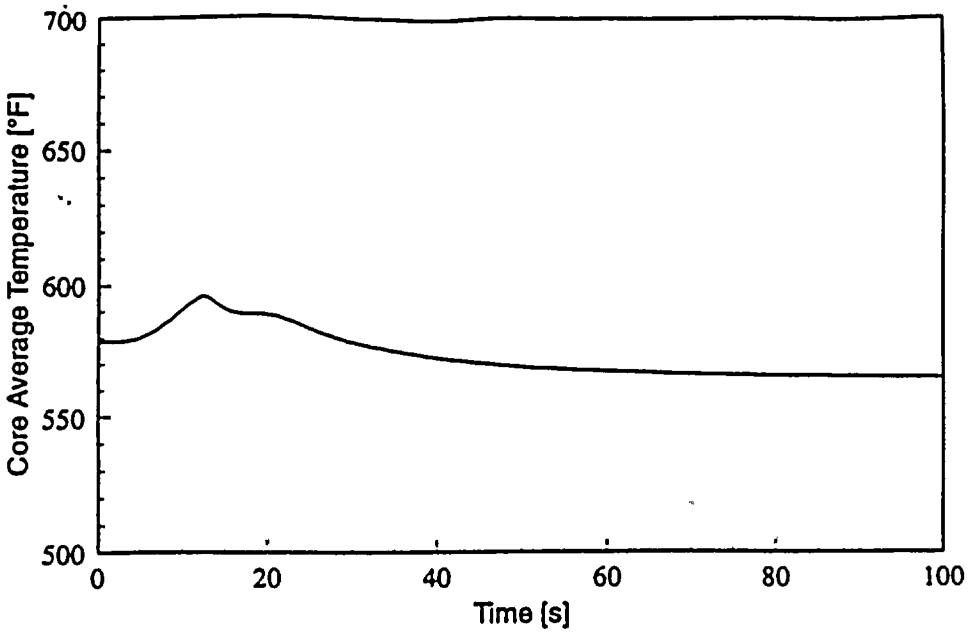
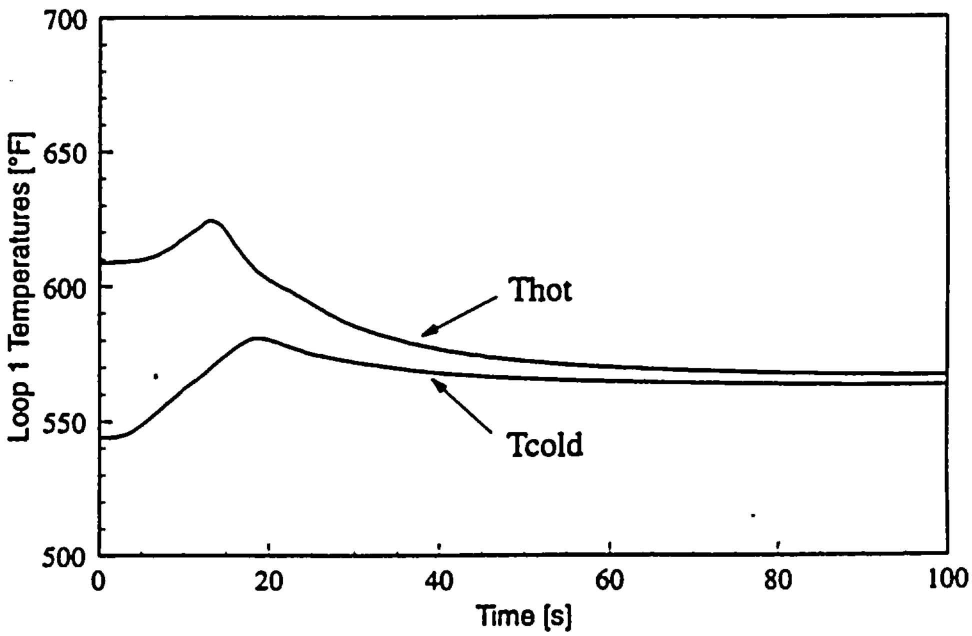
FIGURE 3.3-38

Pressurizer Pressure and Pressurizer Water Volume vs.Time
For Loss of Load, Minimum Reactivity Feedback
Without Pressurizer Spray and PORVs





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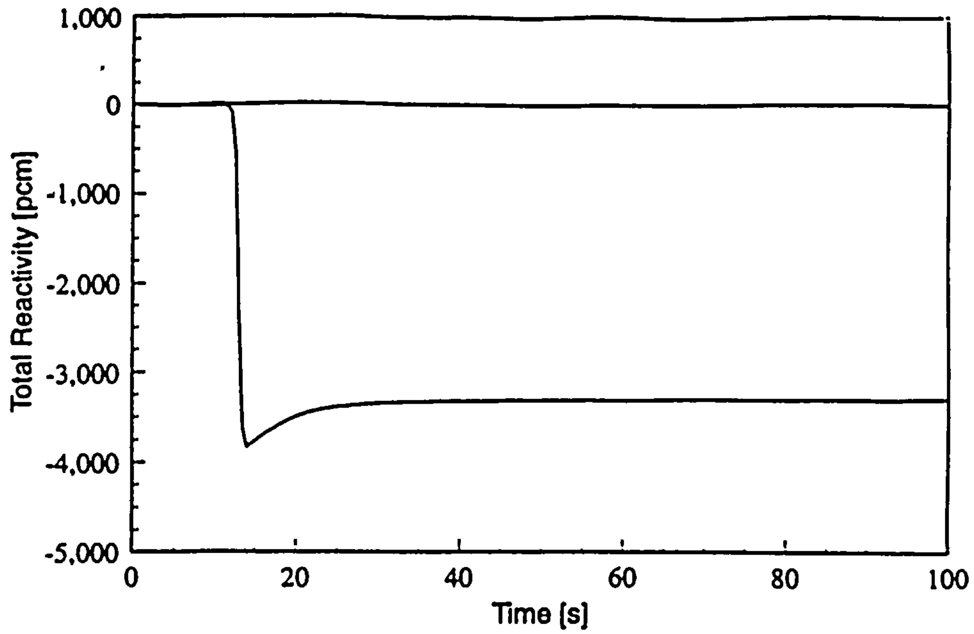
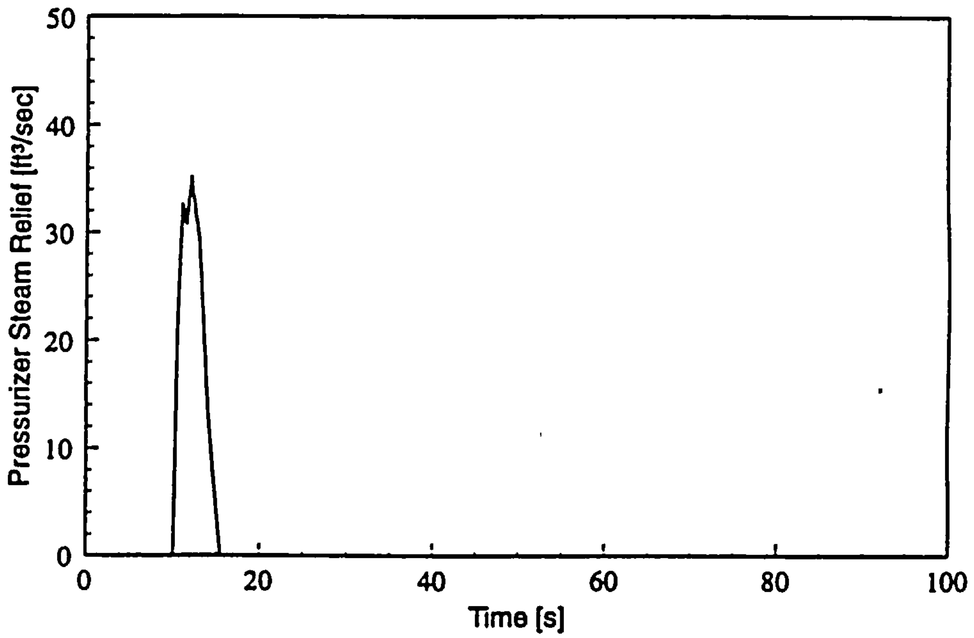
UNIT 1

FIGURE 3.3-39

Core Average and Loop 1 Temperatures vs.Time
For Loss of Load, Minimum Reactivity Feedback
Without Pressurizer Spray and PORVs





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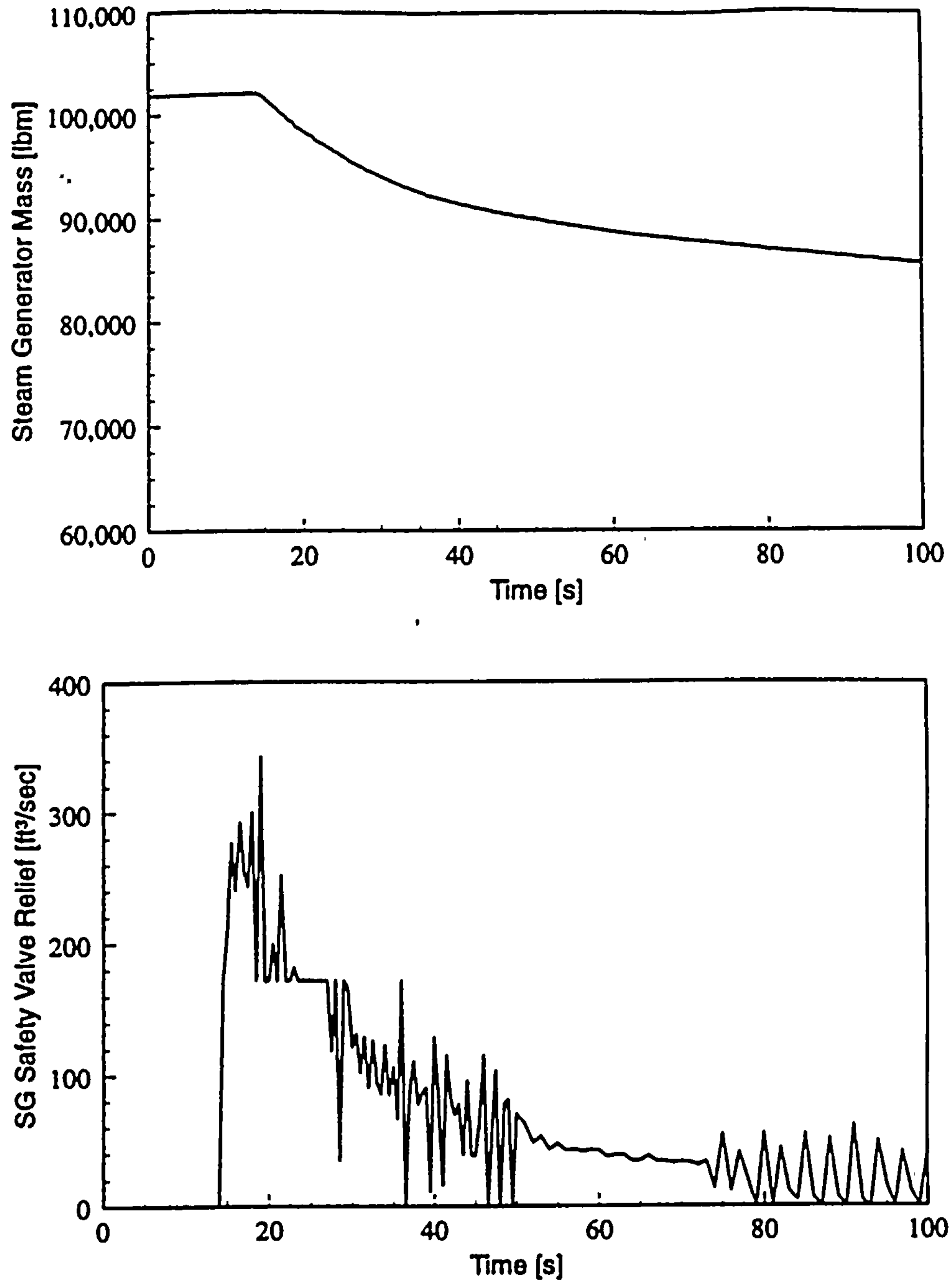
UNIT 1

FIGURE 3.3-40

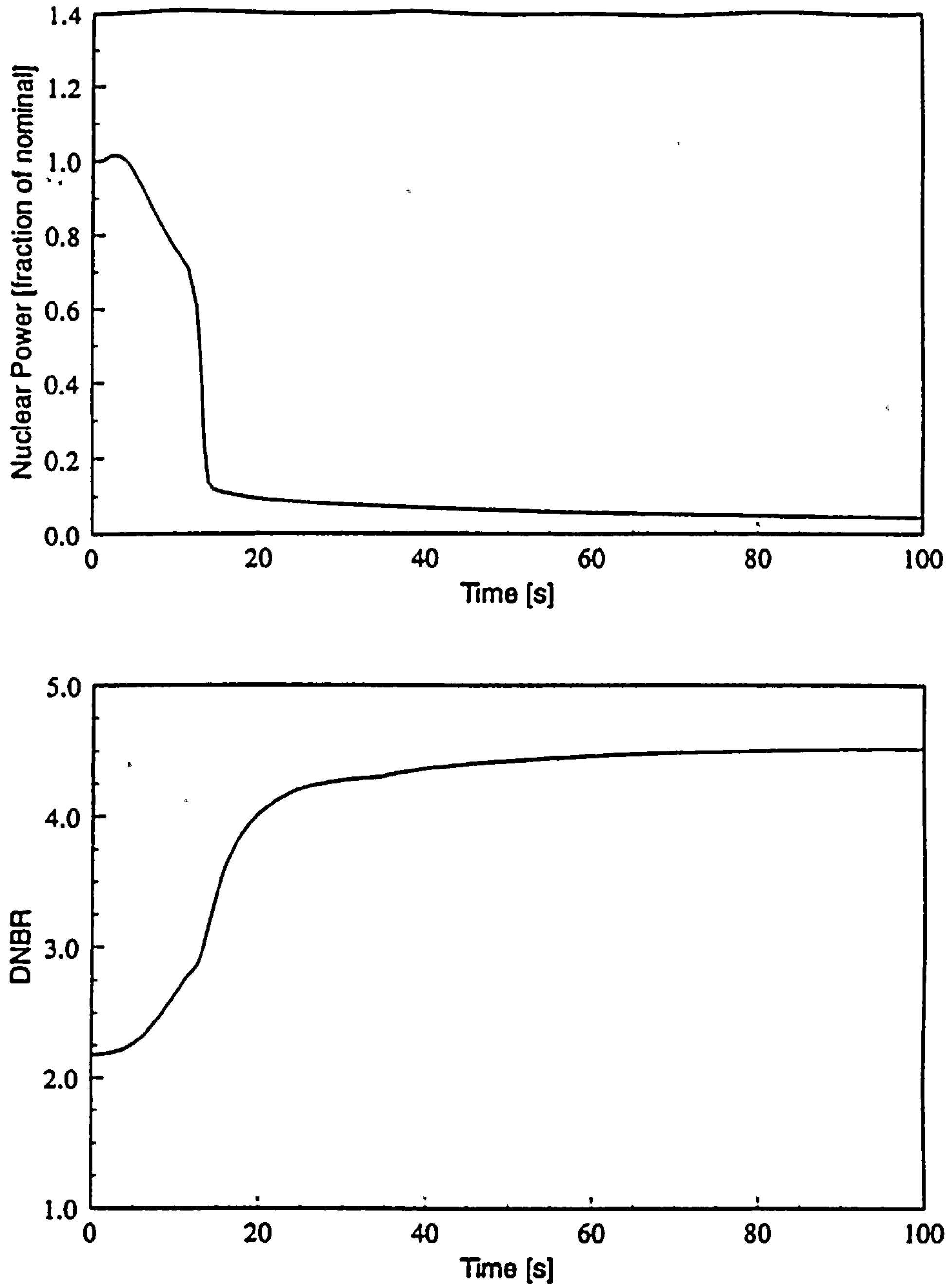
Total Reactivity and Pressurizer Steam Relief vs.Time For Loss of Load, Minimum Reactivity Feedback Without Pressurizer Spray and PORVs



DONALD C. COOK
NUCLEAR PLANT
UNIT 1

FIGURE 3.3-41

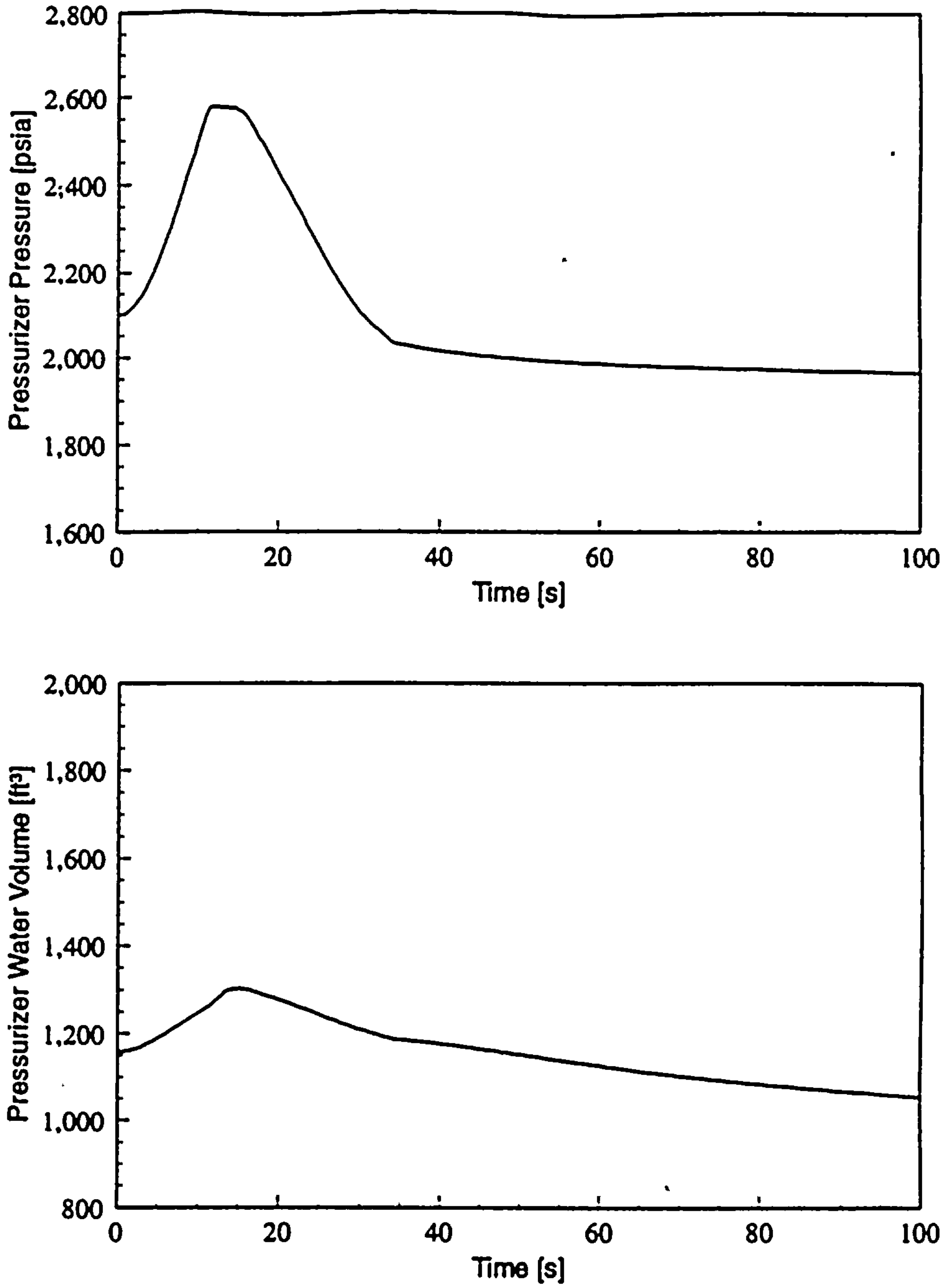
Steam Generator Mass and Safety Valve Relief vs.Time
For Loss of Load, Minimum Reactivity Feedback
Without Pressurizer Spray and PORVs



DONALD C. COOK NUCLEAR PLANT UNIT 1

FIGURE 3.3-42

Nuclear Power and DNBR vs.Time For Loss of Load, Maximum Reactivity Feedback Without Pressurizer Spray and PORVs



DONALD C. COOK
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UNIT 1

FIGURE 3.3-43

Pressurizer Pressure and Pressurizer Water Volume vs.Time
For Loss of Load, Maximum Reactivity Feedback
Without Pressurizer Spray and PORVs

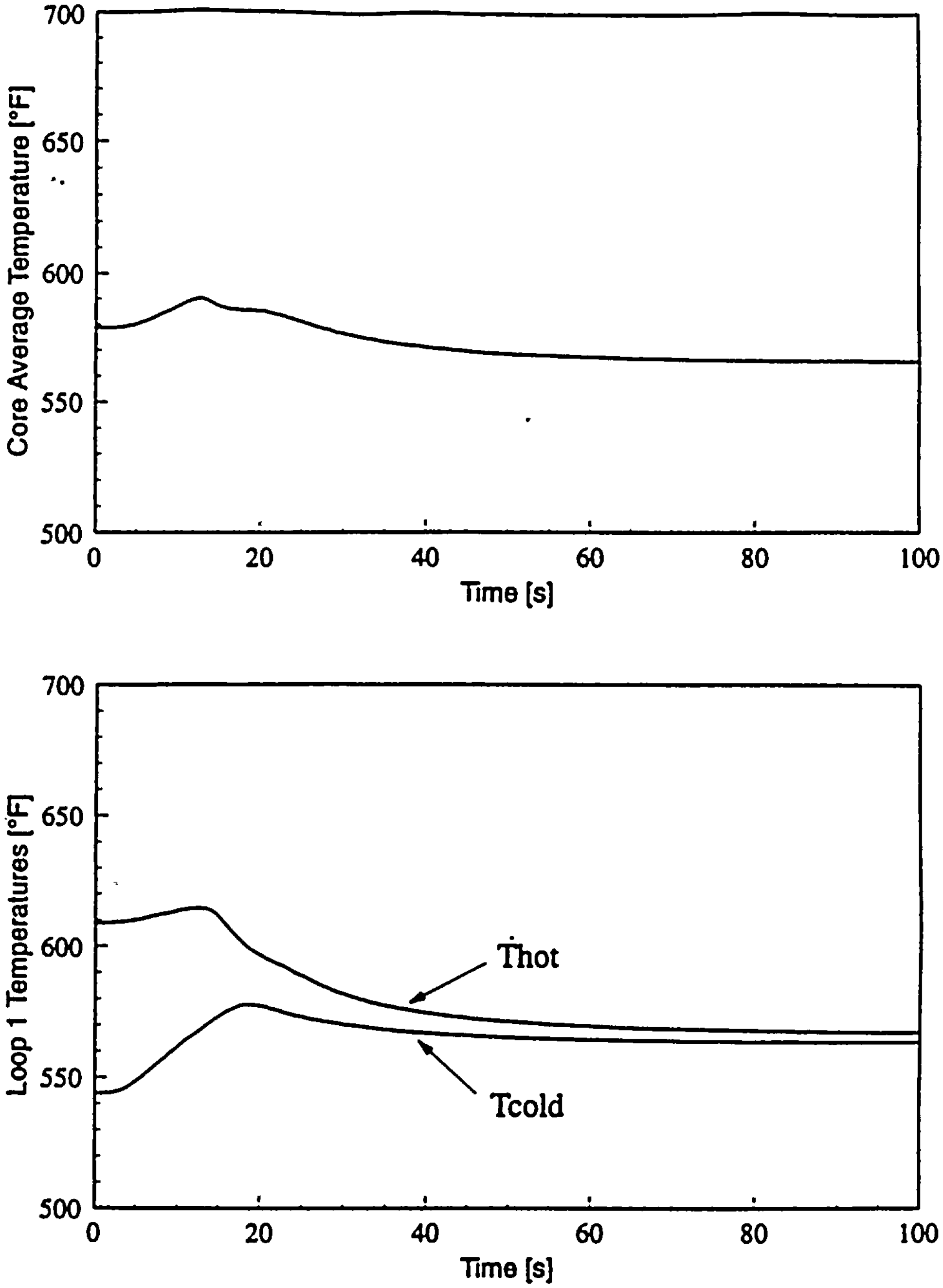

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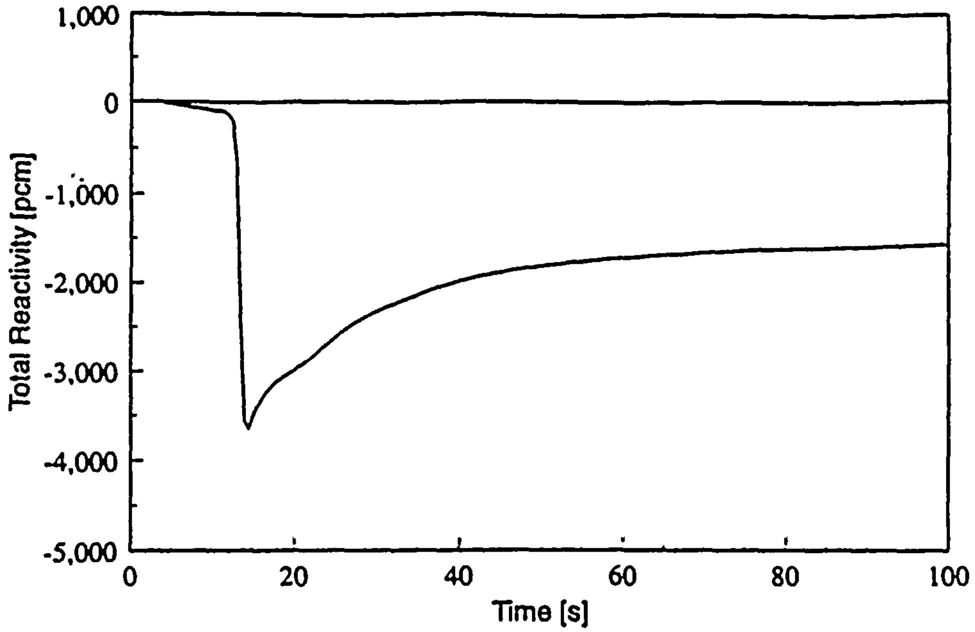
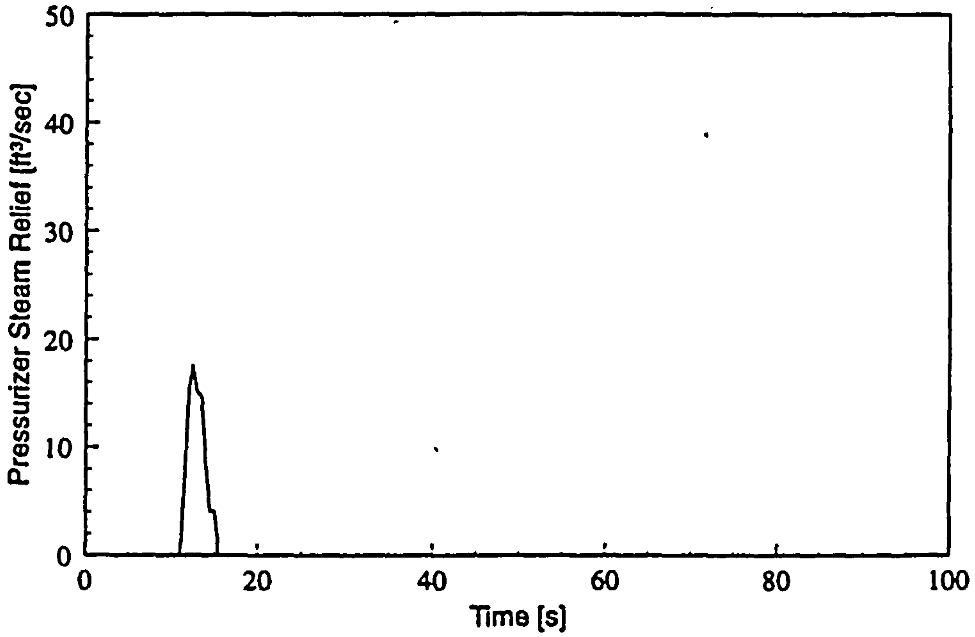
FIGURE 3.3-44

Core Average and Loop 1 Temperatures vs. Time For Loss of Load, Maximum Reactivity Feedback Without Pressurizer Spray and PORVs





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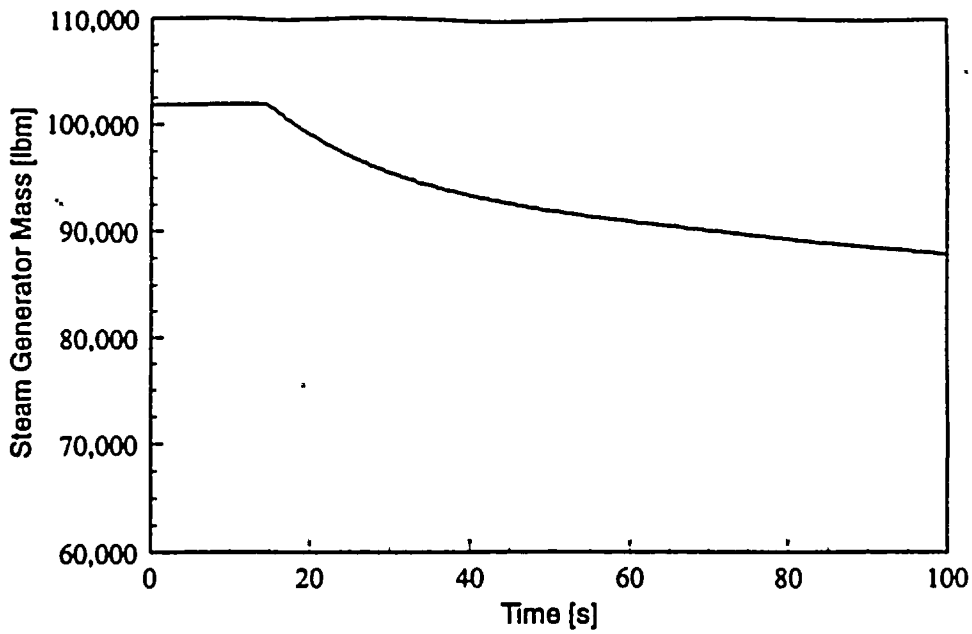
UNIT 1

FIGURE 3.3-45

Total Reactivity and Pressurizer Steam Relief vs.Time
For Loss of Load, Maximum Reactivity Feedback
Without Pressurizer Spray and PORVs



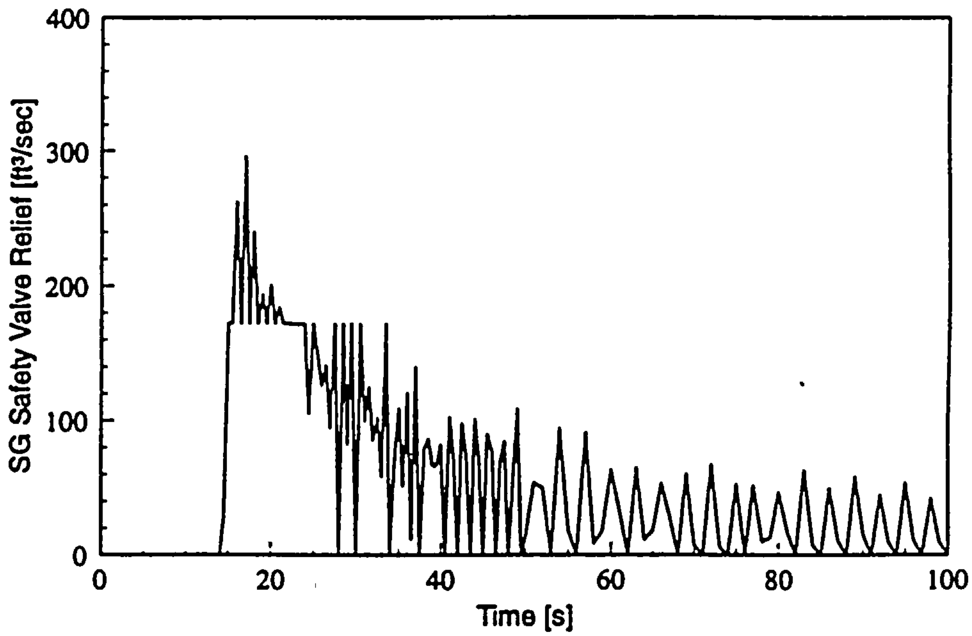


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FIGURE 3.3-46

Steam Generator Mass and Safety Valve Relief vs.Time
For Loss of Load, Maximum Reactivity Feedback
Without Pressurizer Spray and PORVs

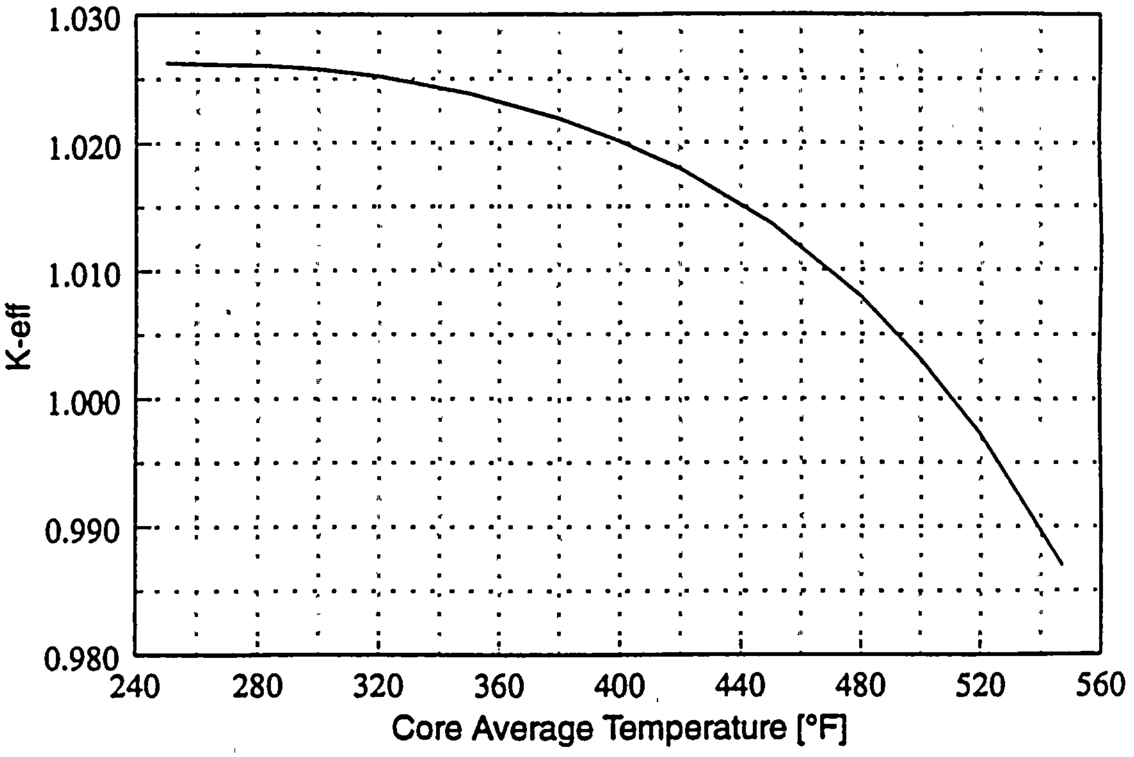

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FIGURE 3.3-47
Variation of Reactivity With Core Temperature At 1050 psia For The End Of Life Rodded Core With One Control Rod Assembly Stuck (Zero Power) For The Steamline Break Double Ended Rupture Event

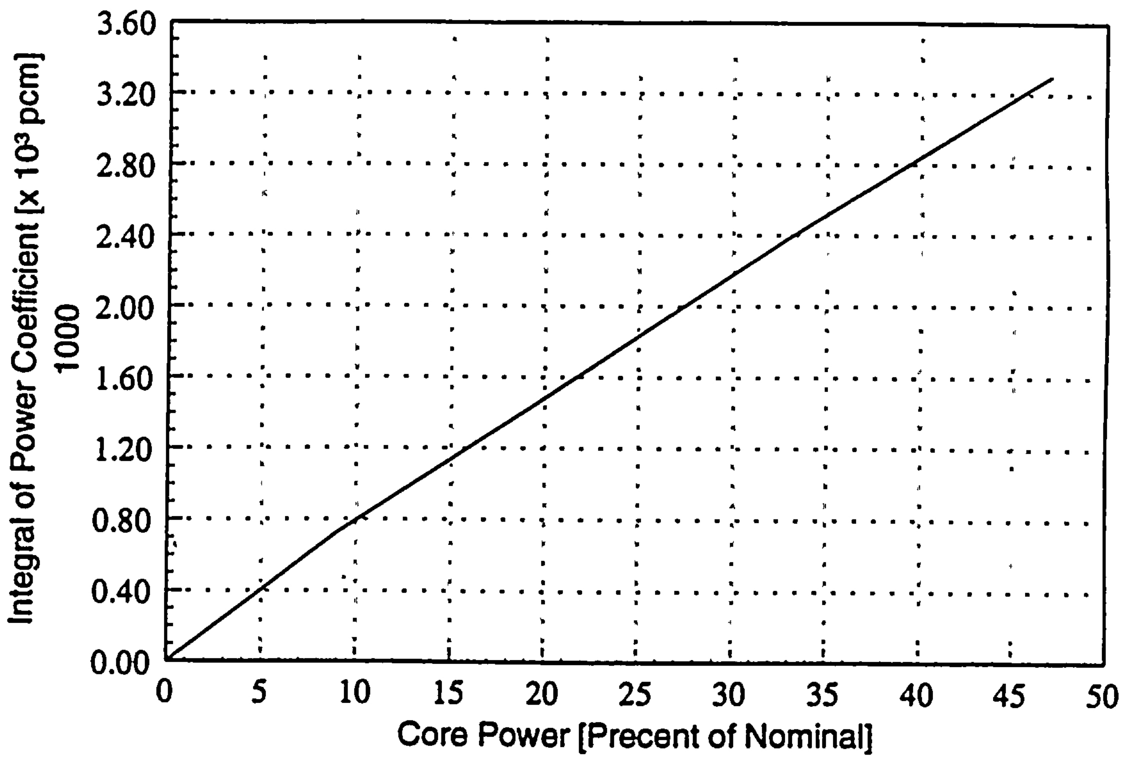

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FIGURE 3.3-48

Doppler Power Feedback
For The Steamline Break Double Ended Rupture Event

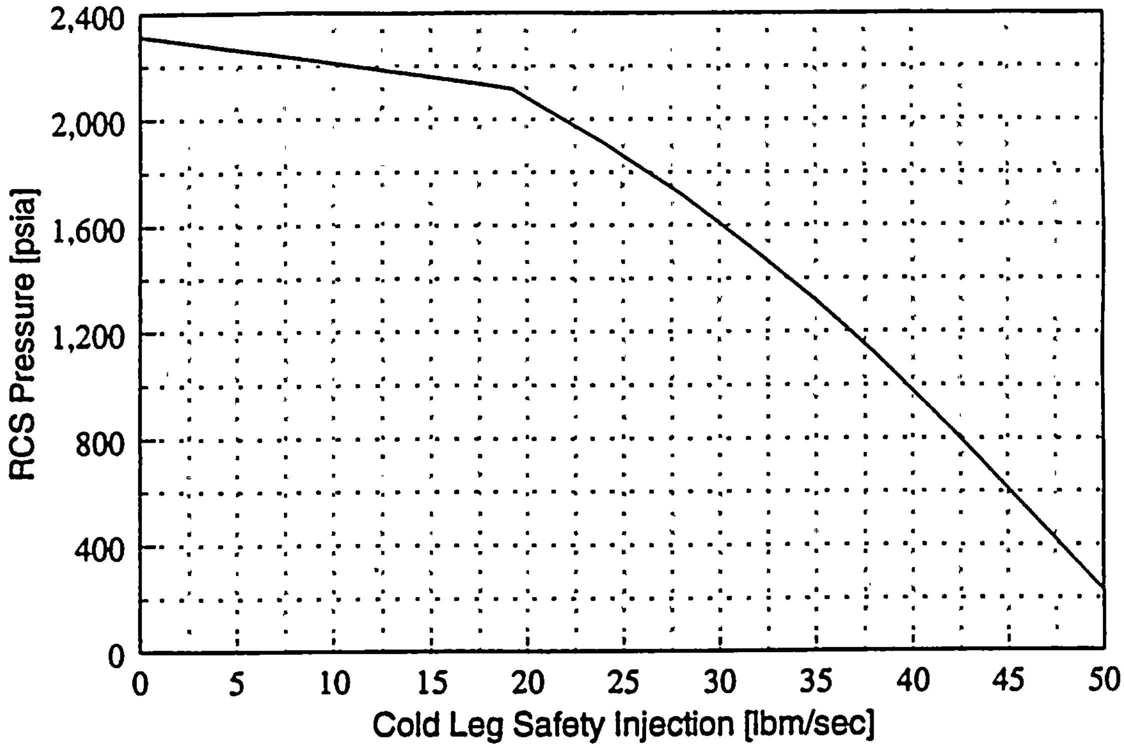

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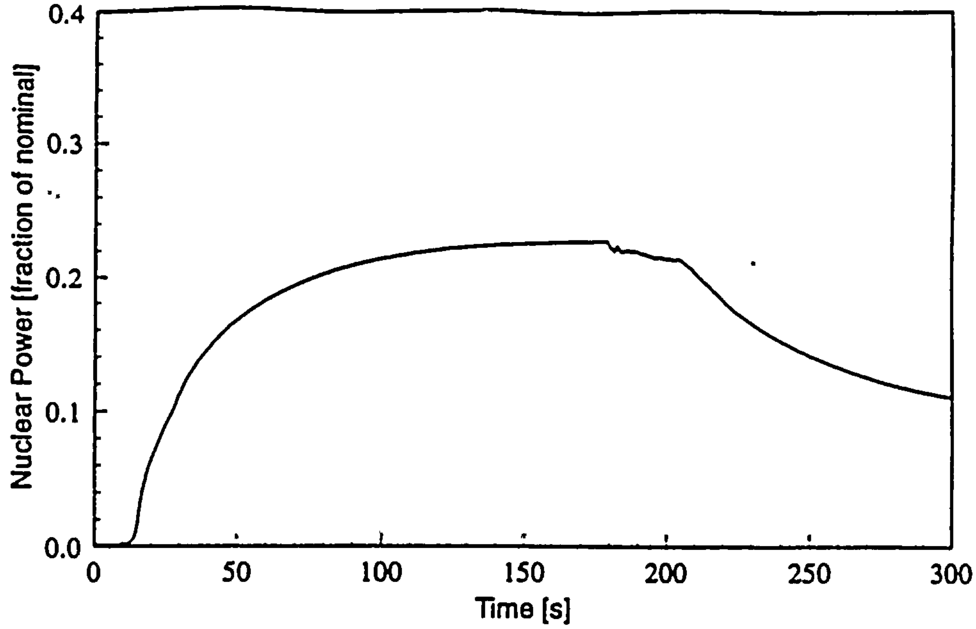
FIGURE 3.3-49

Safety Injection Flow Supplied By One Charging Pump For The Steamline Break Double Ended Rupture Event



