15.1 Increase in Heat Removal by the Secondary (Cold Water Addition)

The SCS is the principal heat removal system for the ATR. Decreasing the secondary temperature or increasing the secondary flow rate will result in an increased heat removal rate from the ATR. Opening of the secondary coolant control valve with the resultant increase in coolant flow is an example of a condition that increases the heat removal rate. The secondary temperature and flow can also be influenced by changes in support systems such as the cooling tower operation and the number of operating secondary pumps.

Depending upon the core configuration, the temperature reactivity feedback coefficient can be either positive or negative for isothermal changes.[8] The flux trap loading can determine the sign of the isothermal temperature coefficient. If each flux trap is occupied by an independently cooled facility (e.g., IPT with associated experiment loop), the coefficient is usually negative since the fuel behavior will dominate. If there are several flux traps cooled by primary coolant, the coefficient could be positive since the flux trap behavior will dominate. If the coefficient is negative, the cold water addition will result in a positive reactivity insertion. Experience has shown that the coefficient is typically near zero, and the effect is therefore relatively small and not limiting for safety analysis considerations. The consequences of these temperature changes are limited power transients similar to those discussed in Section 15.4.

An increase in the heat removal from the PCS results in a decreased inlet temperature. Cold water added to the primary coolant can also result in a decrease in the inlet temperature.

A number of faults which could result in addition of cold water to the reactor core are postulated. These events are discussed in this section assuming that the temperature coefficient is negative. If the coefficient is positive, a heatup of the coolant would result in the power transient. The mechanism to obtain the power transient from the secondary behavior can be reversed (i.e., close the temperature control valve rather than open it). Therefore, the consequences of the limiting event will still envelop the sequences.

A number of faults which could result in addition of cold water to the reactor core are postulated. These events are discussed in this section.

A. SCS FCV fails to the open position

B. Excessive heat removal from the SCS

C. Starting a standby PCP

(Note: If the temperature coefficient is positive, then the resulting reactivity insertion would be due to a "hot" instead of a "cold" water addition, and then event A becomes closing of the valve from the open position, and event B becomes insufficient heat removal.)

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[8]. Isothermal in this case refers to uniform fluid temperature in the core. Uniform fluid temperature changes or isothermal changes are accomplished by changing the reactor vessel coolant inlet temperature.

Based on the analysis in the DBR (EG&G Idaho 1994), items A and B are considered Condition 2 events and item C is considered a Condition 3 event. These sequences have been evaluated in the DBR (EG&G Idaho 1994) and are bounded by the limiting reactivity insertions (Section 15.4).

The discussion of these events is based on information contained in the DBR (EG&G Idaho 1994).

15.1.1 Secondary Coolant System Flow Control Valve Fails to the Open Position

Opening of the secondary FCV results in increased flow rate to the secondary side of the primary heat exchangers. A bounding analysis with the secondary coolant at 32°F and an increase to full flow from an initial flow rate of 5% of full flow indicates a reactivity insertion of 1.7$ added at a rate of 0.03$/sec (EG&G Idaho 1994).

This event results in a very limited overpower as indicated in Figure 15.4-2. The overpower is less than 25% and is acceptable as described in Section 15.4.2.

15.1.2 Starting a Standby Primary Coolant Pump

The PCS consists of one coolant loop with five heat exchangers and four PCPs. The heat exchangers do not have isolation valves; therefore, the only source for cold water is the standby pump(s). The enclosed volume of a pump and corresponding dead legs is small. The reactivity worth of the volume corresponds to less than 0.10$ (EG&G Idaho 1994) and is bounded by the 0.50$ step insertion discussed in Section 15.4.5.

15.1.3 Excessive Heat Removal From the Secondary Coolant System

The secondary coolant heat load is dissipated to the atmosphere via a cooling tower. A change in fan speed, number of operating fans, and volume of water over the tower will result in a change to the off-tower water temperature. However, the large basin volume will make temperature changes to the heat exchangers occur over a long time frame compared to other events discussed in Section 15.4 and thus, this event is bounded. The event is also bounded by the sequence discussed in Section 15.1.1.

15.1.4 References

EG&G Idaho, 1994, *ATR PPS and Technical Specifications Design Basis Report*, (SAR-39), Issue 030, EG&G Idaho, Inc., August 2, 1994.

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15.2 Decrease in Heat Removal by the Secondary

The SCS is the principal heat removal system for the ATR. Increasing the secondary temperature or decreasing the secondary flow rate will result in a decreased heat removal rate for the ATR. A break in the secondary piping with a resultant loss of secondary coolant inventory from the primary heat exchangers is a severe example. The secondary temperature and flow can also be influenced by failure of support systems such as instrument air. Oxide or crud buildup on the heat exchanger tubes can also decrease the rate of heat removal. These phenomena are normally slow to develop (days or weeks for significant changes).

Decrease in heat removal from the primary system results in an increased primary coolant temperature. High primary coolant temperatures result in:

- Lower margins to CHF
- Increased liquid volume due to thermal swell
- Increased pipe wall and reactor vessel temperatures.

A number of faults which could result in a decrease in heat removal are postulated and discussed in this section. A detailed analysis is presented for the most limiting of the following events which result in a decrease in heat removal from the primary coolant system:

- SCS FCV closed against stop.
- Cooling tower failure.
- Loss of commercial power.
- Failure of secondary pumps.
- Loss of secondary flow from either a secondary flow control valve complete closure or closure of all six motor operated cooling tower return valves.
- Failure of cooling tower fans.
- Loss of cooling tower basin water.
- Excessive heat exchanger tube fouling.
- Long-term loss of commercial power.
- Secondary system rupture upstream of primary coolant heat exchangers (complete loss of secondary).
- Extended loss of commercial, diesel, and utility DC power.

Based on analyses in the DBR (EG&G Idaho 1994), items A, B, and C, are considered to be Condition 2 events, items D, E, F, G, and H, are Condition 3 events, and items I, J, and K are Condition 4 events.

The piping failure upstream of the heat exchangers (event J) is analyzed as a complete and abrupt loss of secondary coolant. It becomes the enveloping event for all the other sequences that result in the degradation of the heat sink.

15.2.1 Complete Loss of Secondary (Complete Loss of Heat Sink)

15.2.1.1 Identification of Causes and Accident Description. Catastrophic failure of the piping between the secondary pumps and the heat exchangers would result in complete loss of flow to the heat exchangers with a rapid depletion of the coolant inventory from the secondary of the heat exchangers. The combined events would result in essentially no heat removal from the primary coolant. (Some heat loss occurs through the uninsulated primary coolant piping; however the effect is conservatively neglected in the analysis.)

The necessary protection for this event is provided by diverse protective systems or functions dependent upon the time interval. Initial protection for the reactor fuel is provided by a reactor trip due to high inlet temperature. This trip is actuated by two out of three logic driven by the inlet temperature PPS subsystem. Diverse protection is provided by the high outlet coolant temperature PPS subsystem.

As the temperature of the primary coolant continues to rise due to the decay heat, the volume will increase, resulting in a pressure rise. The pressure relief valves will maintain the pressure below damage levels.

If alternate cooling is not established by the time the primary coolant reaches approximately 200°F, the vessel vent valves are assumed to be manually opened to limit the coolant temperature to the boiling temperature at atmospheric pressure. The EFIS can be used at atmospheric pressure to maintain the coolant inventory.

15.2.1.2 Analysis of Effects and Consequences

15.2.1.2.1 Method of Analysis—The thermal-hydraulic response for the ATR system as a result of this fault was analyzed using RELAP5/MOD 2.5 with four sets of initial conditions (Pafford 1994):

1. A total reactor power of 250 MW with a balanced 50 MW lobe power split, three PCPs operating, and a core inlet temperature of 125°F.
2. A total reactor power of 230 MW with the SAR 70 lobe power split, three PCPs operating, and a core inlet temperature of 125°F.
3. A total reactor power of 250 MW with a balanced 50 MW lobe power split, three PCPs operating, and a core inlet temperature of 94°F (the lowest expected inlet temperature for full power operation).
4. A total reactor power of 250 MW with the 60/40 MW lobe power split, two PCPs operating, and a core inlet temperature of 125°F.

These reactor power levels and inlet temperatures were evaluated to ensure event consequences were bounded.

The RELAP5 model had either two or three PCPs operating with a five channel core model (four quadrant channels and one "hot" fuel element channel). The secondary side coolant was removed at the time of initiation with the heat exchanger structure retained. Radiative and convective heat losses from the PCS piping and reactor vessel exterior surfaces were not modeled. This results in a conservative prediction for primary coolant temperature response.

Reactor trip was from the high vessel inlet temperature subsystem at 150°F, with a 40-sec response time. The PCPs were assumed to be manually tripped at approximately 10 minutes; the vessel vent system was manually initiated when the PCS temperature reached 200°F. The loss of heat sink results in a pressure increase that was conservatively assumed limited by the relief valve opening rather than a high pressure scram which stops the pressurizing pumps (also, the pressure control system could limit the pressure to the normal range).

The reactor protection criteria margins were evaluated using ATR-SINDA and SINDA-SAMPLE with boundary conditions from the RELAP5 results (Polkinghorne 1994). The analysis was completed for fuel plate 19. Figure 15.0-5 depicts the ATR-SINDA fuel plate model.

15.2.1.2.2 Results—The sequence of major events from the RELAP5 calculations is shown in Table 15.2-1 for the three-pump cases and in Table 15.2-2 for the two-pump case. The vessel relief valves lift shortly after transient initiation. The peak hot channel temperature which occurs for two-pump operation is 246.4°F with a peak vessel outlet temperature of 205.6°F. The maximum vessel inlet temperature was 203.2°F. The peak vessel temperature occurs late in the transient which is dominated by decay heat and is insensitive to the number of pumps operated. The time response for the vessel inlet and outlet temperature is illustrated in Figures 15.2-1, -2 and -3.

For three-pump operation, the peak fuel plate temperature calculated with ATR-SINDA is 575.0°F occurring just prior to reactor scram. The minimum CHF and FI margins, 5.14σ and 7.68σ, respectively, occur just prior to reactor scram at 143 sec. The maximum fuel-plate temperature and CHF and FI margins are shown in Figures 15.2-4 and -5, respectively. For two-pump operation, the peak fuel plate temperature calculated with ATR-SINDA is 525.0°F occurring just prior to reactor scram. The minimum CHF and FI margins, 5.46σ and 8.31σ, respectively, occur just prior to reactor scram at 72 sec. The maximum fuel plate temperature and minimum CHF and FI margins are shown in Figures 15.2-6 and -7, respectively.