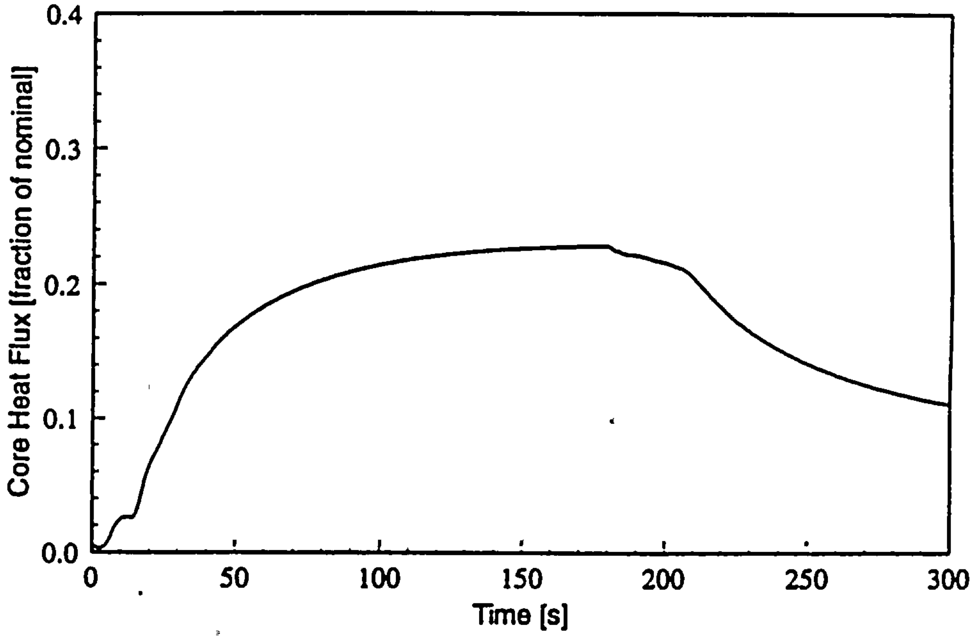


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FIGURE 3.3-50

Nuclear Power and Core Heat Flux vs. Time For
The Steamline Break Double Ended Rupture Event
[Inside Containment With Power]

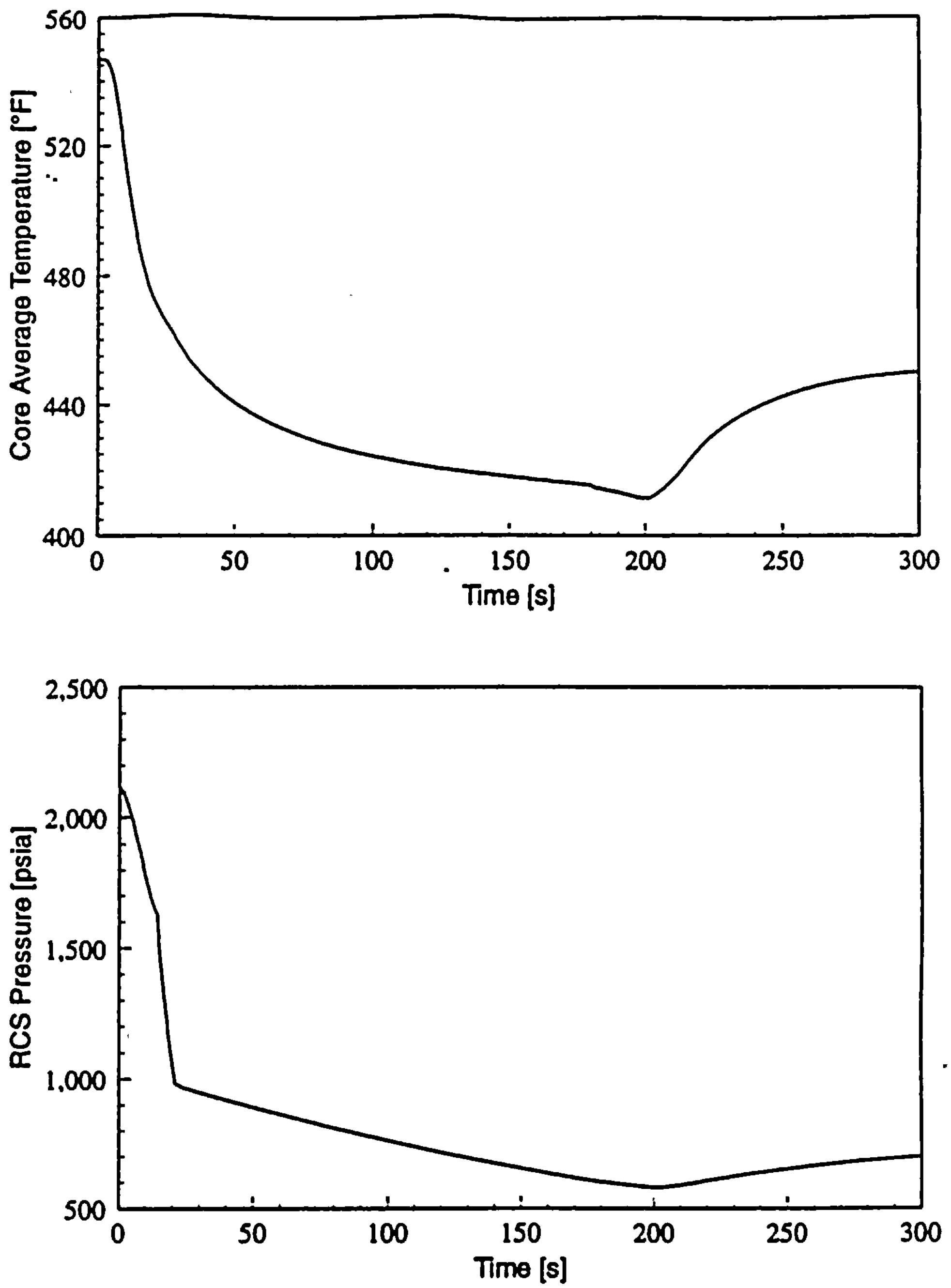

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FIGURE 3.3-51

Core Average Temperature and RCS Pressure vs. Time For The Steamline Break Double Ended Rupture Event [Inside Containment With Power]

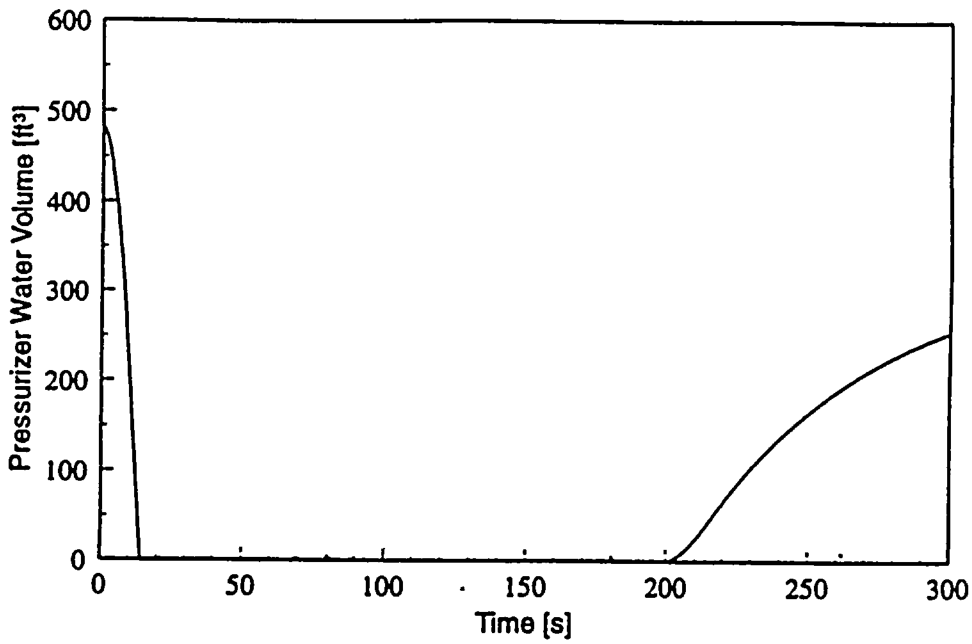

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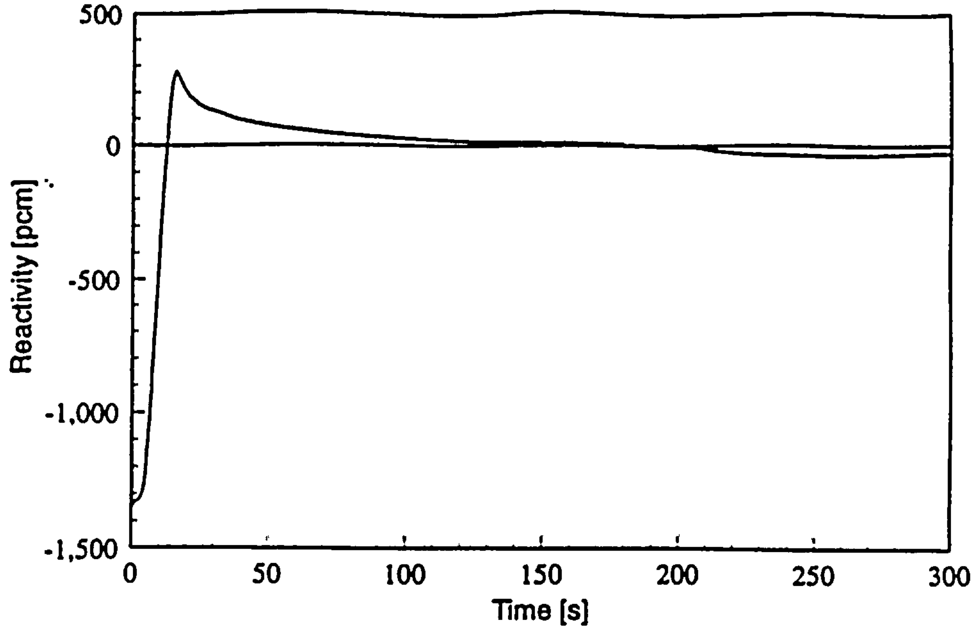
FIGURE 3.3-52

Pressurizer Water Volume vs. Time
For The Steamline Break Double Ended Rupture Event
[Inside Containment With Power]



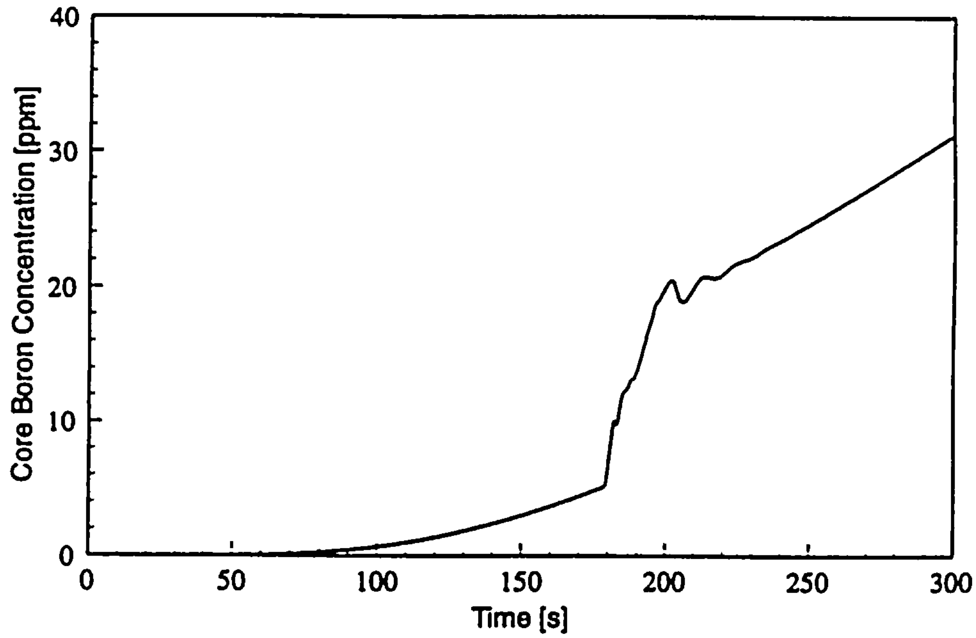


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FIGURE 3.53

Reactivity and Core Boron Concentration vs. Time
For The Steamline Break Double Ended Rupture Event
[Inside Containment With Power]

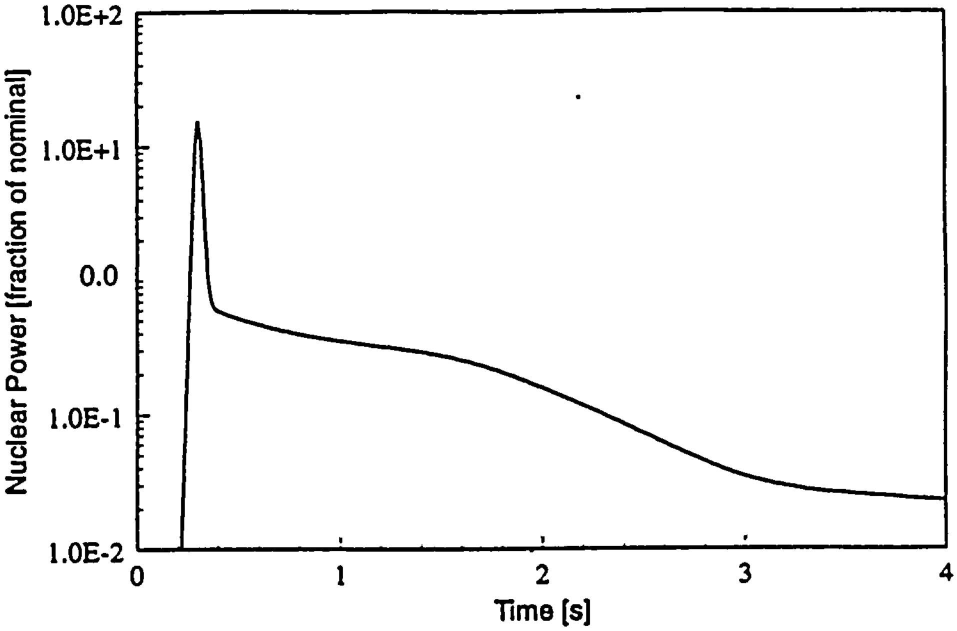

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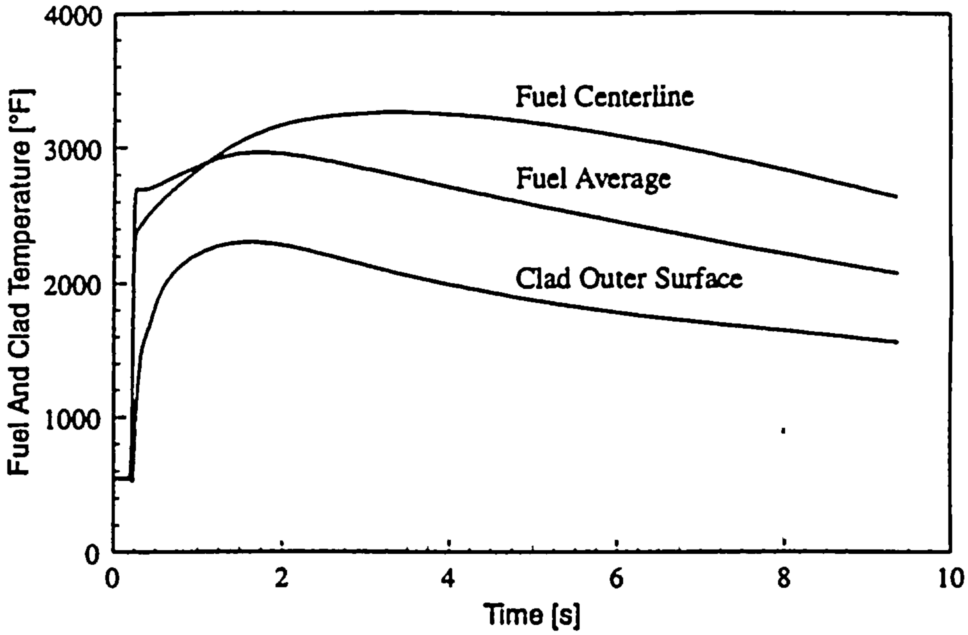
FIGURE 3.3-54

Nuclear Power vs.Time For The Rod Ejection Event, Hot Zero Power, End Of Life



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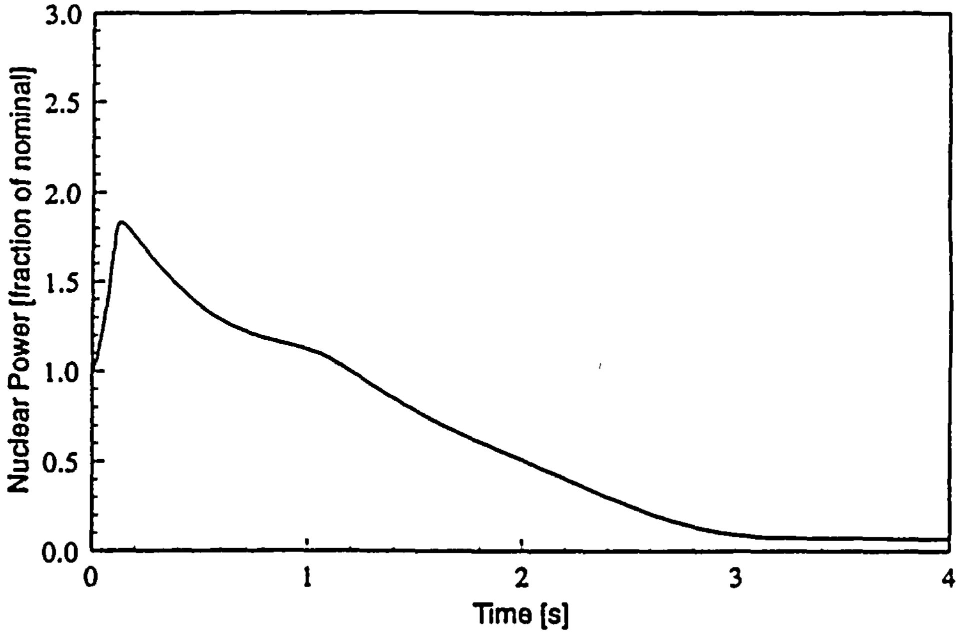
UNIT 1

FIGURE 3.3-55

Fuel Centerline, Fuel Average, and Clad Outer Surface Temperature vs.Time For The Rod Ejection Event, Hot Zero Power, End Of Life



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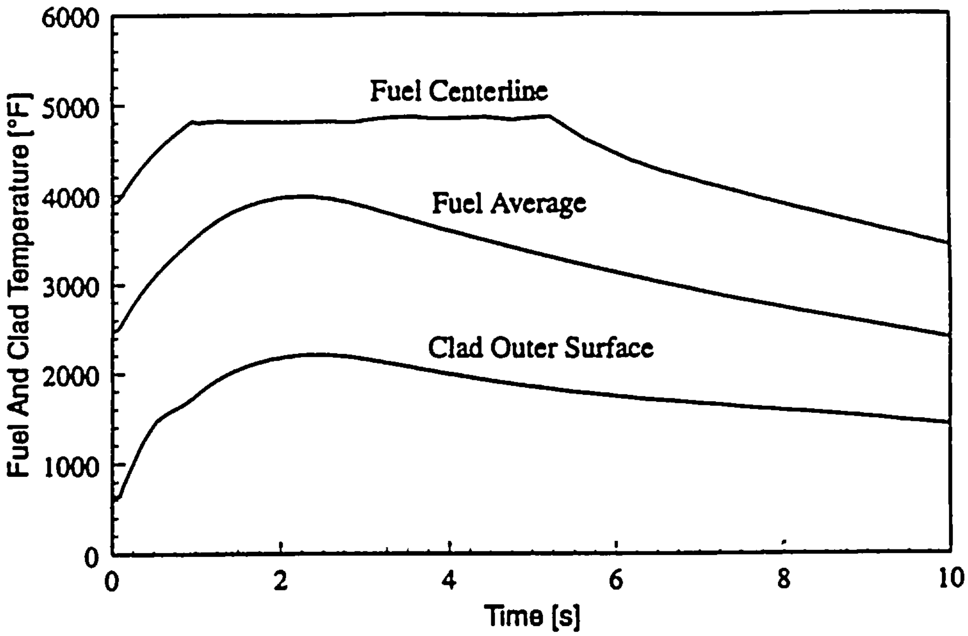
UNIT 1

FIGURE 3.3-56

Nuclear Power vs.Time For The Rod Ejection Event, Hot Full Power, End Of Life



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UNIT 1

FIGURE 3.3-57

Fuel Centerline, Fuel Average, and
Clad Outer Surface Temperature vs.Time For The Rod
Ejection Event, Hot Full Power, End Of Life

3.4 POST LOCA HYDROGEN PRODUCTION

As part of the Rerating Program, Westinghouse provided hydrogen generation rates and inventories inside containment from the various sources; including core radiolysis, sump radiolysis, corrosion-generated hydrogen and the zirconium/water reaction. A comparison of the key parameters and assumptions employed in these analyses were compared to the data provided for the SGTP Program. This comparison indicates that the maximum normal operation containment temperature of 120° F (maximum) remains unchanged. Since a containment temperature of 120° F prior to the accident is the bases for the current analysis of record, the Rerating Program results remain applicable. However, the post-accident time-temperature profile inside containment and the fraction of the core that undergoes a zirc-water reaction resulting from this analysis have been reviewed in order to ensure that the values employed in the Rerating Program analysis remain bounding for the SGTP Program. This review has been completed with the following conclusions:

1. The analysis for the Rerating Program considered zirconium-water reactions of 1.5%, 3.0%, and 5.0%. The limiting PCT calculations for the small break LOCA occurring within the pressurizer doghouse shows that only 0.128% of the core clad is oxidized. Applying the 10CFR50.44(d)(1) factor of 5 increase results in a zirc-water reaction percentage of 0.06%. The analyses remain highly conservative, since the value is less than the minimum value considered in the calculations (i.e., 1.5%). Also, the updated values do not contradict the UFSAR statement that "..zirconium-water reaction is calculated to be a maximum of 0.1% by weight of the total quantity of zirconium in the core." (UFSAR, page 14.3.6-2 dated July, 1982) For the large break LOCA, the zirc-water reaction is 4.93%. Since this value is less than the value that was considered in the Rerating Program (i.e., 5%), the results from the analysis remain applicable.

2. The impact of the revised post-LOCA temperatures on the post-accident hydrogen generation has been reviewed and found to be negligible.

3. Another consideration is the current inventory of corrodible materials inside containment. AEPSC provided the changes to the inventory of corrodible materials inside containment and confirmed that UFSAR Tables 14.3.6-3 and 14.3.6-7 remain valid. Therefore, the Rerating Program analysis remains bounding for the SGTP Program.

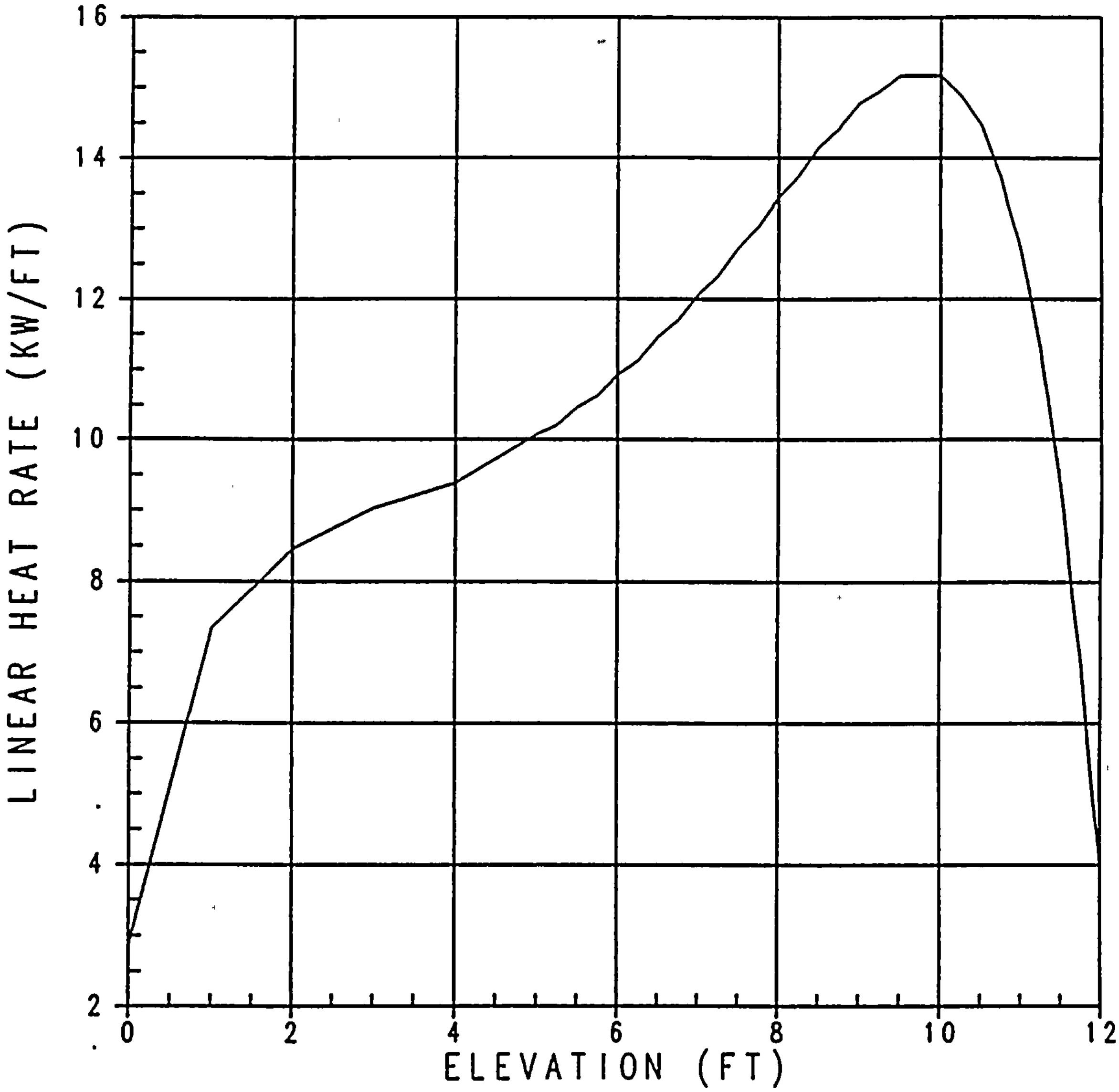


Figure 3.1-86 Hot Rod Power Distribution (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Unit 1

3.5 CONTAINMENT ANALYSES

3.5.1 Short-Term Containment Analysis

The short term containment integrity analysis is used to verify the adequacy of interior structures and walls by demonstrating that calculated differential pressures are less than design limits. The functionality of the ice condenser is demonstrated and containment integrity is also verified. The efforts performed for the short term containment analysis, applicable to the Pressurizer Enclosure, as described in Section 3.4.1 of WCAP-11902, support operation of Cook Nuclear Plant Units 1 & 2 over the full range of rerated parameters. The major impacts on LOCA short term mass and energy release rate calculations and containment subcompartment response analysis, are the effects due to RCS temperature changes. For the steam generator enclosure, mass and energy releases and the subsequent containment response are performed at zero power, which maximizes effects because steam pressure is maximum. All relevant analyses and evaluations performed for the Rerating Program assumed values which would bound both Units 1 & 2 at the rerated power levels and revised temperatures and pressures described in Table 2.1-1 of WCAP-11902, Supplement 1. The results of the short term containment analyses and evaluations for the SGTP Program demonstrate that, for the pressurizer enclosure, the fan accumulator room and the steam generator enclosure, the resulting peak pressures remain below the allowable design peak pressures. Since the calculated pressures in WCAP-11902, Supplement 1 for the loop compartments exceeded the design pressure, demonstration of structural adequacy was required. This issue was addressed by AEPSC and is documented in UFSAR Section 14.3.4.2.3.4.

3.5.2 Loss-of-Coolant Mass and Energy Release

3.5.2.1 Purpose

The purpose of this analysis was to calculate the long term LOCA mass and energy releases with the proposed revised plant conditions and increased operating margins. The increased operating margins include increased EDG start time to 30 seconds and the revised RHR and HHSI pump flow rates.

This section provides the analytical basis with respect to the LOCA containment mass and energy release for the operation of the Donald C. Cook Nuclear Plant Units 1 and 2 at the SGTP Program conditions. This containment integrity analyses bounds both units.

Rupture of any of the piping carrying pressurized high temperature reactor coolant, termed a LOCA, will result in release of steam and water into the containment. This, in turn, will result in an increase in the containment pressure and temperature. The mass and energy release rates described in this document form the basis of further computations to evaluate the structural integrity of the containment following a postulated accident to satisfy the Nuclear

Regulatory acceptance criteria, General Design Criterion 38, which is more restrictive than the GDC criteria in Appendix H of the original FSAR, to which the Donald C. Cook Nuclear Plants are licensed. Section 3.5.3 presents the long term containment integrity analysis for containment pressurization evaluations.

3.5.2.2 System Characteristics and Modeling Assumptions

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Some of the most critical items are the RCS initial conditions, core decay heat, safety injection flow, and metal and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed next. Tables 3.5-1 and 3.5-2 present key data assumed in the analysis.

For the long term mass and energy release calculations, operating temperatures to bound the highest average coolant temperature range were used as bounding analysis conditions. The modeled core rated power of 3413 MWt adjusted for calorimetric error (+2 percent of power) was the basis in the analysis. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures which are at the maximum levels attained in steady state operation. Additionally, an allowance of +5.1°F is reflected in the temperatures in order to account for instrument error and deadband. The initial RCS pressure in this analysis is based on a nominal value of 2250 psia. Also included is an allowance of +67 psi, which accounts for the measurement uncertainty on pressurizer pressure. The selection of 2250 psia as the limiting pressure is considered to affect the blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2317 psia initial pressure was selected as the limiting case for the long term mass and energy release calculations. These assumptions conservatively maximize the mass and energy in the RCS.

The selection of the fuel design features for the long term mass and energy calculation is based on the need to conservatively maximize the core stored energy. The margin in core stored energy was chosen to be +15 percent. Thus, the analysis very conservatively accounts for the stored energy in the core. The fuel conditions were adjusted to provide a bounding analysis for current Cook Nuclear Plant Units 1 and 2 fuel design features. The following items serve as the basis to ensure conservatism in the core stored energy calculation: time of maximum fuel densification and highest BOL temperatures.

Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion and 1.4% for uncertainty) is modeled.

Regarding safety injection flow, the mass and energy calculation considered the limiting scenario of minimum safety injection flow, with the RHR crosstie valve is assumed to be closed, in conjunction with a 15% pump head degradation for the RHR and SI pumps and 10% pump head degradation for the charging pumps. This configuration conservatively bounds other respective alignments. Closure of the RHR crosstie was considered over the HHSI crosstie because this would have a more severe impact on the analysis (i.e., closure of the RHR crosstie would bound closure of the HHSI crosstie). This results in the conservative minimum safety injection flowrate used.

The following assumptions were employed to ensure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment:

1. Maximum expected operating temperature of the RCS (100% full power conditions)

2. An allowance in temperature for instrument error and dead band (+5.1°F)

3. Margin in volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty)

4. Core rated power of 3413 MWt

5. Allowance for calorimetric error (+2 percent of power)

6. Conservative coefficient of heat transfer (i.e., steam generator primary/secondary heat transfer and RCS metal heat transfer)

7. Allowance in core store energy for effect of fuel densification

8. A margin in core stored energy (+15 percent included to account for manufacturing tolerances)

9. An allowance for RCS initial pressure uncertainty (+67 psi)

10. Steam generator tube plugging leveling (0% uniform)

    - Maximizes reactor coolant volume and fluid release

    - Maximizes heat transfer area across the SG tubes

    - Reduces coolant loop resistance, which reduces delta-p upstream of break and increases break flow

Thus, based on the above conditions and assumptions, a bounding analysis of Cook Nuclear Plant Units 1 and 2 is made for the release of mass and energy from the RCS in the event of a LOCA to support the SGTP Program.

3.5.2.3 Long Term Mass and Energy Release Analysis

3.5.2.3.1 Introduction

The evaluation model used for the long term LOCA mass and energy release calculations was the March 1979 model described in Reference 1. This evaluation model has been reviewed and approved by the NRC, and has been used in the analysis of other ice condenser plants.

This report section presents the long term LOCA mass and energy releases that were generated in support of the SGTP Program. These mass and energy releases are then subsequently used in the LOTIC-1 containment integrity analysis peak pressure calculation.

3.5.2.3.2 LOCA Mass and Energy Release Phases

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA mass and energy analysis, is typically divided into four phases:

1. Blowdown - the period of time from accident initiation (when the reactor is at steady state operation) to the time that the RCS and containment reach an equilibrium state at containment design pressure.

2. Refill - the period of time when the lower plenum is being filled by accumulator and ECCS water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

3. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-reflood (Froth) - describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes two phase.

3.5.2.3.3 Computer Codes

The Reference 1 mass and energy release evaluation model is comprised of mass and energy release versions of the following codes: SATAN VI, WREFLOOD, and FROTH. These codes were used to calculate the long term LOCA mass and energy releases for Cook Nuclear Plant Units 1 and 2.

SATAN calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, mass and energy flowrates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the Emergency Core Cooling refills the reactor vessel and provides cooling to the core. The most important feature is the steam/water mixing model (See Section 3.5.2.6.2).

FROTH models the post-reflood portion of the transient. The FROTH code is used for the steam generator heat addition calculation from the broken and intact loop steam generators.

3.5.2.4 Break Size and Location

Generic studies have been performed with respect to the effect of postulated break size on the LOCA mass and energy releases. The double ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the RCS loop can be postulated for pipe rupture:

1. Hot leg (between vessel and steam generator)
2. Cold leg (between pump and vessel)
3. Pump suction (between steam generator and pump)

The break location analyzed for the SGTP Program is the pump suction double ended rupture, DEPS (10.48 ft^2). Break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for each case analyzed. The following information provides a discussion on each break location.

The hot leg double ended rupture has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary is minimal because the majority of the fluid which exits the core bypasses the steam generators venting directly to containment. As a result, the reflood mass and energy

releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break. For the hot leg break, generic studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure would continually decrease). The mass and energy releases for the hot leg break have not been included in the scope of this containment integrity analysis because for the hot leg break only the blowdown phase of the transient is of any significance. Since there are no reflood and post-reflood phases to consider, the limiting peak pressure calculated would be the compression peak pressure and not the peak pressure following ice bed meltout.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment energy releases. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold leg break is not included in the scope of the SGTP Program.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the RCS in calculating the releases to containment. This break location has been determined to be the limiting break for all ice condenser plants.

In summary, the analysis of the limiting break location for an ice condenser containment has been performed and is shown in this report. The double-ended pump suction (DEPS) guillotine break has historically been considered to be the limiting break location, by virtue of its consideration of all energy sources present in the RCS. This break location provides a mechanism for the release of the available energy in the RCS, including both the broken and intact loop steam generators.

3.5.2.5 Application of Single Failure Criteria

An analysis of the effects of the single failure criteria has been performed on the mass and energy release rates for the DEPS break. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety injection system. This is not an issue for the blowdown period which is limited by the compression peak pressure.

The limiting minimum safety injection case has been analyzed for the effects of a single failure. In the case of minimum safeguards, the single failure postulated to occur is the loss of an emergency diesel generator. This results in the loss of one pumped safety injection train,

thereby minimizing the safety injection flow. An additional conservatism has been included in this analysis in that the closure of the RHR crosstie valve has been considered because it results in a further reduction in safety injection flow. The analysis further considers the RHR and SI pump head curves to be degraded by 15% and the charging pump head curve to be degraded by 10%. This results in the greatest SI flow reduction for the minimum safeguards case.

3.5.2.6 Mass and Energy Release Data

3.5.2.6.1 Blowdown Mass and Energy Release Data

A version of the SATAN-VI code is used for computing the blowdown transient, which is the code used for the ECCS calculation in Reference 2.

The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in Reference 1.

Table 3.5-3 presents the calculated mass and energy releases for the blowdown phase of the DEPS break. For the pump suction breaks, break path 1 in the mass and energy release tables refers to the mass and energy exiting from the steam generator side of the break; and, break path 2 refers to the mass and energy exiting from the pump side of the break.

3.5.2.6.2 Reflood Mass and Energy Release Data

The WREFLOOD code used for computing the reflood transient, is a modified version of that used in the 1981 ECCS evaluation model (Reference 2).

The WREFLOOD code consists of two basic hydraulic models - one for the contents of the reactor vessel, and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator release are included as auxiliary equations which interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths

available for discharging steam and entrained water from the core to the break; i.e. the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and ECCS injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the Reference 1 mass and energy release evaluation model in the Rerating Program analyses. Even though the Reference 1 model credits steam/mixing only in the intact loop and not in the broken loop, justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 3). This assumption is justified and supported by test data, and is summarized as follows:

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two phase interaction with condensation of steam by cold ECCS water. The second is a single phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data is that generated in 1/3 scale tests (Reference 4), which are the largest scale data available and thus most clearly simulates the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

From the entire series of 1/3 scale tests, a group corresponds almost directly to containment integrity reflood conditions. The injection flowrates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 1. For all of these tests, the data clearly indicate the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is therefore wholly supported by the 1/3 scale steam/water mixing data.

Additionally, the following justification is also noted. The post-blowdown limiting break for the containment integrity peak pressure analysis is the pump suction double ended rupture break. For this break, there are two flowpaths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam which is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECCS injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam which is condensed that is taken credit for in this

analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECCS injection nozzle. A description of the test and test results is contained in References 1 and 4.

Table 3.5-4 presents the calculated mass and energy release for the reflood phase of the pump suction double ended rupture with minimum safety injection.

The transients of the principal parameters during reflood are provided in Table 3.5-5.

3.5.2.6.3 Post-Reflood Mass and Energy Release Data

The FROTH code (Reference 5) is used for computing the post-reflood transient.

The FROTH code calculates the heat release rates resulting from a two-phase mixture level present in the steam generator tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken loop and intact loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure, but the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two phase fluid exits the core, flows through the hot legs and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two phase. The methodology for the use of this model is described in Reference 1.

After containment depressurization, the mass and energy release available to containment is generated directly from core boiloff/decay heat.

Table 3.5-6 presents the two phase post-reflood (froth) mass and energy release data for the pump suction double ended case.

3.5.2.7 Decay Heat

On November 2, 1978 the Nuclear Power Plant Standards Committee (NUPPSCO) of the American Nuclear Society (ANS) approved ANS standard 5.1 for the determination of decay heat. This standard was used in the mass and energy release model with the following input:

Significant assumptions in the generation of the decay heat curve:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and N_p-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Table 10 of Reference 6.

5. Operation time before shutdown is 3 years.

6. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

7. Two sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

3.5.2.8 Steam Generator Equilibration and Depressurization

Steam generator equilibration and depressurization is the process by which secondary side energy is removed from the steam generators in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is the saturation temperature (T_{sat}) at the containment design pressure. After the FROTH calculations, steam generator secondary energy is removed based on first and second stage rates. The first stage rate is applied until the steam generator reaches T_{sat} at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment pressure. Then, the second stage rate is used until the final depressurization. The heat removal of the broken loop and intact loop steam generators are calculated separately.

During the FROTH calculations, steam generator heat removal rates are calculated using the secondary side temperature, primary side temperature and a secondary side heat transfer coefficient determined using a modified McAdam's correlation (Reference 7). Steam generator energy is removed during the FROTH transient until the secondary side temperature reaches saturation temperature at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The SG energy available to be released during the first stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user specified intermediate equilibration pressure, assuming saturated conditions. This energy is then divided by the first stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second stage rate. The second stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization.

3.5.2.9 Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy release analysis are given in Table 3.5-7. These sources are the RCS, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in Table 3.5-8. The energy sources include:

1. RCS Water
2. Accumulator Water
3. Pumped Injection Water
4. Decay Heat
5. Core Stored Energy
6. RCS Metal - Primary Metal (includes SG tubes)
7. Steam Generator Metal (includes transition cone, shell, wrapper, and other internals)
8. Steam Generator Secondary Energy (includes fluid mass and steam mass)
9. Secondary Transfer of Energy (feedwater into and steam out of the steam generator secondary)

Energy Reference Points:

Available Energy: 212°F; 14.7 psia
Total Energy Content: 32°F; 14.7 psia

It should be noted that the inconsistency in the energy balance tables from the end of Reflood to 3600 seconds, i.e., "Total Available" data versus "Total Accountable", resulted from the omission of the reactor upper head in the analysis following blowdown. It has been concluded that the results are more conservative when the upper head is neglected. This does not affect the instantaneous mass and energy releases, or the integrated values, but causes an increase in the total accountable energy within the energy balance table.

The mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)
2. End of blowdown time
3. End of refill time
4. End of reflood time
5. Time of broken loop steam generator equilibration to pressure setpoint
6. Time of intact loop steam generator equilibration to pressure setpoint

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the clad temperature did not rise high enough for the rate of the Zirc-water reaction heat to be of any significance.

The consideration of the various energy sources in the mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Although Cook Nuclear Plant Unit 1 is not a Standard Review Plan Plant, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied.

3.5.2.10 References

1. "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version", WCAP-10325-P-A, May 1983 (Proprietary), WCAP-10326-A (Non-Proprietary).

2. "Westinghouse ECCS Evaluation Model - 1981 Version", WCAP-9220-P-A, Rev. 1, February 1982 (Proprietary), WCAP-9221-A, Rev.1 (Non-Proprietary)

3. Docket No. 50-315, "Amendment No. 126, Facility Operating License No. DPR-58 (TAC No. 7106), for D.C. Cook Nuclear Plant Unit 1", June 9, 1989.

4. EPRI 294-2, Mixing of Emergency Core Cooling Water with Steam; 1/3 Scale Test and Summary, (WCAP-8423), Final Report June 1975.

5. "Westinghouse Mass and Energy Release Data For Containment Design", WCAP-8264-P-A, Rev. 1, August 1975 (Proprietary), WCAP-8312-A (Non-Proprietary).

6. ANSI/ANS-5.1 1979, American National Standard for Decay Heat Power in Light Water Reactors", August 1979.

7. W. H. McAdam, Heat Transmission , McGraw-Hill 3rd edition, 1954, p.172.

3.5.3 LOCA Containment Integrity Analysis

3.5.3.1 Description of LOTIC-1 Model

Early in the ice condenser development program, it was recognized that there was a need for modeling of long term ice condenser performance. It was realized that the model would have to have capabilities comparable to those of the dry containment (COCO) model. These capabilities would permit the model to be used to solve problems of containment design and optimize the containment and safeguards systems. This has been accomplished in the development of the LOTIC code, described in Reference 1.

The model of the containment consists of five distinct control volumes, the upper compartment, the lower compartment, the portion of the ice bed from which the ice has melted, the portion of the ice bed containing unmelted ice, and the dead ended compartment. The ice condenser control volume with unmelted and melted ice is further subdivided into six subcompartments to allow for maldistribution of break flow to the ice bed.

The conditions in these compartments are obtained as a function of time by the use of fundamental equations solved through numerical techniques. These equations are solved for three phases in time. Each phase corresponds to a distinct physical characteristic of the problem. Each of these phases has a unique set of simplifying assumptions based on test results from the ice condenser test facility. These phases are the blowdown period, the depressurization period, and the long term.

The most significant simplification of the problem is the assumption that the total pressure in the containment is uniform. This assumption is justified by the fact that after the initial blowdown of the RCS, the remaining mass and energy released from this system into the containment are small and very slowly changing. The resulting flow rates between the control volumes will also be relatively small. These flow rates then are unable to maintain significant pressure differences between the compartments.

In the control volumes, which are always assumed to be saturated, steam and air are assumed to be uniformly mixed and at the control volume temperature. The air is considered a perfect gas, and the thermodynamic properties of steam are taken from the ASME steam table.

The condensation of steam is assumed to take place in a condensing node located, for the purpose of calculation, between the two control volumes in the ice storage compartment. The exit temperature of the air leaving this node is set equal to a specific value which is equal to the temperature of the ice filled control volume of the ice storage compartment. Lower compartment exit temperature is used if the ice bed section is melted.

3.5.3.2 Containment Pressure Calculation

The following are the major input assumptions used in the LOTIC analysis for the pump suction pipe rupture case with the steam generators considered as an active heat source for the Cook Nuclear Plant Containment:

1. Minimum safeguards are employed in all calculations, e.g., one of two spray pumps and one of two spray heat exchangers; one of two RHR pumps and one of two RHR heat exchangers providing flow to the core; one of two safety injection pumps and one of two centrifugal charging pumps; and one of two air retum fans.

2. 2.11×10^6 lbs. of ice initially in the ice condenser.

3. The blowdown, reflood, and post reflood mass and energy releases described in Section 3.5.2 are used.

4. Blowdown and post-blowdown ice condenser drain temperatures of 190°F and 130°F are used, respectively.

5. Nitrogen from the accumulators in the amount of 4510 lbs. is included in the calculations.

6. Essential service water temperature of 87.5°F is used on the spray heat exchanger and the component cooling heat exchanger.

7. The air retum fan is effective 10 minutes after the transient is initiated.

8. No maldistribution of steam flow to the ice bed is assumed. (This assumption is conservative since it contributes to early ice bed melt out time.)

9. No ice condenser bypass is assumed. (This assumption depletes the ice in the shortest time and is thus conservative.)

10. The initial conditions in the containment are a temperature of 57°F in the upper compartment volume, and 60°F in the lower and dead-ended compartment volumes. All volumes are at a pressure of 0.3 psig.

11. Containment structural heat sinks are assumed with conservatively low heat transfer rates. (See Tables 3.5-11 and 3.5-12)

12. The operation of one containment spray heat exchanger ($UA = 3.107 \times 10^6$ Btu/hr-°F), for containment cooling and the operation of one RHR heat exchanger ($UA = 2.22 \times 10^6$ Btu/hr-°F) for core cooling. The component cooling heat exchanger was modeled at 3.58×10^6 Btu/hr-°F.

13. The air return fan returns air at a rate of 39,000 cfm from the upper to the lower compartment.

14. An active sump volume of 40,600 ft^3 is used.

15. 102% of 3413 MWt power is used in the calculations.

16. Subcooling of ECCS water from the RHR heat exchanger is assumed.

17. Essential service water flow to the containment spray heat exchanger was modeled as 2000 gpm. Also the nuclear service water flow to the component cooling heat exchanger was modeled as 5000 gpm.

18. RHR Spray initiation is assumed after switchover from injection to recirculation has been completed and containment pressure is greater than or equal to 8 psig.

3.5.3.3 Structural Heat Removal

Provision is made in the containment pressure analysis for heat storage in interior and exterior walls. Each wall is divided into a number of nodes. For each node, a conservation of energy equation expressed in finite difference forms accounts for transient conduction into and out of the containment structural heat sinks used in the analysis. The material property data used is found in Tables 3.5-11 and 3.5-12.

The heat transfer coefficient to the containment structure is based primarily on the work of Tagami (Reference 2). When applying the Tagami correlations, a conservative limit was placed on the lower compartment stagnant heat transfer coefficients. They were limited to a steam-air ratio of 1.4 according to the Tagami correlation. The imposition of this limitation is to restrict the use of the Tagami correlation within the test range of steam-air ratios where the correlation was derived.

With these assumptions, the heat removal capability of the containment is sufficient to absorb the energy releases and still keep the maximum calculated pressure below the design pressure.

3.5.3.4 Analysis Results

The results of the analysis shows that the maximum calculated containment pressure is 11.49 psig for the DEPS minimum safeguards break case. This pressure peak occurs at approximately 7752 seconds, with ice bed meltout at approximately 5423 seconds.

The following plots show the containment integrity transient, as calculated by the LOTIC-1 code.

Figure 3.5-1: Containment Pressure Transient
Figure 3.5-2: Upper Compartment Temperature Transient
Figure 3.5-3: Lower Compartment Temperature Transient
Figure 3.5-4: Active and Inactive Sump Temperature Transient
Figure 3.5-5: Ice Melt Transient

Tables 3.5-9 and 3.5-10 give energy accountings at various points in the transient.

3.5.3.5 Relevant Acceptance Criteria

The LOCA mass and energy analysis has been performed in accordance with the criteria shown in the Standard Review Plan (SRP) Section 6.2.1.3. In this analysis, the relevant requirements of General Design Criteria (GDC) 50 and 10 CFR Part 50 Appendix K have been included by confirmation that the calculated pressure is less than the design pressure, and because all available sources of energy have been included, which is more restrictive than the GDC criteria in Appendix H of the original FSAR, to which the Donald C. Cook Nuclear Plants are licensed. These sources include reactor power, decay heat, core stored energy, energy stored in the reactor vessel and internals, metal-water reaction energy, and stored energy in the secondary system.

The containment integrity peak pressure analysis has been performed in accordance with the criteria shown in the SRP Section 6.2.1.1.b for ice condenser containments. Conformance to GDC's 16, 38, and 50 is demonstrated by showing that the containment design pressure is not exceeded at any time in the transient. This analysis also demonstrates that the containment heat removal systems function to rapidly reduce the containment pressure and temperature in the event of a LOCA.

3.5.3.6 Conclusions

Based upon the information presented, it may be concluded that operation with the revised plant conditions and increased operating margins for Donald C. Cook Nuclear Plant is acceptable. Operation with the RHR crosstie valve closed was also shown to be more limiting than operation with the valve open since there is less safety injection water available for steam condensation. Operation with the revised plant conditions, increased operating margins

and the RHR crosstie valve closed results in a calculated peak containment pressure of 11.49 psig, as compared to the design pressure of 12.0 psig. Thus, the most limiting case has been considered, and has been demonstrated to yield acceptable results.

3.5.3.7 References

1. "Long Term Ice Condenser Containment Code - LOTIC Code", WCAP-8354-P-A, April 1976 (Proprietary), WCAP-8355-A (Non-Proprietary).

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

3.5.4 Steamline Break Mass/Energy Releases Inside Containment

The mass/energy releases for the inside containment analysis of record at the time the 30% SGTP Program was underway is based on the Rerating Program, which was performed to bound both units. The calculation of the mass/energy releases following a steamline break is described in the Cook Nuclear Plant Unit 1 UFSAR Section 14.3.4.4.

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steamline rupture is dependent upon the many possible configurations of the plant steam system and containment designs as well as the plant operating conditions and the size of the rupture. These variations make it difficult to reasonably determine the single "worst case" for both containment pressure and temperature evaluations following a steambreak. The analysis performed as part of the Rerating Program determined that the limiting scenario of the steambreak cases analyzed for the containment response evaluation were a break size of 0.86 ft^2 occurring at full power for the split rupture scenario and a break size of 4.6 ft^2 occurring at full power for the double-ended rupture scenario.

As part of the 30% SGTP Program, several of the limiting cases of the steamline break mass and energy release calculations inside containment have been reanalyzed to assess a longer feedwater motor operated (FMO) valve stroke time, larger unisolatable feedline and steamline volumes, and revised maximum AFW flow rates. A relaxation in the EDG start-up time from 10 to 30 seconds, and an increase in the upper containment and lower compartment spray delay time are also addressed. However, these latter analysis assumption changes only affect the containment response analysis.

It should be noted that the changes associated only with the SGTP Program for Unit 1, i.e., RCS flow reduction, reduced primary-to-secondary heat transfer capability, and reduction in the rated thermal power, are less limiting parameters relative to the assumptions currently made for the mass/energy release calculations following a steamline break inside

containment. The parameter changes associated with the SGTP Program do not warrant reanalysis of this event. However, evaluations are currently in place (References 1 and 2) to address several non-conservative assumptions in the analysis. A reanalysis effort was undertaken for the steamline break mass and energy releases inside containment as part of the SGTP Program, so that the Reference 1 and 2 evaluations will no longer be required and to support the increased operating margins included in the discussion of the previous paragraph.

This section discusses a series of steamline breaks, consistent with the cases presented in the UFSAR, which were analyzed to determine the mass and energy releases from a variety of postulated pipe breaks encompassing wide variations in plant operation, safety system performance, and break sizes. The mass and energy release data is subsequently used as input to the containment integrity analysis discussed in Section 3.5.5.

3.5.4.1 Method of Analysis

The LOFTRAN computer code (Reference 3) was used to calculate the break flows and enthalpy of the releases through the steamline break as a function of time. Blowdown mass/energy releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant thick metal heat storage, and reverse steam generator heat transfer.

A bounding analysis was performed to address the range of conditions possible for Unit 1 operation and the potential Unit 2 uprating. The assumptions on the initial conditions are taken to maximize the mass and total energy released. The higher primary temperatures along with the higher uprated power level associated with the Unit 2 uprating parameters are conservative for the mass/energy release calculations. The upper bound temperature of Table S-2.1-1, Case 8 of WCAP-11902, Supplement (Reference 4), was used. Since the mass blowdown rate is dependent on steam pressure, and the steam pressure is less for the lower bound temperature case, the steam pressure of the upper bound temperature case is limiting for the range of operating conditions possible for the uprating of Unit 2.

The functions which actuate safety injection and steamline isolation during a steamline rupture event are commonly referred to as the Steamline Break Protection System. A plant's steamline break protection system design can have a large effect on steamline break results. The current steamline break protection system designs for Unit 1 and Unit 2 are different. The current Unit 1 design is referred to as an "OLD" steamline break protection system design. The Unit 2 design (and the proposed modified Unit 1 design; see Section 3.3.2.5) is referred to as a "HYBRID" steamline break protection system design. The two systems have the following characteristics:

Current Unit 1 - "OLD" Steamline Break Protection

Safety Injection Signals

1. High steam flow coincident with low steamline pressure (two out of four lines)

2. High steam flow coincident with low-low T_{avg} (two out of four lines)

3. Two out of three differential pressure signals between a steam line and the remaining steam lines

4. Two out of three low pressurizer pressure signals

5. Two out of three high containment pressure signals

Steamline Isolation Signals

1. High steam flow coincident with low steamline pressure (two out of four lines)
2. High steam flow coincident with low-low T_{avg} (two out of four lines)
3. Two out of four high-high containment pressure signals

Unit 2 - "HYBRID" Steamline Break Protection

Safety Injection Signals

1. Low steamline pressure (two out of four lines)

2. Two out of three differential pressure signals between a steam line and the remaining steam lines

3. Two out of three low pressurizer pressure signals

4. Two out of three high containment pressure signals

Steamline Isolation Signals

1. Low steamline pressure (two out of four lines)
2. High steam flow coincident with low-low T_{avg} (two out of four lines)
3. Two out of four high-high containment pressure signals

The only differences between the current Unit 1 and Unit 2 steamline break protection logic designs are the actuations from a high steam flow and low-low T_{avg} signal and the logic associated with the low steamline pressure signal required to actuate safety injection and

steamline isolation. Currently, for Unit 1, a high steam flow coincident with low-low T_{avg} signal actuates both safety injection and steamline isolation. For Unit 2, a high steam flow coincident with low-low T_{avg} signal actuates only steamline isolation. However, the difference is not significant for the calculation of the mass/energy releases since the analysis does not take credit for any ESF actuations on a high steam flow coincident with low-low T_{avg} signal.

The current Unit 1 design requires a coincidence between the low steamline pressure and high steam flow for protection actuation. The Unit 2 design only requires the low steamline pressure signal for protection actuation; no coincidence with steam flow is required.

The coincidence logic required for safety injection initiation and steamline isolation on high steam flow and low steam pressure for the current Unit 1 design is more limiting for the calculation of mass/energy releases inside containment than Unit 2's design. Actuation of safety injection and steamline isolation will limit the mass/energy released to the containment. Delaying the safeguards initiation will result in a conservative calculation of the mass/energy releases for the containment pressure and temperature evaluation. The coincidence requirement for high steam flow with low steam pressure of the current Unit 1 design increases the likelihood that safeguards initiation might be delayed compared to Unit 2's design where only a low steam pressure signal is required. In the case where the coincidence logic prohibits safety injection and steamline isolation on high steam flow with low steam pressure, one of the other signals must be received before the safeguards are initiated. As such, the current Unit 1 steamline break protection system design was assumed in this bounding analysis for the calculation of the mass/energy releases inside containment.

3.5.4.2 Assumptions

Several steamline breaks were analyzed to determine a limiting break condition for the containment temperature and pressure response. The following assumptions were used in the analysis:

a. Double-ended pipe breaks were assumed to occur at the nozzle of one steam generator and also downstream of the flow restrictor. (Since neither Unit 1 nor Unit 2 has integral flow restrictors.) Split ruptures were assumed to occur at the nozzle of one steam generator.

b. The blowdown is assumed to be dry saturated steam.

c. As discussed previously, the Unit 1 steamline break protection system design is assumed. However, credit was not taken for safeguards actuation on high steam line differential pressure or high steam flow coincident with low-low T_{avg}.

d. Steamline isolation is assumed complete 11 seconds after the setpoint is reached for either high steam flow coincident with low steam pressure or

high-high containment pressure. The isolation time allows 8 seconds for valve closure plus 3 seconds for electronic delays and signal processing.

e. As part of the Rerating Program, 4.6 ft^2 and 1.4 ft^2 double-ended pipe breaks were analyzed at 102, 70, 30, and zero percent power levels.

For the SGTP Program, a sub-set of these double-ended pipe breaks have been re-analyzed. This sub-set, listed below, corresponds to the most limiting double-ended pipe breaks, as determined during the Rerating Program effort.

4.6 ft^2, 102% power, with an MSIV failure;
4.6 ft^2, 70% power, with an MSIV failure;
1.4 ft^2, 102% power, with an MSIV failure;
1.4 ft^2, 70% power, with an MSIV failure.

f. As part of the Rerating Program, split pipe ruptures were analyzed at 0.86 ft^2, 102% power; 0.908 ft^2, 70% power; 0.942 ft^2, 30% power; and 0.4 ft^2, hot shutdown.

These split break sizes for each power level were modeled because they reflect the largest breaks for which ESF actuations (i.e., steamline isolation, feedwater isolation, and safety injection) must be generated by high containment pressure trips. The high steam flow coincident with low steam pressure is not reached for these break sizes or smaller break sizes at the respective power levels (Reference 5).

For the SGTP Program, a sub-set of these split breaks have been re-analyzed. This sub-set, listed below, corresponds to the most limiting split break sizes, as determined during the Rerating Program effort.

0.86 ft^2, 102% power,with an auxiliary feedwater runout protection (AFWRP) failure;

0.942 ft^2, 30% power, with an AFWRP failure;

0.942 ft^2, at 30% power, with an MSIV failure.

g. Failure of an MSIV, failure of a feedwater isolation valve or main feed pump trip, and failure of auxiliary feedwater runout control were considered. As part of the Rerating Program two cases for each break size and power level scenario were analyzed with one case modeling the MSIV failure and the other case modeling the AFW runout control failure. Each case assumed conservative main feedwater addition to bound the feedwater isolation valve or

main feed pump trip failure. For the SGTP Program, the sub-set of case re-analyzed assumed a failure as previously noted (items e. and f.). The case re-analyzed for the SGTP Program also assumed. Feedline isolation via FMO valves complete 44 seconds after the setpoint is reached for either high steam flow coincident with low steam pressure or high containment pressure. The isolation time allows 41 seconds for valve closure plus 3 seconds for electronic delays and signal processing.

h. The auxiliary feedwater system is manually re-aligned by the operator after 10 minutes.

i. A shutdown margin of 1.3% Δk/k is assumed.

j. A moderator density coefficient of 0.54 Δk/gm/cc is assumed.

k. Minimum capability for injection of boric acid (2400 ppm) solution corresponding to the most restrictive single failure in the safety injection system. The ECCS consists of the following systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, 3) the high head (intermediate head) safety injection system, and 4) the charging safety injection system. Only the charging safety injection system and the passive accumulators are modeled for the steam line break accident analysis.

The modeling of the safety injection system in LOFTRAN is described in Reference 3. Figure 3.3-49 of this report presents the safety injection flow rates as a function of RCS pressure assumed in the analysis. The flow corresponds to that delivered by one charging pump delivering its full flow to the cold legs. The safety injection flows assumed in this analysis take into account the 10% degradation of the charging pump performance. No credit has been taken for any borated water that might exist in the injection lines, which must be swept from the lines downstream of the boron injection tank isolation valves prior to the delivery of boric acid to the RCS loops. For this analysis, a boron concentration of 0 ppm for the boron injection tank is assumed.

After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the safety injection charging pump starts. In 27 seconds, the valves are assumed to be in their final position (VCT charging pump suction valve has closed following opening of RWST charging pump suction valve) and the pump is assumed to be at full speed and to draw suction from the RWST. The volume containing the low concentration borated water is swept into the core before the 2400 ppm borated water reaches the core. This delay, described above, is inherently included in the modeling. Note that the relaxed

EDG start-up time is not reflected in the steamline mass/energy releases, as conservative releases are obtained if offsite power is maintained (see item m. below).

l. For the at-power cases, reactor trip is available by safety injection signal, overpower protection signal (high neutron flux reactor trip or OPΔT reactor trip), and low pressurizer pressure reactor trip signal.

m. For RCP operation, offsite power is assumed available. Continued operation of the RCPs maximizes the energy transferred from the RCS to the steam generators.

n. No steam generator tube plugging is assumed to maximize the heat transfer characteristics.

3.5.4.3 Single Failure Effects

a. Failure of an MSIV increases the volume of steam piping which is not isolated from the break. When all valves operate, the piping volume capable of blowing down is located between the steam generator and the first isolation valve. If this valve fails, the volume between the break and upstream of the isolation valves in the other steamlines, including safety and relief valve headers and other connecting lines, will feed the break. For the cases which modeled a failure of an MSIV, the steamline volumes associated with Unit 1 were assumed since the volume available for blowdown for this scenario is greater than that for Unit 2. For the cases which did not model a failure of an MSIV, the steamline volumes associated with Unit 1 were assumed since the volume available for blowdown for this scenario is greater than that for Unit 2.

b. Failure of a diesel generator would result in the loss of one containment safeguards train resulting in minimum heat removal capability.

c. Failure of a feedwater isolation valve would result in additional inventory in the feedwater line which would not be isolated from the faulted steam generator. The mass in this volume can flash into steam and exit through the break. For consistency with the UFSAR steamline break mass/energy release analysis, all cases conservatively assumed failure of the feedwater isolation valve, which resulted in the additional inventory available for release through the steambreak and in higher than normal main feedwater flows.

d. Failure of the auxiliary feedwater runout control equipment would result in a higher auxiliary feedwater flow entering the faulted steam generator prior to re-alignment of the AFW system. For cases where the runout control operates properly, a bounding

constant AFW flow of 775 gpm to the faulted steam generator was assumed. This value was increased to 1375 gpm to simulate a failure of the runout control.

3.5.4.4 Results

The steamline break mass/energy releases inside containment were calculated to account for the range of conditions possible for the potential uprating of Unit 2. One set of mass/energy releases were calculated to bound both Units incorporating the limiting steamline break protection design currently installed in Unit 1.

Section 3.5.5 presents the containment integrity evaluation for a main steamline break using the mass/energy releases calculated here. As discussed in Section 3.5.5, the limiting scenarios of the steambreak cases analyzed for the containment response evaluation were a break size of 1.4 ft^2 occurring at 102% power with an MSIV failure for the double-ended rupture scenario and a break size of 0.942 ft^2 occurring at 30% power with an MSIV failure for the split rupture scenario. Table 3.5.13 presents the mass/energy releases for these limiting steambreak cases of the containment response evaluation.

3.5.4.5 References

1. "American Electric Power Service Corporation, Donald C. Cook Nuclear Power Plant Units 1 and 2, Increased Upper & Lower Compartment Spray Delivery Times," W Letter AEP-94-712, June 13, 1994.

2. "American Electric Power Service Corporation, Donald C. Cook Nuclear Power Plant Units 1 and 2, Feedwater Isolation Valve Evaluation Support," W Letter AEP-93-528, April 8, 1993.

3. Burnett, T, W. T., et al., "LOFTRAN Code Description," WCAP-7907-A, April 1, 1984.

4. "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 & 2 Licensing Report," WCAP-11902, Supplement, September 1989.

5. Land, R. E., "Mass and Energy Releases Following a Steam Line Rupture," WCAP-8860, September 1976.

3.5.5 Main Steam Line Break Containment Integrity

3.5.5.1 Introduction and Background

A series of main steam line split and double-ended breaks were analyzed as part of the Rerating Program for Cook Nuclear Plant Units 1 and 2 to determine the most severe break condition for containment temperature and pressure response for this design basis event. The analysis and evaluation conducted are discussed in Reference 1. The results from the Rerating Program, which are documented in the FSAR, show that the worst case of the double-ended breaks was a 4.6 square foot break, occurring at 102% power with a main steam isolation valve failure. The worst case for the split breaks was the 0.86 square foot break, occurring at 102% power, with the failure of auxiliary feedwater runout protection. The calculated peak containment temperature was 324.9°F and 324.4°F, respectively.

3.5.5.2 Purpose of Analysis

An analysis was performed as a part of the SGTP Program, to demonstrate that the peak containment temperature resulting from a design basis main steam line break will not exceed the equipment qualification criterion for the plant. The analysis was performed to bound Cook Nuclear Plant Units 1 and 2 operation under uprated conditions (3600 MWt NSSS). The containment pressure response generated for the LOCA double-ended pump suction break (Section 3.5.3) is calculated to be more severe, and therefore is not a concern here.

3.5.5.3 Major Analytical Assumptions

An analysis consistent with the Reference 1 analysis was performed. The analytical effort provides bounding calculations for both Units 1 & 2 at a power level of 3600 MWt.

A spectrum of the limiting split breaks from Reference 1 were analyzed:

> 0.86 ft^2, 102% power, with an AFWRP failure;
> 0.942 ft^2, 30% power, with an AFWRP failure;
> 0.942 ft^2, at 30% power, with an MSIV failure.

Also, the following double-ended breaks from Reference 1 were analyzed:

> 4.6 ft^2, 102% power, with an MSIV failure;
> 4.6 ft^2, 70% power, with an MSIV failure;
> 1.4 ft^2, 102% power, with an MSIV failure;
> 1.4 ft^2, 70% power, with an MSIV failure.

The mass and energy release to containment as a result of the postulated steam line break were calculated using the LOFTRAN computer code (Reference 2). Consistent with the

Reference 1 analysis, no credit was taken for entrainment. Section 3.5.4 presents additional details regarding the calculation of the inside containment steam line break mass and energy releases.

The consequences of these releases; in particular the peak containment temperature, was calculated using the LOTIC-3 computer code (Reference 3).

The following are the major input assumptions used in LOTIC-3:

1. The containment integrity calculations were performed with an additional failure of one of the containment safeguards trains, e.g., one of two spray pumps, which results in the minimum spray flow and one of two air return fans. Where applicable, plant data consistent with the LOCA containment integrity analysis (Section 3.5.3) was used.

2. The total initial ice mass used is 2.11×10^6 lbs.

3. The initial conditions in the containment are a temperature of 120°F in the lower and dead-ended compartments, a temperature of 27°F in the ice condenser, and a temperature of 57°F in the upper compartment. All volumes are at a pressure of 0.3 psig and a relative humidity of 15%.

4. The RWST temperature was assumed to be 105°F.

5. A containment spray pump flow of 2075 gpm to the upper compartment and 1006 gpm to the lower compartment was assumed.

6. Containment spray response time following high-high containment pressure setpoint is 115 seconds.

7. The mass and energy releases are given in Section 3.5.4.

3.5.5.4 Results

Large Break

The limiting case among the double-ended ruptures, which yielded a calculated peak temperature of 322.7°F, is the 1.4 ft^2 double-ended rupture, 102% power, MSIV failure case. Figures 3.5-6 and 3.5-7 provide the upper and lower compartment temperature profiles. Figures 3.5-8 and 3.5-9 illustrate the upper and lower compartment pressure transients.

Small Break

The most limiting case in terms of peak calculated temperature is the 0.942 ft^2 split break, 30% power, with an MSIV failure. This case resulted in a calculated peak temperature of 326°F. Figures 3.5-10 and 3.5-11 provide the upper and lower compartment temperature profiles. Figures 3.5-12 and 3.5-13 present the upper and lower compartment pressure transients.

3.5.5.5 Conclusion

The main steam line break containment integrity analysis has been performed consistent with the current licensing basis analysis and SGTP Program, considering the present operating plant conditions. The results of this analysis show that the Environmental Acceptance Criteria (Reference 4) applicable for Cook Nuclear Plant Units 1 and 2 are met. This analysis therefore demonstrates that the containment heat removal systems function to rapidly reduce the containment pressure and temperature in the event of a main steam line break accident. GDC 50 and 10CFR Part 50 Appendix K are satisfied, which is more restrictive than the GDC criteria in Appendix H of the original FSAR, to which the Donald C. Cook Nuclear Plants are licensed.

3.5.5.6 References

1. WCAP-11902, Supplement 1, September 1989, "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 & 2 Licensing Report."

2. WCAP-7907-P-A (Proprietary), "LOFTRAN Code Description", April 1984.

3. WCAP-8354-P-A (Proprietary), Supplement 2, "Long Term Ice Condenser Containment Code - LOTIC-3 Code", February 1979.

4. AEP/W SGTP-19, "Donald C. Cook Nuclear Plant Steam Generator Tube Plugging Analysis Technical Documentation Transmittal", August 10, 1994.

TABLE 3.5-1
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
SYSTEM PARAMETERS
-INITIAL CONDITIONS

PARAMETERS	VALUE
Core Thermal Power (MWt)	3413
Reactor Coolant System Flowrate, per Loop (gpm)	79000
Vessel Outlet Temperature* (°F)	615.2
Core Inlet Temperature* (°F)	547.4
Vessel Average Temperature (°F)	581.3
Initial Steam Generator Steam Pressure (psia)	836.3
Steam Generator Design	Model 51
Steam Generator Tube Plugging (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	114075
Accumulator	
Water Volume (ft^3)	946
N_2 Cover Gas Pressure (psia)	600
Temperature (°F)	120
Safety Injection Delay (sec) (includes time to reach pressure setpoint)	48.0

* (analysis value includes an additional +5.1°F allowance for instrument error and deadband)

TABLE 3.5-2
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
SAFETY INJECTION FLOW
Minimum SI

INJECTION MODE

RCS Pressure (psig)	Total Flow (gpm)
0	3635.5
20	3447.2
40	3235.3
60	3003.7
80	2738.0
100	2425.6
120	2041.3
140	1493.3
160	889.5
180	883.0
200	876.4

RECIRCULATION MODE
(w/o RHR Spray)

Total Flow
(gpm)

3011

RECIRCULATION MODE
(w/ RHR Spray)

Total Flow
(gpm)

764

TABLE 3.5-3
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
MINIMUM SI
BLOWDOWN MASS AND ENERGY RELEASE

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
.000	.0	.0	.0	.0
.101	40932.2	22363.5	21842.5	11892.7
.201	41809.9	22984.5	23698.7	12911.5
.301	46791.4	25944.9	23586.2	12864.8
.401	47304.8	26505.7	23083.2	12605.8
.601	44920.5	25777.3	21588.6	11803.0
.800	45062.1	26395.2	19910.9	10889.6
1.10	41715.9	25037.9	18871.6	10327.0
1.40	38993.8	23888.1	18466.4	10106.3
2.30	31603.5	20634.8	17983.4	9842.5
2.80	26248.9	17695.2	17008.4	9311.5
3.00	21790.0	14847.8	16632.6	9107.5
3.40	19301.6	13347.2	15908.2	8714.5
3.90	18078.7	12537.0	15034.4	8239.8
4.60	15511.6	10751.8	13993.7	7674.4
5.20	13955.8	9663.8	13348.7	7323.9
6.20	12621.1	8669.3	12570.9	6900.4
6.60	12409.8	8475.8	13180.5	7237.4
8.00	12926.3	8608.2	12562.2	6907.8
8.40	12289.7	8355.6	12442.3	6843.7
8.80	10124.1	7526.0	12205.8	6713.4
9.40	9779.3	7313.0	11841.3	6514.6
11.0	9523.6	6842.3	10832.1	5959.4
13.8	7137.6	5509.2	9219.1	5074.5
16.4	5492.8	4593.4	7813.7	4311.0
18.4	4381.2	3819.6	6705.5	3474.5
18.8	4172.2	3729.5	9594.3	4972.0
19.0	4012.7	3668.6	5136.1	2661.2
19.2	3961.1	3685.6	8561.9	4245.4
19.4	3820.3	3688.4	8618.3	4403.8
19.6	3734.8	3730.0	5453.2	2703.5
19.8	3554.0	3671.6	8394.0	4148.0
20.0	3322.7	3582.1	4907.9	2449.3
20.2	3066.4	3451.8	6807.8	3152.7
20.6	2642.1	3188.3	4697.0	2086.8
20.8	2411.6	2951.7	5832.6	2567.9
21.0	2237.3	2755.6	3815.9	1683.9
21.4	1909.0	2365.5	4740.7	1974.2
21.8	1705.5	2122.8	2922.4	1187.0
22.0	1562.8	1947.9	6501.6	2457.3
22.2	1445.6	1805.1	4567.9	1744.3
22.4	1337.0	1672.2	5030.5	1926.8
22.8	1176.0	1475.1	3300.6	1231.8
23.6	914.5	1151.5	1329.9	459.4
23.8	833.4	1050.2	2964.0	796.5
24.4	475.4	600.7	2402.3	633.4
24.6	413.0	523.1	2843.3	754.2
25.8	140.8	179.4	2309.9	648.0
27.0	66.5	85.2	299.7	93.9
28.0	19.0	24.5	126.0	45.3

TABLE 3.5-4
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
MINIMUM SI
REFLOOD MASS AND ENERGY RELEASE

TIME	BREAK PATH NO. 1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
28.0	.0	.0	.0	.0
28.3	143.2	168.1	1818.3	162.7
28.5	122.8	144.3	1791.4	160.3
28.7	110.9	130.3	1778.9	159.2
29.7	108.0	126.8	1717.9	153.8
30.0	119.6	140.4	1696.2	151.8
30.7	125.7	147.6	1658.1	148.4
34.1	148.1	174.1	1508.0	135.0
36.2	160.6	188.8	1431.7	128.1
38.1	170.9	200.9	1371.5	122.8
39.1	213.4	251.1	2017.3	223.3
40.1	351.3	414.5	3809.9	531.6
41.2	371.3	438.4	4023.2	581.6
42.2	368.0	434.6	3989.7	581.3
43.2	362.3	427.7	3929.7	574.9
45.2	350.6	413.8	3807.4	560.8
47.2	339.7	400.8	3691.6	547.3
48.2	355.6	419.7	3902.6	563.8
49.2	355.3	419.3	3885.2	561.6
51.2	346.0	408.2	3787.0	549.9
53.2	337.3	397.9	3694.5	538.9
55.2	329.2	388.3	3607.3	528.5
57.2	321.6	379.3	3524.8	518.7
59.2	314.5	370.8	3446.8	509.4
61.2	307.8	362.9	3372.8	500.5
62.3	313.7	369.5	238.7	160.2
63.3	456.3	539.8	301.8	244.5
64.3	448.9	531.0	298.4	240.1
68.3	414.7	490.1	282.9	219.4
70.3	399.6	472.1	276.1	210.4
74.3	372.0	439.2	263.7	194.1
78.3	347.1	409.5	252.5	179.5
81.3	329.9	389.1	244.9	169.7
82.3	324.5	382.6	242.6	166.5
86.3	304.0	358.3	233.6	154.9
93.3	272.8	321.1	220.1	137.4
97.3	257.3	302.8	213.5	128.9
101.3	243.3	286.3	207.6	121.3
105.3	230.9	271.5	202.4	114.6
107.3	225.1	264.7	200.0	111.6
117.3	201.1	236.3	190.1	98.9
127.3	183.5	215.4	183.0	89.8
135.3	173.0	203.0	178.8	84.4
143.3	165.1	193.7	175.8	80.4
153.3	158.1	185.5	173.1	76.9
163.3	153.5	180.0	171.4	74.5
175.3	150.1	176.0	170.1	72.7
193.3	148.0	173.5	169.4	71.4
223.3	148.4	173.8	169.6	70.9
249.7	150.3	175.9	170.5	71.4

TABLE 3.5-5
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
MINIMUM SI
PRINCIPAL PARAMETERS DURING REFLOOD

TIME	FLOODING		CARRYOVER	CORE	DOWNCOMER	FLOW	INJECTION			ENTHALPY
	TEMP	RATE	FRACTION	HEIGHT	HEIGHT	FRACTION	TOTAL	ACCUMULATOR	SPILL	
SECONDS	DEGREE F	IN/SEC		FT	FT		(POUNDS MASS PER SECOND)			BTU/LBM
28.0	219.9	.000	.000	.00	.00	.250	.0	.0	.0	.00
28.5	217.1	19.962	.000	.56	.35	1.000	7168.8	7168.8	.0	89.50
28.9	215.1	8.452	.000	1.05	.55	1.000	7070.4	7070.4	.0	89.50
29.2	214.8	3.412	.018	1.18	.97	1.000	6999.3	6999.3	.0	89.50
29.4	214.8	4.303	.041	1.24	1.27	1.000	6953.1	6953.1	.0	89.50
30.5	215.1	2.570	.314	1.50	3.00	.598	6672.9	6672.9	.0	89.50
33.1	216.7	2.141	.534	1.77	7.00	.485	6194.7	6194.7	.0	89.50
36.2	218.7	2.220	.635	2.00	11.55	.445	5728.8	5728.8	.0	89.50
41.2	221.6	3.530	.710	2.36	15.99	.578	4713.3	4713.3	.0	89.50
43.0	222.6	3.401	.724	2.50	16.00	.575	4549.3	4549.3	.0	89.50
44.2	223.4	3.319	.731	2.60	16.00	.573	4455.0	4455.0	.0	89.50
47.2	225.3	3.149	.742	2.81	16.00	.566	4240.3	4240.3	.0	89.50
48.2	226.0	3.256	.745	2.87	16.00	.576	4472.3	4049.6	.0	87.94
50.1	227.2	3.177	.750	3.00	16.00	.575	4397.4	3971.2	.0	87.90
58.2	232.9	2.899	.761	3.50	16.00	.562	3981.5	3548.8	.0	87.71
61.2	235.1	2.817	.764	3.67	16.00	.558	3851.3	3416.7	.0	87.64
62.3	235.8	3.092	.765	3.73	15.98	.612	437.7	.0	.0	72.99
63.3	236.5	3.715	.763	3.80	15.79	.628	403.2	.0	.0	72.99
66.2	238.4	3.515	.766	4.01	15.23	.626	408.8	.0	.0	72.99
74.3	242.8	3.035	.770	4.52	13.95	.621	421.8	.0	.0	72.99
83.3	244.3	2.628	.771	5.00	12.92	.614	431.8	.0	.0	72.99
95.3	242.3	2.224	.769	5.56	12.02	.604	441.3	.0	.0	72.99
106.3	243.4	1.949	.768	6.00	11.55	.594	447.3	.0	.0	72.99
121.3	244.0	1.694	.766	6.53	11.31	.581	452.4	.0	.0	72.99
136.3	243.2	1.537	.764	7.00	11.36	.571	455.8	.0	.0	72.99
155.3	244.3	1.423	.765	7.55	11.68	.563	458.8	.0	.0	72.99
171.8	243.5	1.375	.765	8.00	12.07	.560	460.6	.0	.0	72.99
191.3	244.3	1.347	.768	8.52	12.61	.559	462.4	.0	.0	72.99
210.0	243.8	1.338	.770	9.00	13.17	.581	463.9	.0	.0	72.99
229.3	244.3	1.334	.774	9.49	13.75	.563	465.3	.0	.0	72.99
231.3	244.2	1.334	.774	9.54	13.81	.563	465.5	.0	.0	72.99
249.7	244.1	1.337	.776	10.00	14.37	.565	466.7	.0	.0	72.99

TABLE 3.5-6
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
MINIMUM SI
POST REFLOOD MASS AND ENERGY RELEASE

TIME	BREAK PATH NO. 1 FLOW		BREAK PATH NO 2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
249.7	200.8	251.0	285.3	103.3
254.7	201.2	251.5	284.9	103.2
259.7	200.4	250.6	285.7	103.2
264.7	200.8	251.0	285.3	103.0
269.7	200.0	250.0	286.1	103.1
274.7	200.3	250.5	285.8	102.9
279.7	199.6	249.5	286.6	103.0
284.7	199.9	249.8	286.3	102.8
289.7	199.0	248.8	287.1	102.9
299.7	199.6	249.5	286.6	102.5
304.7	198.7	248.5	287.4	102.6
309.7	199.0	248.7	287.2	102.5
314.7	198.1	247.7	288.0	102.5
324.7	198.5	248.1	287.6	102.2
329.7	197.6	247.0	288.5	102.3
339.7	197.9	247.4	288.2	102.0
344.7	197.0	246.2	289.2	102.1
359.7	197.2	246.5	289.0	101.7
364.7	196.2	245.3	289.9	101.8
399.7	195.8	244.7	290.4	101.2
404.7	194.8	243.5	291.4	101.3
424.7	194.4	243.1	291.7	100.9
449.7	193.3	241.6	292.9	100.7
454.7	193.8	242.3	292.3	100.4
469.7	192.5	240.6	293.7	100.4
474.7	192.8	241.0	293.3	100.2
489.7	191.7	239.7	294.4	100.2
499.7	191.7	239.7	294.4	99.9
519.7	190.7	238.4	295.5	99.7
529.7	190.9	238.7	295.2	99.5
534.7	190.1	237.7	296.0	99.5
544.7	190.1	237.7	296.0	99.3
549.7	189.5	236.9	296.7	99.4
554.7	189.9	237.4	296.2	99.2
574.7	188.6	235.8	297.5	99.0
579.7	188.6	235.8	297.5	98.9
609.7	187.3	234.1	298.8	98.5
614.7	82.0	102.5	404.1	120.8
870.8	82.0	102.5	404.1	120.8
870.9	79.9	94.4	406.3	88.7
874.7	79.8	99.4	406.3	155.7
1979.7	64.8	75.2	421.4	84.8
1982.3	64.7	80.5	103.3	96.5
2222.2	64.7	80.5	103.3	96.5
2222.3	63.4	78.8	332.2	158.0
2316.8	63.4	78.8	332.2	158.0

TABLE 3.5-7
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
MINIMUM SI

MASS BALANCE

	TIME (SECONDS)	.00	28.00	28.00	249.66	870.94	2316.82
		MASS (THOUSAND LBM)					
INITIAL	IN RCS AND ACC	771.32	771.32	771.32	771.32	771.32	771.32
ADDED MASS	PUMPED INJECTION	.00	.00	.00	91.01	393.02	1087.36
	TOTAL ADDED	.00	.00	.00	91.01	393.02	1087.36
*** TOTAL AVAILABLE ***		771.32	771.32	771.32	862.33	1164.34	1858.68
DISTRIBUTION	REACTOR COOLANT	537.32	57.74	67.87	135.93	135.93	135.93
	ACCUMULATOR	234.00	171.20	161.07	.00	.00	.00
	TOTAL CONTENTS	771.32	228.94	228.94	135.93	135.93	135.93
EFFLUENT	BREAK FLOW	.00	542.36	542.36	726.39	1028.39	1722.73
	ECCS SPILL	.00	.00	.00	.00	.00	.00
	TOTAL EFFLUENT	.00	542.36	542.36	726.39	1028.39	1722.73
TOTAL ACCOUNTABLE		771.32	771.30	771.30	862.31	1164.32	1858.66

TABLE 3.5-8
DONALD C COOK NUCLEAR PLANT UNITS 1 AND 2
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
MINIMUM SI

ENERGY BALANCE

TIME (SECONDS)		.00	28.00	28.00	249.66	870.94	2316.82
		ENERGY (MILLION BTU)					
INITIAL ENERGY	IN RCS,ACC,S GEN	901.43	901.43	901.43	901.43	901.43	901.43
ADDED ENERGY	PUMPED INJECTION	.00	.00	.00	6.64	28.69	84.57
	DECAY HEAT	.00	8.96	8.96	34.20	87.00	181.86
	HEAT FROM SECONDAR	.00	-5.10	-5.10	-5.10	-2.19	4.03
	TOTAL ADDED	.00	3.87	3.87	35.75	113.49	270.46
*** TOTAL AVAILABLE ***		901.43	905.30	905.30	937.18	1014.92	1171.89
DISTRIBUTION	REACTOR COOLANT	318.00	12.74	13.64	30.54	30.54	30.54
	ACCUMULATOR	20.94	15.32	14.42	.00	.00	.00
	CORE STORED	28.06	13.71	13.71	3.19	3.16	2.92
	PRIMARY METAL	178.97	168.74	168.74	143.60	92.94	63.41
	SECONDARY METAL	84.19	84.08	84.08	77.83	59.46	35.57
	STEAM GENERATOR	271.26	275.82	275.82	252.03	189.66	116.14
	TOTAL CONTENTS	901.43	570.41	570.41	507.19	375.76	248.57
EFFLUENT	BREAK FLOW	.00	334.41	334.41	421.73	630.90	916.99
	ECCS SPILL	.00	.00	.00	.00	.00	.00
	TOTAL EFFLUENT	.00	334.41	334.41	421.73	630.90	916.99
TOTAL ACCOUNTABLE		901.43	904.82	904.82	928.92	1006.66	1165.56

TABLE 3.5-9
ENERGY ACCOUNTING IN MILLIONS OF BTU

	Approx. End of Blowdown (t = 10.0 sec.)	Approx. End of Reflood (t = 249.7 sec.)
Ice Heat Removal*	207.7	250.3
Structural Heat Sinks*	17.37	44.73
RHR Heat Exchanger Heat Removal*	0	0
Spray Heat Exchanger Heat Removal*	0	0
Energy Content of Sump	188.94	250.0
Ice Melted (Pounds) (10^6)	0.67	0.84

* Integrated Energies

TABLE 3.5-10
ENERGY ACCOUNTING IN MILLIONS OF BTU

	Approx. Time of Ice Melt Out (t = 5423 sec.)	Approx. Time of Peak Pressure (t = 7752 sec.)
Ice Heat Removal*	567.21	567.21
Structural Heat Sinks*	82.52	112.68
RHR Heat Exchanger Heat Removal*	49.0	77.31
Spray Heat Exchanger Heat Removal*	58.31	92.3
Energy Content of Sump	583.6	599.3
Ice Melted (Pounds)(10^6)	2.11	2.11

* Integrated Energies

TABLE 3.5-11
STRUCTURAL HEAT SINK TABLE

SURFACES		AREA (Ft^2)	THICKNESS (Ft)
Upper Compartment Material			
1.	Paint	32500.	0.001083
	Carbon Steel	32500.	0.0469
	Concrete	32500.	2.0
2.	Paint	10086.	0.001083
	Concrete	10086.	2.0
3.	Paint	5880.	0.00125
	Concrete	5880.	1.5
4.	Paint	11970.	0.00125
	Concrete	11970.	1.0
Lower Compartment Material			
5.	Paint	5069.	0.00125
	Concrete	5069.	2.0
6.	Paint	13660.	0.00125
	Concrete	13660.	1.5
7.	Paint	16730.	0.00125
	Concrete	16730.	1.0
8.	Paint	8665.	0.00125
	Concrete	8665.	2.0
Ice Condenser			
9.	Steel	180600.	0.00663
10.	Steel	76650.	0.0217
11.	Steel	28670.	0.0267
12.	Paint	3336.	0.000833
	Concrete	3336.	0.333