The two-pump calculation was terminated after 200 s since sufficient data for evaluation of the fuel element performance was generated. The results in Table 15.2-1 show that peak fuel plate temperatures occur at the time of the scram (see also Figure 15.2-4). The long-term effect of concern is the heating of the piping; the results for the three-pump calculation are sufficient to determine limits and controls.

15.2.1.2.3 Conclusions—A high vessel inlet temperature subsystem, which initiates reactor trip at 150°F with a 40-sec response time, will preserve the reactor protection criteria for CHF and FI. The vessel relief valves with a nominal lift pressure setpoint of ≤ 390 psig will provide overpressure protection. The PCS must be depressurized before the primary coolant inlet temperature exceeds 200°F. Depressurizing the PCS ensures that the 240°F design temperature for the PCS is not exceeded and that the fuel plate temperature remains a few degrees above the local fluid saturation temperature. Depressurization can be accomplished in several ways (normal pressure control system, opening the vessel vent valves, or bleeding air from the surge tank); however, the vessel vent valves comprise the safety system for depressurizing to allow EFIS addition. The scram at 150°F ensures that the margins to CHF in the hot fuel element are at least 3σ.

The analysis included actuation of the EFIS following opening of the vent valves before the inlet temperature exceeds 228°F; the analysis demonstrates that EFIS addition will terminate the temperature transient. However, the manual actuation of EFIS is not necessary if full depressurization of the PCS occurs before exceeding 200°F. The calculation indicated that the addition of cooler water can reduce temperatures below those determined by the saturation temperature. The limits on inlet temperature (≤ 200°F and ≤ 228°F) can be conservatively applied to the outlet temperature since outlet temperature exceeds the inlet temperature during this phase of the transient. Assuming full depressurization occurs, actuation of firewater should be done only to maintain inventory since addition of fire water drastically degrades chemistry as well as causing building flooding because the firewater flow exceeds the capability to dispose of liquid effluents. The evaluation of vessel level setpoints in Section 15.9 and EFIS setpoints in Section 15.3 are sufficient to maintain the inventory and ensure piping temperatures are limited.

15.2.2 Bounded Events - Decrease in Heat Removal by the Secondary

15.2.2.1 Secondary Coolant System Flow Control Valve Closed Against Stop. Closure of the SCS flow control valve, TCV-1-10, results in a secondary flow reduction to not less than 2,000 gpm. Closure can occur due to operator error, controller failure, or valve component mechanical failure.

Temperature rise rate for the primary coolant will be slower than for the event discussed in Section 15.2.1 (loss of heat sink assumed to be instantaneous), and after reactor scram, the decay energy will be removed. Thus, the sequence is bounded by the discussion in Section 15.2.1.

15.2.2.2 Failure of Secondary Pumps. One through four pumps, in any combination, supply coolant to the secondary system. Two lower capacity UCW pumps are also available.

Pump failure may be caused by blockage, by failure in the pump/motor/coupling assembly, or by a loss of power (local equipment failure). These result in a range of consequences from partial loss to total loss of secondary coolant flow. The PCS response is bounded by the response discussed in Section 15.2.1 for these conditions.

15.2.2.3 Loss of Secondary Flow from Either a Secondary Flow Control Valve Complete Closure or Closure of All Six Motor Operated Cooling Tower Return Valves. These events are Condition 3 events or below. There are flow paths in the secondary coolant system upstream and downstream of the primary coolant heat exchangers that could be blocked by valve closure(s). The secondary flow control valve, TCV-1-10, if completely closed would block the secondary flow supply to the primary heat exchangers. A mechanical stop in the valve prevents complete closure; however, failure of the stop or valve shaft would allow complete closure. The closure of all six motor operated cooling tower return valves could also block secondary flow return to the cooling tower or basin. Multiple valve closures would be required for complete flow blockage. Inadvertent operation of all valves to the closed position resulting from control system faults is prevented by administrative controls that are in place to disable the motor operator of at least one cooling tower riser valve from being inadvertently controlled by the DCS. The events are bounded by the complete loss of secondary discussed in Section 15.2.1.

15.2.2.4 Failure of Cooling Tower Fans. The cooling tower has four bays with a fan on each bay. Partial or total failure of the fans will result in a slow temperature increase in the supply to the heat exchangers. This sequence is bounded by the loss of heat sink discussed in Section 15.2.1 since the increase is slow and flow is maintained.

15.2.2.5 Loss of Cooling Tower Basin Water. Loss of the basin water inventory may occur due to an open drain valve or due to failure of the secondary coolant makeup. Several hours are required for total loss. The sequence is bounded by the loss of heat sink discussed in Section 15.2.1 since the delay time is long before complete loss, even if undetected by the operators.

15.2.2.6 Cooling Tower Failure. Tower icing, failure of the cell packing, flow distribution system, or flow being bypassed to the basin will result in a slow temperature rise of the basin inventory. Again, the sequence is bounded by the complete loss of secondary discussed in Section 15.2.1.

15.2.2.7 Excessive Heat Exchanger Tube Fouling. The process of fouling requires appreciable time. Gradual increase in the primary coolant temperature would restrict reactor operation if the 125°F limit could not be maintained. The sequence is bounded by the complete loss of heat sink discussed in Section 15.2.1.

15.2.2.8 Loss of Commercial Power and Long-Term Loss of Commercial Power. Loss of commercial power will result in a loss of secondary flow since the secondary coolant pumps are powered from the commercial bus. The secondary flow will coast down to UCW flow, and the sequence approaches the complete loss of heat sink. The sequence is bounded by the complete loss of heat sink discussed in Section 15.2.1. The more global effects of loss of commercial power are discussed in Sections 15.3 and 15.11.