TABLE 3.5-11(continued)
STRUCTURAL HEAT SINK TABLE

SURFACES	AREA (Ft^2)	THICKNESS (Ft)
Ice Condenser		
13. Steel and Insulation	19100.	1.0
Steel	19100.	0.0625
14. Steel and Insulation	13055.	1.0
Concrete	13055.	1.0

TABLE 3.5-12
MATERIAL PROPERTIES TABLE

Material	Conductivity Btu/hr -°ft -°F	Volumetric Heat Capacity Btu/ft^3 - °F
Concrete	0.8	22.6
Steel	26.0	56.4

TABLE 3.5-13
UNIT 1/UNIT 2 STEAMLINE BREAK MASS/ENERGY RELEASES INSIDE CONTAINMENT

30% Power, 0.942 ft^2 Split Break
Failure - MSIV

TIME (sec)	MASS (lbm/sec)	ENERGY (MBtu/sec)
.0000	.0000	.0000
.2000	1873.	2.234
5.600	1744.	2.085
7.000	1734.	2.073
10.00	1718.	2.054
13.00	1703.	2.036
13.60	1698.	2.031
14.80	1688.	2.020
15.60	1681.	2.011
16.00	1677.	2.007
18.00	1629.	1.950
20.00	1522.	1.825
26.00	1284.	1.544
35.00	1061.	1.277
40.00	974.2	1.173
45.00	905.9	1.091
50.00	853.1	1.028
60.00	782.0	.9418
70.00	741.1	.8925
80.00	719.1	.8659
90.00	707.4	.8517
100.0	701.3	.8444
110.0	698.3	.8408
120.0	696.7	.8388
150.0	695.1	.8369
200.0	694.1	.8357
270.0	692.9	.8342
290.0	691.3	.8323
292.5	667.1	.8028
295.0	607.8	.7312
297.5	554.0	.6658
320.0	476.8	.5711
337.5	403.4	.4820
352.5	344.5	.4106
367.5	296.7	.3531
395.0	183.5	.2174
410.0	136.6	.1609
432.5	114.3	.1345
495.0	106.6	.1252
605.0	109.5	.1282

TABLE 3.5-13 (continued)
UNIT 1/UNIT 2 STEAMLINE BREAK MASS/ENERGY RELEASES INSIDE CONTAINMENT

102% Power, 1.4 ft^2 Double Ended Rupture
Failure - MSIV

TIME (sec)	MASS (lbm/sec)	ENERGY (MBtu/sec)
.0000	.0000	.0000
.2000	9753.	11.68
1.400	8708.	10.45
3.800	7436.	8.940
6.000	7228.	8.693
8.000	7069.	8.504
10.00	6882.	8.281
11.60	6658.	8.014
12.00	6441.	7.752
12.20	6224.	7.490
13.00	5353.	6.443
14.20	4047.	4.871
15.20	2959.	3.562
16.00	2090.	2.516
16.40	1657.	1.995
17.00	1482.	1.784
22.00	1306.	1.572
24.00	1253.	1.509
26.00	1208.	1.455
28.00	1169.	1.408
30.00	1137.	1.369
32.00	1110.	1.337
34.00	1087.	1.309
36.00	1068.	1.285
45.00	1006.	1.211
75.00	878.0	1.056
100.0	851.6	1.024
200.0	831.4	.9998
280.0	825.3	.9924
282.5	789.4	.9485
285.0	695.6	.8349
287.5	619.9	.7431
292.5	455.2	.5429
300.0	241.3	.2850
320.0	112.0	.1308
350.0	106.5	.1244
610.0	3.231	.0038

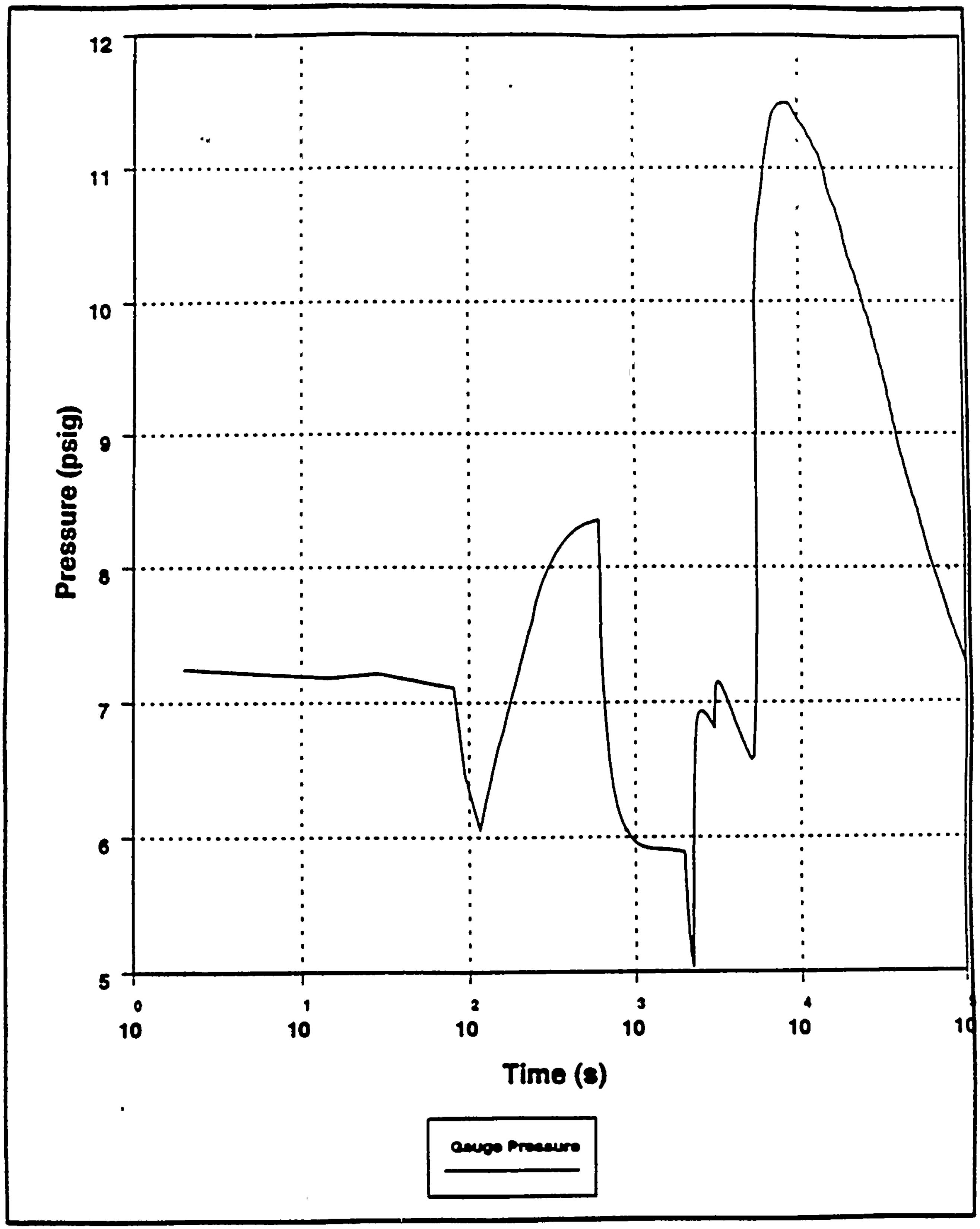


Figure 3.5-1 LOCA Mass Energy Release Containment Integrity Containment Pressure

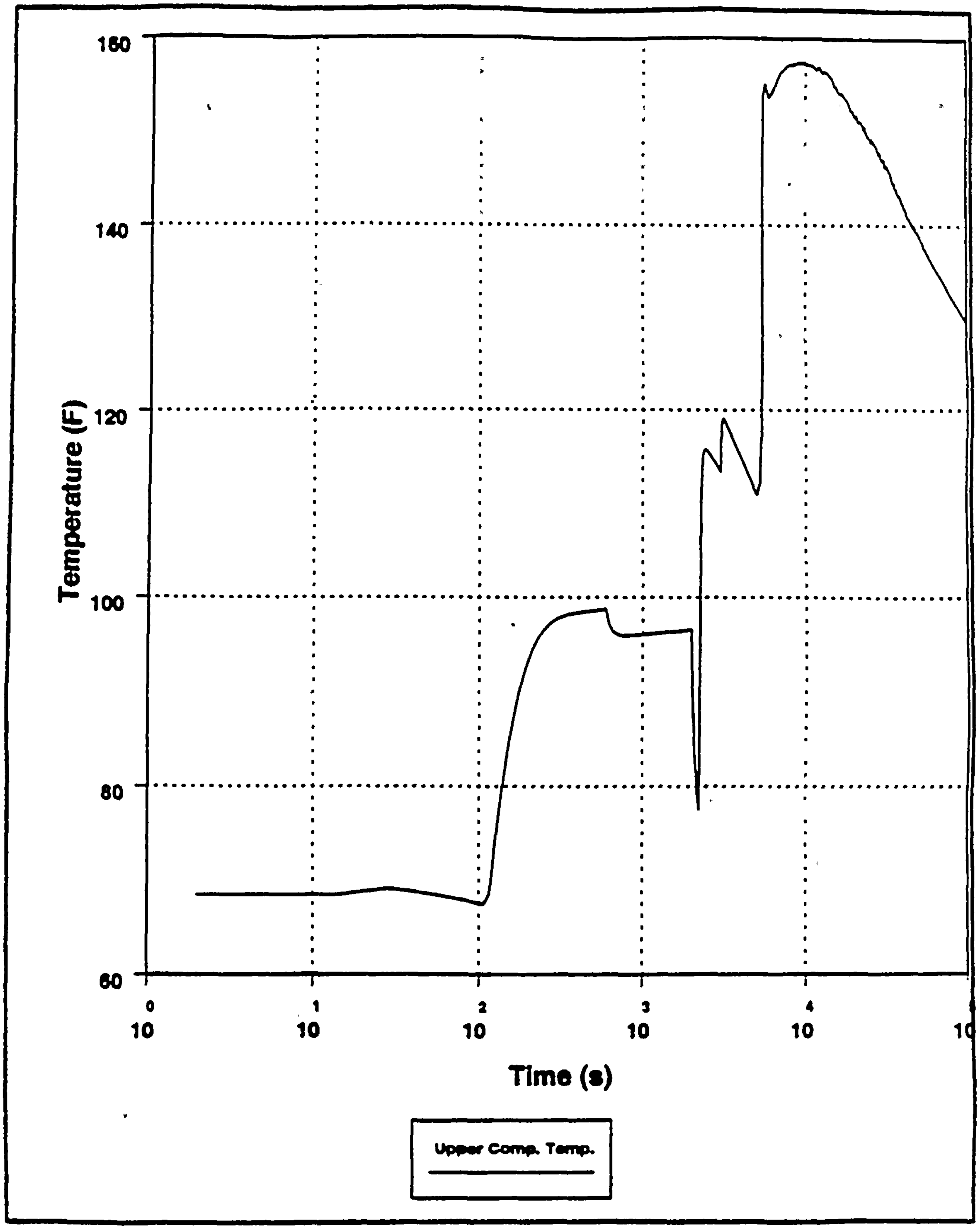


Figure 3.5-2 LOCA Mass Energy Release Containment Integrity
Upper Containment Temperature

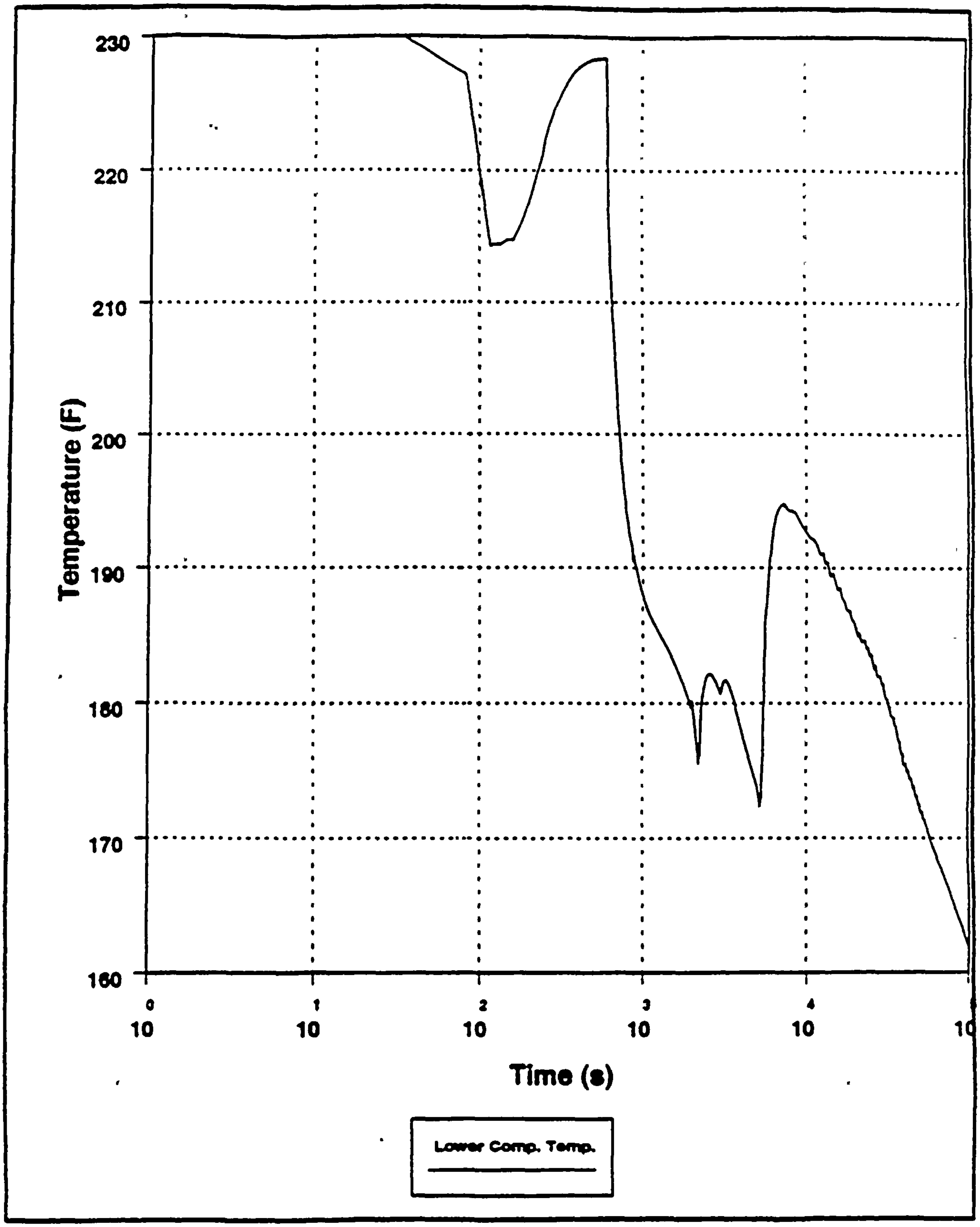


Figure 3.5-3 LOCA Mass Energy Release Containment Integrity
Lower Containment Temperature

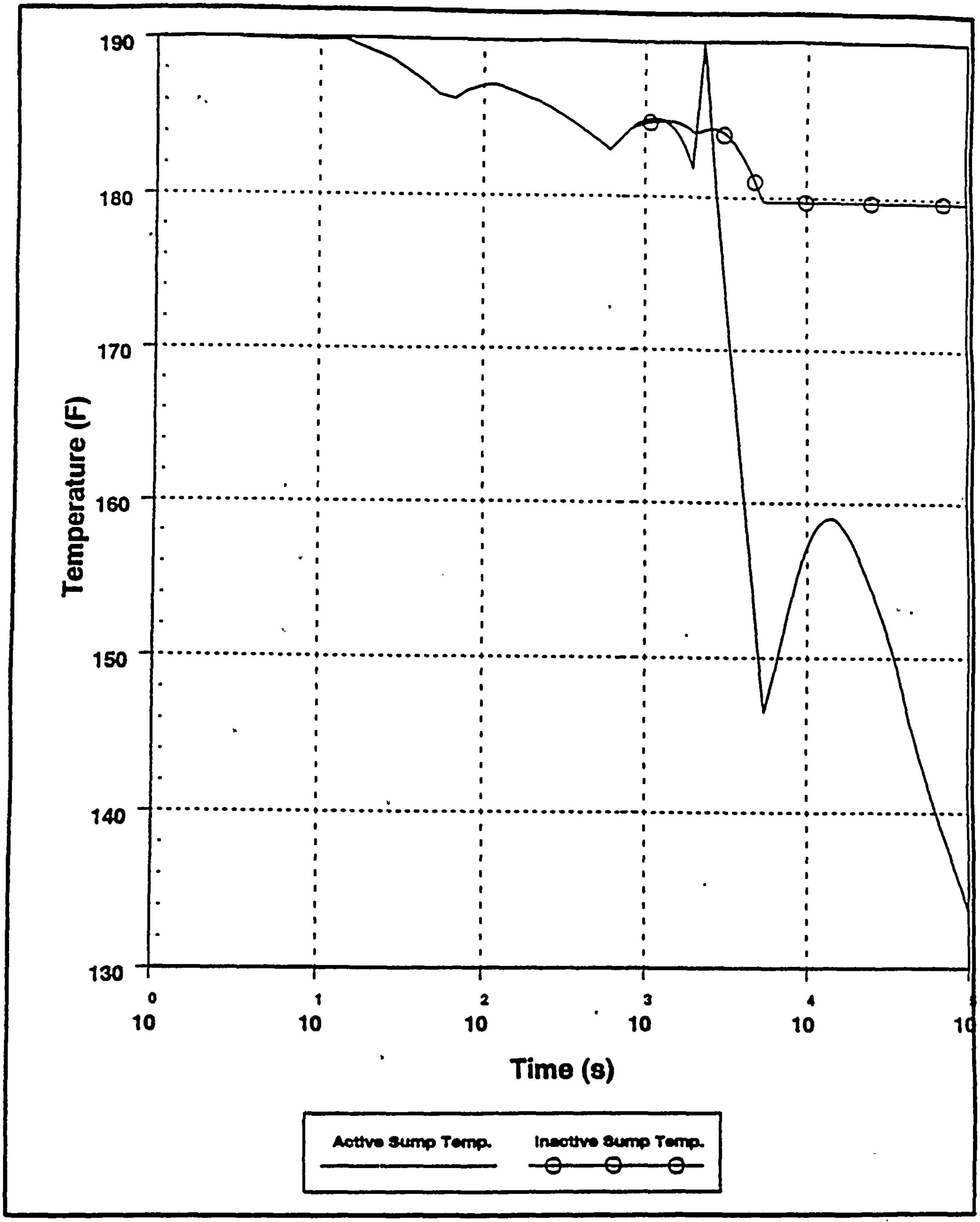


Figure 3.5-4 LOCA Mass Energy Release Containment Integrity Active Sump and Inactive Sump Temperature Transient

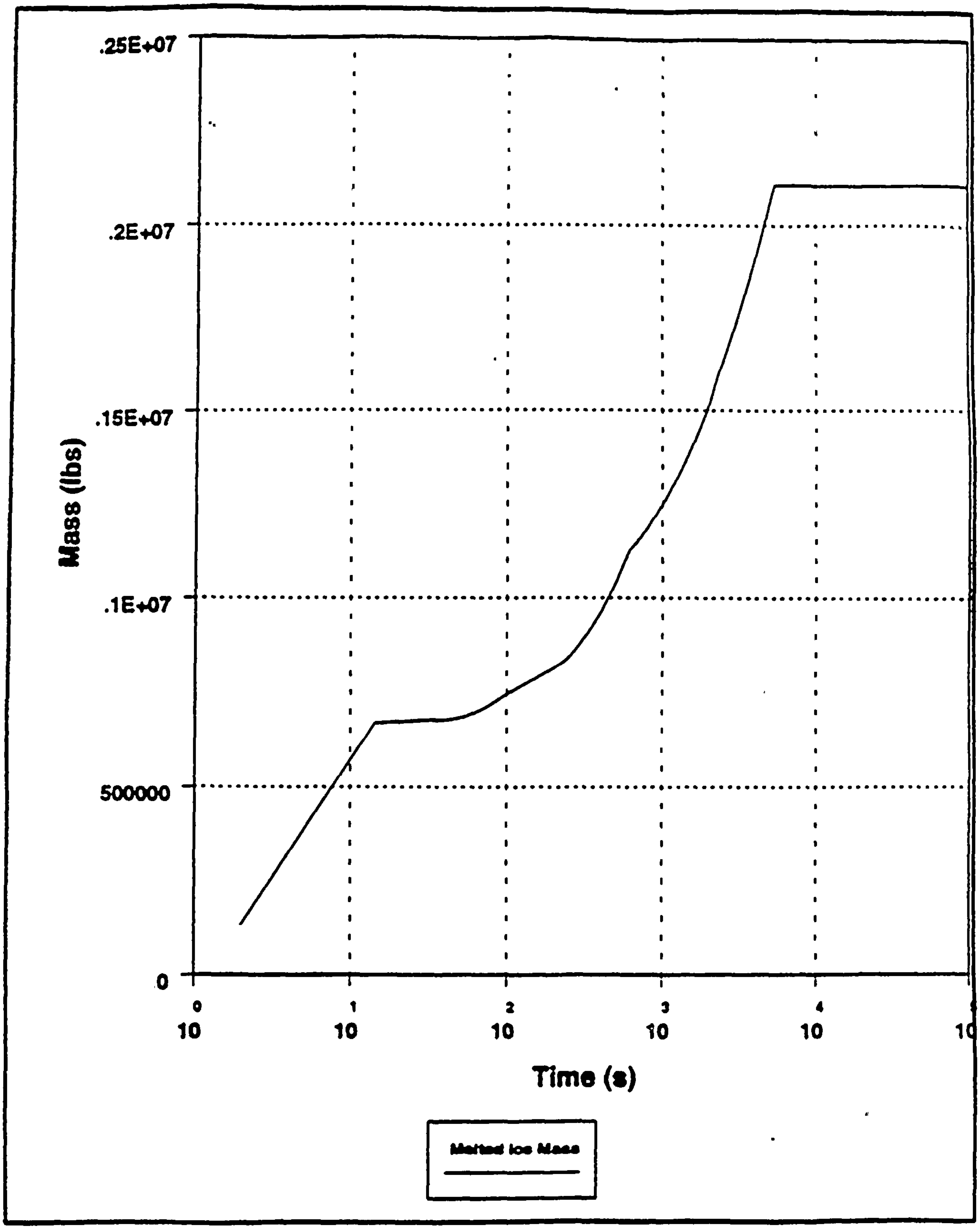


Figure 3.5-5 LOCA Mass Energy Release Containment Integrity
Ice Melt Transient

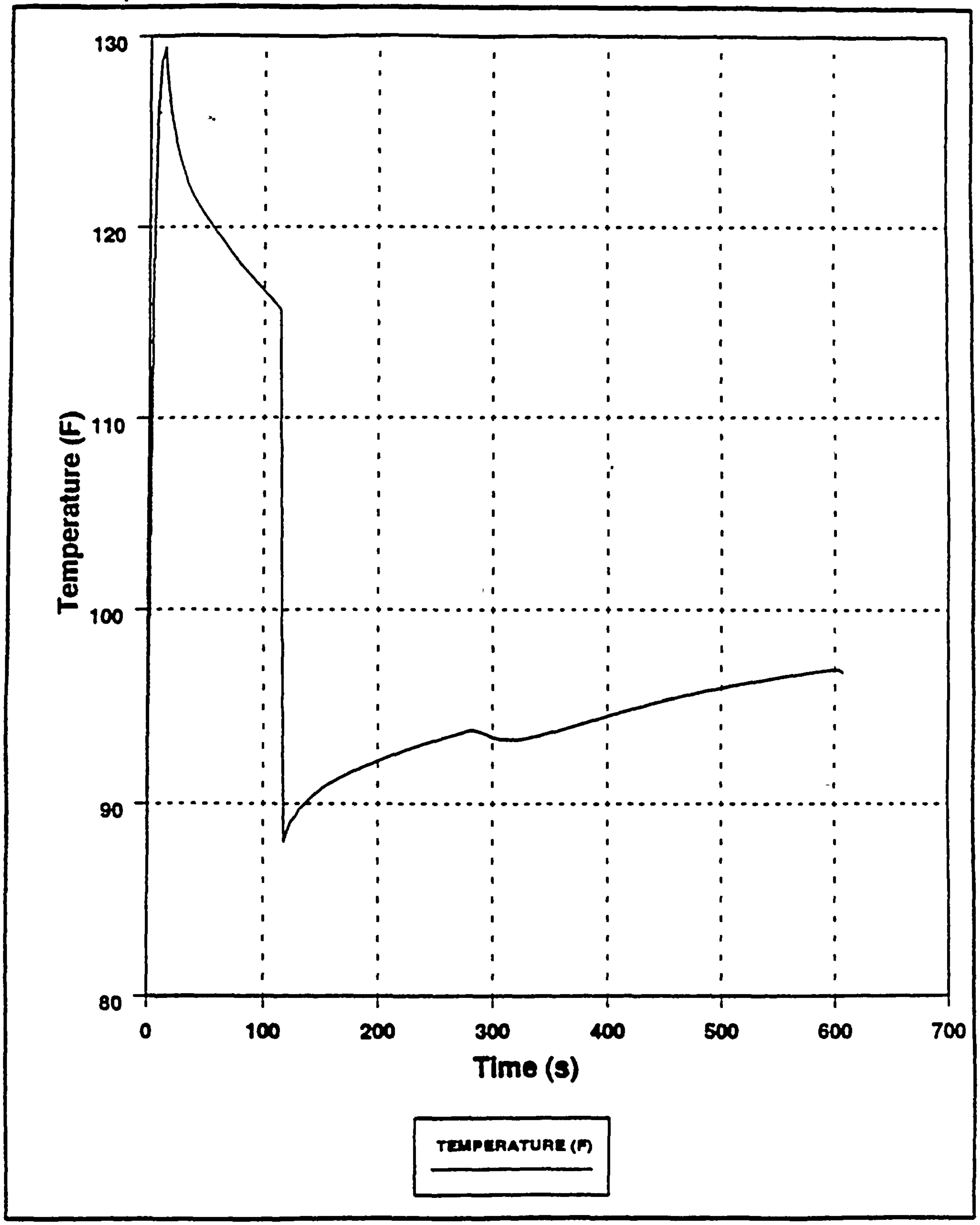


Figure 3.5-6 102% Power, 1.4 sq. ft. Double Ended Rupture - MSIV Failure
Upper Compartment Temperature

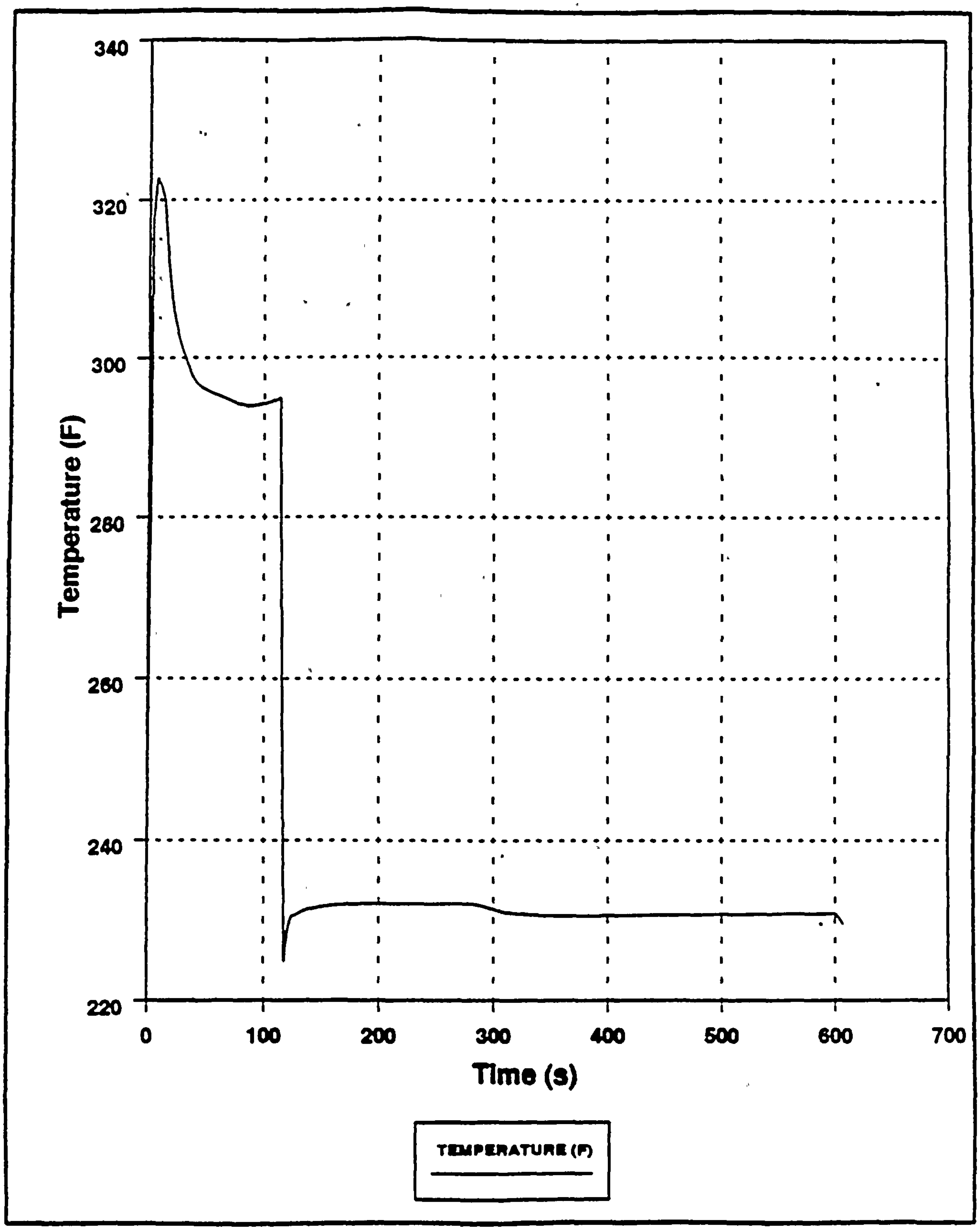


Figure 3.5-7 102% Power, 1.4 sq. ft. Double Ended Rupture - MSIV Failure Lower Compartment Temperature

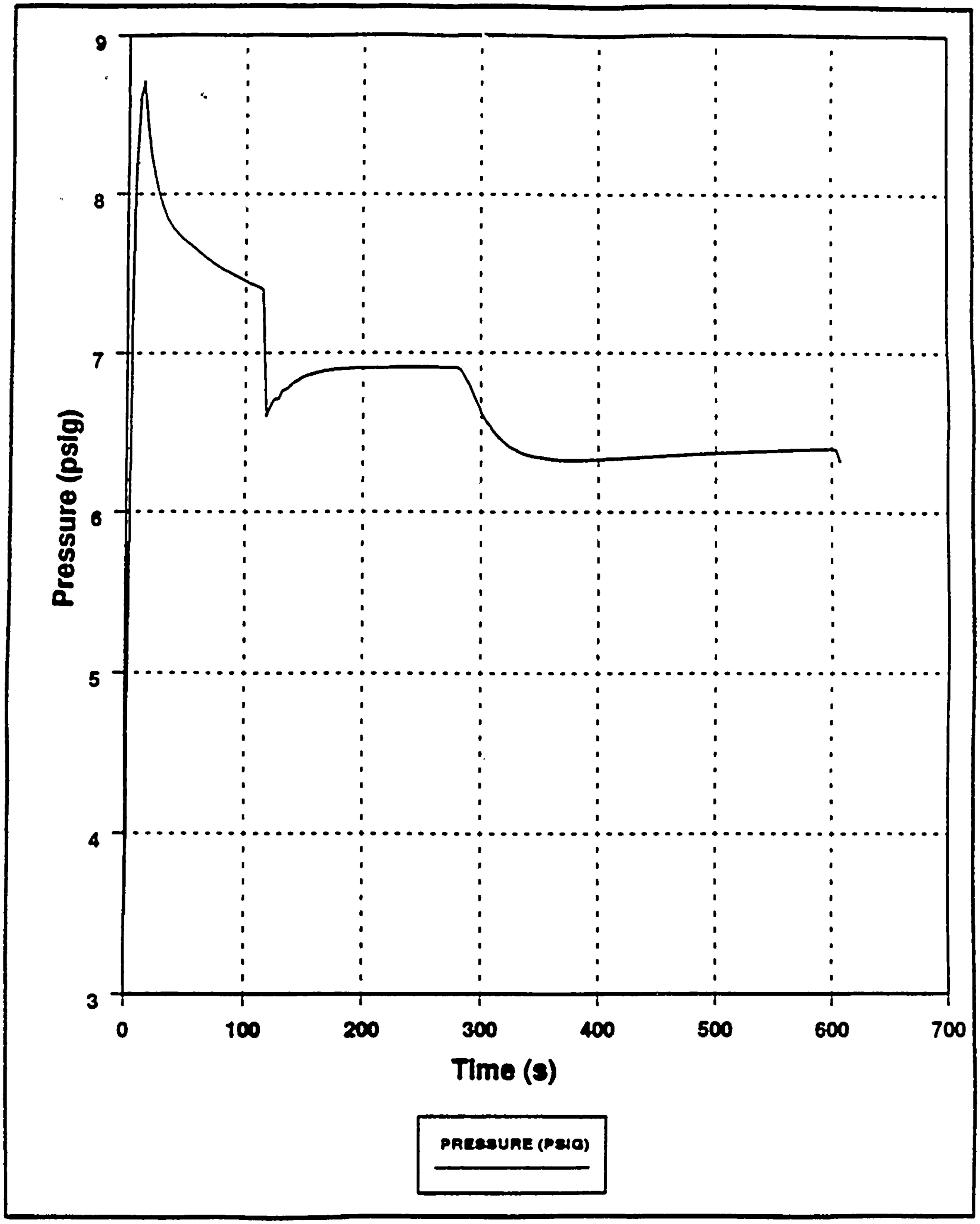


Figure 3.5-8 102% Power, 1.4 sq. ft. Double Ended Rupture - MSIV Failure
Upper Compartment Pressure

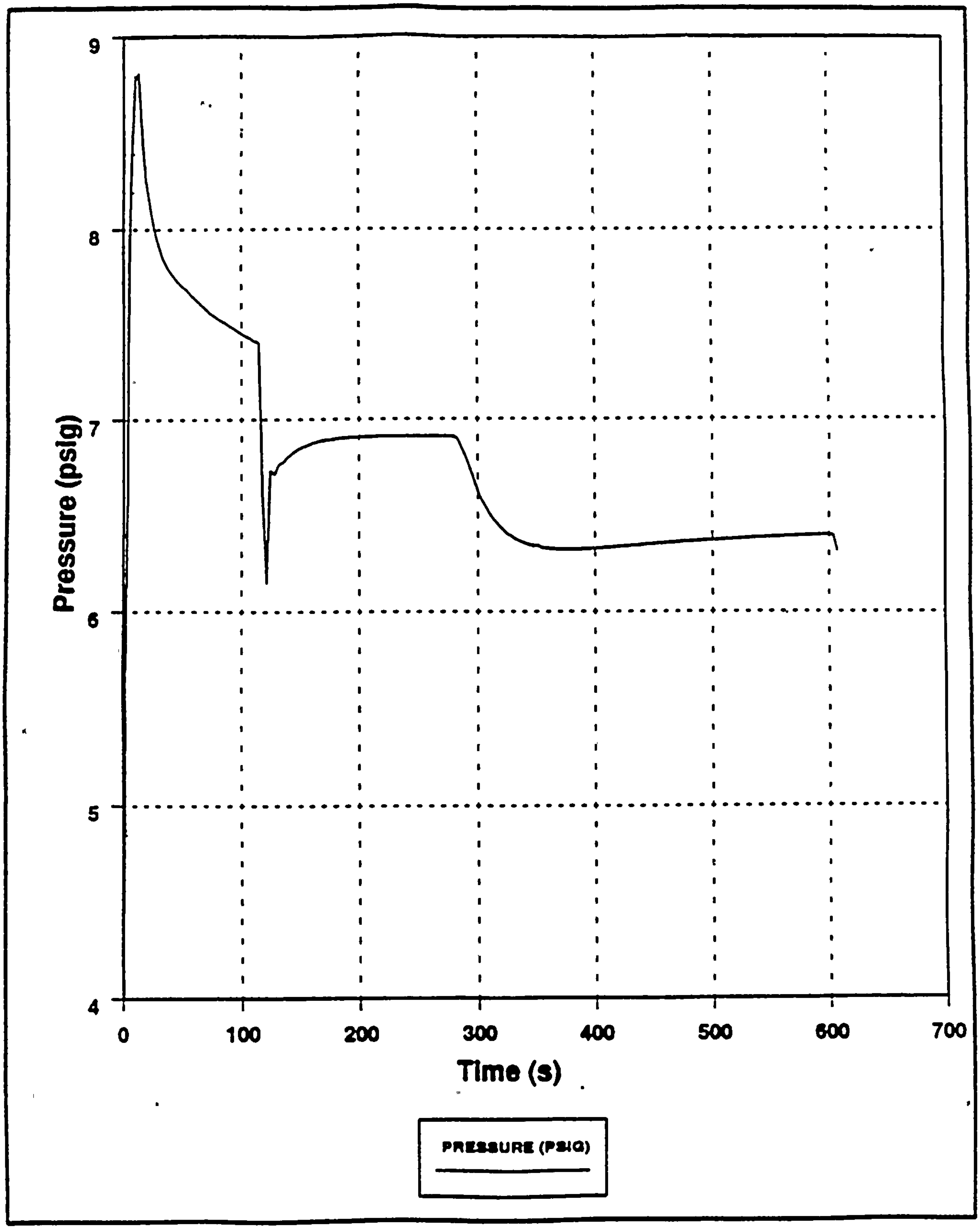


Figure 3.5-9 102% Power, 1.4 sq. ft. Double Ended Rupture - MSIV Failure
Lower Compartment Pressure

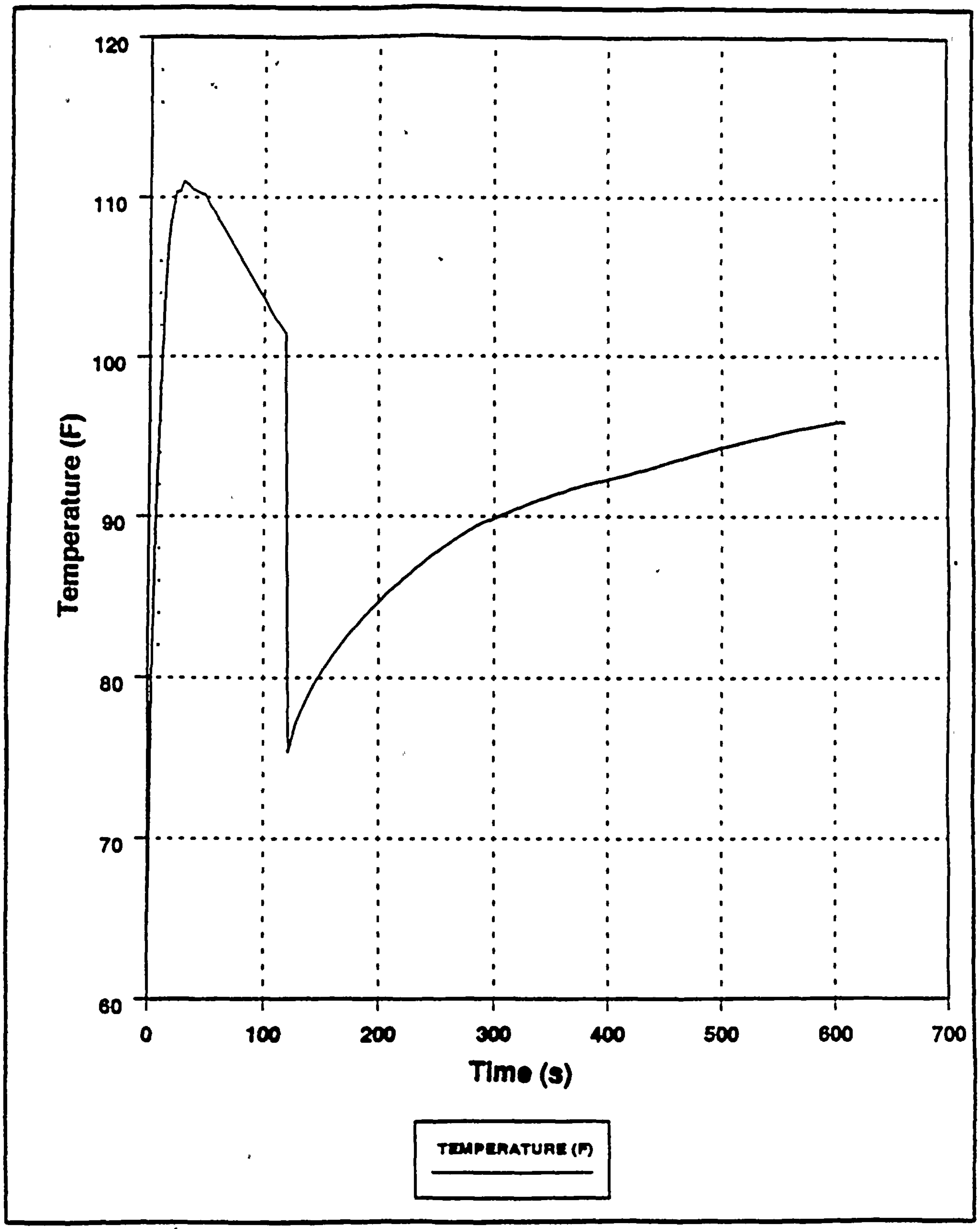


Figure 3.5-10 30% Power, 0.942 sq. ft. Split Break - MSIV Failure
Upper Compartment Temperature