15.2.2.9 Extended Loss of Commercial, Diesel, and Utility DC Power. Loss of all power sources for an extended period of time will result in loss of all secondary flow as is considered in Section 15.2.1. The global effects of the failure of these power sources are considered in Section 15.3. The event is bounded by the loss of PCS flow analysis discussed in Section 15.3.

15.2.3 Failure of Inlet Temperature Subsystem

The outlet temperature PPS subsystem has a reactor trip at 188°F for three-pump operation and 185°F for two-pump operation and a response time of 25 sec. The difference between the inlet and outlet setpoints is very nearly the core differential temperature at 250 MW. Since the transit time for the coolant from the inlet sensor to the outlet sensor is about 25 sec, the inlet scram and the outlet scram signals are generated with about a 10-sec differential (setpoints are reached with a 25-sec differential but the outlet response time is 15 sec shorter). As shown in Figures 15.2-5 and -7, the margins to CHF change little if any in 10 sec. Therefore, the margins to CHF are greater than 3.0σ, and there is no approach to CHF. The piping and fuel plate temperatures are little affected by the 10 sec delay in the scram; the long term temperature is limited by the depressurization to atmospheric pressure. The time at which depressurization is required may be reached a few seconds earlier, but the change is not significant relative to the actual time (perhaps a 10 sec change in 2600 sec). Since the event is adequately limited by an outlet temperature scram at 188°F or 185°F (3-PCP and 2-PCP operation, respectively), the outlet temperature PPS subsystem is diverse to the inlet temperature PPS subsystem.

Table 15.2-1. Sequence of events for the loss of heat sink transient calculations (three-pump operation).

	Time(s)		
Event	250 MW[a]	250 MW[b]	230 MW[a]
Transient initiated	0.0	0.0	0.0
Safety relief valves first lift	< 3.0	< 2.0	< 3.0
Reactor scram on high inlet temperatures (includes 40.0 second delay)	71.8	142.9	73.3
Peak hot channel exit temperature reached	71.8	142.9	73.3
(Peak temperature)	(218.5°F)	(219.8°F)	(241.7°F)
PCPs manually tripped	671.8	742.9	673.3
PCS temperature reaches 200°F (at core inlet)	2672.	3020.	3347.
Vent valves manually opened	2672.	3020.	3347.
Lower firewater injection started	2747.	3095.	3422.
Peak vessel outlet temperature reached	2778.	3118.	3452.
(Peak temperature)	(205.6°F)	(205.6°F)	(202.8°F)
Peak vessel inlet temperature reached	3148.	3490.	3812.
(Peak temperature)	(203.2°F)	(203.2°F)	(204.8°F)
Transient calculation terminated	4500.	4500.	4500.

a. Core inlet temperature of 125°F.

b. Core inlet temperature of 94°F.

Table 15.2-2. Sequence of events for the loss of heat sink transient calculations (two-pump operation with 125°F inlet).

Event	Times (s)
Transient initiated	0.0
Safety relief valves first lift	< 2.0
Reactor scram on high inlet temperature	72.4
(includes 40.0 second delay)	
Peak hot channel exit temperature reached	72.4
(Peak temperature)	(246.4°F)
Transient calculation terminated	200