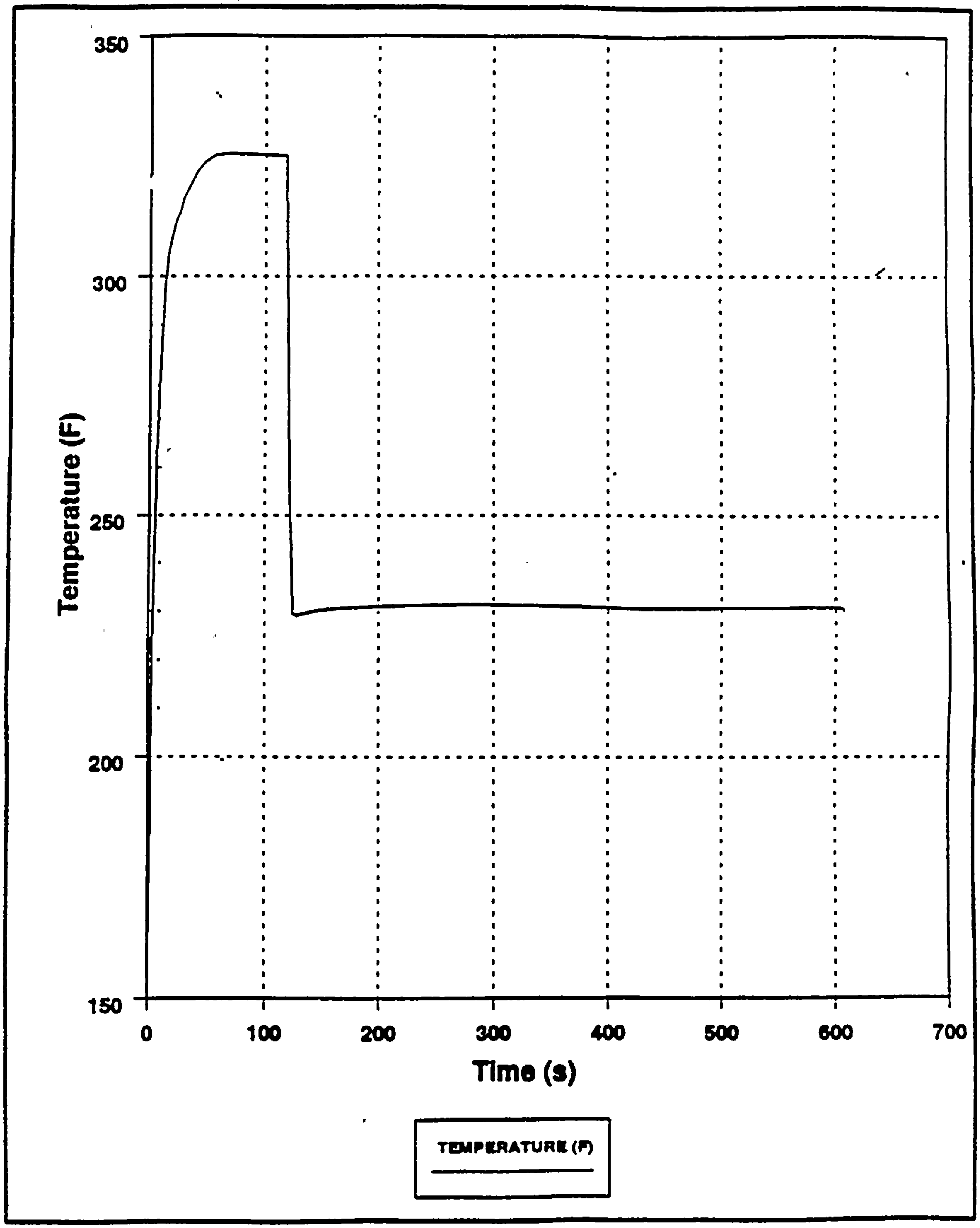


Figure 3.5-11 30% Power, 0.942 sq. ft. Split Break - MSIV Failure
Lower Compartment Temperature

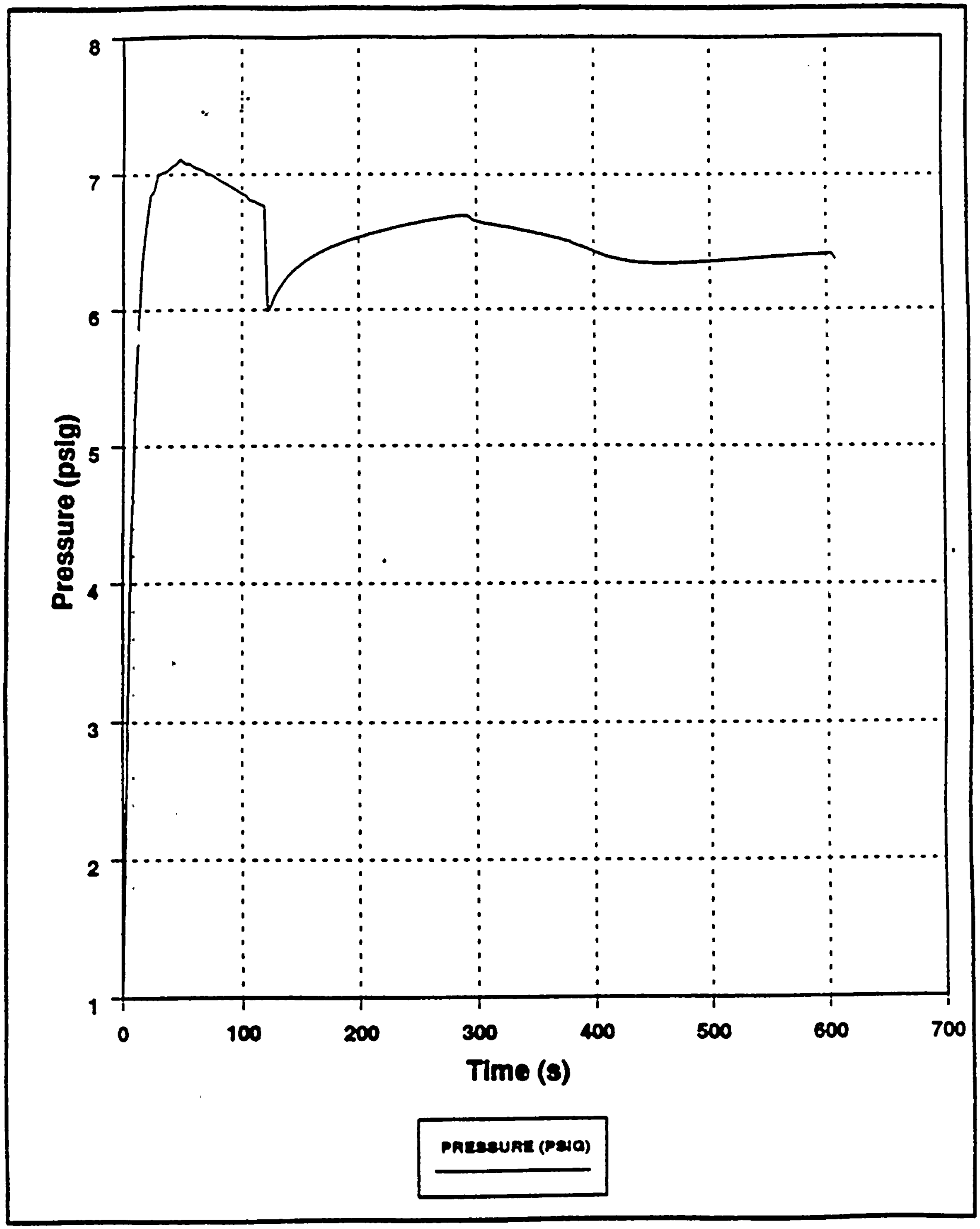


Figure 3.5-12 30% Power, 0.942 sq. ft. Split Break - MSIV Failure
Upper Compartment Pressure

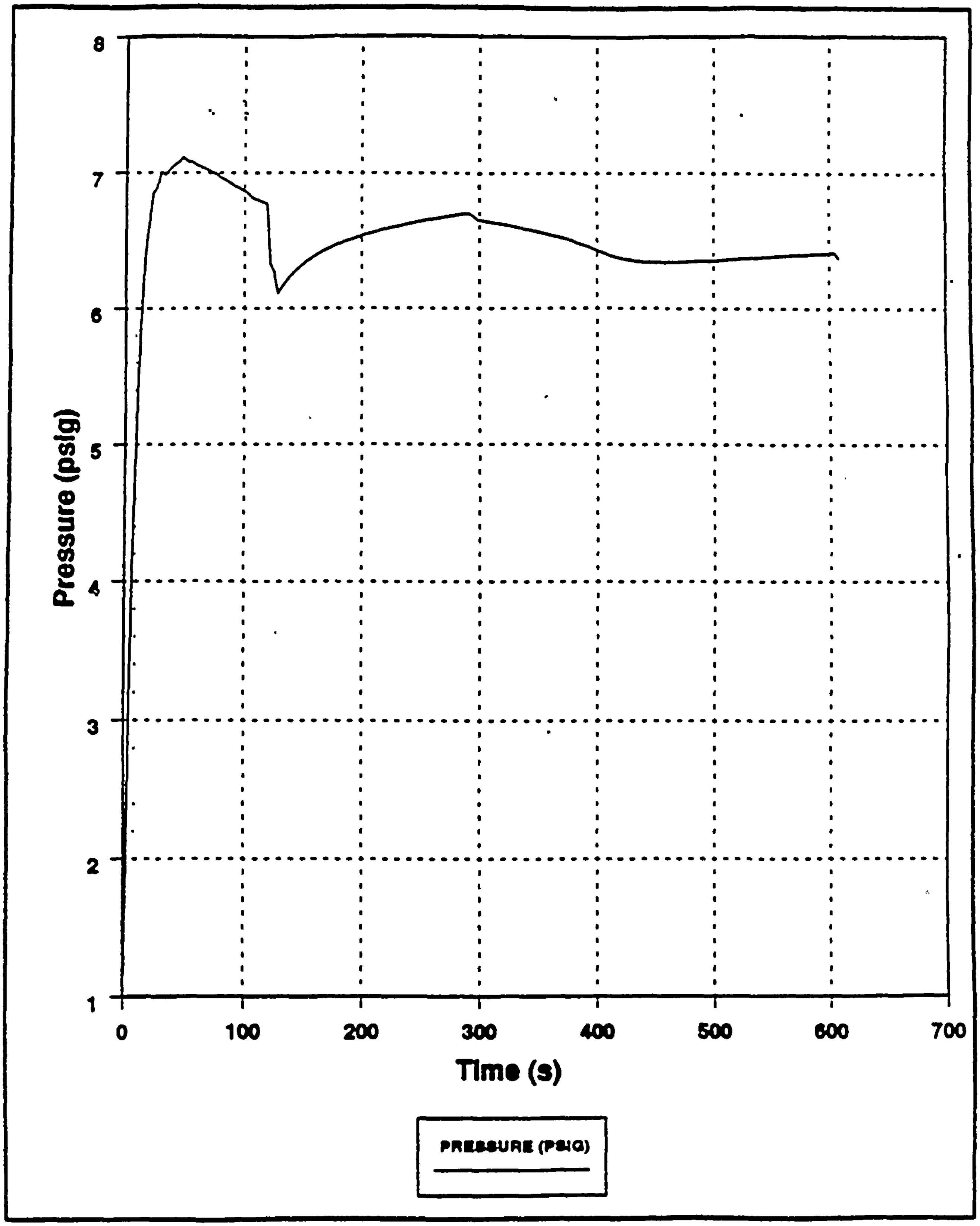


Figure 3.5-13 30% Power, 0.942 sq. ft. Split Break - MSIV Failure
Lower Compartment Pressure

3.6 STEAM GENERATOR TUBE RUPTURE ACCIDENT ANALYSIS

The UFSAR analysis of a steam generator tube rupture (SGTR) transient is performed to conservatively predict the radiological consequences of such an event. An evaluation of this transient, supporting an increase to 30% SGTP for Donald C. Cook Nuclear Plant Unit 1, has been completed to determine the impact on the dose releases.

The primary thermal hydraulic parameters which affect the calculated offsite radiation doses for a SGTR event are the assumed radioactivity level in the reactor coolant, the reactor coolant released through the ruptured tube to the secondary steam generator volume, and the steam released from the ruptured steam generator to the atmosphere. With respect to the UFSAR analysis, a change in the SGTP level does not impact the reactivity level of the reactor coolant. However, both the potential primary coolant release to the secondary and the secondary steam release to the atmosphere are impacted by the assumed tube plugging level.

To evaluate the effect of a 30% SGTP level, the mass releases from the RCS and from the secondary volume to the atmosphere were calculated. Four cases were considered assuming a nominal RCS temperature of 533.0°F and 576.3°F with both symmetric and asymmetric RCS flow conditions. A nominal full power level of 3262 MWt (NSSS) was also assumed.

The thyroid and whole body doses estimated for Cook Nuclear Plant Unit 1, based on the 30% SGTP evaluation, remain within a "small fraction" (10%) of the 10CFR100 exposure limit guidelines. Small fraction is the smallest of the exposure guide lines defined in NUREG-0800 (Standard Review Plan). Therefore, the conclusions of the UFSAR remain valid.

3.7 POST-LOCA HOT LEG RECIRCULATION TIME

The hot leg switchover calculation performed to preclude boron precipitation during cold leg recirculation following a LOCA is not adversely affected by the changes associated with the SGTP Program. The analysis is not affected since the proposed changes do not affect the normal plant operating parameters, the safeguards systems actuations, or the accident mitigation capabilities important to the calculation. The SGTP Program does not significantly change the assumptions used in the analysis. Any variation in the initial RCS fluid inventory due to tube plugging is judged to be insignificant with respect to the calculated time for hot leg switchover to preclude boron precipitation. Other fluid inventories and boron concentrations remain unchanged. Furthermore, the proposed changes do not create conditions more limiting than those assumed in the hot leg switchover calculation.

3.8 REACTOR CAVITY PRESSURE ANALYSIS

The Reactor Cavity Pressure Analysis is performed to calculate the initial pressure response in the reactor cavity to a loss of coolant accident. The Reactor Cavity Pressure Analysis that was performed for the Rerating Program was reviewed and it was determined that the conclusions provided for the Rerating Program (WCAP-11902) remain valid for the SGTP Program.

The SGTP Program parameters affect the Reactor Cavity Pressure Analysis through the mass and energy releases provided as input into the analysis. There is no direct impact of SGTP level on short-term mass and energy release rate calculations and containment subcompartment response analysis (See Section 3.5.1). The major impact results from changes to RCS temperature. For short-term effects, higher release rates typically result from cooler RCS conditions. The mass and energy releases used as input for the Reactor Cavity Pressure Analysis reflected limiting conditions and therefore, the NSSS performance parameters for the SGTP Program did not impact the results.

3.9 RADIOLOGICAL ANALYSIS

A review was performed of the radiological analysis in the UFSAR for Cook Nuclear Plant Unit 1 to determine the effects of the SGTP Program. The source terms for LOCA and the fuel handling accident are unaffected by the increase in SGTP level. However, a reanalysis of the offsite doses following a large break LOCA was performed for the increase in EDG start time to 30 seconds. The EDG start time delay resulted in a delay in spray injection flow to containment of 115 seconds, whereas the previous analysis assumed no delay in spray injection flow to containment. While there was a slight increase in the offsite thyroid doses, the doses are within the applicable limits.

Source terms for the SGTR event were recalculated at the SGTP Program conditions. The radiological consequences of the SGTR are summarized in Section 3.6 of this report.

For the steamline break radiological analysis, the offsite thyroid dose and the corresponding steam generator primary-to-secondary leak rate determined for the alternate steam generator tube plugging criteria program (APC) is, by design of the methodology, relatively insensitive to the amount of steam released to the environment. For the steam generator in the ruptured loop, all of the initial iodine activity along with all of the primary-to-secondary leakage activity is released to the environment. Any additional steam release from this steam generator would be due to the introduction of clean aux feedwater, which would not increase the activity released to the environment (i.e., can't release more than 100% of the activity from this steam generator). Steam released form the unaffected steam generators due to boiling of the secondary coolant accounts for approximately 1% of the total activity release. Additional steam released from these steam generators will have no significant impact on either the calculated offsite dose or the allowable primary-to-secondary leak rate. Therefore, a steamline break radiological analysis was not required for the SGTP Program.

3.10 FLUID AND AUXILIARY SYSTEMS EVALUATIONS

3.10.1 Fluid Systems Evaluation

3.10.1.1 Introduction

This section addresses the impact of the SGTP Program on the ability of the Reactor Coolant System and auxiliary fluid systems to perform their required functions. The parameters considered are listed in Table 2.1-1.

In order to support the operation of Cook Nuclear Plant Unit 1 at the SGTP Program conditions, the following systems were evaluated at the new conditions: 1) Reactor Coolant System (RCS), 2) Chemical and Volume Control System (CVCS), 3) Emergency Core Cooling System (ECCS), 4) Residual Heat Removal System (RHR), and 5) Spent Fuel Pool Cooling System (SFPCS). A brief description of each system is provided below.

The Emergency Core Cooling System flowrates were revised as part of the SGTP Program. These ECCS flowrates reflected a charging pump head degradation of 10% (differential pressure of 2290 psid on recirculation), a safety injection and RHR pump head degradation of 15% (differential pressures of 1326 psid and 150 psid, respectively, on recirculation). The ECCS flowrates were used in the safety analyses and evaluations for the SGTP Program.

3.10.1.2 Description of Fluid Systems

Reactor Coolant System

The RCS consists of four identical heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump and a steam generator. In addition, the system includes a pressurizer, a pressurizer relief tank, interconnecting piping, and instrumentation necessary for operational control.

During operation, the RCPs circulate pressurized water through the reactor vessel and the four coolant loops. The water, which serves both as a coolant, moderator, and solvent for boric acid (chemical shim control), is heated as it passes through the core. It then flows to the steam generators, where the heat is transferred to the steam system, and returns to the RCPs to repeat the cycle.

RCS pressure is controlled by the use of the pressurizer where water and steam are maintained in equilibrium by electrical heaters and water sprays. Steam can be formed (by the heaters) to increase RCS pressure or condensed (by the pressurizer spray) to reduce the pressure. Three spring loaded safety valves and three power operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank, where the steam is condensed and cooled by mixing with water.

Chemical and Volume Control System

The CVCS provides for boric acid addition, chemical additions for corrosion control, reactor coolant clean-up and degasification, reactor coolant make-up, reprocessing of water letdown from the RCS, and RCP seal water injection. During plant operation, reactor coolant flows through the shell side of the regenerative heat exchanger and then through a letdown orifice. The regenerative heat exchanger reduces the temperature of the reactor coolant and the letdown orifice reduces the pressure. The cooled, low pressure water leaves the reactor containment and enters the auxiliary building. A second temperature reduction occurs in the tube side of the letdown heat exchanger followed by a second pressure reduction due to the low pressure letdown valve. After passing through one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filter and enters the Volume Control Tank (VCT).

Emergency Core Cooling System

The ECCS injects borated water into the reactor following a break in either the reactor or steam systems in order to cool the core and prevent an uncontrolled return to criticality. Two each safety injection pumps and residual heat removal pumps take suction from the RWST and deliver borated water to four cold leg connections via the accumulator discharge lines. In addition, two centrifugal charging pumps take suction from the RWST on SI actuation and provide flow to the RCS via separate SI connections on each cold leg. This arrangement of SI pumps can provide safety injection flow at any RCS pressure up to the set pressure of the pressurizer safety valves.

Residual Heat Removal System

The RHRS is designed to remove sensible and decay heat from the core and reduces the temperature of the RCS during the second phase of plant cooldown. As a secondary function, the RHRS is used to transfer refueling water between the RWST and the refueling cavity at the beginning and end of refueling operations.

The RHRS consists of two residual heat exchangers, two RHR pumps and associated piping, valves and instrumentation. During system operation, coolant flows from one hot leg of the RCS to the RHR pumps, through the tube side of the residual heat exchangers and back to two RCS cold legs. The residual heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell.

Spent Fuel Pool Cooling System

The SFPCS removes the decay heat generated by spent fuel elements stored in the spent fuel pool. A secondary function is to maintain the clarity and purity of the spent fuel pool

water. The SFPCS serves the spent fuel pool which is shared between the two Cook Nuclear Plant units. The system design incorporates two cooling trains. Each of the two cooling trains in the SFPCS consists of a pump, heat exchanger, strainer, associated piping, valves and instrumentation, and a shared discharge. The pump draws water from the pool, circulates it through the heat exchanger and returns it to the pool. The heat exchangers are of the shell and U-tube type; component cooling water circulates through the shell, and spent fuel pool water circulates through the tubes.

3.10.1.3 Fluid Systems Evaluation

The impact of the SGTP Program on the ability of the following fluid systems to perform their required functions has been evaluated for the RCS, CVCS, ECCS, RHRS, and SFPCS.

Reactor Coolant System

The capability of the RCS to operate was evaluated at the SGTP Program conditions listed in Table 2.1-1.

The capacities of the pressurizer spray and power operated relief valves, pressurizer surge line, relief line, RTD bypass delay times and pressurizer relief tank setpoints were evaluated. It was concluded that the design pressurizer spray flow rate of 750 gpm can not be achieved at the 30% SGTP Program conditions. The pressurizer spray flow rate was calculated to be 596 gpm. This was determined to be adequate such that the PORVs are not actuated following a 10% step load decrease from full power. Results of this analysis indicate that the reduced spray valve capacity is adequate to prevent PORV actuation for the nominal operating pressures of 2100 and 2250 psia as well as over the range of full load average RCS temperatures between 553°F and 576.3°F.

The pressurizer surge line pressure drop was evaluated during a design basis surge. The design basis surge results from three safety valves relieving at the design capacity. It was determined that the RCS maximum pressure at the discharge of the RCP is 2745 psia which is below the ASME maximum allowable pressure for the RCS.

The pressurizer relief line pressure drop calculation was unaffected by the revised NSSS parameters for the SGTP Program.

The results of the evaluations showed that the installed PORV capacity of 630,000 lbm/hr is adequate for the design basis load swings for operation at all SGTP operating conditions.

The RTD Bypass delay time calculations indicate that the fluid transport delay times for the existing piping network remain below 1.0 second in all loops and are, therefore, acceptable.

The pressurizer relief tank setpoints were found to be acceptable. The PRT pressure will be maintained below the rupture disc set pressure following a design basis discharge with the current level setpoints.

Chemical and Volume Control System

The regenerative and letdown heat exchangers are designed to cool letdown flow from T_{cold} to 115°F. This reduction in temperature is required to ensure that the normal RCP seal injection temperature requirement of 130°F will be maintained, including an allowance for a 15°F temperature rise across the centrifugal charging pump. The variations in T_{cold} considered for the SGTP Program are bounded by the design inlet temperature of 547°F for the regenerative heat exchanger. Therefore, the cooling requirements of the letdown function are met with the revised operating parameters.

The letdown function is designed to reduce the static pressure of the reactor letdown stream from the RCP suction pressure to VCT operating pressure, such that the design pressure of intervening piping and components is not exceeded and fluid is maintained in a subcooled condition throughout the system. The majority of the pressure reduction is taken across the letdown orifices. The pressure control valve, QRV-301, ensures that adequate back pressure is maintained on the letdown orifices to ensure subcooled fluid conditions. The pressurizer pressures considered (2100 or 2250 psia) are bounded by the design pressurizer operating pressure. In addition, it has been verified that QRV-301 is capable of maintaining sufficient backpressure on the letdown orifices to ensure subcooled fluid conditions when the pressurizer pressure is reduced to 2100 psia. Therefore, the pressure reduction requirements of the letdown function are met with the revised operating parameters.

Emergency Core Cooling System

The primary system pressures considered for this program are less than or equal to the primary system pressure against which the original system was designed to deliver. The required core cooling flow rate is proportional to reactor power level which has not changed as a result of this program. Therefore, the revised primary system parameters do not require an increase in either the motive pressure or core cooling capacity of the ECCS.

Residual Heat Removal System

The RHRS is normally placed in operation approximately four hours after reactor shutdown when the pressure and temperature of the RCS are approximately 400 psig and 350°F, respectively. Under normal operating conditions, the RHRS is designed to reduce the temperature of the reactor coolant to 140°F within 20 hours following reactor shutdown, with both trains operating. In the event of a train failure, the RHRS is designed to reduce the reactor coolant temperature to 200°F within 36 hours after reactor shutdown. Since the initiation temperature and decay heat generation rates (power level) have not changed from

those previously evaluated for the Rerating Program, the demands on the RHRS are not affected. Therefore, the RHRS is still capable of reducing the reactor coolant temperature to 140°F within the 20 hour limit for normal operating conditions, when both trains are operating. In the event of a train failure, the RHRS is still capable of reducing the reactor coolant temperature to 200°F within the 36 hour limit.

Spent Fuel Pool Cooling System

The primary function of the SFPCS is to remove decay heat which is generated by the spent fuel pool elements stored in the pool. Decay heat generation is proportional to plant power level. Since the plant power level of 3262 MWt remains unchanged from that previously evaluated for the Rerating Program, the demands on the SFPCS are not increased. The purification function is controlled by SFPCS demineralization and filtration rates, which are not affected by the SGTP Program.

3.10.2 NSSS/Balance of Plant Interface Systems Evaluation

The proposed NSSS Performance Parameters for the SGTP Program were compared with those of the Rerating Program. The results of the evaluation show that a SGTP level of 30% will have no adverse effects on the Balance Of Plant (BOP) systems performance.

The Donald C. Cook Nuclear Plant Unit 1 BOP fluid systems and components have been evaluated to assess the effects of increasing the SGTP level up to 30%. The evaluation compared the bounding NSSS performance parameters with the current bounding Unit 1 Rerating Program parameters (Cases 1,3,4,5, & 6 of WCAP-11902, Supplement 1) to determine the impact on the following BOP systems:

- Main Steam System
- Condensate and Feedwater System
- Auxiliary Feedwater System
- Steam Generator and Blowdown System

The proposed performance parameters which affect the BOP systems and components, compared to the Unit 1 Rerating Program parameters, either do not change, or change in a favorable direction with increased SGTP levels of up to 30%. For example, the SGTP Program power level of 3262 MWt corresponds to the minimum power level evaluated for the Unit 1 Rerating Program. The final feedwater temperature remains unchanged as well as no load Tavg and secondary steam pressure; Also, the steam mass flowrates are bounded by the Unit 1 Rerating Program parameters. One significant change in parameters is the change in the full power steam pressure where the lower bounding full power steam pressure (589 psia - Case 4 of Table 2.1-1) is below a lower bounding rerating full power steam pressure (603 psia - Case 4, Table 2.1-1 of WCAP 11902, Supplement 1). This would result in an increase in volumetric steam flow (cubic feet/sec) for the same power rating. However,

since the Rerating Program incorporated a power uprating (3425 MWt) and corresponding mass flow increase, the volumetric flow increase at the reduced power level (3262 MWt) also falls within the bounds of the Unit 1-Rerating Program parameters. Consequently, the changes in steam flow rates and the design considerations associated with steam flow rates are not significant. Therefore, it was concluded that an increase in SGTP levels of up to 30% will have either no impact or an insignificant impact on the NSSS/BOP fluid systems. They will continue to perform acceptably at the conditions associated with the SGTP Program.

3.11 PRIMARY COMPONENT EVALUATIONS

Evaluations were performed for all NSSS primary and auxiliary components to support the SGTP Program for Cook Nuclear Plant Unit 1. In some cases, structural reanalysis was performed. In general, the evaluations and analyses were performed assuming the associated NSSS performance parameters case(s) (from Table 2.1-1) most limiting for the particular component.

The NSSS components reviewed for the SGTP Program are as follows:

Section	Component
3.11.1	Steam Generators
3.11.2	Reactor Vessel
3.11.3	Reactor Internals
3.11.4	Control Rod Drive Mechanisms
3.11.5	Reactor Coolant Pumps
3.11.6	Pressurizer
3.11.7	Reactor Coolant Loop Piping and Supports
3.11.8	Auxiliary Components

A summary of the evaluations and analyses is provided below.

3.11.1 Steam Generators

The following sections describe the analyses and evaluations performed under the Cook Nuclear Plant Unit 1 SGTP Program for the Unit 1 Steam Generators. The Steam Generators evaluated are the original Model 51-series. Three separate areas of evaluation are addressed for the SGTP Program:

- Thermal-hydraulic performance characteristics (including moisture separator performance)

- U-bend tube fatigue

- Structural integrity

<u>Thermal-Hydraulic Performance Evaluation</u>

The factors governing the thermal-hydraulic performance of steam generators can be reduced to the thermal power and steam pressure. Other factors such as primary temperature, primary flow and plugging level are important only insofar as they affect the steam pressure. Primary pressure, in the range under consideration, does not affect thermal hydraulic performance.

As part of the Rerating Program, thermal hydraulic performance parameters were evaluated for a range of thermal powers and steam pressures. The pressure range is bounded by the Rerating Program when all powers are considered. At the 3262 MWt power rating, applicable to the 30% plugging evaluation, the lowest steam pressure is slightly below the value analyzed during the Rerating Program. This will be shown to be of no consequence.

The conclusion of the Rerating Program was that the performance characteristics of the steam generators, including moisture carryover, continue to be acceptable at all the Rerating Program conditions. This conclusion continues to apply for the 30% SGTP conditions.

Moisture Separator Limits

Modifications to the moisture separators at Unit 1 were completed in the Spring of 1989. These modifications include the following elements:

- primary separator "top hats" which diffuse the jet issuing from the primary separators,
- "steam chimneys" which vent steam from below the mid deck plate without entraining liquid drops, and
- additional upper tier dryer drains.

With these modifications, moisture carryover values measured in the field have been near or below 0.1% over a wide range of power levels and steam pressures. In earlier separator systems, including the unmodified Model 51, moisture strongly increased with power. The top hats eliminate a primary cause of this dependence which is the jets issuing from the primary separators. The trend of increasing moisture with decreasing steam pressure remains, but its effect is small and the moisture level remains low to the lowest limit of the data, 700 psia. Based on the available field data, moisture carryover is expected to remain comfortably below 0.25% for steam pressures down to 700 psi and below.

Other Thermal Hydraulic Characteristics

In addition to moisture carryover, the Rerating Program evaluated circulation ratio, hydrodynamic stability, and steam generator mass as additional indicators of acceptable performance. The change in these parameters from the design value to the values at each of the Rerating Program conditions were calculated. Steam pressures analyzed for the rating of 3262 MWt had a range of 610 to 820 psia. Variation of the three parameters over this range is presented in Table 3.11-1 along with the design value at the rated power.

It is evident from Table 3.11-1 that the parameters listed are minimally affected by steam pressure at constant power. Circulation ratio is essentially unaffected. Damping factor is the

measure of hydrodynamic stability, a large negative value indicating a stable unit. This parameter, too, is essentially unaffected by the steam pressure. Steam generator mass is slightly affected by reduced steam pressure. As steam pressure decreases, the voids in the bundle increase reducing the mass inventory. The effect is small and does not affect operability.

Table 3.11-1 displays the parameter variation down to a steam pressure of 610 psia. For the 30% plugging conditions, the minimum steam pressure is 589 psia. The 21 psi pressure change represented in the table was shown to have minimal affect on the parameters reviewed. The additional 21 psi pressure change to 589 will also be small. Steam generator operating characteristics will be acceptable down to the minimum steam pressure of 589 psia.

U-bend Fatigue Evaluation

A complete U-bend fatigue evaluation is documented in WCAP-13814, December 1993, "D. C. Cook Unit 1 - Evaluation for Tube Vibration Induced Fatigue" (Reference 1). The evaluation was performed to determine the susceptibility to fatigue-induced cracking, consistent with NRC Bulletin 88-02. The evaluations were performed for the current operating conditions as well as for a level of 30% tube plugging. The analysis identified preventative actions for tubes identified as potentially susceptible to U-bend vibration induced fatigue.

Structural Integrity Evaluation

Structural integrity evaluation of steam generator components performed for the Donald C. Cook Nuclear Plant Unit 1 Rerating Program included NSSS performance parameter cases that bounded steam generator tube plugging (SGTP) levels of up to 15%. The NSSS design transients developed for the Rerating Program continue to apply to Donald C. Cook Nuclear Plant Unit 1 at the 30% SGTP conditions. Since the performance parameters and the design transients still apply, the evaluations performed for 15% SGTP would be applicable for all components of the steam generator except the divider plate. A new evaluation of the divider plate, therefore, was performed for a higher pressure differential across the divider plate caused by a higher (30%) tube-plugging level. This analysis demonstrated the structural acceptability of the divider plate.

It was therefore concluded that the Cook Nuclear Plant Unit 1 steam generator structural integrity would be maintained for operations with a tube-plugging level of up to 30%.

Reference

1. WCAP-13814, "Donald C. Cook Unit 1 Evaluation for Tube Vibration Induced Fatigue," December 1993

3.11.2 Reactor Vessel

3.11.2.1 Introduction

The Addendum Report prepared for the Unit 1 reactor vessel (reference 1) to evaluate the stress and fatigue effects of the operating parameters and RCS transients associated with the Rerating Program remains applicable. Therefore, no new reactor vessel stress calculations were performed for the SGTP Program. This report evaluates the maximum primary plus secondary stress intensity ranges and maximum cumulative fatigue usage factors resulting from the Rerating Program conditions which bound the 30% SGTP conditions. The calculated stress intensity range and usage factor values are compared with the applicable limits of Section III of the ASME Boiler and Pressure Vessel Code.

The operating parameters shown in Table 3.11-2 were used as a basis for the evaluation. These parameters bound the 30% SGTP parameters contained in Table 2.1-1 of this report.

3.11.2.2 Summary of Results

The results of the reactor vessel analyses and evaluations are summarized below. Based on these results, all of the stress intensity and fatigue usage limits (with the exception of the $3S_m$ maximum range of primary plus secondary stress intensity limit for the control rod drive mechanism housings and outlet nozzle safe end) of the applicable ASME Code version for Unit 1 (reference 2) are met. Exceeding the $3S_m$ limit for the CRDM housings and outlet nozzle safe end is reconciled by simplified elastic-plastic analyses in accordance with reference 3. Therefore, the reactor vessel for Cook Nuclear Plant Unit 1 continues to remain in compliance with the applicable Code for the conditions associated with the Rerating and SGTP Programs.

<u>Control Rod Drive Mechanism Housing (Adapter)</u>

The maximum range of primary plus secondary stress intensity is calculated to be 77.76 ksi which exceeds the $3S_m$ limit of 69.9 ksi. However, a simplified elastic-plastic analysis was performed in accordance with paragraph NB-3228.3 of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code, and the higher range of stress intensity is reconciled. The maximum cumulative fatigue usage factor is 0.1687 which is below the ASME Code limit of 1.0.

<u>Main Closure Region</u>

The main closure region of the reactor vessel consists of the vessel flange, the closure head flange and the closure stud assemblies which couple the head to the vessel. The maximum ranges of stress intensity in the closure head flange and the vessel flange are 65.26 ksi and

61.04 ksi, respectively, compared to an ASME Code $3S_m$ limit of 80.1 ksi. The maximum service in the closure studs is 91.8 ksi which compares favorably to the $3S_m$ limit 107.7 ksi.

The maximum cumulative fatigue usage factor for the closure head flange, vessel flange and closure studs are 0.018, 0.029 and 0.99, respectively. The usage factors are all less than the 1.0 ASME Code limit. However, it should be noted that the closure stud usage factor of 0.99 was calculated under the assumption that the first 25 percent of the 11,680 occurrences of plant loading and plant unloading at 5 percent of full power per minute (2,920 occurrences of each) occurred during the first 10 years of operation when the vessel outlet temperature (T_{hot}) was 599.3°F. If the 0.99 usage factor is unacceptably high or if cycle counting indicates that 1.00 may be exceeded, the closure studs are readily replaceable.

Outlet Nozzle

The maximum range of primary plus secondary stress intensity in the outlet nozzle safe end is calculated to be 59.58 ksi compared to the $3S_m$ limit for the austenitic stainless steel material of 50.1 ksi. Since the maximum range exceeds $3S_m$, a simplified elastic-plastic analysis per paragraph NB-3228.3 of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code was perfomed which justified the higher maximum range of stress intensity. The maximum usage factor at the safe end is 0.021 which is less than 1.0.

The maximum range of stress intensity in the outlet nozzle and nozzle-to-shell juncture is 57.09 ksi compared to the $3S_m$ allowable of 80.1 ksi. The maximum cumulative usage factor in the nozzle and nozzle-to-shell juncture is 0.0631 which is also less than 1.0.

Inlet Nozzle

The maximum range of stress intensity in the inlet nozzle safe end is 49.65 ksi which is less than $3S_m$ = 50.1 ksi. The maximum range of stress intensity in the inlet nozzle and nozzle-to-shell juncture is 49.86 ksi which compares favorably with a $3S_m$ limit of 80.1 ksi. The maximum cumulative usage factors in the nozzle safe end and nozzle-to-shell juncture are 0.0174 and 0.0977, respectively, which are both less than 1.0.

Vessel Wall Transition

The maximum range of stress intensity and cumulative fatigue usage factor for the vessel wall transition, between the nozzle shell and the vessel beltline, are 33.57 ksi and 0.0066. These values are less than the ASME Code limits of 80.1 ksi and 1.0, respectively.

Bottom Head-to-Shell Juncture

The maximum range of primary plus secondary stress intensity at the juncture between the vessel bottom hemispherical head and the vessel beltline shell is 34.53 ksi compared to a $3S_m$

allowable of 80.1 ksi. The maximum cumulative fatigue usage factor at the juncture was calculated to be 0.0182 which is less than 1.0.

Bottom Head Instrumentation Penetrations

The bottom head instrumentation penetrations are acceptable for the SGTP Program based upon a maximum range of primary plus secondary stress intensity of 51.49 ksi and a maximum cumulative fatigue usage factor of 0.1220. These values compare favorably with ASME Code allowables of 69.9 ksi ($3S_m$) and 1.0, respectively.

Core Support Pads

The core support pads were evaluated to have a maximum range of stress intensity of 69.70 ksi compared to a $3S_m$ limit of 69.9 ksi. The maximum cumulative fatigue usage factor was calculated to be 0.693 which is less than the 1.0 ASME Code limit.

3.11.2.3 Conclusions

The results of the evaluations demonstrate that operation of the reactor vessel in accordance with the conditions associated with the 30% SGTP Program does not result in stress intensities or fatigue usage factors which exceed the acceptance criteria of the applicable ASME Code version for Cook Nuclear Plant Unit 1 (reference 2). Some of the stress intensity ranges are higher than the original stress report. However, all of the stress intensity limits specified in the applicable ASME Code version are still satisfied with the incorporation of 30% SGTP conditions, with the exception of the $3S_m$ maximum stress intensity range limit for the CRDM housings and outlet nozzle safe ends. Exceeding $3S_m$ in the CRDM housings and outlet nozzle safe ends is reconciled by simplified elastic-plastic analyses in accordance with the requirements of paragraph NB-3228.3 of the 1971 Edition of Section III of the ASME Code (reference 3).

3.11.2.4 Reactor Vessel Integrity Evaluation

The 30% SGTP conditions for Donald C. Cook Nuclear Plant Unit 1 will not result in an increase in the fast neutron fluence values calculated for the Rerating Program. Based on this information, the reactor vessel integrity analyses performed per the methodology of Regulatory Guide 1.99, Revision 2, as part of the Rerating Program will remain applicable after 30% SGTP. These analyses are applicable for reactor vessel inlet temperatures (Tcold) above 525°F. Operation below 525°F down to 510°F has been further evaluated and found to be acceptable (Reference 4). The increase in SGTP to 30% does not impact the technical support provided in Reference 4 for operation below 525°F.

3.11.2.5 References

1. WCAP-11967, "T_{hot} Reduction/Rerating Reactor Vessel Evaluation Addendum to Analytical Report for Indiana and Michigan Electric Company, Donald Cook Nuclear Power Plant Unit No. 1 Reactor Vessel", August, 1988, by S. L. Abbott.

2. ASME Boiler and Pressure Vessel Code, Section III, American Society of Mechanical Engineers, New York. "Nuclear Vessels", 1965 Edition with Addenda through the Winter of 1966.

3. ASME Boiler and Pressure Vessel Code, Section III, American Society of Mechanical Engineers, New York. "Nuclear Power Plant Components", 1971 Edition.

4. AEP-93-582, "American Electric Power Service Corporation, Donald C. Cook Nuclear Plant Unit 1, Operation Below 525 degree F", Keith F. Matthews to John Jensen, October 18, 1993.

3.11.3 Reactor Internals

3.11.3.1 Introduction

This section documents the results and conclusions of the evaluations performed to investigate the impact of the SGTP Program on the Cook Nuclear Plant Unit 1 reactor vessel internals. In order to assess the impact of the SGTP Program, the following evaluations were performed.

- Review and Evaluation of Thermal Transients
- Review and Evaluation of Power Level
- Thermal-Hydraulic Analysis - The analyses included:
  - Evaluation of the effects on core bypass flow
  - Pressure drop distribution in the reactor vessel
  - Component hydraulic lift forces
- Mechanical System Evaluations which include:
  - Asymmetric flow evaluation
  - Flow induced vibrational
- Components Thermal Stress and Fatigue Evaluations

3.11.3.2 Thermal Transients and Power Level Review

<u>Thermal Transients</u>

Per Section 2.2, the thermal transients and number of occurrences used in the Rerating Program remain unchanged; therefore the thermal transients evaluation for the Rerating Program remain applicable.

<u>Power Level</u>

The power level for the SGTP Program per Table 2.1-1 is 3250 MWt reactor power which is the original design basis for Cook Nuclear Plant Unit 1. The Rerating Program used a power of 3588 MWt reactor power. A change in power level will affect the thermal loads on various reactor internals components such as:

- Lower Core Support Structure
- Baffle-Barrel Region
- Thermal Shield

The decrease in power level from 3588 MWt to 3250 MWt will not have an adverse effect on the structural evaluation performed for the above components.

3.11.3.3 Thermal-Hydraulic Analyses

Thermal-hydraulic analyses, as part of the reactor internals qualification for the SGTP Program were performed. The thermal-hydraulic analyses input parameters were taken from Table 2.1-1. Four different conditions were evaluated using the input parameters for Cases 2 and 3 from Table 2.1-1 (Case 2 @ 2250 psi, Case 2 @ 2100 psi, Case 3 @ 2250 psi, and Case 3 @ 2100 psi). The most conservative results for pressure drops, lift forces, and core bypass flow was obtained using Case 2 input parameters @ 2250 psi from Table 2.1-1.

Core Bypass Flow

Core bypass flow is defined as the total amount of reactor coolant flow that bypasses the core region, and is not considered effective in the core heat transfer process. Consequently, the effect of increasing bypass flow is a reduction in core power capability. Evaluations show that the input parameters from Case 2 of Table 2.1-1, provide results with the highest total bypass flow of 4.4%. The resulting total core bypass flow is still within the allowable limit of 4.5% specified for the SGTP Program.

System Pressure Losses and Hydraulic Lift Forces

The Rerating Program used a conservative evaluation for the system pressure losses, and hydraulic lift forces. The system pressure losses and hydraulic lift forces for the SGTP Program are considered bounded by the Rerating Program. The input parameters, which influence the system pressure losses and hydraulic lift forces for the SGTP Program are the same or lower than those used for the Rerating Program. The SGTP parameters will yield system pressure losses and hydraulic lift forces which are bounded by those used in the Rerating Program. The only parameter which would increase the system pressure losses and hydraulic lift forces is the change in power level, and the change in power level is considered to have an insignificant effect for these parameters. Therefore, the SGTP Program does not have an adverse effect on the system pressure losses and hydraulic lift forces.

3.11.3.4 Mechanical System Evaluation

Flow Induced Vibration

The parameters which can influence the flow induced vibration characteristics of the reactor vessel internals is the flow and temperature. The mechanical design flow for the SGTP Program are not changing, only the thermal design flow is changing and it is decreasing. The mechanical design flow is unchanged and the temperature range is enveloped by the temperature range in the Rerating Program. Therefore, flow induced vibration will not be

adversely impacted by the SGTP Program since the flow induced vibration loadings are enveloped by the work performed for the Rerating Program.

Asymmetric Flow Evaluation

The effect of asymmetric flow on the reactor vessel internals was evaluated. The asymmetric flow loads from Table 2.1-1 were used to evaluate the effect of the new flow condition. Test data from another plant was also used since the internals for Cook Nuclear Plant Unit 1 are similar to the internals of that plant. The evaluation concluded that the maximum displacements for the core barrel beam mode n=1, and shell modes n=2 and n=3 were enveloped by the test data. Therefore, the assymmetric flow condition is considered acceptable.