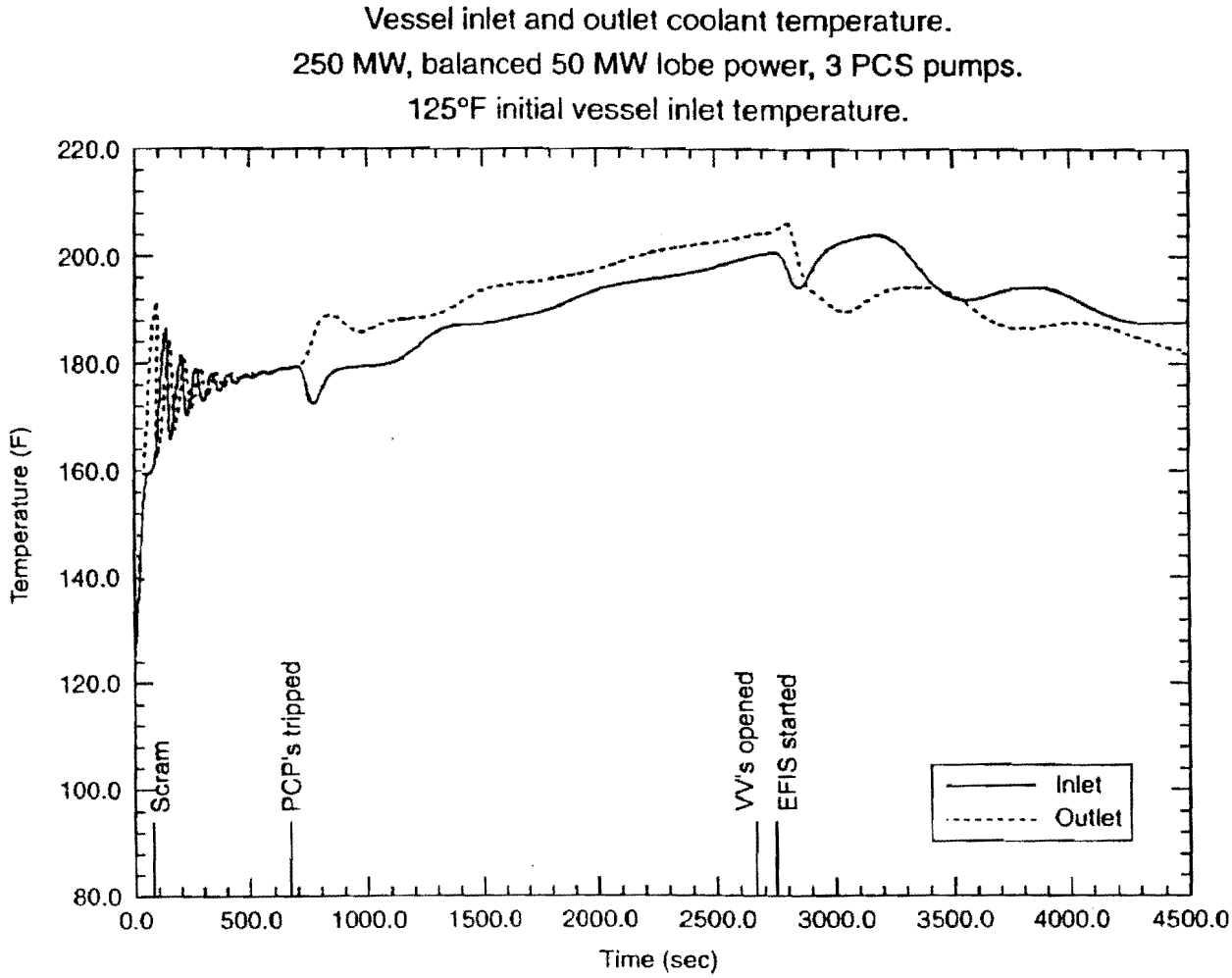

Figure 15.2-1. Vessel inlet and outlet coolant temperatures: 250 MW, balanced 50 MW lobe power distribution; 125°F initial vessel temperature; three-pump operation. **(For Information Only)**

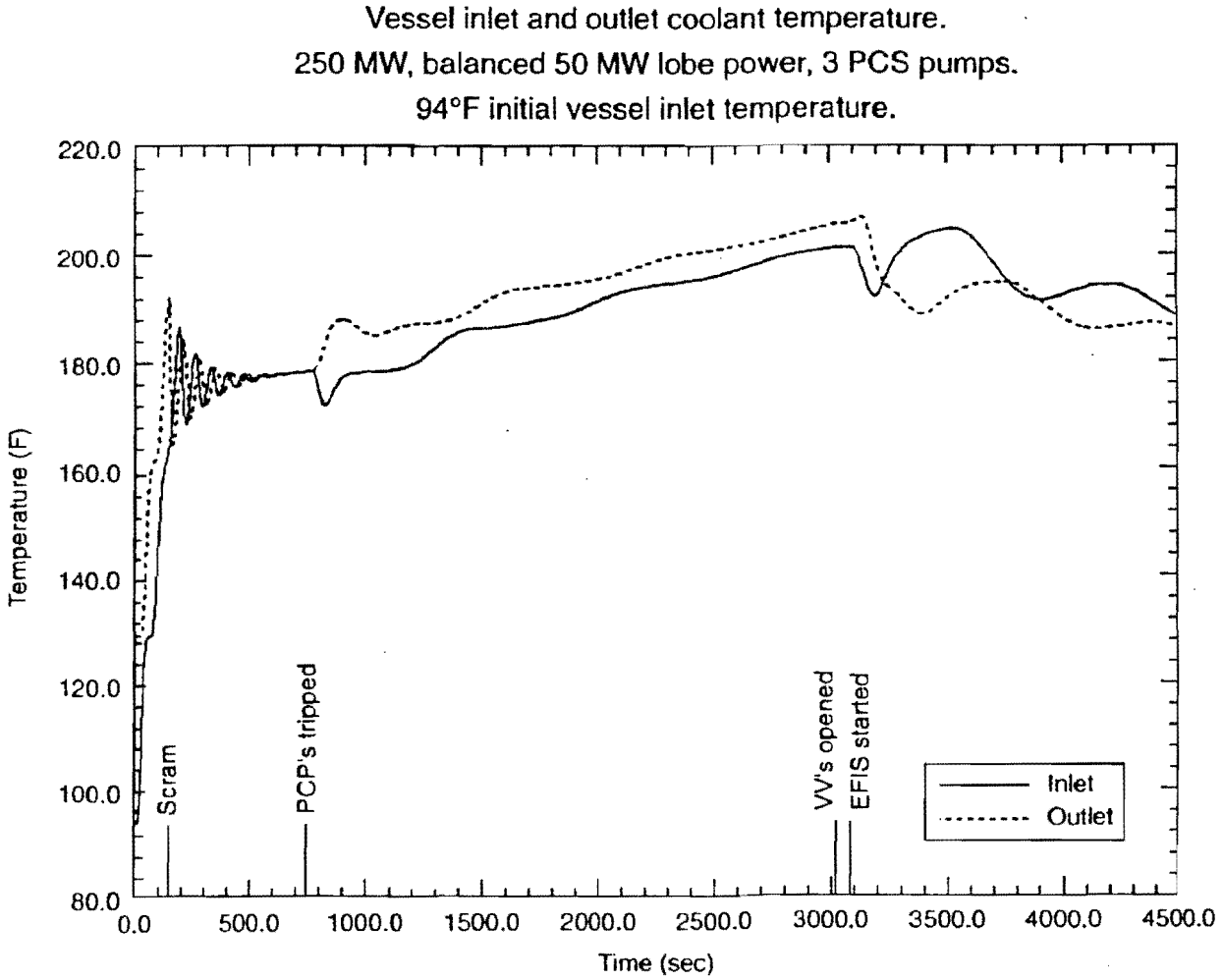

Figure 15.2-2. Vessel inlet and outlet coolant temperatures: 250 MW, balanced 50 MW lobe power distribution; 94°F initial vessel temperature; three-pump operation. **(For Information Only)**

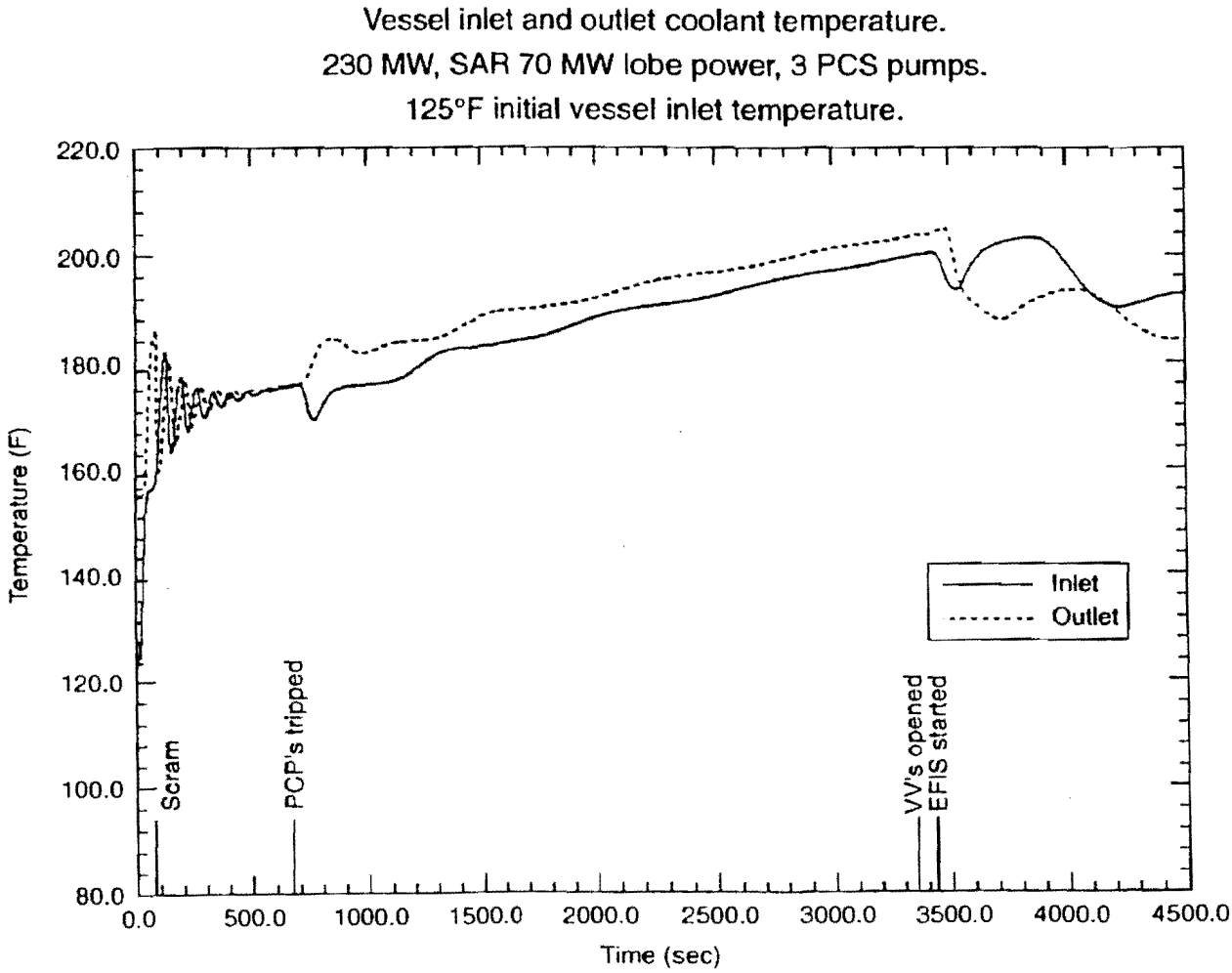

Figure 15.2-3. Vessel inlet and outlet coolant temperatures: 230 MW, SAR 70 MW lobe power distribution; 125°F initial vessel inlet temperature; three-pump operation. **(For Information Only)**