3.11.3.5 Component Evaluation

Reactor Internals Thermal/Stress and Fatigue Evaluation

The reactor internals thermal/stress and fatigue evaluation was performed by using the Rerating Program evaluation as the last qualified operating conditions and evaluating the change in loadings due to the SGTP Program on reactor internals. Loadings which can impact the evaluation performed are:

- Thermal Transients and the Number of Occurrences
- Power Level
- Gamma Heating Rates
- Mechanical Loadings
- Flow Rates
- Seismic Loads (OBE)
- Operating Temperature

The new loadings for the reactor vessel internals are evaluated in the following section.

Load Evaluation

The NSSS design transients, Section 2.2, for the SGTP Program remain the same as those previously analyzed for the Rerating Program. Seismic loads, mechanical loads, and gamma heating rates are not affected by steam generator tube plugging. The operating temperatures from Table 2.1-1 were chosen such that they would be enveloped by the operating temperatures used in the Rerating Program. Therefore, for this evaluation, power level and flow rates are the only parameters which are changing that can affect the reactor vessel internals evaluation. The reduction in power level, from 3588 MWt to 3250 MWt, will decrease the gamma heating levels for various reactor vessel components. A lower gamma heating value will cause the metal temperature due to gamma heating to decrease, which will bring

the metal temperature closer to the fluid temperature. This will cause a smaller thermal gradient on the various portions of the reactor vessel internals which are affected by gamma heating. The smaller thermal gradient will cause a lower stress in the affected components. Since the reduced power level will result in a lower stress state, the previous evaluation for the Rerating Program is considered bounding for thermal stress and fatigue.

The reduction in thermal design flow is approximately 6% for the normal operating conditions. The temperature range for the 30% SGTP are within the range of temperatures evaluated and are considered bounded by the Rerating Program. Since there is only a small change in flow rate, this will cause the pressure drop in the reactor vessel to decrease. The decreased pressure drop will result in a smaller calculated stress level for various components, (core barrel and baffle-former plates). The reduction in Thermal Design Flow translates a 5% or less reduction in the forced convection heat transfer coefficients. A 5% reduction in the film coefficients is not expected to significantly affect the heat convection between the reactor vessel internals and the reactor coolant. Therefore, the thermal evaluation performed for the Rerating Program is considered applicable.

Conclusion

The reactor vessel internals stress and fatigue evaluation is considered bounded by the Rerating Program evaluation. The SGTP Program does not have an adverse effect on the reactor vessel internals since the loads which are changing are actually improving the margins for the reactor vessel internals when compared to the Rerating Program results.

Rod Drop Time

An assessment was made to confirm the present RCCA drop time limit of 2.4 seconds remains applicable with the SGTP Program conditions. Based on the analysis performed, it is concluded that the 2.4 second RCCA drop time remains applicable.

3.11.4 Control Rod Drive Mechanisms

An evaluation was performed to evaluate the effects of 30% SGTP for Donald C. Cook Nuclear Plant Unit 1. A review of the NSSS Performance Parameters, given in Table 2.1-1, shows that these conditions are bounded and have been evaluated. Since the NSSS Performance Parameters and the NSSS Design Transients for the SGTP Program are bounded by those of the Rerating Program, the conclusion of the generic analysis performed for the Rerating Program remains valid for the SGTP Program.

Varying the hot leg reactor coolant temperature will have no impact on the structural and thermal analysis of the CRDM. Varying the hot leg temperature will affect certain CRDM material properties. However, the effect will be insignificant. Since the hot leg temperature range for the SGTP Program is within the bounds of the hot leg temperature range for the

Rerating Program, the Rerating Program analysis continues to apply and the design requirements for the CRDM pressure boundary are still met.

3.11.5 Reactor Coolant Pumps and Motors

The Model 93A reactor coolant pumps (RCPs) used in the Cook Nuclear Plant Unit 1 NSSS were reviewed to determine the impact of the NSSS parameters for the SGTP Program provided in Table 2.1-1 of this report. Because the NSSS parameters for the SGTP Program are bounded by those of the Rerating Program and the NSSS design transients are also bounded by the Rerating Program, no additional thermal or structural analysis were required to demonstrate compliance with the codes and standards in effect at the time of the original contract.

Varying the cold leg temperature from 536.3°F up to 543.2°F or down to 517.2°F will have no impact on the structural and thermal analysis of the reactor coolant pump. The only difference is related to the material properties used in the analysis. The difference in material properties at the subject temperature is considered negligible. Therefore, the design requirements for the RCP pressure boundary are still met.

The RCP motors were evaluated to determine the worst case loading. The performance of the motors at these loads has been evaluated and the results are as follows:

1. Continuous operation at the new hot loop rating at 6420 HP.

   This represents a 7.0% increase over the nameplate rating of the motor. The change in stator winding temperature resulting from the increase will be less than 5°C. Original test data indicates that with this temperature increase included, the NEMA design limits for a Class B winding will not be exceeded. Therefore, continuous operation of the motors under hot loop conditions with 30% SGTP is acceptable.

2. Operation at the new cold loop rating of 8020 HP.

   The revised load represents a 6.9% increase over the nameplate rating of the motor. Analysis indicates this load increase will cause the stator winding temperature to increase about 7°C. The resulting winding temperature will be less than the Class F NEMA design limits. Therefore, operation of the motors under cold loop conditions with 30% SGTP is acceptable.

3. Starting with revised load torque under the worst case conditions (maximum reverse flow, cold loop, 80% voltage).

The increase in rotor cage winding temperature due to the increased load is small and the total winding temperature is well below the design limit. Therefore, starting under the worst case scenario is acceptable.

The review for the SGTP Program of the Cook Nuclear Plant Unit 1 reactor coolant pumps demonstrates that the SGTP conditions are acceptable for the Model 93A RCP. The design requirements of the RCP pressure boundary are still met. The RCP Motor evaluation determined that the Cook Nuclear Plant Unit 1 motors are acceptable for operation with the 30% SGTP conditions.

3.11.6 Pressurizer

3.11.6.1 Introduction

The functions of the pressurizer are to absorb any expansion or contraction of the primary reactor coolant due to changes in temperature and pressure and to keep the RCS at the desired pressure. The first function is accomplished by keeping the pressurizer approximately half full of water and half full of steam at normal conditions, connecting the pressurizer to the RCS at the hot leg of one of the reactor coolant loops and allowing inflow or outflow to or from the pressurizer as required. The second function is accomplished by keeping the temperature in the pressurizer at the water saturation temperature (T_{SAT}) corresponding to the desired pressure. The temperature of the water and steam in the pressurizer can be raised by operating electric heaters at the bottom of the pressurizer and can be lowered by introducing relatively cool water spray into the steam space at the top of the pressurizer.

The limiting locations from a structural standpoint on the pressurizer are the surge nozzle, the spray nozzle, and the upper shell at the point of spray impingement. The limiting operating condition (relative to the SGTP conditions) of the pressurizer occurs when the RCS pressure is high and the RCS hot leg temperature (T_{HOT}) and cold leg temperature (T_{COLD}) are low. This is explained as follows: Due to inflow and outflow to and from the pressurizer during various transients the surge nozzle alternately sees water at the pressurizer temperature (T_{SAT}) and water from the RCS hot leg at T_{HOT}. If the RCS pressure is high (which means that T_{SAT} is high) and T_{HOT} is low, then the surge nozzle will see maximum thermal gradients and thus experience the maximum thermal stress. Likewise the spray nozzle and upper shell temperatures alternate between steam at T_{SAT} and spray which for many transients is at T_{COLD}. Thus, if RCS pressure is high (T_{SAT} is high) and T_{COLD} is low, then the spray nozzle and upper shell will also experience the maximum thermal gradients and thermal stresses.

3.11.6.2 Description of Analysis and Results

The updated analysis performed for the Donald C. Cook Nuclear Plant Unit 1 SGTP Program for the pressurizer is based on the NSSS design transients provided for the Rerating Program. The design transients are also applicable for the SGTP Program (see Section 2.2).

The analysis was performed by modifying the original Cook Pressurizer analysis (Reference 1), which was performed to the requirements of the ASME Code, 1968 Edition (Reference 2). The original analysis was performed using finite element techniques. Finite element models were constructed for the various parts of the pressurizer. These were then subjected to the pressure loads, external loads (such as piping loads on the nozzles) and thermal transients. The models then calculate the primary, secondary and peak stresses for the various conditions.

The pressurizer maximum pressure and maximum external loads did not increase due to the SGTP Program. Thus, the primary stresses from the original analysis are still valid. Also, the conditions that cause maximum primary plus secondary stress (inadvertent auxiliary spray for spray nozzle and upper shell, and DBE for the surge nozzle) have not changed. Therefore, the only ASME Code requirement affected by the transient modifications was fatigue. The fatigue usage factors are shown in Table 3.11-3 for the critical components.

3.11.6.3 Conclusions

A fatigue analysis was performed for the Cook Nuclear Plant Unit 1 pressurizer, incorporating the most conservative conditions of the SGTP Program. The results of this analysis demonstrate that the pressurizer remains in compliance with the applicable ASME Code criteria.

3.11.6.4 References

1. Model 51 Series Pressurizer Report, Westinghouse Electric Corporation, October 1974.

2. ASME Boiler and Pressure Vessel Code, 1968 Edition, Section III, Article 4.

3.11.7 Reactor Coolant Loop Piping and Supports

As part of the Rerating Program, an evaluation of the reactor coolant loop piping, primary equipment nozzles, and the primary equipment supports was performed for a set of thermal parameter cases that included 15% SG tube plugging. The program reported results for a rerating and also addressed a number of additional cases to cover various temperatures in the loop piping. In that report the analysis for the loop piping, the primary equipment nozzles, and the primary equipment supports were reconciled to the Rerating Program as well as the SGTP Program conditions.

The 30% SGTP loop piping temperatures are enveloped on both the lower and upper ends by the loop piping temperatures already considered in the Rerating Program. The LOCA hydraulic forcing functions generated for the Rerating Program bound the proposed 30% SGTP conditions. The NSSS thermal design transients are applicable for both the Rerating and SGTP Program conditions. The Rerating Program transients and the plant parameters associated with both the Rerating and the SGTP Programs were reviewed for impact on the WCAP-14070 (Reference 1) evaluation for NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to the Reactor Coolant System". The WCAP specifically addressed the auxiliary spray piping. The defined transients are primary loop piping transients and are far enough removed from the auxiliary spray piping to have a negligible impact. The operating parameters for both the Rerating and the SGTP Program conditions have normal operating cold leg temperatures that deviate from the existing design basis values. The range of normal operating cold leg temperatures have been reviewed for impact on the NRC Bulletin 88-08 evaluation and were found to have no impact on the conclusions stated in WCAP-14070.

The Rerating Program transients for Cook Nuclear Plant Unit 1 were reviewed for potential impact on the existing evaluation for the pressurizer surge line thermal stratification analysis. The report that was prepared to demonstrate compliance with NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification" is WCAP-12850 (Reference 2). The reconciliation of the referenced transients applies to both the Rerating Program and the SGTP Program because the transients cover both programs. The results of the evaluation indicate that the fatigue usage factor increases by a small amount (from 0.275 to 0.277). Since the maximum usage factor reported in the WCAP was rounded to a value of 0.30, the result does not change. Because the allowable fatigue usage factor is 1.0, the results are acceptable. As part of the surge line stratification analysis, a set of pressurizer nozzle loadings due to stratification was used as input to the pressurizer evaluation. Our evaluation shows that the changes to the pressurizer nozzle loadings are not significant, and need not be evaluated further (there are no increases greater that 2% and load decreases were ignored).

In conclusion, the reactor coolant loop piping, the primary equipment nozzles, and the primary equipment support loads are acceptable for the SGTP Program conditions because these conditions are already enveloped by the evaluation performed for the Rerating Program. All design basis analysis performed for these components applies to the 30% SGTP condition.

The Rerating Program transients and plant parameters associated with the Rerating and the SGTP Programs for Donald C. Cook Nuclear Plant have been reviewed, and the impact on the design basis analysis for the NRC Bulletin 88-08 evaluation of the auxiliary spray piping and NRC Bulletin 88-11 evaluation of the pressurizer surge line piping is insignificant.

References

1. WCAP-14070, "Evaluation of Donald C. Cook Units 1 and 2 Auxiliary Spray Piping per NRC Bulletin 88-08," July 1994.

2. WCAP-12850, "Structural Evaluation of Donald C. Cook Nuclear Plant Units 1 and 2 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," January 1991.

3.11.8 Auxiliary Components

The auxiliary components (pumps, valves, tanks and heat exchangers) were reviewed to determine the impact of the NSSS parameters for the SGTP Program, provided in Table 2.1-1 of this report. Because the NSSS parameters of the SGTP Program are bounded by those of the Rerating Program and the Auxiliary Equipment Transients are either unchanged or still bounded, there is no effect on the auxiliary components of Cook Nuclear Plant Unit 1.

TABLE 3.11-1
PERFORMANCE CHARACTERISTICS AT 3262 MWT

Parameter	Design Value	Maximum Steam Pressure	Minimum Steam Pressure
Steam Pressure (psia)	812	820	610
Circulation Ratio	5.49	5.45	5.42
Damping Factor (hr^{-1})	-453	-434	-482
Secondary Mass (lbm x 10^{-3})	113	113	108

TABLE 3.11-2
ASSUMED OPERATING PARAMETERS FOR REACTOR VESSEL
STRUCTURAL EVALUATION FOR COOK NUCLEAR PLANT UNIT 1

Design Pressure (psig)	2485
Normal Operating Pressure (psig)	
Upper Bound	2235
Lower Bound	1985
Design Temperature (°F)	650
*Normal Operating Vessel Inlet Temperature (°F)	511.7
*Normal Operating Vessel Outlet Temperature (°F)	615.2
Zero Load Temperature, (°F)	547

**Design Life: The design life of the reactor vessel is 40 years. The design life is the period of anticipated plant service which is used as a basis for defining the number of occurrences of design transients and external loads to be used in the design fatigue analysis. The design life is not to be considered as a warranty but is used strictly for determining fatigue usage factors for the reactor vessel components.

* The reactor vessel is analyzed to operate with normal operating vessel inlet temperatures (T_{cold}) from 511.7°F to 547°F and normal operating vessel outlet temperatures (T_{hot}) from 582.3°F to 615.2°F.

** The reactor vessel closure studs were analyzed for fatigue usage assuming a normal vessel outlet temperature of 599.3°F for the first 10 years of operation and the maximum normal vessel outlet temperature of 615.2°F for the remaining 30 years.

TABLE 3.11-3
DONALD C. COOK 1 PRESSURIZER COMPONENTS, CALCULATED
FATIGUE USAGES CONSIDERING 30% SGTP

COMPONENT	FATIGUE USAGE
Surge Nozzle	0.3323
Spray Nozzle	0.99
Safety and Relief Nozzle	0.148
Seismic Analysis	–
Lower Head - Heater Well	0.07
Lower Head Perforations	0.0165
Upper Head and Shell	0.97
Support Skirt/Flange	0.011
Heater Vibrations	–
Baffle Vibrations	–
Support Lug	0.048
Manway	0.0
Instrument Nozzle	0.1084
Immersion Heater	0.004
Valve Support Bracket	0.01

3.12 FUEL STRUCTURAL EVALUATION

Evaluations were performed of the fuel for Cook Nuclear Plant Unit 1 under the Rerating Program in the areas fuel rod and fuel assembly structural integrity, core design and thermal-hydraulic design. These evaluations assumed a maximum core power level of 3250 MWt and the associated range of operating conditions from Table 2.1-1.

3.12.1 Fuel Assembly Structural Evaluation

Fuel assemblies are designed to perform as described in the Technical Specifications. The combined effects of design basis loads are considered in the verification of the fuel assembly and its components to maintain the fuel assembly structural integrity. This is necessary so that the fuel assembly functional requirements are met, the core coolable geometry is maintained, and the reactor core can be shut down safely.

A structural evaluation of the fuel assembly was performed for the SGTP Program for Cook Nuclear Plant Unit 1, considering the range of operating parameters described in Table 2.1-1. This evaluation assumed 15 x 15 optimized fuel for Unit 1.

The summary of the maximum LOCA and DBE grid load results are presented in Table 3.12-1 with consideration of the requirement of grid load combination, the SRSS of the DBE and LOCA maximum loads is less than 2040 lbs. This maximum load is 33.6% of the grid strength for the 15 x 15 OFA fuel assembly design. Thus, the 15 x 15 OFA design has ample margin for resisting faulted conditional loading. The fuel assembly design is structurally acceptable for Donald C. Cook Nuclear Plant Unit 1.

In conclusion, the SGTP Program for Cook Nuclear Plant Unit 1 does not significantly increase the operating and postulated transient loads such that they will adversely affect the fuel assembly functional requirements. The fuel assembly structural integrity is not affected and the core coolable geometry is maintained for the assumed fuel type for Cook Nuclear Plant Unit 1.

3.12.2 Fuel Rod Structural Evaluation

An evaluation was performed under the SGTP Program of the impact of NSSS performance parameters in Table 2.1-1 on the ability of fuel to satisfy fuel rod design criteria for Cook Nuclear Plant Unit 1. While fuel rod design analyses are not directly impacted by steam generator tube plugging levels, they are sensitive to core inlet temperature, mass flow rates, and other related parameters. Table 3.13-2 provides a comparison of the parameters assumed for the Rerating Program against those of the SGTP Program.

A review of the thermal models indicates that the ~5% reduction in power, or heat flux will generally offset the ~6% reduction in mass flow rate, especially when combined with the 2.7°F

reduction in the maximum inlet temperature. Other fuel performance models, e.g., fission gas release, thermal creep, etc., dependent upon the core power and fuel temperatures, will also be offset by these effects. As a result, fuel rod design analyses performed for the 30% SGTP parameters would not be anticipated to be more limiting than the Rerating Program analyses for any of the impacted fuel rod design criteria, and the conclusions of Rerating Program will remain valid for SGTP Program for Donald C. Cook Nuclear Plant Unit 1.

Finally, as in the past, cycle-specific fuel performance analyses will continue to be performed for each fuel region to confirm that this assessment, and all fuel rod design criteria, are satisfied for the operating conditions specific to each cycle of operation. These evaluations support the Reload Safety Evaluation (RSE), which is transmitted to AEPSC prior to each cycle of operation.

3.12.3 Core Design

The results of the core design evaluation indicated that the increased steam generator tube plugging level and reduced Thermal Design Flow result in no impacts to the core design except for the values of the statepoint for the Steamline Break Analysis, and the Dropped Rod Analysis. See Section 3.3 for the new statepoint values for the Steamline Break Analysis. See Section 3.12.4 for the new limits concerning Dropped Rod Limit Lines.

3.12.4 Thermal Hydraulic Design

3.12.4.1 Purpose of Analysis

The purpose of this section is to describe the thermal-hydraulic analysis necessary to support the decrease in flow associated with an increase in SGTP level to 30% over a range of RCS temperatures.

3.12.4.2 Assumptions

Table 3.12-3 summarizes the thermal-hydraulic design parameters used in this analysis. The core inlet temperature is consistent with the high temperature 30% SGTP case. Use of high inlet temperature bounds the range of RCS Tavg with regard to the Departure from Nucleate Boiling (DNB) analysis. Included in Table 3.12-3, for comparison, are the thermal-hydraulic parameters currently in the Donald C. Cook Nuclear Plant Unit 1 Safety Analysis.

3.12.4.3 Discussion of Evaluation

3.12.4.3.1 Calculation Methods

The thermal hydraulic design criteria and methods remain the same as those presently in the Donald C. Cook Nuclear Plant Unit 1 UFSAR with the exceptions described in the following paragraphs.

DNB Methodology

The existing thermal-hydraulic analyses use the Improved Thermal Design Procedure (ITDP) (Reference 1). For this methodology, 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 at a 95 percent confidence level that the minimum DNBR of the limiting fuel rod is greater than or equal to the DNBR limit of the DNB correlation being used. Plant parameter uncertainties are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR limit, establishes Design Limit DNBR values which must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the Design Limit DNBR values, the plant safety analyses are performed using values of input parameters without uncertainties. In addition, the Design Limit DNBR values are increased to values designated as the Safety Analysis Limit DNBRs. The DNBR margin available between the Safety Analysis Limit DNBR values and the Design Limit DNBR values is used to offset DNBR penalties.

The analysis of the 30% SGTP conditions uses the Revised Thermal Design Procedure (RTDP) (Reference 2). This methodology gives improved DNBR performance over ITDP by statistically combining the DNB correlation uncertainties with the ITDP uncertainties listed above, i.e., uncertainties in plant operating parameters (vessel coolant flow, core power, coolant temperature, system pressure and effective core flow fraction), nuclear and thermal parameters ($F^N_{\Delta H}$), fuel fabrication parameters ($F^E_{\Delta H,1}$), THINC IV, and transient codes. The uncertainty factor obtained is used to define the Design Limit DNBR which satisfies the DNB design criterion. The DNB design criterion is that the probability that DNB will not occur on the most limiting fuel rod is at least 95 percent at a 95 percent confidence level during normal operation and operational transients (Condition I events) and during transient conditions arising from faults of moderate frequency (Condition II events). Condition I and II events are defined in ANSI 18.2.

As was done with ITDP, the design limit DNBR values are increased to values designated as the Safety Analysis Limit DNBR and the DNBR margin available between these limits is used for flexibility of design and operation of the plant and to offset DNBR penalties such as rod bow. The DNBR limits, current penalties, and margin associated with RTDP analysis are listed in Table 3.12-4.

THINC IV Modeling

An improved THINC IV model was used in the DNB analysis of this core. This model is described in Reference 3 and has been approved for use by the NRC.

3.12.4.3.2 Design Evaluation

DNB Performance

The change in design parameters in going from the current analysis to the 30% SGTP conditions included decreasing the power, flow and inlet temperatures as shown in Table 3.12-3. This affects the DNB performance of the core. The DNB methodology was changed from ITDP to RTDP to generate DNBR margin. The DNBR Safety Analysis Limits (Table 3.12-4) were set to keep the DNBR limiting portion of the core limits unchanged. The associated axial offset limits were recalculated. The DNB events not protected by core limits that were analyzed were Loss of Flow, Locked Rotor, Static Rod Misalignment, Dynamic Dropped Rod (RCCA), and RCCA Bank Withdrawal from subcritical (Rod Withdrawal from Subcritical). The results of the analyses showed that the thermal hydraulic design criteria were met for each event.

Fuel Temperatures

The limiting values of the fuel average and centerline temperatures will not change due to the 30% SGTP conditions.

3.12.4.4 Conclusions

Thermal-hydraulic analyses were made for the fuel for the limiting 30% SGTP parameters using RTDP methodology. The analysis showed that the DNBR design basis was met for the limiting DNB events. This analysis caused the available DNBR margin to increase. This margin can be used for flexibility of design and to offset unanticipated DNBR penalties.

3.12.4.5 References

1. WCAP-8567-P-A, "Improved Thermal Design Procedure," H. Chelemer, L. H. Boman, D. L. Sharp, February, 1989.

2. WCAP-8567-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1989.

3. WCAP-12330-P, "Improved THINC IV Modelling for PWR Core Design," A. J. Friedland and S. Ray, August, 1989.

TABLE 3.12-1
MAXIMUM LOCA AND DBE GRID LOAD RESULTS

Case	Grid Load (lbs) X	Grid Load (lbs) Z	Grid Strength
Accumulator	394	75	> 6080 lbs.
DBE	< 2000		
SRSS (DBE & LOCA)	< 2040		< 33.6% of grid strength

TABLE 3.12-2
FUEL ROD DESIGN ANALYSIS PARAMETERS

Parameter	Rerating Program	SGTP Program
Core Power, MWt	3413	3250
Minimum System Pressure, psia	2100	2100
Maximum Inlet Temperature, °F	546.2	543.5
Thermal Design Flow, gpm	354,000	332,800
Bypass Flow, %	4.5	4.5
FDH	1.55	1.55

TABLE 3.12-3
DONALD C. COOK NUCLEAR UNIT 1 30% SGTP PROGRAM
THERMAL AND HYDRAULIC DESIGN PARAMETERS

	Design Parameters			
	Current Analysis		30% SGTP Program[b]	
Reactor Core Heat Output, MWt	3413[a]		3250	
Reactor Core Heat Output, 10^6 Btu/hr	11,646		11,090	
Heat Generator in Fuel, %	97.4		97.4	
Pressurizer Pressure, Nominal, psia	2100		2100	
Radial Power Distribution	1.55 [1+0.3(1-P)]		1.55[1+0.3(1-P)]	
Limit DNBR for Design Transients Flow Channel	Typical 1.45	Thimble 1.45	Typical 1.40	Thimble 1.42
DNB Correlation	WRB-1		WRB-1	
HFP Nominal Conditions				
Vessel Thermal Design Flow, 10^6 lbm/hr	133.4		125.9	
Core Flow Rate, 10^6 lbm/hr	127.4		120.3	
Bypass Flow, %	4.5		4.5	
Normal Vessel/Core Inlet Temp, °F	546.4		543.5	
Vessel Average Temp, °F	578.7		576.3	
Core Average Temp, °F	581.8		579.4	
Vessel Outlet Temp, °F	611.0		609.1	
Average Temp Rise in Vessel, °F	64.6		65.6	
Average Temp Rise in Core, °F	67.3		68.4	
Heat Transfer[c]				
Average Heat Transfer Area, ft^2	52,200		52,200	
Average Heat Flux, Btu/hr-ft^2 [c]	217,400		207,000	
Average Linear Power, kw/ft[c]	7.04		6.70	
Peak Linear Power for Normal Operation, kw/ft[c]	16.5[d]		15.7	
Temperature at Peak Linear Power for Prevention of Centerline Melt, °F	4700		4700	

(a) Cook Nuclear Plant Unit 1 is currently licensed to operate at 3250 MWt
(b) High inlet temperature bounds the proposed temperature range with respect to DNB
(c) Based on nominal 144 inch active fuel length
(d) Based on 2.35 F_Q Peaking Factor

TABLE 3.12-4
DONALD C. COOK NUCLEAR PLANT UNIT 1 SGTP PROGRAM
RTDP DNBR LIMITS AND MARGIN SUMMARY

DNB Correlation	WRB-1	
Cell Type	Typical	Thimble
Design Limit	1.23	1.22
Safety Analysis Limit	1.40	1.42
Total DNBR Margin	12.1	14.1
DNBR Penalty - Rod Bow	2.6	2.6
1°F Temperature Bias	1.5	1.5
Net Remaining DNBR Margin	8.0	10.0

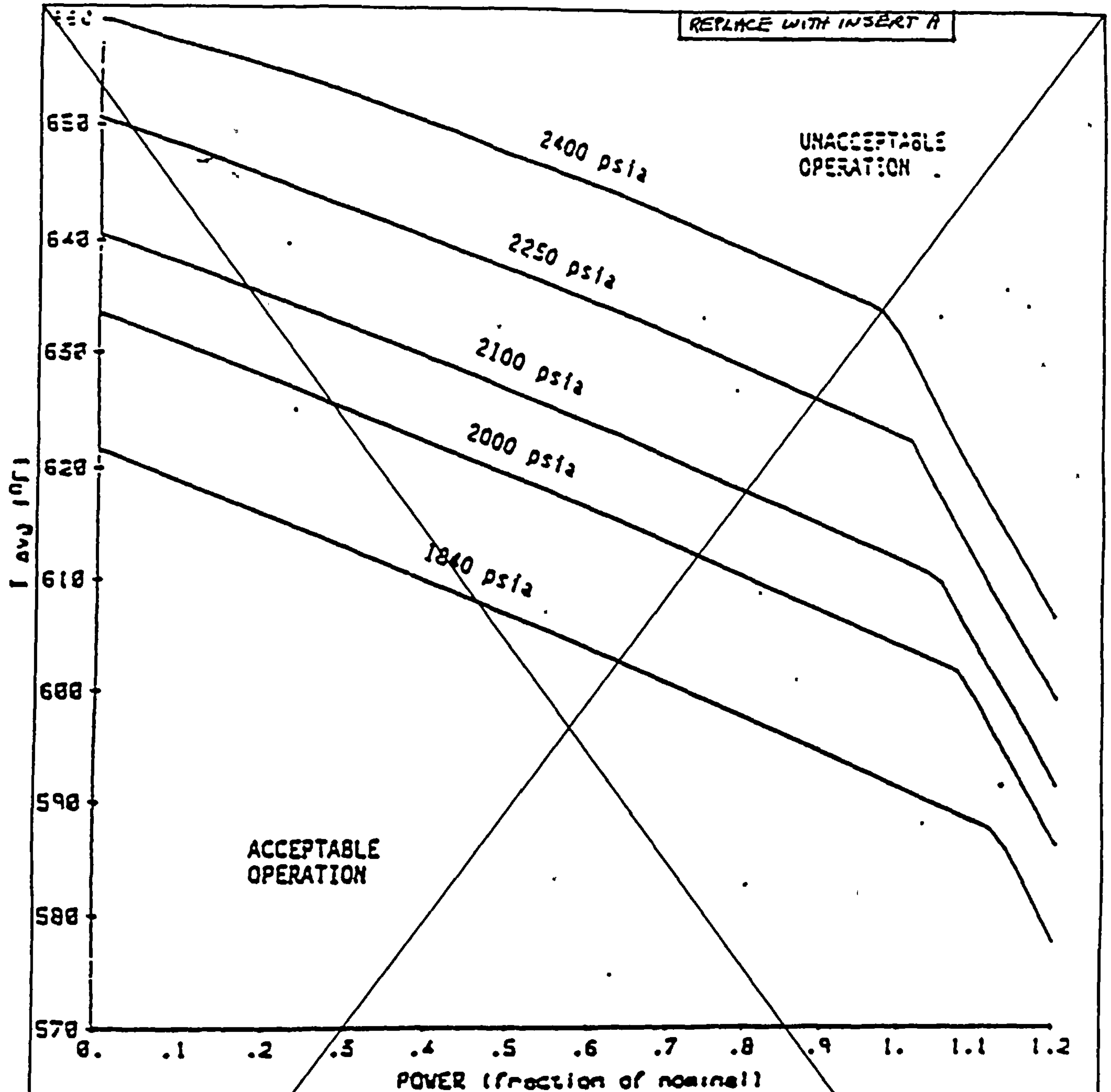
4.0 CONCLUSIONS

Provided in this document are the results and conclusions of the safety analyses and evaluations to support the implementation of the SGTP Program and the revised Technical Specification changes for Cook Nuclear Plant Unit 1. The safety analyses, evaluations, and supporting documentation provided in this submittal demonstrate acceptable results in each case, incorporating the revised operating conditions associated with the SGTP Program. A brief summary of the results of each analysis and evaluation is provided in the "Summary and Conclusions" section of this report.

APPENDIX A

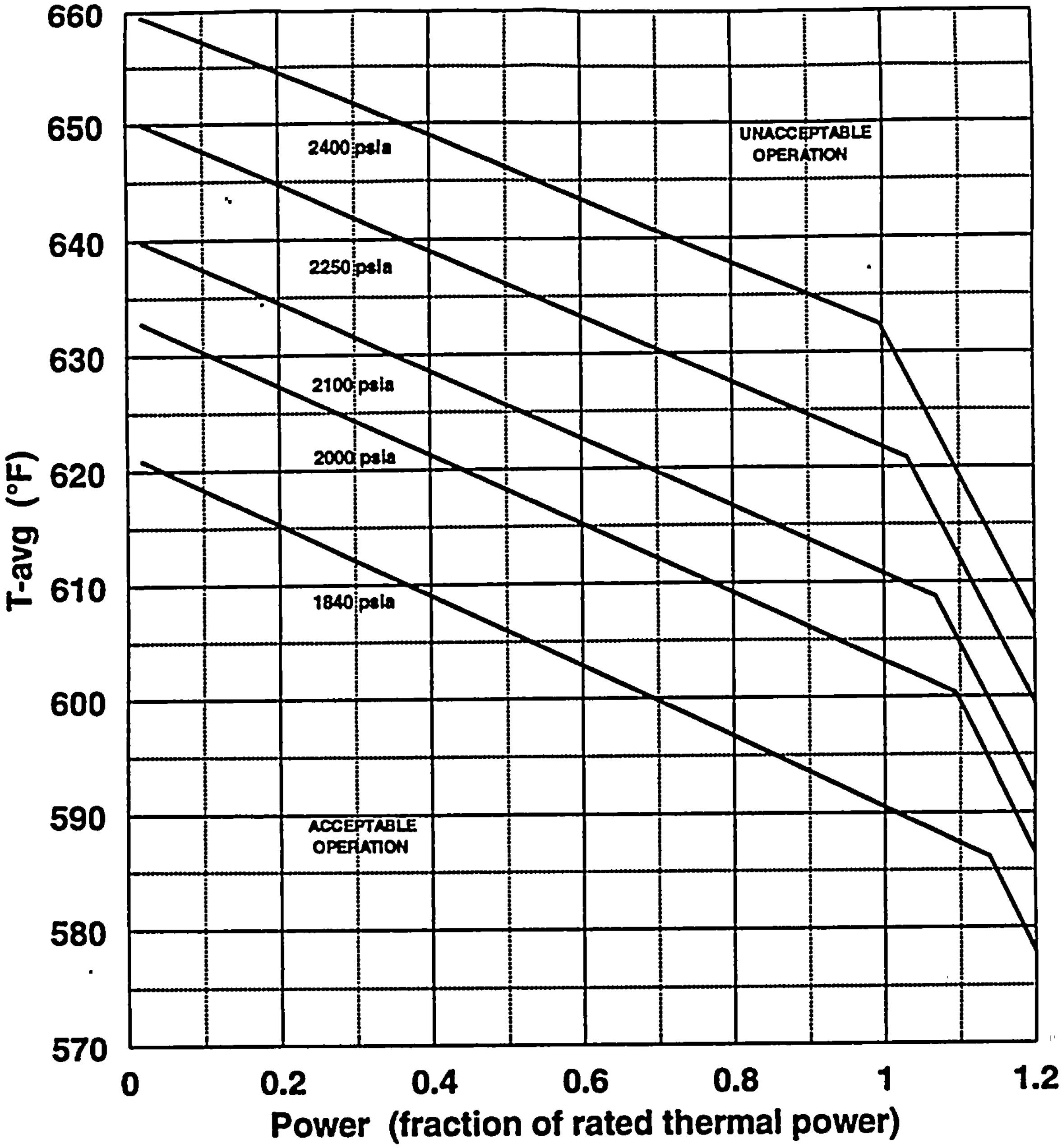
Proposed Technical Specification Changes

REPLACE WITH INSERT A



PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T-AVG IN DEGREES F)
1840	(0.0, 622.1), (1.13, 587.3), (1.20, 577.5)
2000	(0.0, 633.8), (1.08, 601.4), (1.20, 586.0)
2100	(0.0, 640.8), (1.06, 609.8), (1.20, 591.3)
2250	(0.0, 650.7), (1.02, 621.9), (1.20, 598.9)
2400	(0.0, 660.1), (0.98, 633.7), (1.20, 606.2)

FIGURE 2.1-1 REACTOR CORE SAFETY LIMITS



PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T-AVG IN °F)		
1840	(0.02, 620.86),	(1.136, 586.17),	(1.2, 577.94)
2000	(0.02, 632.79),	(1.094, 600.31),	(1.2, 586.52)
2100	(0.02, 639.85),	(1.068, 608.72),	(1.2, 591.77)
2250	(0.02, 649.96),	(1.031, 620.83),	(1.2, 599.40)
2400	(0.02, 659.52),	(0.996, 632.42),	(1.2, 606.63)

FIGURE 2.1-1 REACTOR CORE SAFETY LIMITS

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - less than or equal to 25% of RATED THERMAL POWER High Setpoint - less than or equal to 109% of RATED THERMAL POWER	Low Setpoint - less than or equal to 26% of RATED THERMAL POWER High Setpoint - less than or equal to 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
5. Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3×10^5 counts per second
7. Overtemperature Delta T	See Note 1	See Note 3
8. Overpower Delta T	See Note 2	See Note 4
9. Pressurizer Pressure -- Low	Greater than or equal to 1875 psig	Greater than or equal to 1865 psig
10. Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11. Pressurizer Water Level -- High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12. Loss of Flow	Greater than or equal to 90% of ~~design~~ flow per loop*	Greater than or equal to 89.1% of ~~design~~ flow per loop*

85,275

*~~Design~~ flow is ~~90,400~~ gpm per loop. MINIMUM MEASURED

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left[\frac{1+\tau_1 S}{1+\tau_2 S}\right](T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where:

ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, °F
T'	=	Indicated T_{avg} at RATED THERMAL POWER ($\leq$ ~~567.8~~ 576.3 °F)
P	=	Pressurizer pressure, psig
P'	=	Indicated RCS nominal operating pressure (2235 psig or 2085 psig)
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 22 secs. τ_2 = 4 secs.
S	=	Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Operation with 4 Loops

K_1 = ~~1.12~~ 1.17

K_2 = 0.0230

K_3 = 0.00110

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -37 percent and + ~~2~~ 3 percent, $f_1(\Delta I)=0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) For each percent that the magnitude of $(q_t - q_b)$ exceeds -37 percent, the ΔT trip setpoint shall be automatically reduced by 0.33 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds + ~~2~~ 3 percent, the ΔT trip setpoint shall be automatically reduced by ~~2.17~~ 2.34 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S} \right] T - K_6 (T - T'') - f_2(\Delta I) \right]$

where:

ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, ^oF
T''	=	Indicated T_{avg} at RATED THERMAL POWER ($\leq$ ~~567.8~~ 563.0°F ^oF)
K_4	=	1.083
K_5	=	$0.0177/^oF$ for increasing average temperature and 0 for decreasing average temperature
K_6	=	0.0015 for $T > T''$; $K_6 = 0$ for $T \leq T''$
$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	=	Time constant utilized in the rate lag controller for T_{avg}; τ_3 = 10 secs.
S	=	Laplace transform operator
$f_2(\Delta I)$	=	0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than ~~1.2~~ percent ΔT span.
(AEPSC RESPONSIBILITY)

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than ~~2.1~~ percent ΔT span.

2.1 SAFETY LIMITS

BASES

4 Loop Operation

Westinghouse Fuel
(15x15 OFA)

(WRB-1 Correlation)

	Typical Cell*	Thimble Cell**
Correlation Limit	1.17	1.17
Design Limit DNBR	~~1.33~~ 1.23	~~1.32~~ 1.22
Safety Analysis Limit DNBR	~~1.45~~ 1.40	~~1.45~~ 1.42

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the applicable design DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

----

* represents typical fuel rod

** represents fuel rods near guide tube

SAFETY LIMITS

BASES

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor.. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBR's will be greater than the applicable design limit DNBR value for each fuel type.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The source Range Channels will initiate a reactor trip at about 10^{+5} counts per second, unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature delta T

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. ~~The reference average temperature (T') and the reference operating pressure (P') are set equal to the full power indicated Tavg and the nominal RCS operating pressure, respectively, to ensure protection of the core limits and to preserve the actuation time of the Overtemperature delta T trip for the range of full power average temperatures assumed in the safety analyses.~~ With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

DELETE

LIMITING SAFETY SYSTEM SETTINGS

BASES

Overpower Delta T

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. ~~The reference average temperature (T") is set equal to the full power indicated Tavg to ensure fuel integrity during overpower conditions for the range of full power average temperatures assumed in the safety analysis.~~ DELE The overpower delta T reactor trip provides protection or back-up protection for at power steamline break events. Credit was taken for operation of this trip in the steam line break mass/energy releases outside containment analysis. In addition, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the reactor protection system.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. The pressurizer high water level trip precludes water relief for the Uncontrolled RCCA Withdrawal at Power event.

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - TAVG GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to ~~1.6%~~ 1.3 Delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than ~~1.6%~~ 1.3 Delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~1.6%~~ 1.3 Delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5.

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*See Special Test Exception 3.10.1.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3

a. One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

b. One charging flowpath associated with support of Unit 2 shutdown functions shall be available.*

APPLICABILITY: Specification 3.1.2.3.a. - MODES 5 and 6
Specification 3.1.2.3.b. - At all times when Unit 2 is in MODES 1, 2, 3, or 4.

ACTION:

a. With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.**

b. With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 152°F, unless the reactor vessel head is removed, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within one hour.

c. The provisions of Specification 3.0.3 are not applicable.

d. In addition to the above, when Specification 3.1.2.3.b is applicable and the required flow path is not available, return the required flow path to available status within 7 days, or provide equivalent shutdown capability in Unit 2 and return the required flow path to available status within the next 60 days, or have Unit 2 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours.

e. The requirements of Specification 3.0.4 are not applicable when Specification 3.1.2.3.b applies.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a DIFFERENTIAL ~~discharge~~ pressure of greater than or equal to 2290 ~~2390~~ psi~~g~~d when tested pursuant to Specification 4.0.5.

*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F.

**For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a DIFFERENTIAL ~~discharge~~ pressure of greater than or equal to ~~2405~~ 2290 psi~~g~~d when tested pursuant to Specification 4.0.5.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum usable borated water volume of 4300 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum usable borated water volume of 90,000 gallons,
2. A minimum boron concentration of 2400 ppm, and
3. A minimum solution temperature of ~~80~~ 70°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the water level volume of the tank, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water.

* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

BORATED WATER SOURCES - OPERATIONS

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum usable borated water volume of 5650 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 350,000 gallons of water,
2. Between 2400 and 2600 ppm of boron, and
3. A minimum solution temperature of ~~80~~ 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

TABLE 3.2-1
DNB PARAMETERS

PARAMETER	LIMITS 4 Loops in Operation at RATED THERMAL POWER
Reactor Coolant System Tavg	≤ ~~570.9°F~~* (576.3 + 5.1) - (READABILITY ERROR, AEPSC RESPONSIBILITY)*
Pressurizer Pressure	≥ 2050 psig**
Reactor Coolant System Total Flow Rate	≥ ~~361,600~~ 341,100 gpm***

* Indicated average of at least three OPERABLE instrument loops.

** Limit not applicable during either a THERMAL POWER ramp increase in exce of 5 percent RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10 percent RATED THERMAL POWER.