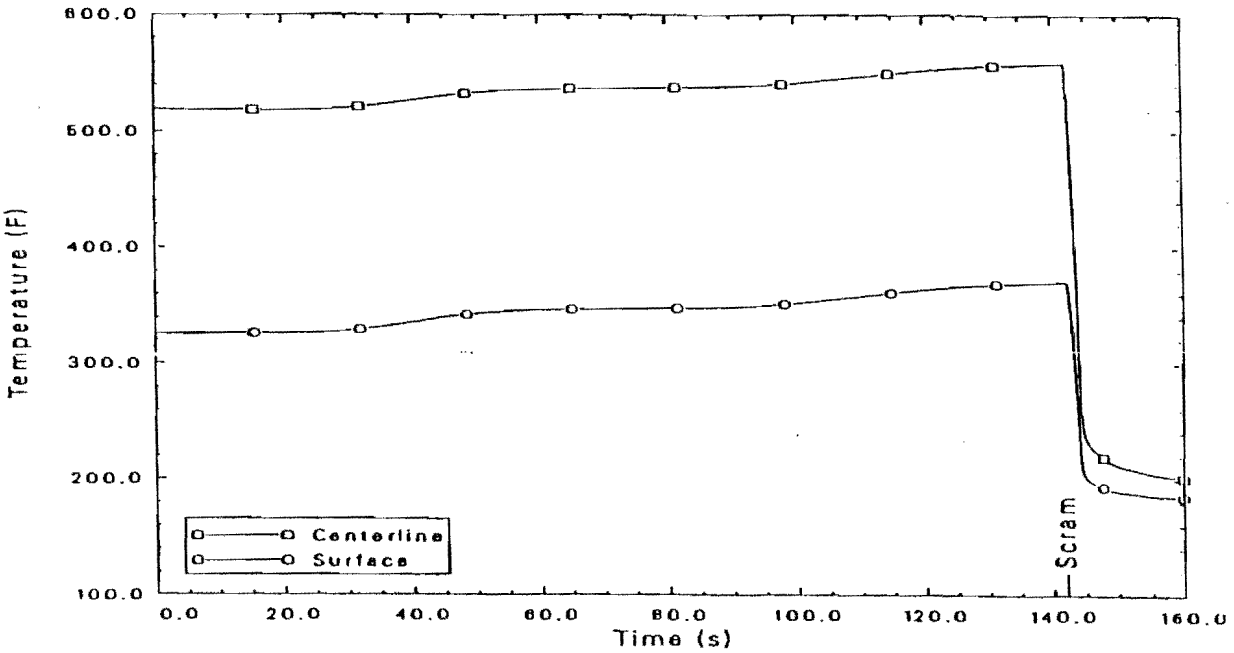

Figure 15.2-4. Maximum fuel plate temperature (LOHS accident, three-pump operations). **(For Information Only)**

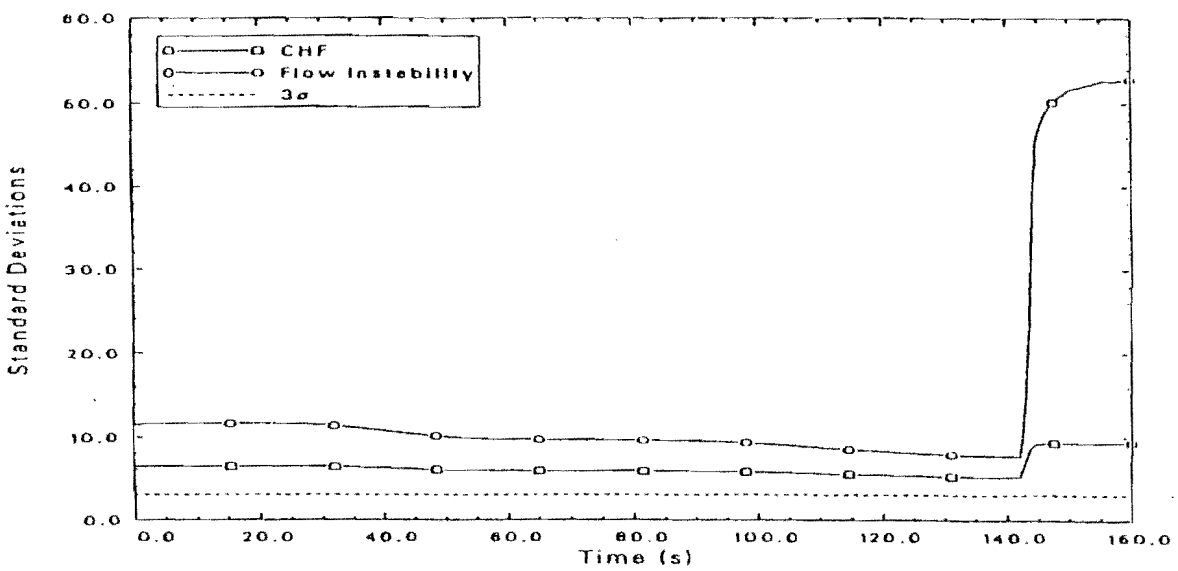

Figure 15.2-5. CHF and FI margins (LOHS accident, three-pump operation). **(For Information Only)**