*** Indicated value.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux (High and Low Setpoint)	Less than or equal to 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	Less than or equal to 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature delta T	Less than or equal to 6.0 seconds*
8. Overpower delta T	Less than or equal to 6.0 seconds*
9. Pressurizer Pressure--Low	Less than or equal to 2.0 ~~1.0~~ seconds
10. Pressurizer Pressure--High	Less than or equal to 2.0 ~~1.0~~ seconds
11. Pressurizer Water Level--High	Less than or equal to 2.0 seconds

* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Loss of Flow - Single Loop (Above P-8)	Less than or equal to 1.0 seconds
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	Less than or equal to 1.0 seconds
14. Steam Generator Water Level--Low-Low	Less than or equal to 2.0 ~~1.5~~ seconds
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	NOT APPLICABLE
16. Undervoltage-Reactor Coolant Pumps	Less than or equal to 1.5 ~~1.2~~ seconds
17. Underfrequency-Reactor Coolant Pumps	Less than or equal to 0.6 seconds
18. Turbine Trip	
A. Low Fluid Oil Pressure	NOT APPLICABLE
B. Turbine Stop Valve	NOT APPLICABLE
19. Safety Injection Input from ESF	NOT APPLICABLE
20. Reactor Coolant Pump Breaker Position Trip	NOT APPLICABLE

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

DELETE

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Flow in Two Steam Lines-High					
Four Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line	1, 2, 3**	14*
Three Loops Operating	2/operating steam line	1***/any operating steam line	1/operating steam line	3**	15
COINCIDENT WITH EITHER					
T_{avg}--Low-Low					
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops	1, 2, 3**	14*
Three Loops Operating	1 T_{avg}/ operating loop	1*** T_{avg} in any operating loop	1 T_{avg} in any two operating loops	3**	15
OR, COINCIDENT WITH					
f. Steam Line Pressure-Low					
Four Loops Operating	1 pressure/ loop	2 pressures any loops	1 pressure any 3 loops	1, 2, 3**	14*
Three Loops Operating	1 pressure/ operating loop	1*** pressure in any operating loop	1 pressure in any 2 operating loops	3**	15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
COINCIDENT WITH ~~EITHER~~					
T_{avg} -- Low-Low					
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops	1, 2, 3**	14*
Three Loops Operating	1 T_{avg}/ operating loop	1*** T_{avg} in any operating loop	1 T_{avg} in any two operating loops	3**	15
~~OR, COINCIDENT WITH~~					
e. Steam Line Pressure-Low					
Four Loops Operating	1 pressure/ loop	2 pressures any loops	1 pressure any 3 loops	1, 2, 3**	14*
Three Loops Operating	1 pressure/ operating loop	1*** pressure in any operating loop	1 pressure in any 2 operating loops	3**	15
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1,2,3	14*

ENGINEERED SAFETY FEATURES INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-11	With 2 of 3 pressurizer pressure channels greater than or equal to 1915 psig.	P-11 prevents or defeats manual block of safety injection actuation on low pressurizer pressure.
P-12	With 2 of 4 T_{avg} channels less than or equal to Setpoint. Setpoint greater than or equal to 541°F	P-12 allows the manual block of safety injection ACTUATION ON ~~from high steam flow coincident with either~~ low steam line pressure. CAUSES STEAM LINE ISOLATION ON HIGH STEAM FLOW. ~~or low-low T_{avg}. P-12 in coincidence with high steam flow will result in a steam line isolation. P-12~~ Affects steam dump blocks. With 3 of 4 T_{avg} channels above the reset ~~value, the manual block of safety injection from high steam flow coincident with either low steam line pressure or low-low T_{avg} is prevented or defeated.~~ POINT, PREVENTS OR DEFEATS THE MANUAL BLOCK OF SAFETY INJECTION ACTUATION ON LOW STEAM LINE PRESSURE.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN FEEDWATER PUMPS		
a.	Manual Initiation	------------- See Functional Unit 9 ------------	
b.	Automatic Actuation Logic	Not Applicable	Not Applicable
c.	Containment Pressure--High	Less than or equal to 1.1 psig	Less than or equal to 1.2 psig
d.	Pressurizer Pressure--Low	Greater than or equal to 1815 psig	Greater than or equal to 1805 psig
e.	Differential Pressure Between Steam Lines--High	Less than or equal to 100 psi	Less than or equal to 112 psi
f.	~~Steam Flow in Two Steam Lines--High Coincident with Tavg --Low-Low or~~ Steam Line Pressure--Low	~~Less than or equal to 1.42×10^6 lbs/hr from 0% load to 20% load. Linear from 1.42×10^6 lbs/hr at 20% load to 3.88×10^6 lbs/hr at 100% load~~ ~~T_{avg} greater than or equal to 541°F~~ G ~~g~~reater than or equal to 500 psig steam line pressure	~~Less than or equal to 1.56×10^6 lbs/hr from 0% load to 20% load. Linear from 1.56×10^6 lbs/hr at 20% load to 3.93×10^6 lbs/hr at 100% load~~ ~~T_{avg} greater than or equal to 539°F~~ G ~~g~~reater than or equal to 480 psig steam line pressure

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Containment Radioactivity--High Train A (VRS-1101, ERS-1301, ERS-1305)	See Table 3.3-6	Not Applicable
3. Containment Radioactivity--High Train B (VRS-1201, ERS-1401, ERS-1405)	See Table 3.3-6	Not Applicable
4. STEAM LINE ISOLATION		
a. Manual	------------- See Functional Unit 9 -------------	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	Less than or equal to 2.9 psig	Less than or equal to 3 psig
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg} --Low-Low ~~or Steam Line Pressure--Low~~	Less than or equal to 1.42×10^6 lbs/hr from 0% load to 20% load. Linear from 1.42×10^6 lbs/hr at 20% load to 3.88×10^6 lbs/hr at 100% load.	Less than or equal to 1.56×10^6 lbs/hr from 0% load to 20% load. Linear from 1.56×10^6 lbs/hr at 20% load to $3.93\ 10^6$ lbs/hr at 100% load.
	T_{avg} greater than or equal to 541°F	T_{avg} greater than or equal to 539°F
e. STEAM LINE PRESSURE-LOW	Greater than or equal to 500 psig steam line pressure	Greater than or equal to 480 psig steam line pressure
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level--High-High	Less than or equal to 67% of narrow-range instrument span each steam generator	Less than or equal to 68% of narrow-range instrument span each steam generator

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
1. Manual		
a.	Safety Injection (ECCS)	Not Applicable
	Feedwater Isolation	Not Applicable
	Reactor Trip (SI)	Not Applicable
	Containment Isolation-Phase "A"	Not Applicable
	Containment Purge and Exhaust Isolation	Not Applicable
	Auxiliary Feedwater Pumps	Not Applicable
	Essential Service Water System	Not Applicable
b.	Containment Spray	Not Applicable
	Containment Isolation-Phase "B"	Not Applicable
	Containment Purge and Exhaust Isolation	Not Applicable
	Containment Air Recirculation Fan	Not Applicable
c.	Containment Isolation-Phase "A"	Not Applicable
	Containment Purge and Exhaust Isolation	Not Applicable
d.	Steam Line Isolation	Not Applicable
2. Containment Pressure-High		
a.	Safety Injection (ECCS)	47.0 Less than or equal to 27.0@@/~~27.0++~~
b.	Reactor Trip (from SI)	Less than or equal to 3.0
c.	Feedwater Isolation	Less than or equal to 8.0
d.	Containment Isolation-Phase "A"	Less than or equal to 18.0#/28.0##
e.	Containment Purge and Exhaust Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	Not Applicable
g.	Essential Service Water System	Less than or equal to 13.0#/48.0##

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	Less than or equal to 27.0##/~~27.0++~~ 47.0
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation-Phase "A"	Less than or equal to 18.0#
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Less than or equal to 48.0++/13.0#
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	Less than or equal to 27.0##/~~37.0#~~ 57.0
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation-Phase "A"	Less than or equal to 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Less than or equal to 13.0#/48.0##
5. Steam Flow in Two Steam Lines - High Coincident with Tavg--Low-Low	NOT APPLICABLE
a. Safety Injection (ECCS)	~~Less than or equal to 29.0##/39.0#~~
b. Reactor Trip (from SI)	~~Less than or equal to 5.0~~
c. Feedwater Isolation	~~Less than or equal to 10.0~~
d. Containment Isolation-Phase "A"	~~Less than or equal to 20.0#/30.0##~~
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	~~Less than or equal to 15.0#/50.0##~~
h. Steam Line Isolation	Less than or equal to 13.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. ~~Steam Flow in Two Steam Lines-High Coincident With~~ Steam Line Pressure-Low	
a. Safety Injection (ECCS)	Less than or equal to 27.0@@/37.0@
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation-Phase "A"	Less than or equal to 18.0*/28.0**
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Less than or equal to 14.0*/48.0**
h. Steam Line Isolation	Less than or equal to 11.0
7. Containment Pressure--High-High	
a. Containment Spray	Less than or equal to ~~45.0~~ 65.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	Less than or equal to 10.0
d. Containment Air Recirculation Fan	Less than or equal to 600.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	Less than or equal to 2.5
b. Feedwater Isolation	Less than or equal to 11.0
9. Steam Generator Water Level--Low-Low	≤80.0**
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
10. 4160 volt Emergency Bus Loss of Voltage	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
11. Loss of Main Feedwater Pumps	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
12. Reactor Coolant Pump Bus Undervoltage	
a. Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0

TABLE 3.3-5 (Continued)

TABLE NOTATION

* Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

** Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

++ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging, SI, and RHR pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is NOT included.

@ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

@@ Diesel generator starting and sequence loading delays NOT included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

i RESPONSE TIMES USED FOR OFFSITE POWER AVAILABLE ANALYSIS.

ii DIESEL GENERATOR STARTING AND SEQUENCE LOADING DELAYS INCLUDED. RESPONSE TIME INCLUDES OPENING OF VALVES TO ESTABLISH FLOW PATH AND ATTAINMENT OF DISCHARGE PRESSURE FOR PUMPS.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Manual Initiation	-------------------	See Functional Unit 9	-------------------		
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure-High	S	R	M(3)	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	S	R	M	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	N.A.	1, 2, 3
f. ~~Steam Flow in Two Steam Lines--High Coincident with Tavg --Low or Steam Line~~ STEAM LINE Pressure--Low	S	R	M	N.A.	1, 2, 3
2. CONTAINMENT SPRAY					
a. Manual Initiation	-------------------	See Functional Unit 9	-------------------		
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure-High-High	S	R	M(3)	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURED ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION					
a. Manual	------------------- See Functional Unit 9 -------------------				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3
c. Containment Pressure--High-High	S	R	M(3)	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with Tavg--Low-Low ~~Pressure--Low~~	S	R	M	N.A.	1, 2, 3
e. STEAM LINE PRESSURE-LOW	S	R	M	N.A.	1,2,3
5. TURBINE TRIP AND FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	S	R	M	N.A.	1, 2, 3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level--Low-Low	S	R	M	N.A.	1, 2, 3
b. 4 kv Bus Loss of Voltage	S	R	M	N.A.	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	N.A.	1, 2, 3
d. Loss of Main Feed Pumps	N.A.	N.A.	R	N.A.	1, 2

REACTOR COOLANT SYSTEM

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± ~~1~~ 3%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE:

a. Immediately suspend all operations involving positive reactivity changes** and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. Immediately render all Safety Injection pumps and all but one charging pump inoperable by removing the applicable motor circuit breakers from the electric power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Requirement 4.4.3.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ±~~1~~3%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional surveillance requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection signal:

a) Centrifugal charging pump

b) Safety injection pump

c) Residual heat removal pump

DIFFERENTIAL

f. By verifying that each of the following pumps develops the indicated ~~discharge~~ pressure on recirculation flow when tested pursuant to Specification 4.0.5.

1. Centrifugal charging pump greater than or equal to ~~2405~~ 2290 psig d

2. Safety injection pump greater than or equal to ~~1409~~ 1326 psig d

3. Residual heat removal pump greater than or equal to ~~190~~ 150 psig d

g. By verifying the correct position of each mechanical stop for the following Emergency Core Cooling System throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 350,000 gallons of borated water.

b. Between 2400 and 2600 ppm of boron, and

c. A minimum water temperature of ~~80~~ 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank,
2. Verifying the fuel level in the fuel storage tank,
3. Verifying that the fuel transfer pump can be started and that it transfers fuel from the storage system to the day tank,
4. Verifying that the diesel starts from ~~ambient~~ STANDBY condition and ~~that it accelerates to at least 514 rpm in less than or equal to 10 seconds,*~~ ACHIEVES IN ≤ 30 SECONDS, VOLTAGE @ 4160 ± 420 VOLTS AND FREQUENCY AT 60 ± 1.2 Hz.
5. Verifying that the generator is loaded to greater than or equal to 1750 kw and that it operates for greater than or equal to 60 minutes and verifying that the generator output breaker to the emergency bus is OPERABLE, and
6. Verifying that the diesel generator is aligned to provide standby power to the associated emergency busses.

b. By removing accumulated water**:

1) From the day tank at least once per 31 days and after each occasion when the diesel is operated for greater than 1 hour, and

2) From the storage tanks at least once per 31 days.

c. By sampling new fuel oil** in accordance with the applicable guidelines of ASTM D4057-81 prior to adding new fuel to the storage tanks and

1) By verifying, in accordance with the tests specified in ASTM D975-81 and prior to adding the new fuel to the storage tanks, that the sample has:

~~*The diesel generator start (10 seconds) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing and compensatory action may be at reduced acceleration rates as recommended by the manufacturer so that mechanical stress and wear on the diesel engine are minimized.~~

**The actions to be taken should any of the properties be found outside of specified limits are defined in the Bases.

2. Verifying the generator capability to reject a load greater than or equal to 600 kw while maintaining voltage at 4160 $\pm$ 420 volts and frequency at 60 $\pm$ 1.2 Hz,

3. Verifying the generator capability to reject a load of 3500 kw without exceeding 75% of the difference between nominal speed and the overspeed trip setpoint,

4. Simulating a loss of offsite power by itself, and:

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses,

   b) Verifying that the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within ~~10~~ 30 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After load sequencing is completed, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz during the test.

5. Verifying that, on a Safety Injection actuation test signal (without loss of offsite power), the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes,

6. Simulating a loss of offsite power in conjunction with a Safety Injection actuation test signal, and by:

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses,

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within ~~10~~ 30 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After load sequencing is completed, the steady state voltage and frequency of the emergency busses shall be 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz. The voltage and frequency shall be maintained within these limits for the remainder of this test, and

SURVEILLANCE REQUIREMENTS (Continued)

b) Agitate the fuel oil in the storage tank while pumping the oil from the bottom of the tank through a 5-micron filter, and back to the opposite end of the tank. Three successive samples shall be taken and analyzed according to ASTM D2276-83. If the contaminant level in any of the samples is greater than 10 mg per liter, the agitation, filtration, and sampling processes shall be repeated. If the contaminant level remains above 10 mg per liter after 3 iterations, the draining and cleaning method described in surveillance requirement 4.8.1.1.2.f.1.a shall be employed.

2) Performing a precision leak detection test to verify that the leakage rate from the fuel oil system is less than or equal to .05 gallons per hour.

3) Starting both diesel generators simultaneously, FROM STANDBY CONDITION during shutdown, and verifying that both diesel generators ~~accelerate to at least 514 RPM in less than or equal to 10 seconds.~~*

ACHIEVE IN ≤ 30 SECONDS, VOLTAGE AT 4160 ± 420 VOLTS, AND FREQUENCY AT 60 ± 1.2 Hz.*

*Shall be performed after any modifications which could affect diesel generator interdependence.

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3 ~~1.6%~~ Delta k/k is initially required to control the reactivity transient and automatic ESF is assumed to be available. With Tavg less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% Delta k/k SHUTDOWN MARGIN provides adequate protection for this event.

The SHUTDOWN MARGIN requirements are based upon the limiting conditions described above and are consistent with FSAR safety analysis assumptions.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 2000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 plus or minus 100 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC at the beginning, and near the end of each fuel cycle is adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above ~~1.69~~ THE SAFETY ANALYSIS LIMIT during all normal operations and anticipated transients. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (31 percent of RATED THERMAL POWER).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE. Three loops are required to be OPERABLE and to operate if the control rods are capable of withdrawal and the reactor trip breakers are closed. The requirement assures adequate DNBR margin in the event of an uncontrolled rod withdrawal in this mode.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 152°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCP's to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a loss of coolant accident or a steam line rupture. The limits on RWST minimum volume and boron concentration ensure th 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) ~~the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with a control rods inserted except for the most reactive control assembly.~~ These assumptions are consistent with the LOCA analyses.

REPLACE WITH INSERT A

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST als ensure a pH value of between 7.6 and 9.5 for the solution recirculated withi containment after a LOCA. This pH band minimizes the evolution of iodine an minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 70°F. This temperature value of the RWS water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50. ~~The value of th minimum RWST temperature in Technical Specification 3.5.5 has been conservatively changed to 80°F to increase the consistency between Units 1 and 2. The lower RWST temperature results in lower containment pressure fro containment spray and safeguards flow assumed to exit the break. Lower containment pressure results in increased flow resistance of steam exiting the core thereby slowing reflood and increasing PCT.~~

DELETE

INSERT A

the reactor will remain subcritical in the cold condition following a LOCA assuming mixing of the RWST, RCS, ECCS water, and other sources of water that may eventually reside in the sump, with all control rods assumed to be out.

CONTAINMENT SYSTEMS

BASES

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

The maximum peak pressure resulting from a LOCA event is calculated to be ~~11.89~~ 11.49 psig, which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 60°F will limit the peak pressure to ~~11.39~~ 11.49 psig which is less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

DESIGN FEATURES

a. In accordance with the code requirements specified in Section 4.1.6 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

APPROXIMATELY 12,466 CUBIC FEET AT 0% STEAM GENERATOR TUBE PLUGGING AND 11,551 CUBIC FEET AT 30% STEAM GENERATOR PLUGGING.

5.4.2 The total contained volume of the reactor coolant system is ~~12,612 ± 100 cubic feet~~ at a nominal T_{avg} of 70°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements, with one exception. This exception is the CVCS boron makeup system and the BIT.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water.

b. A nominal 8.97 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. 1. The fuel assemblies will be classified as acceptable for Region 1, Region 2, or Region 3 storage based upon their assembly average burnup versus initial nominal enrichment. Cells acceptable for Region 1, Region 2, and Region 3 assembly storage are indicated in Figures 5.6-1 and 5.6-2. Assemblies that are acceptable for storage in Region 1, Region 2, and Region 3 must meet the design criteria that define the regions as follows:

ATTACHMENT 7 TO AEP:NRC:1207

DESCRIPTION OF ANALYSES PERFORMED BY
WESTINGHOUSE ELECTRIC CORPORATION FOR
DONALD C. COOK NUCLEAR PLANT UNIT 2

Vantage 5 Reload Transition Safety Report

for

Donald C. Cook Nuclear Plant Unit 2

6.0 SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES

Table 6.1 presents a list of the Technical Specifications changes. The changes noted in Table 6.1 are given in the proposed Technical Specifications page changes in Appendix A.

TABLE 6.1

SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES

SECTION, PAGE	CHANGE	REASON FOR CHANGE
1.0, pg I	Add COLR to index	COLR implementation
1.12a, pg 1-3	Add COLR	COLR implementation
Figure 2.1-1, pg 2-2	Revised safety limits	Reanalysis supports VANTAGE 5 reload
2.2.1, pg 2-5	Design flow change & trip setpoint	Change in design flow due to VANTAGE 5 fuel reload, RTDP implementation
Table 2.2-1, pg 2-7 & 2-8	Revise Overtemperature ΔT limits	Reanalysis supports VANTAGE 5 reload
Table 2.2-1, pg 2-9	Revise Overpower ΔT limits	Reanalysis supports VANTAGE 5 reload
2.1.1 Bases, pg B 2-1 & B 2-2	Update to bases	VANTAGE 5 fuel reload and COLR implementation (relocation of $F^N_{\Delta H}$)
2.1.1 Bases, pg B 2-4	Update to bases	VANTAGE 5 fuel reload and delete Cycle 6 specific information
2.1.1 Bases, pg B 2-5	Revise bases	Reanalysis supports VANTAGE 5 reload
2.1.1 Bases, pg B 2-7	Revise bases circuit breaker time	Reanalysis supports VANTAGE 5 reload
3/4.1.1.1, pg 3/4 1-1 & 1-2	Decrease shutdown margin	Reanalysis with reduced SDM
3/4.1.1.2, pg 3/4 1-3 & 1-3b	Decrease shutdown margin	Reanalysis with reduced SDM. Change to Westinghouse dilution accident methodology

TABLE 6.1
SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES
(continued)

SECTION, PAGE	CHANGE	REASON FOR CHANGE
3/4.1.1.4, pg 3/4 1-5 & 3/4 1-6	MTC relocated to COLR & revised EOL limit	VANTAGE 5 fuel reload and COLR implementation (relocation of MTC)
3/4.1.1.5, pg 3/4 1-7	Minimum temperature for surveillance req.	Reanalysis with reduced temp
3/4.1.2.3, pg 3/4 1-11	Change ch. pump discharge head	Make consistent with the analysis
3/4.1.2.4, pg 3/4 1-12	Change ch. pump discharge head	Make consistent with the analysis
3/4.1.2.7, pg 3/4 1-15	Change 80 °F to 70 °F	Make spec consistent with the analysis limit
3/4.1.2.8, pg 3/4 1-16	Change volume from 5650 to 7715 gallons & change 80 °F to 70 °F	Make spec consistent with the VANTAGE 5 reload analysis limit to accommodate reduced rod worth and management flexibility
3/4.1.3.1, pg 3/4 1-19	Delete reference to Fig. 3.1-1	COLR implementation
3/4.1.3.4, pg 3/4 1-23	Change rod drop time from 2.2 to 2.7 sec Relocate steps withdrawn to COLR	Make spec consistent with the analysis limit & COLR implementation
3/4.1.3.5, pg 3/4 1-24	Relocate shutdown rod insertion limits to COLR	COLR implementation (relocation of shutdown rod insertion limits)

TABLE 6.1
SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES
(continued)

SECTION, PAGE	CHANGE	REASON FOR CHANGE
3/4.1.3.6, pg 3/4 1-25	Relocate control rod insertion limits to COLR	COLR implementation (relocation of control rod insertion limits)
3/4.1.3.6, pg 3/4 1-26	Delete figure 3.1-1	COLR implementation
3/4.3.2.1, pg 3/4 2-1 & 2-3	Relocate axial flux difference limits to COLR	COLR implementation (relocation of AFD limits)
3/4.3.2.1, pg 3/4 2-4	Relocate axial flux difference allowable deviation Fig. to COLR	COLR implementation (relocation of AFD allowable deviation)
3/4.3.2.2, pg 3/4 2-5	Relocate F_Q limits to COLR	COLR implementation (relocation of F_Q limit)
3/4.3.2.2, pg 3/4 2-8, 2-8a & 2-8b	Relocate K(Z) & V(Z) figures to COLR	COLR implementation (relocation of F_Q limit)
3/4.3.2.3, pg 3/4 2-9	Relocate $F^N_{\Delta H}$ limits to COLR	COLR implementation (relocation of $F^N_{\Delta H}$ limit)
3/4.2.5.1, pg 3/4 2-15	Reformat DNB spec Change DNB parameter values and add low Tavg window	Adopt planned Cook Nuclear Plant Unit 1 spec format consistent with VANTAGE 5 reload
3/4.2.5.1, pg 3/4 2-16 & 2-17 & 2-18	Delete tables 3.2-1 and 3.2-2 Delete 3.2.5.2	Adopt planned Cook Nuclear Plant Unit 1 spec format Not required

TABLE 6.1
SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES
(continued)

SECTION, PAGE	CHANGE	REASON FOR CHANGE
3/4.3.2.6, pg 3/4 2-19	Relocate F_Q limits to COLR Changed definition of F_Q	COLR implementation (relocation of F_Q limit) Westnghouse CAOC methodology
Table 3.3-2, pg 3/4 3-9 & 3-10	Changed and added RPS response times	Make consistent with the analysis limits
Table 3.4-4, pg 3/4 3-25	Change ESFAS setpoint	Make consistent with analysis
Table 3.3-5, pg 3/4 3-26 & 3/4 3-27 & 3/4 3-28	Changed ESF response time times	Make consistent with the analysis limits
3/4.4.1.2, pg 3/4 4-2 & 4-3	Reduce number of RCPs required operable in mode 3	Make consistent with the analysis limits
3/4.4.4, pg 3.4 4-6	Change water volume from 62% to 92%	Make consistent with the analysis limit
3/4.4.6.2, pg 3/4 4-15 & 3/4 4-16	Controlled leakage in terms of resistance	Consistent with analysis
3/4.5.1b, pg 3/4 5-1	Revise minimum contained borated water volume & min/max cover-pressure	Make consistent with analysis limits
3/4.5.2.f, pg 3/4 5-5	Revised SI pump performance	Reanalysis with degraded SI performance

TABLE 6.1
SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES
(continued)

SECTION, PAGE	CHANGE	REASON FOR CHANGE
3/4.5.2.h, pg 3.4 5-6	Revised SI pump flow balance limits	Adopt limits similar to Cook Nuclear Plant Unit 1
3/4.5.5, pg 3/4 5-11	Reduce RWST min temp to 70 °F	Make spec consistent with analysis limit
3/4.1.1.1, pg B 3/4 1-1	Decrease shutdown margin	Reanalysis with reduced shutdown margin
B 3/4.1, pg B 3/4 1-3	Revise concentrations and volumes	Make spec consistent with analysis limits
B 3/4.2.1, pg B 3/4 2-1 & 2-2 & 2-3	Revise to reflect COLR implementation Changed to WCAP-8385	COLR implementation (relocation of AFD limits) Westinghouse methodology
B 3/4.2.2 & 3, pg B 3/4 2-4 thru 2-4b	Revised to reflect COLR implementation & VANTAGE 5 reload	VANTAGE 5 reload T-H analysis and COLR implementation (relocation of F_Q and $F^N_{\Delta H}$ limits)
B 3/4.2.5, pg B 3/4 2-5	Revise to reflect reduced temp DNB limit	Reanalysis with reduced temp
B 3/4.2.6, pg B 3/4 2-5	Revise to reflect CAOC control	Make spec consistent with analysis
B 3/4.5.5, pg B 3/4 5-3	Reduce RWST temp to 70 °F	Make spec consistent with the analysis limit
B 3/4.7.1, pg B 3/4 7-1	Reformat valve lift criteria	Make consistent with the analysis limit

TABLE 6.1
SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES
(continued)

SECTION, PAGE	CHANGE	REASON FOR CHANGE
3.4.9.1, pg B 3/4 9-1	Delete reference to refueling reactivity calcs at 2000 ppm	Reanalysis of refueling reactivity at 2400 ppm boron
6.9.1.11, pg 6-18	Add COLR to section 6	COLR implementation

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.*

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

a) Centrifugal charging pump

b) Safety injection pump

c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated differential ~~discharge~~ pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1. Centrifugal charging pump ≥ ~~2405 psig~~ 2290 psid

2. Safety Injection pump ≥ ~~1445 psig~~ 1385 psid

3. Residual heat removal pump ≥ ~~195 psig~~ 160 psid

g. By verifying the correct position of each mechanical stop for the following Emergency Core Cooling System throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

* The provisions of Specification 4.0.7 are applicable.

APPENDIX B

NON-LOCA ANALYSES

FOR THE

DONALD C. COOK NUCLEAR PLANT UNIT 2

TRANSITION TO 17X17 VANTAGE 5 FUEL

B.3.11 Rupture of a Steamline (Steamline Break)

B.3.11.1 Introduction

Although the no load temperature does not change due to the plant rerating and VANTAGE 5 fuel, the impact of the various fuel parameter changes as well as various temperature and pressure operation was addressed. Also, the nominal low steam pressure setpoint for steamline isolation and safety injection actuation is revised (lowered from 600 psig to 500 psig) to provide operating margin. As such, the rupture of a steam pipe event was analyzed. Included in the analysis are the design changes associated with the VANTAGE 5 transition and other modified safety analysis assumptions as discussed in Section B.1.

A rupture of a steam pipe results in an uncontrolled steam release from a steam generator. The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant temperature and pressure. In the presence of a negative coolant temperature coefficient,

the cooldown results in a reduction of core shutdown margin. If the most reactive RCCA is assumed stuck in its fully withdrawn position, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential concern mainly because of the high hot channel factors which exist when the most reactive RCCA is assumed stuck in its fully withdrawn position. The core is ultimately shut down by boric acid delivered by the Emergency Core Cooling System.

The analysis of a steam pipe rupture is performed to demonstrate that:

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

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

B.3.11.2 Method of Analysis

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

A. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN Code (Reference 5) has been used.

B. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital computer code, THINC, has been used to determine if DNB occurs for the limiting core conditions computed in item A above.

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

A. End-of-life shutdown margin (1.3 $\%\Delta k/k$) at no load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position.

B. A negative moderator temperature coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position: The variation of the coefficient with temperature and pressure has been included. The keff versus temperature at 1050 psia corresponding to the negative moderator temperature coefficient used plus the Doppler temperature effect, is shown in Figure B.3-55. The Doppler power feedback assumed for this analysis is presented in Figure B.3-56.

The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculation. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod. To verify the conservatism of this method, the reactivity as well as the power distribution was checked for the limiting conditions for the cases analyzed. This core analysis considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high enthalpy water near the stuck RCCA, power redistribution and non-uniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated including the above local effects for the statepoints. These results verify conservatism; i.e., underprediction of negative reactivity feedback from power generation.

C. Minimum capability for injection of boric acid (2400 ppm) solution from the RWST corresponding to the most restrictive single failure in the safety injection system. The Emergency Core Cooling System (ECCS) consists of the following systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, 3) the intermediate head safety injection system, and 4) the high head safety injection (charging) system. Only the high head safety injection (charging) system and the passive accumulators are modeled for the steam line break accident analysis. Centrifugal Charging pump flow degradation of 10% was assumed.

The modeling of the safety injection system in LOFTRAN is described in Reference 5. Figure B.3-57 presents the safety injection flow rates as a function of RCS pressure assumed in the

analysis. The flow corresponds to that delivered by one charging pump delivering its full flow to the cold leg header. No credit has been taken for the low concentration borated water which must be swept from the lines downstream of the RWST isolation valves prior to the delivery of boric acid to the reactor coolant loops. For this analysis, a boron concentration of 0 ppm for the boron injection tank is assumed.

For the cases where offsite power is assumed, the sequence of events in the safety injection system is the following. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high head safety injection pump starts. In 27 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed and to draw suction from the RWST. The volume containing the low concentration borated water is swept into core before the 2400 ppm borated water reaches the core. This delay, described above, is inherently included in the modeling.

In cases where offsite power is not available, an additional 10 second delay is assumed to start the diesel generators and to commence loading the necessary safety injection equipment onto them.

D. Design value of the steam generator heat transfer coefficient.

E. Four combinations of break sizes and initial plant conditions have been considered in determining the core power transient which can result from large area pipe breaks.

   a. Complete severance of a pipe downstream of the steam flow restrictor with the plant initially at no load conditions and all reactor coolant pumps running.

   b. Complete severance of a pipe inside the containment at the outlet of the steam generator (upstream of the steam flow restrictor) with the same plant conditions as above.

   c. Case (a) above with loss of off-site power simultaneous with the generation of the Safety Injection Signal (loss of AC power results in reactor coolant pump coastdown).

d. Case (b) above with the loss of offsite power simultaneous with the Safety Injection Signal.

A fifth case was analyzed to show that the DNBR remains above the limit value in the event of the spurious opening of a steam dump or relief valve.

e. A break equivalent to a steam flow of 265 lbs per second at 1100 psia from one steam generator with offsite power available.

F. Power peaking factors corresponding to one stuck RCCA are determined at end of core life assuming non-uniform core inlet coolant temperatures. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow, and are thus different for each case studied.

The analyses assumed initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power the reactor coolant system contains more stored energy than at no-load, the average coolant temperature is higher than at no-load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no-load conditions at time zero.

In addition, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of RCS cooldown are more severe than steam line breaks occurring at power.

G. In computing the steam flow during a steam line break, the Moody Curve (Reference 11) for fL/D = 0 is used.

H. The total delay time assumed for the steamline isolation is 11 seconds from receipt of actuation signal. The 11 second steamline isolation time includes valve closure time, and electronics and sensor delay. The Technical Specifications require a maximum 8 second valve closure time. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location following steamline isolation, no more than one steam generator would experience an uncontrolled blowdown even if one of the isolation valves fails to close.

Plant characteristics and initial conditions are shown in Table B.2.4.

B.3.11.3 Results

The limiting case for Cases a through e was shown to be the double-ended rupture located upstream of the flow restrictor with offsite power available (case b). Table B.3-10 lists the limiting statepoint for this worst case. The results presented are a conservative indication of the events which would occur assuming a steam line rupture.

Figures B.3-58 through B.3-60 show the RCS transient and core heat flux following a main steam line rupture (complete severance of a pipe), upstream of the flow restrictor at initial no-load conditions.

Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by high differential pressure between any steamline and the remaining steamlines or by low steam line pressure in two steamlines will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast acting isolation valves in the steam lines by high-high containment pressure signals or low steamline pressure or high steam flow coincident with low-low T-avg. Even with the failure of one valve, release from the other steam generators is terminated by steamline isolation while the one generator blows down. The steam line stop valves are assumed to be fully closed in less than 11 seconds from receipt of a closure signal.

As shown in Figure B.3-60, the core attains criticality with the RCCAs inserted (with the design shutdown margin assuming one stuck RCCA) before boron solution (2400 ppm from RWST) enters the RCS. A peak core power less than the nominal full power value is attained.

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

The assumed steam release for an accidental depressurization of the main steam system (case e) is the maximum capacity of any single steam dump, relief, or safety valve. Safety injection is initiated automatically by low pressurizer pressure. Operation of one centrifugal charging pump is assumed. Boron solution at 2400 ppm enters the RCS providing sufficient negative reactivity to prevent core damage. The transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements or the energy stored in the other steam generators. Since the transient occurs over a period of about 5 minutes, the neglected stored energy is likely for this event to have a significant effect in slowing the cooldown. The DNB transient is bounded by the limiting case for a steamline rupture.

The DNB analysis for the limiting case (double-ended rupture located upstream of the flow restrictor) showed that the minimum DNBR remained above the limit value. The DNBR design basis limit for the hypothetical steamline break event is 1.45. The pressures for this event fall in the low pressure range (500-1000 psia) where the W-3 based DNB correlation is used with a 1.45 limit DNBR. This design limit for low pressure applications of the W-3 correlation has been approved by the NRC in Reference 15. Although the low pressure limit was approved in conjunction with WCAP-9227-NP, which is not referenced in the Cook Nuclear Plant Unit 2 UFSAR, the SER is an applicable reference for reload designs.

The calculated sequence of events for the limiting case (double-ended rupture located upstream of the flow restrictor) are shown in Table B.3-11.

B.3.11.4 Conclusions

The analysis has shown that the criteria stated earlier are satisfied.

Although DNB and possible clad perforation following a steam pipe rupture can be acceptable and is not precluded by the criteria, the above analysis, in fact, shows that no DNB occurs for the rupture (including an accidental depressurization of the main steam system) assuming the most reactive RCCA stuck in its fully withdrawn position.

TABLE B.3-10

LIMITING STEAMLINE BREAK STATEPOINT

DOUBLE ENDED RUPTURE INSIDE CONTAINMENT

WITH OFFSITE POWER AVAILABLE

			Inlet Temp					
Time	Pressure	Heat Flux	Cold	Hot	Flow	Boron	Reactivity	Density
Sec	Psia	Fraction	°F	°F	Frac	PPM	Percent	GM/CC
100.2	598.7	0.107	330.2	441.8	1.0	1.51	0.044	0.863

TABLE B.3-11

TIME SEQUENCE OF EVENTS

Accidents	Events	Time (sec)
Rupture of a Steamline		
1. Inside Containment With Offsite Power available	Steam line ruptures	0.0
	Low steamline pressure setpoint reached	0.26
	Feedwater Isolation (All loops)	8.26
	Steamline Isolation (Loops 2, 3 and 4)	11.26
	Pressurizer empties	13.8
	SI flow starts	27.26
	Criticality attained	29.4
	Boron from SI reaches cores	38.2
	Peak heat flux attained	100.2
	Core becomes subcritical	116.2

TABLE B.3-11
(continued)
TIME SEQUENCE OF EVENTS

Accidents	Events	Time (sec)
Rupture of a Steamline		
2. Inside Containment Without Offsite Power available	Steam line ruptures	0.0
	Low steamline pressure setpoint reached	0.26
	Feedwater Isolation (All loops)	8.26
	Steamline Isolation (Loops 2, 3 and 4)	11.26
	Pressurizer empties	15.4
	Criticality attained	37.0
	SI flow starts	37.26
	Boron from SI reaches cores	51.4
	Peak heat flux attained	236.0
	Core becomes subcritical	291.7

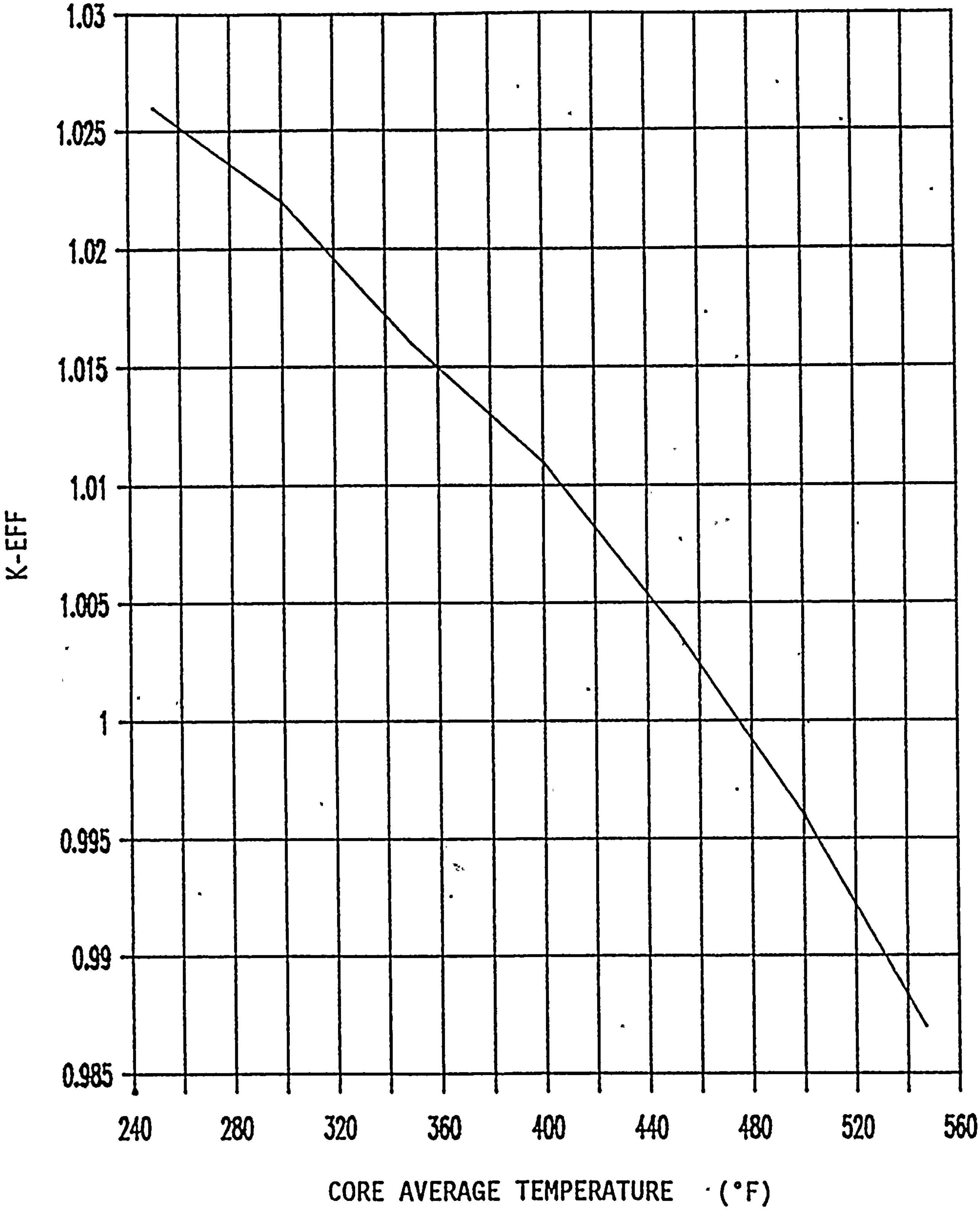


Figure B.3-55 Variation of Reactivity with Core Temperature at 1050 PSIA for the End of Life Rodded Core with One Control Rod Assembly Stuck (Assumes Zero Power)

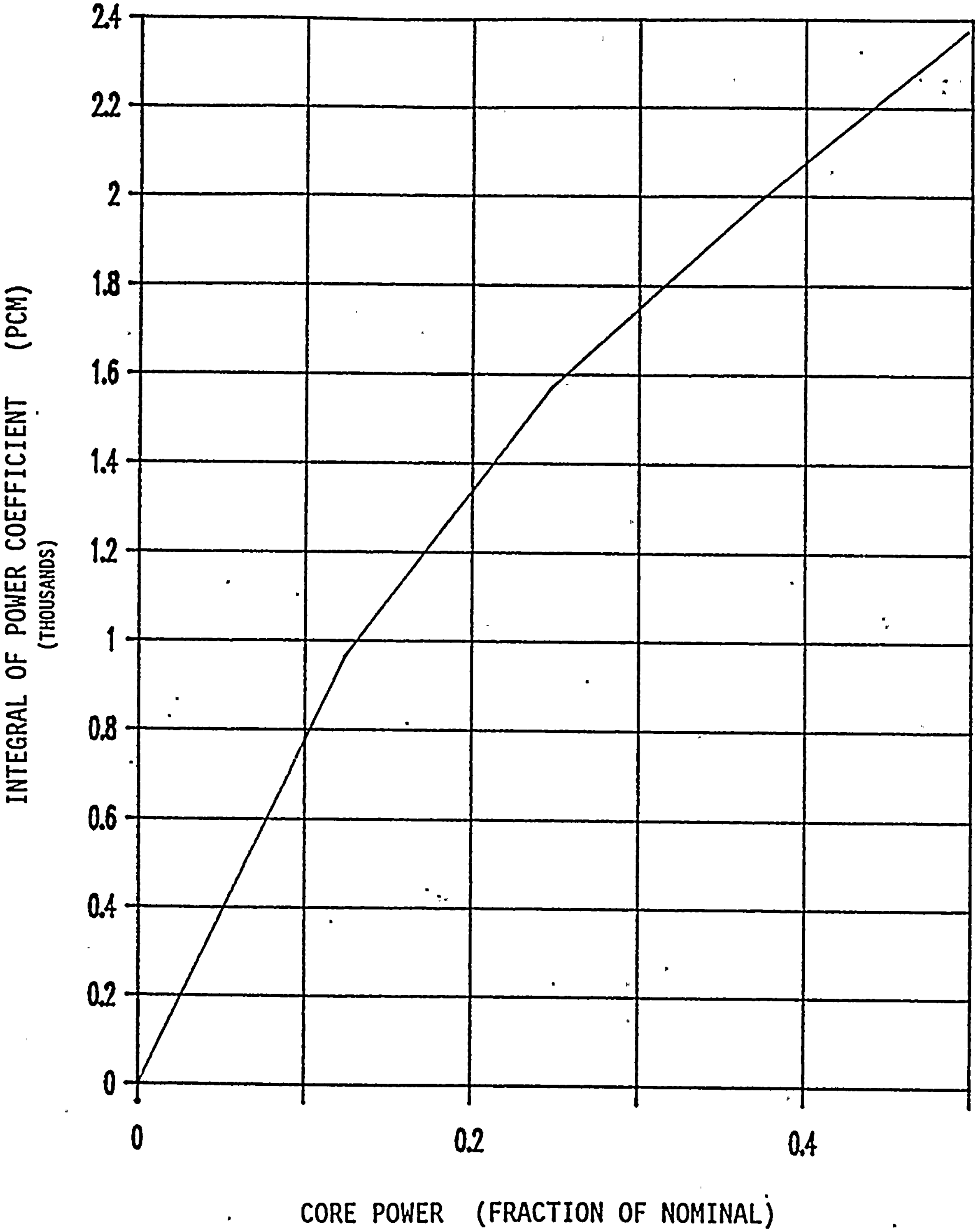


Figure B.3-56 Doppler Power Feedback for Steamline Break

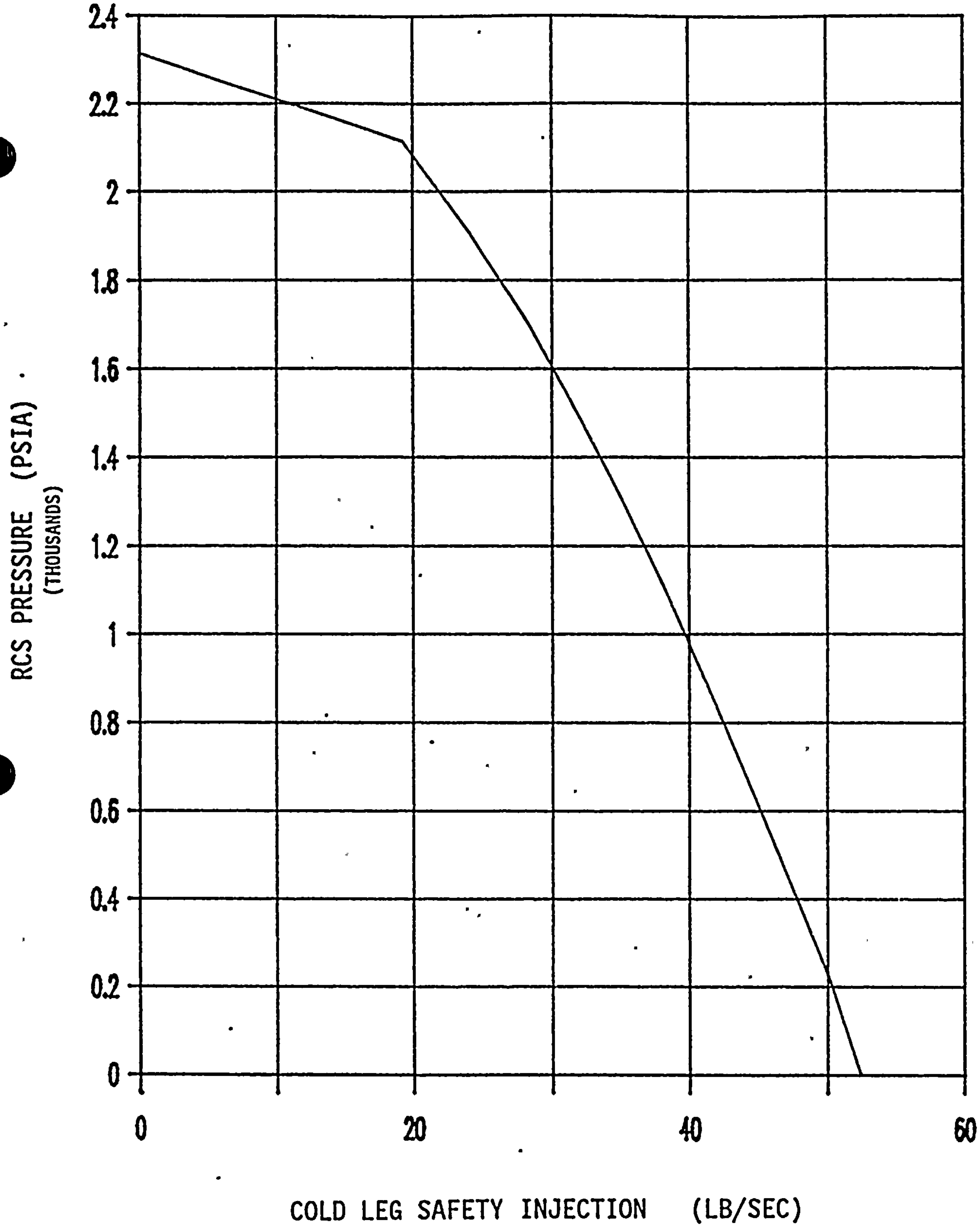
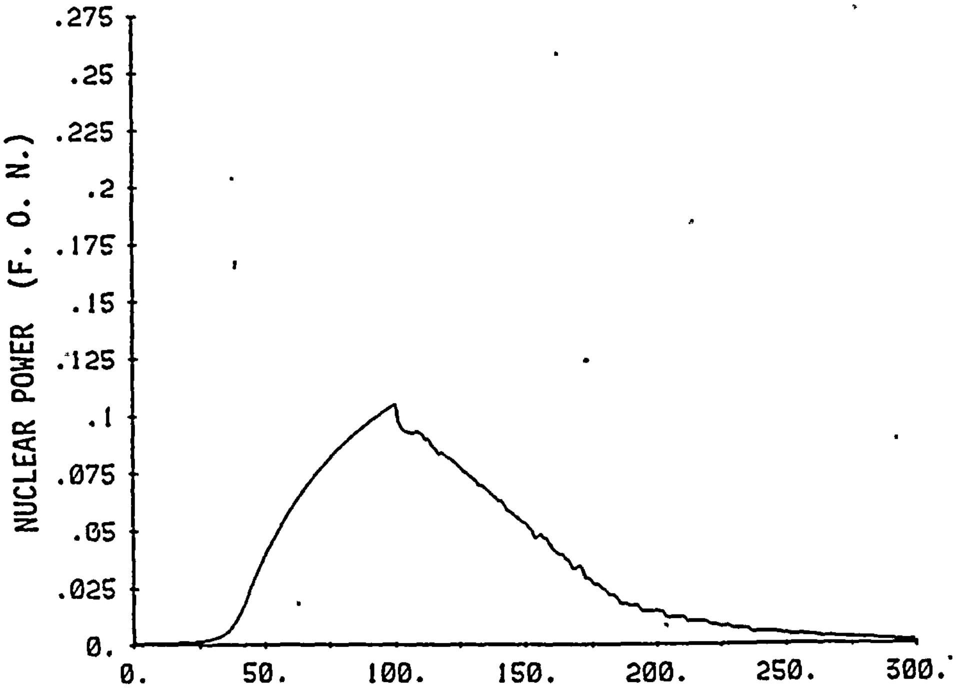


Figure B.3-57 Safety Injection Flow Supplied by One Charging Pump



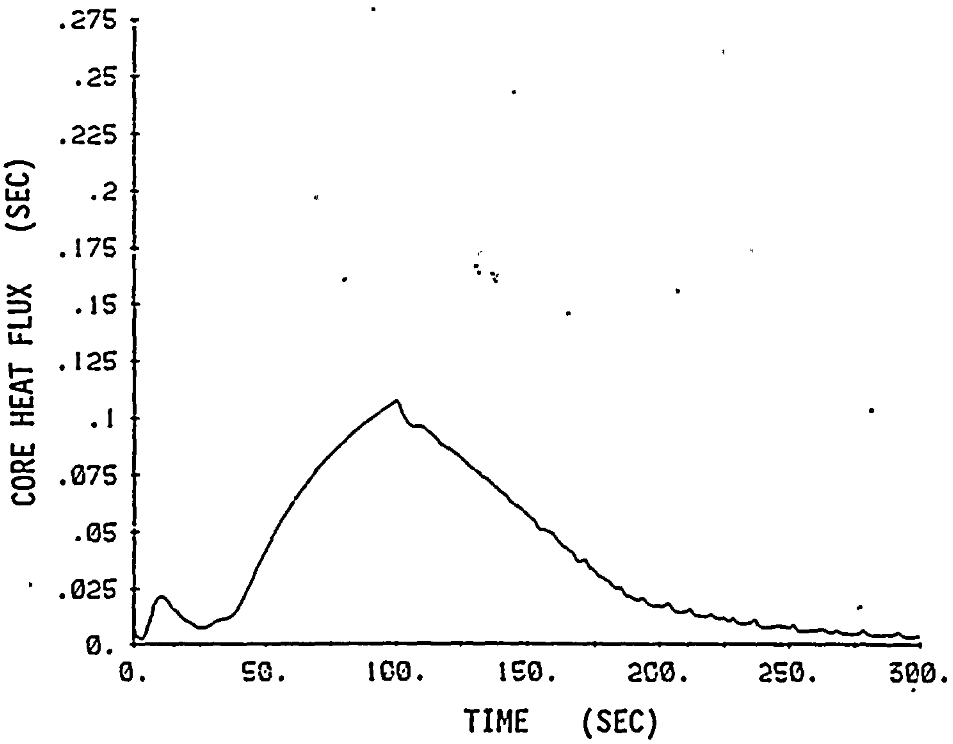


Figure B.3-58 Nuclear Power and Core Heat Flux Versus Time Steamline Break DER Inside Containment with Power

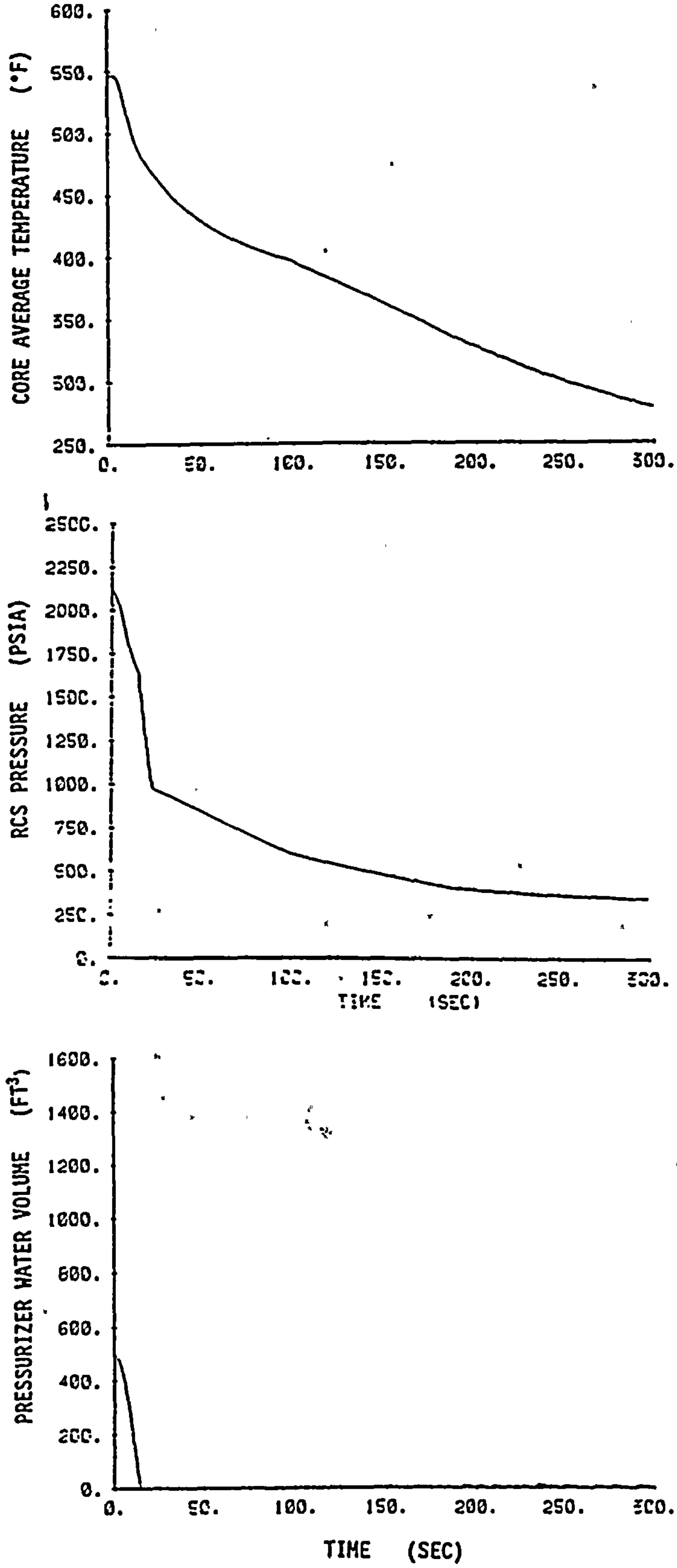


Figure B.3-59 Core Average Temperature, RCS Pressure, and Pressurizer Water Volume Versus Time Steamline Break DER Inside Containment with Power

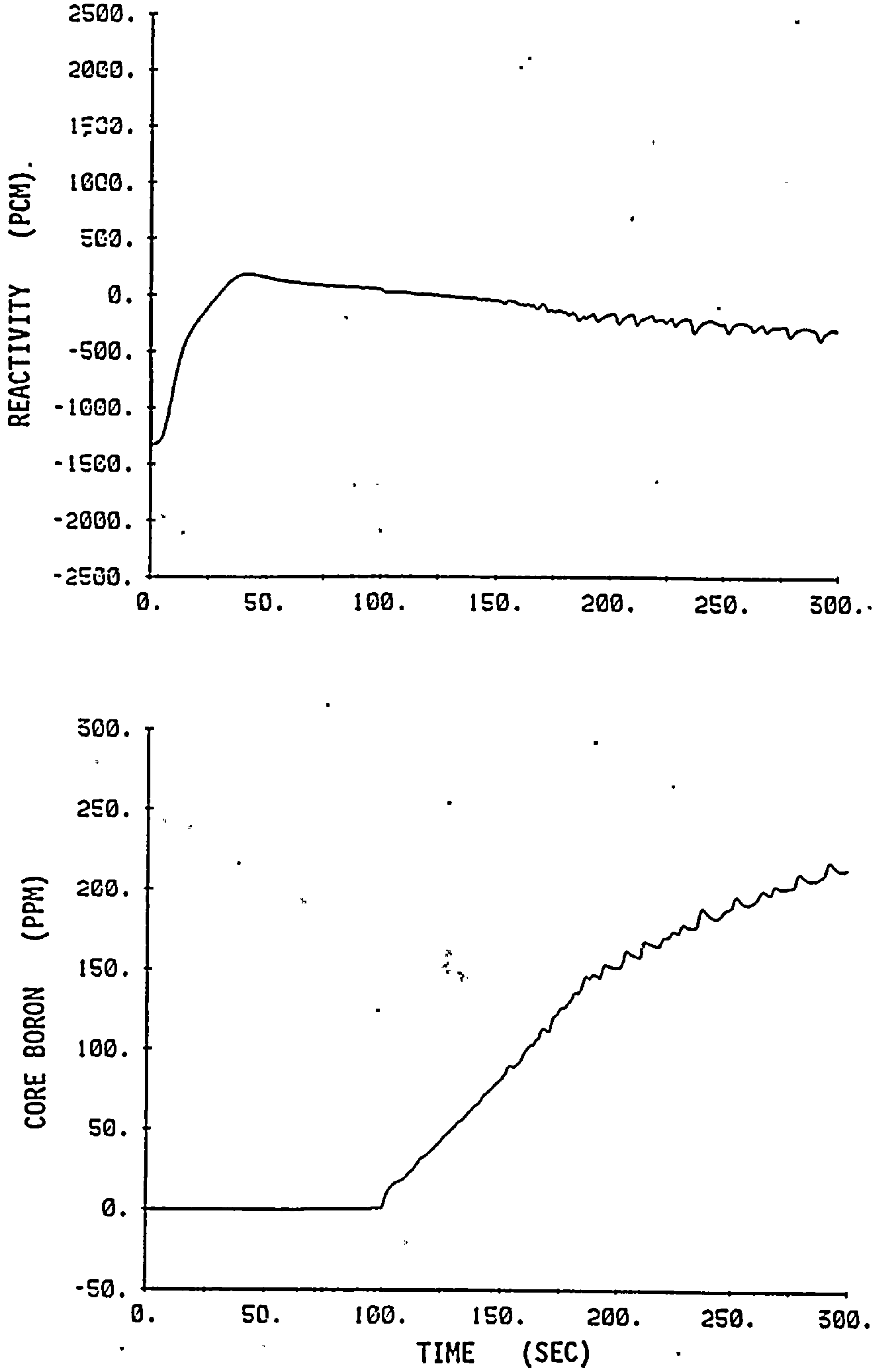


Figure B.3-60 Reactivity and Core Boron Concentration Versus Time Steamline Break DER Inside Containment with Power

APPENDIX C

LOCA ANALYSES

FOR THE

DONALD C. COOK NUCLEAR PLANT UNIT 2

TRANSITION TO 17x17 VANTAGE 5 FUEL

C.3.1.2 MAJOR LOCA ANALYSES APPLICABLE TO WESTINGHOUSE FUEL

Identification of Causes and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, 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 an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of the Donald C. Cook Nuclear Plant Unit 2, but is postulated as a conservative design basis.

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 (10 CFR 50.46 and Appendix K of 10 CFR 50 1974)(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 to generate hydrogen, does not exceed 1 percent of the total amount of Zircaloy in the fuel rod cladding.

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

These criteria were established to provide a significant margin in emergency core cooling system (ECCS) performance following a LOCA. WASH-1400 (USNRC 1975)(10) presents a study in regards to the probability of occurrence of RCS pipe ruptures.

Sequence of Events and Systems Operations

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. Loss-Of-Offsite Power (LOOP) is assumed coincident with the occurence of the break. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint is reached.

These countermeasures will limit the consequence of the accident in two ways:

1. Reactor trip and borated water injection supplement void formation in causing rapid reduction of power to the residual level corresponding to fission product decay heat. No credit is taken in the LOCA analysis for the boron content of the injection water, however an average RCS/sump mixed boron concentration is calculated to ensure that the core remains subcritical. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis.

2. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

In the present Westinghouse design, the large break single failure is the loss of one RHR (low head) pump. This means that credit could be taken for two high head charging pumps, two safety injection pumps, and one low head pump. The following is a discussion of the modelling procedure for the minimum safeguards and the flow splitting from a break of an ECCS branch injection line (i.e., the spilling line assumptions).

The current procedure for large break analyses assumes that at least one train of ECCS is available for delivery of water to the RCS. Although the single failure is an RHR pump, only one pump in each subsystem is assumed to deliver to the primary loops. However, both Emergency Diesel Generators (EDGs) are assumed to start in the modelling of the containment deck fans and sprays. Modelling full containment heat removal systems operation is required by Branch Technical Position CSB 6-1 and is conservative for the large break LOCA. The high head charging pump starts and delivers flow through the injection lines (one per loop) with one branch injection line spilling to the containment backpressure. To minimize delivery to the reactor, the branch line chosen to spill

is selected as the one with the minimum resistance. When one safety injection pump and one low head residual heat removal pump start, flow is delivered to the reactor coolant system through the accumulator injection lines. Again, one line, with the minimum resistance, is assumed to spill to containment backpressure. In addition, all safety injection pump performance curves were degraded by 10% and a 25 gpm flow imbalance was assumed for the high head charging pumps.

Therefore, in the large break ECCS analysis performed by Westinghouse, single failure is conservatively accounted for via the loss of an ECCS train, and the spilling of the minimum resistance injection line despite full containment active heat removal system operation (i.e., two EDGs).

The time sequence of events following a large break LOCA is presented in Table C.3.1-5. The safety injection performance, as modelled for the large break LOCA, is presented in Figures C.3.1.1 and C.3.1.2.

Before the break occurs, the unit 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 RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10 CFR 50[1]. Thereafter, the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes uncovered, both turbulent and laminar forced convection and radiation are considered as core heat transfer mechanisms.

The heat transfer between the RCS 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, the 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 signal actuates a feedwater isolation signal which isolates main feedwater flow by closing the main feedwater isolation valves, and also initiates auxiliary feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is made that accumulator water injected bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10 CFR 50. Since loss of offsite power (LOOP) is assumed, the RCPs 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 RCS pressure (initial values with uncertainty assumed to range from 2037 to 2313 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowdown, the mechanisms that are responsible for the emergency core cooling water bypassing the core, are calculated not to be effective. At this time (called end-of-bypass) refill of the reactor vessel lower plenum begins. Refill is completed 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 bottom of core recovery until the reactor vessel has been filled with water to the exten~~d~~t that the core temperature rise has been terminated. From the latter stage of blowdown and then the beginning-of-reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation hea~~l~~d provides the driving force required for the reflooding of the reactor core. The RHR (low head), safety injection and high head charging pumps aid in the filling of the downcomer and, subsequently, supply water to maintain a full downcomer and complete the reflooding process.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady state levels associated with the dissipation of residual heat generation. After the water level of the ~~residual~~ refueling water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold leg recirculation phase of operation in which spilled borated water is drawn from the engineered safety features (ESF) containment sumps by the residual heat removal (low head) safety injection pumps and returned to the RCS cold legs. The containment spray system continues to operate to further reduce containment pressure.

Approximately 12 hours after the initiation of the LOCA, the ECCS is realigned to inject water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel. Long-term cooling includes long-term criticality control. Criticality control is achieved by determining the RWST and accumulator concentrations necessary to maintain subcriticality without credit for RCCA insertion. The necessary RWST and accumulator concentrations are a function of each core design and are checked each cycle. The current Technical Specifications value are 2400 to 2600 ppm boron for the RWST and 2400 to 2600 ppm for the accumulators. The accumulators are conservatively modelled at 2300 ppm for the post-LOCA subcriticality requirement.

Core and System Performance

Mathematical Model:

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50(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 RCS, the pressure and temperature transient within the containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

A description of the various aspects of the LOCA analysis methodology is given by Bordelon, Massie, and Zordan (1974)(6). This document describes 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, BASH and LOCBART codes, which are used in the LOCA analysis, are described in detail by Bordelon et al. (1974)(5); Kelly et al. (1974)(9); Young et al. (1987)(4); and Bordelon et al. (1974)(6). Code modifications are specified in References 2, 7, 13, and 17. These codes assess the core heat transfer geometry and determine if the core remains amenable to cooling through and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic

transient in the RCS during blowdown and the WREFLOOD computer code calculates this transient during the refill phase of the accident.

The BASH code is used to determine the RCS response during the reflood phase of the transient. The LOTIC computer code, described by Hsieh and Raymund in WCAP-8355 (1975) and WCAP-8345 (1974)(3), calculates the containment backpressure transient. The containment backpressure transient is input to BASH for the purpose of calculating the reflood transient. The LOCBART computer code calculates the thermal transient of the hottest fuel rod in the three phases. The improved fuel performance model, described in Reference 15, generates the initial fuel rod conditions input to LOCBART.

SATAN-VI calculates the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the break mass and energy flow rates that are assumed to be vented to the containment during blowdown.

At the end of the blowdown, information on the state of the system is transferred to the WREFLOOD code which performs the calculation of the refill period to bottom of core (BOC) recovery time. Once the vessel has refilled to the bottom of the core, the reflood portion of the transient begins. The BASH code is used to calculate the thermal-hydraulic simulation of the RCS for the reflood phase.

Information concerning the core boundary conditions is taken from all of the above codes and input to the LOCBART code for the purpose of calculating the core fuel rod thermal response for the entire transient. From the boundary conditions, LOCBART computes the fluid conditions and heat transfer coefficient for the full length of the fuel rod by employing mechanistic models appropriate to the actual flow and heat transfer regimes. Conservative assumptions ensure that the fuel rods modeled in the calculation represent the hottest rods in the entire core.

The large break analysis was performed with the December 1981 version of the Evaluation Model modified to incorporate the BASH[4] computer code.

Input Parameters and Initial Conditions:

The analysis presented in this section was performed with a reactor vessel upper head temperature equal to the RCS hot leg temperature.

A range of reactor operating temperatures was analyzed in order to justify plant operation at a reactor power level of 3588 Mwt between 582.2 °F to 615.2 °F in the hot legs and 511.7 °F to 547.6 °F in the cold legs. In addition to the temperature range analyzed, initial RCS pressurizer pressure was also varied to justify plant operation between 2037 and 2313 psia. A full spectrum break analysis was done at the high pressure/high temperature RCS conditions (initial RCS pressurizer pressure, with uncertainty, of 2313 psia and initial hot leg temperature of 615.2 °F) from which the limiting break size was determined. The limiting break was then reanalyzed for low temperature and high RCS pressure. The limiting break was also reanalyzed for the high temperature and low initial RCS pressure of 2037 psia. The analysis also considered plant operation at reduced power level with the RHR cross tie valve closed. The reduction in power level was considered necessary to offset the reduction in safety injection flow due to the closure of the RHR cross tie valve. This case assumed a reduced power level of 3413.MWt and minimum safeguards with the RHR cross tie valve closed at the limiting RCS conditions. All cases conservatively assumed 15% steam generator tube plugging in all four steam generators. Table C.3.1-1 describes the cases analyzed. Tables C.3.1-2 and C.3.1-3 summarize the key input parameters and setpoints modelled in the Cook Nuclear Plant Unit 2 large break LOCA analysis.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from extensive sensitivity studies (Westinghouse 1974[12]; Salvatori 1974[11]; Johnson, Massie, and Thompson 1975[8]). In addition, the requirements of Appendix K to 10 CFR 50[1] regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions which were 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, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated as per the requirements of Appendix K to 10 CFR 50[1].

Another input parameter that affects LOCA analysis results is the assumed axial power shape at the beginning of the accident. Power shape sensitivity studies performed with Westinghouse ECCS evaluation models have always demonstrated the chopped cosine shape with the peak at the core midplane to be limiting. Westinghouse has performed "spot check" analyses using the BASH reflood evaluation model for power shapes skewed to the top of the core. Results of these analyses have demonstrated the chopped cosine peaked at the core midplane remains the limiting power shape(18).

A meeting was held at the Westinghouse Licensing Office in Bethesda on December 17, 1981, between members of the U. S. Nuclear Regulatory Commission and members of the Westinghouse Nuclear Safety Department to discuss the impact of maximum safety injection on the large break ECCS analysis on a generic basis. Further discussion of this issue is provided in a letter from E. P. Rahe, Manager of Westinghouse Nuclear Safety Department, to Robert L. Tedesco of the U. S. Nuclear Regulatory Commission(14). A brief description of this issue is given below.

Westinghouse ECCS analyses currently assume minimum safeguards for the safety injection flow, which minimizes the amount of flow to the RCS by assuming maximum injection line resistances, degraded ECCS pump performance, and the loss of one residual heat removal (RHR) pump as the most limiting single failure. This is the limiting single failure assumption when offsite power is unavailable for most Westinghouse plants. However, for some Westinghouse plants, including Cook Nuclear Plant Unit 2, the current nature of the Appendix K ECCS evaluation models is such that it may be more limiting to assume the maximum possible ECCS flow delivery. In that case, maximum safeguards which assume minimum injection line resistances, enhanced ECCS pump performance, and no single failure, result in the highest amount of flow delivered to the RCS.

Therefore, the worst break for Cook Nuclear Plant Unit 2 (C_D=0.6) was reanalyzed, assuming maximum safeguards (Case A vs. Case F of Table C.3.1-1). Examination of the LOCA analysis results in Table C.3.1-6 demonstrates that minimum safeguards assumptions result in the highest peak clad temperature for Cook Nuclear Plant Unit 2.

Transition Core Effects

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 cladding temperature (PCT) than either a complete core of the 17x17 ANF assembly design or a complete core of the Westinghouse 17x17 VANTAGE 5 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. Hydraulic resistance mismatch will exist only for a transition core and is the only unique difference between a complete core of either fuel type and the transition core.

The 17x17 ANF fuel assembly is nearly identical to the Westinghouse 17x17 OFA assembly in terms of hydraulic and geometric characteristics. Therefore, the analyses reported in Reference 19 which demonstrate that the 17x17 VANTAGE 5 fuel features result in a fuel assembly that is more limiting than a Westinghouse 17x17 OFA fuel assembly, with respect to large break LOCA ECCS performance, remain valid as applied at Cook Nuclear Plant Unit 2. The same large break LOCA transition core penalty reported in Section 5.2.3 of Reference 19 will be applied to the transition from 17x17 ANF fuel assemblies to Westinghouse 17x17 VANTAGE 5 fuel assemblies.

Westinghouse transition core designs, including specific 17X17 OFA to 17x17 VANTAGE 5 transition core cases, were analyzed. The increase in hydraulic resistance for the VANTAGE 5 assembly was shown to produce a reduction in reflood steam flow rate for the VANTAGE 5 fuel at mixing vane grid elevations for transition core configurations. The various fuel assembly specific transition core analyses performed resulted in peak cladding temperature increases of up to 50 °F for core axial elevations that bound the location of the PCT. Therefore, the maximum PCT penalty possible for VANTAGE 5 fuel residing in a transition core is 50 °F, Reference 19. As stated earlier, this transition core penalty continues to apply to the transition from 17x17 ANF fuel assemblies to Westinghouse 17x17 VANTAGE 5 fuel assemblies due to the near identical design of 17x17 ANF and Westinghouse 17x17 OFA fuel assemblies. Once a full core of VANTAGE 5 fuel is achieved the large break LOCA analysis will apply without the transition core penalty.

Results:

Based on the results of the LOCA sensitivity studies (Westinghouse 1974[12]; Salvatori 1974[11]; Johnson, Massie, and Thompson 1975[8]), 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 ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. The results of these calculations are summarized in Tables C.3.1-5 and C.3.1-6.

The containment data used to generate the LOTIC backpressure transient are shown in Table C.3.1-4. The mass and energy release data for the minimum and maximum safeguards cases (Case A and F) are shown in Tables C.3.1-7 and C.3.1-8 respectively. In addition, mass and energy release data for Case G (3413 Mwt, RHR cross tie valve closed) are shown in Table C.3.1-9. The mass releases for the remaining cases are not presented, since they do not vary significantly from the data shown in Table C.3.1-7. Nitrogen release rates to the containment are given in Table C.3.1-10.

Figures C.3.1-3a through C.3.1-30 present the results of the cases analyzed for the large break LOCA. The alpha designation in the figure number corresponds to the cases as described in Table C.3.1-1.

Figures C.3.1-3a-g	The system pressure shown is the calculated core pressure.
Figures C.3.1-4a-g	The flow rate from the break is plotted as the sum of both ends of the guillotine break.
Figures C.3.1-5a-g	The core pressure drop shown is from the lower plenum, near the core, to the upper plenum at the core outlet.
Figures C.3.1-6a-g	The core flow rate is shown during the blowdown phase of the transient.
Figures C.3.1-7a-g	The accumulator flow rate during blowdown is plotted as the sum of that injected into the intact cold legs.

Figures C.3.1-8a-g	The core and downcomer collapsed liquid water levels are plotted during the reflood phase of the transient.
Figures C.3.1-9a-g	The core inlet flow rate is shown as it is calculated during the reflood phase.
Figures C.3.1-10a-g	The total pumped ECCS flow rate injecting into the intact cold legs is shown.
Figures C.3.1-11a-g	The integral of the core inlet flow rate as calculated with BASH is plotted.
Figures C.3.1-12a-g	The mass flux is plotted at the hot spot (the node which produced the peak clad temperature) on the hot rod.
Figures C.3.1-13a-g	The heat transfer coefficient is plotted at the hot spot on the hot rod.
Figures C.3.1-14a-g	The fluid temperature at the hot spot on the hot rod is plotted.
Figures C.3.1-15a-g	The clad temperature at the hot spot is shown for the hot rod.
Figures C.3.1-16-18	The containment backpressure transient used in the analysis is provided for Cases A, F and G (the minimum and maximum SI flow cases, and the 3413 Mwt cross tie valve closed case).
Figures C.3.1-19-27	These figures show the heat removal rates of the heat sinks found in the lower and upper compartment and the heat removal by the sump and lower compartment spray for Cases A, F and G.
Figures C.3.1-28-30	These figures show the temperature transients in both the lower and upper compartments of containment and flow from the upper to lower compartments for Cases A, F and G.

The peak clad temperature calculated for a large break is 2140 °F, which is less than the acceptance criterion limit of 2200 °F. The maximum local metal-water reaction is 6.80 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46. The total

metal-water reaction is less than 0.3 percent for all breaks, corresponding to less than 0.3 percent hydrogen generation, as compared with the 1 percent criterion of 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

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TABLE C.3.1-2
INPUT PARAMETERS USED IN THE LARGE BREAK LOCA ECCS ANALYSIS

	Cross Ties Open	RHR Cross Ties Closed
License Core Power[a], (MWt)	3588	3413
Peak Linear Power[a], (kw/ft)	12.714	12.721
Total Peaking Factor, $F_Q{}^T$	2.220	2.335
Axial Peaking Factor, F_Z	1.370	1.420
Hot Channel Enthalpy Rise Factor, $F^N{}_{\Delta H}$	1.620	1.644
Power Shape:	Chopped Cosine	
Fuel Assembly Array	17 X 17 VANTAGE 5	
Accumulator Water Volume, Nominal (ft^3/accumulator)	946	946
Allowance	±25	±25
Accumulator Tank Volume, Nominal (ft^3/accumulator)	1350	1350
Accumulator Gas Pressure, Minimum (psia)	600	600
Safety Injection Pumped Flow Rate (All pumps degraded 10%, Charging pump flow rate imbalance = 25 gpm)	See Figures C.3.1.1 and C.3.1.2	
Containment Parameters	See Table C.3.1-4	
Initial Loop Flow (GPM)	88,500	88,500
Vessel Inlet Temperature (°F)	511.7 to 547.6	513.3 to 546.4
Vessel Outlet Temperature (°F)	582.2 to 615.2	580.6 to 611.2
Average Reactor Coolant Pressure (psia)	2037.4 to 2312.6	2037.4 to 2312.6
Steam Pressure (psia)	587 to 820	603 to 820
Steam Generator Tube Plugging Level (%)	15	15
Refueling Water Storage Tank Temperature (°F)	70[b]	70[b]

(a) Two percent is added to this power to account for calorimetric error.
(b) The BASH computer code models average RWST temperature during core reflooding (85 °F). Other computer codes in the model use 70 °F.

C.3.2 LOSS OF REACTOR COOLANT FROM SMALL RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES WHICH ACTUATES THE EMERGENCY CORE COOLING SYSTEM

Analysis of Effects and Consequences

Method of Analysis

For small breaks (less than 1.0 ft2) the NOTRUMP[10][11] digital computer code is employed to calculate the transient depressurization of the Reactor Coolant System as well as to describe the mass and enthalpy of the fluid flow through the break.

Small Break LOCA Analysis Using NOTRUMP

For loss-of-coolant accidents due to small breaks less than 1 square foot, the NOTRUMP computer code[10][11] is used to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of the fluid flow through the break. The NOTRUMP computer code is a state-of-the-art one-dimensional general network code incorporating a number of advanced features. Among these are 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 NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants".

The reactor coolant system model is nodalized into volumes interconnected by flowpaths. The broken loop is modelled explicitly, while the three intact loops are lumped into a second loop. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multinode capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations. Detailed descriptions of the NOTRUMP code and the evaluation model are provided in References 10 and 11.

After the small break LOCA is initiated, reactor trip occurs due to a low pressurizer pressure signal (1860 psia). Soon after the reactor trip signal is generated, the safety injection signal is actuated due to low pressurizer pressure (1715 psia). Safety injection systems consist of gas pressurized accumulator tanks and pumped injection systems. The small break LOCA analysis assumed an accumulator water volume equal to the average of that allowed in the technical specification with a cover gas pressure of 600 psia. This is the minimum of the cover gas pressure allowed in the Technical Specifications. Minimum emergency core cooling system availability is assumed for the analysis at the maximum RWST temperature. Assumed pumped safety injection characteristics as a function of RCS pressure used as boundary conditions in the analysis are shown in Figure C.3.2-1 and in Table C.3.2-6. The safety injection flow rates presented are based on pump performance curves degraded 10 percent from the design head and an assumed charging system branch line imbalance of 25 gpm. The effect of flow from the RHR pumps is not considered in the small break LOCA analyses since their shutoff head is lower than the RCS pressure during the time portion of the transient considered here. Safety injection (SI) is delayed 27 seconds after the occurrence of the injection signal to account for diesel generator startup and emergency power bus loading in case of a loss of offsite power coincident with a LOCA. The small break LOCA analysis also assumed that the auxiliary feedwater pumps were degraded by 15 percent and that the rod drop time was 2.7 seconds.

Peak clad temperature calculations are performed with the LOCTA-IV[2] code using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. Figure C.3.2-10 depicts the hot rod axial power shape used to perform the small break LOCA analysis. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core. Such a distribution is limiting for small-break LOCAs because it minimizes coolant level swell, while maximizing vapor superheating and fuel rodheat generation at the uncovered elevations. The small break LOCA analysis assumes the core continues to operate at full power until the control rods are completely inserted.

Results

This section presents results of the limiting small break LOCA analysis (as determined by the highest calculated peak clad temperature) for a range of break sizes and RCS pressures and temperatures. The limiting break was found to be a 4-inch diameter cold leg break initiated at

reduced RCS pressurizer pressure (2100 psia) and high temperature (core Tavg = 581.3 °F) conditions. The peak clad temperature attained during the transient was 1357 °F. A list of input assumptions used in the low pressure and high temperature analysis is provided in Table C.3.2-1. The results of a three break spectrum analysis performed at the reduced RCS pressure and high temperature conditions are summarized in Table C.3.2-4, while the key transient event times are listed in Table C.3.2-2.

Figures C.3.2-2 through 9 show for the limiting four-inch break transient, respectively:

- RCS pressure
- Core mixture level
- Peak clad temperature
- Core outlet steam flow rate
- Hot spot rod surface heat transfer coefficient
- Hot spot fluid temperature
- Cold leg break mass flow rate
- Safety injection mass flow rate

During the initial period of the small-break transient the effect of the break flow rate is not strong enough to overcome the flow rate maintained by the reactor coolant pumps as they coast down. Normal upward flow is maintained through the core and core heat is adequately removed. At the low heat generation rates following reactor trip the fuel rods continue to be well cooled as long as the core is covered by a two-phase mixture level. From the clad temperature transient for the 4-inch break calculation shown in Figure C.3.2-4, it is seen that the peak clad temperature occurs near the time when the core is most deeply uncovered and the top of the core is steam cooled. This time is also accompanied by the highest vapor superheating above the mixture level. A comparison of the total break flow rate to containment shown in Figure C.3.2-8 to the safety injection flow rate shown in Figure C.3.2-~~7~~9 shows that at the time the transient was terminated, either when the safety injection flow rate that was delivered to the RCS exceeded the mass flow rate out the break or when the core was covered as in Figure C.3.2-20. Although the inner vessel core mixture level has not yet covered the entire core, there is no longer a concern of exceeding the 10 CFR 50.46 criteria since the pressure is gradually decaying and there is a net mass inventory gain. As the RCS inventory continues to gradually increase, the core mixture level will continue to increase and the fuel clad temperatures will continue to decline.

Conclusions

Analyses presented in this section show that the high head charging and safety injection subsystems of the Emergency Core Cooling System, together with the accumulators, provide sufficient core flooding to keep the calculated peak clad temperatures below the required limit of 10 CFR 50.46. Hence adequate protection is afforded by the Emergency Core Cooling System in the event of a small break loss-of-coolant accident.

Additional Break Cases

Studies documented in Reference 3 determined that the limiting small-break size occurred for breaks less than 10 inches in diameter. To insure that the 4-inch diameter break was limiting, calculations were run with breaks of 3 inches and 6 inches. The results of these calculations are shown in the Sequence of Events Table C.3.2-2, and the Results Table C.3.2-4. Plots of the following parameters are shown in Figures C.3.2-11 through 18 for the 3-inch break, and Figures C.3.2-19 through 26 for the 6-inch break.

- RCS pressure.
- Core mixture level
- Peak clad temperature
- Core outlet steam flow rate
- Hot spot rod surface heat transfer coefficient
- Cold leg break mass flow rate
- Safety injection mass flow rate

Hot spot fluid temperature

As seen in Table C.3.2-4 the peak clad temperatures were calculated to be less than that for the 4-inch break.

Additional Analysis

Calculations were also performed for Cook Nuclear Plant Unit 2 with the NOTRUMP[10][11] and LOCTA-IV[2] codes to examine the influence of initial RCS coolant operating temperatures and operating pressures on small break LOCA peak clad temperature. The analyses performed demonstrated that the reduced pressure and high temperature conditions analyzed resulted in the limiting PCT for the 4-inch diameter break.

To support operation of Cook Nuclear Plant Unit 2 at RCS pressures of 2100 psia and 2250 psia for a range of loop operating temperatures, two additional analyses were performed. Calculations were performed for a four-inch diameter break for an initial RCS pressurizer pressure of 2250 psia at initial RCS coolant operating temperatures corresponding to core Tavg program setpoints of 581.3 °F and at a Tavg of 547.0 °F. The results of these calculations are shown in the Sequence of Events Table C.3.2-3, and the Results Table C.3.2-5.

Plots of the following parameters are shown in Figures C.3.2-27 through 34 for the high pressure, high temperature case, and Figures C.3.2-35 through 42 for the high pressure, low temperature case.

- RCS pressure
- Core mixture level,
- Peak clad temperature,
- Core outlet steam flow rate,
- Hot spot rod surface heat transfer coefficient,
- Hot spot fluid temperature,
- Cold leg break mass flow rate, and
- Safety injection mass flow rate.

As seen in Table C.3.2-5, the peak clad temperatures were calculated to be less than that for the 4-inch break initiated at reduced pressure and high temperature conditions.

Additional calculations were made to support closure of the high head safety injection cross tie valves. Since the amount of pumped injection flow would be reduced with the high head cross tie valves closed, it was necessary to lower core power in order to maintain the peak clad temperature within the 10 CFR 50.46 limit. Thus the calculation which supports plant operation with the high head cross tie valves closed assumed an initial RCS pressurizer pressure at 2100 psia and core Tavg at 581.3 °F at a core power level of 3413 Mwt. This calculation also assumed a charging system flow rate imbalance of 25 gpm. The assumed pumped ECCS flow performance for the high head cross tie valve closed case is listed in Table C.3.2-9.

Past analyses have shown that a reduction in pumped safety injection flow rate increases the peak clad temperature for smaller breaks (3 inches) more than larger small breaks (4 and 6 inches). An important parameter in determining what will be the limiting break size is the reactor power to safety injection flow rate ratio. For the high head cross tie closure case the reactor power to safety injection flow rate ratio was reduced which shifted the limiting break size to the 3-inch diameter cold leg break. Evidence of this effect is the Cook Nuclear Plant Unit 1 small break LOCA analysis which was performed with the high head cross tie valves closed assuming a charging system flow rate imbalance of 10 gpm with a reactor power of 3588 Mwt. The Cook Nuclear Plant Unit 1 small break LOCA analysis had a reactor power to safety injection flow rate ratio approximately equal to Cook Nuclear Plant Unit 2 with the high head cross tie valves closed assuming 25 gpm charging system flow rate imbalance at a reactor power level of 3413 Mwt. It was concluded that with the high head cross tie valves closed and with reduced reactor power the limiting break would be shifted from the 4-inch diameter cold leg break to the 3-inch diameter break size. To verify this conclusion, two calculations were performed which assumed break sizes of 3- and 4-inch diameters at the reduced pressure, high temperature initial conditions. Table C.3.2-8 lists the results of the cross tie closed cases which show that with the reduced safety injection flow the 3-inch diameter break is limiting. The sequence of events for these calculations is listed in Table C.3.2-7. Past small break LOCA analyses that were performed for plants which are similar to Cook Nuclear Plant Unit 2 but have power to safety injection flow rate ratios less than that of Cook Nuclear Plant Unit 2, have shown that an assumed break size of 2 inches did not result in the limiting peak clad temperature. Thus, based on the comparision of power to safety injection flow rate ratio, it was concluded that a 2-inch diameter break would not yield a peak clad temperature more limiting that that of the 3-inch diameter break size. Plots for the 3- and 4-inch break with the HHSI cross tie valves closed are shown in figures C.3.2-43 through C.3.2-50 and C.3.2-51 through C.3.2-58 respectively.

NUREG-0737(13), Section II.K.3.31, required plant-specific small break LOCA analysis using an Evaluation Model revised per Section II.K.3.30. In accordance with NRC Generic Letter 83-65(14), generic analyses using NOTRUMP(10)(11) were performed and are presented in WCAP-11145(17). Those results demonstrate that in a comparison of cold leg, hot leg and pump suction leg break locations, the cold leg break location is limiting.