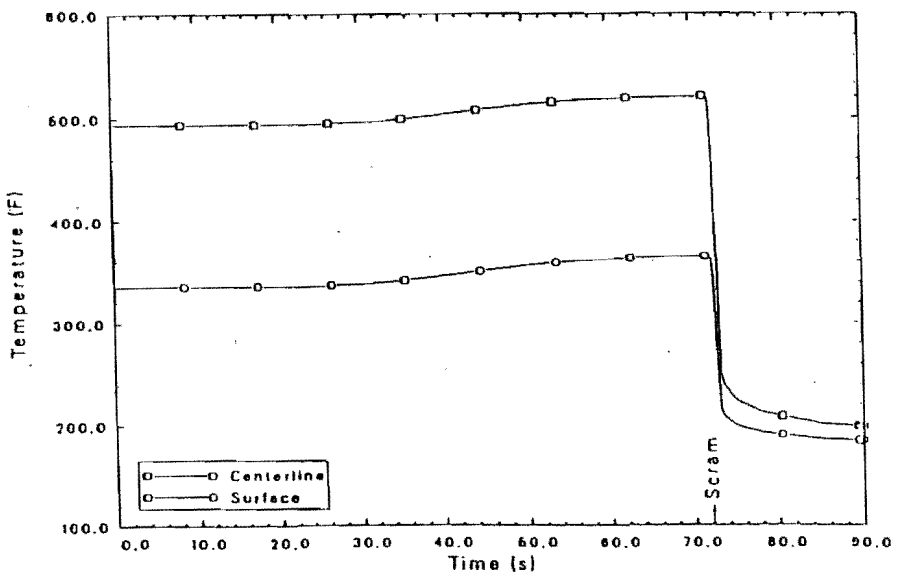

Figure 15.2-6. Maximum fuel plate temperatures (LOHS accident, two-pump operation).
(For Information Only)

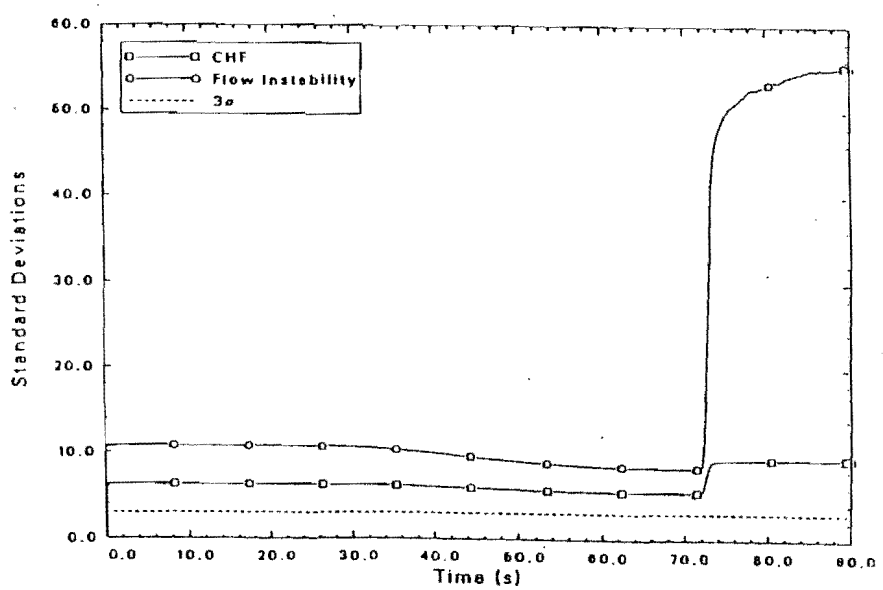

Figure 15.2-7. CHF and FI margins (LOHS accident, two-pump operation).
(For Information Only)

15.2.4 References

EG&G Idaho, 1994, *ATR PPS and Technical Specifications Design Basis Report*, (SAR-39), Issue 030, EG&G Idaho, Inc., August 2, 1994.

Pafford, D. J., 1994, *RELAP5 Analysis of the Loss of Heat Sink Transient for the Advanced Test Reactor Updated Final Safety Analysis Report*, EDF TRA-ATR-846, EG&G Idaho, Inc., June 23, 1994.

Polkinghorne, S. T., 1994, *ATR-SINDA and SINDA-SAMPLE Calculations for Chapter 15 of the ATR's Updated FSAR*, EDF TRA-ATR-840, EG&G Idaho, Inc., February 11, 1994.

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15.3 Decrease in Reactor Primary Coolant Flow Rate

The PCS consists of a single loop with some parallel flow paths to provide for four PCPs, five primary heat exchangers, two inlet pipe segments, and four discharge pipe segments. There are also two emergency coolant pumps which are in parallel to each other and to the PCPs.

The normal mode of operation is with either two or three PCPs and one emergency coolant pump operating. Gross PCS flow is controlled by the number of operating primary pumps, and fine flow adjustments are made with the butterfly FCV.

The PCPs are powered by commercial power. Thus, a loss of commercial power will result in a coastdown of the primary pumps. One emergency coolant pump operates from an uninterruptible power supply (UPS), that is normally supplied by a commercial bus (dc-motor), the other operates on the diesel-commercial bus (ac-motor). If the diesel-commercial power supply were to fail in conjunction with the commercial power, the emergency coolant pump with battery backup will provide short-term forced flow (the pump starts if in standby mode and continues to operate if in service). The battery bank will operate one emergency coolant pump for a minimum of 30 minutes which allows actions to restore the other power sources and allows time for core decay heat to decrease.

The primary coolant flow rate can also be decreased by closure of the butterfly FCV, by closure of block valves, by failure of pumping components, or by partial blockages. A decrease in the primary coolant flow rate results in a decrease in the heat transfer coefficient and an increase in the coolant temperature.

A number of faults which result in a decrease of the reactor primary coolant flow rate are postulated. A discussion of these events is included in this section. Detailed analyses are presented for the limiting faults which result in decreased primary coolant flow rates. The identified faults are as follows:

A. Trip of one (two-pump operation) or two (three-pump operation) of the operating PCPs

B. Loss of commercial power (or a severe commercial power undervoltage or underfrequency)

C. Sleeve failure in a 16-in. outlet pipe

D. Failure of one operating PCP due to mechanical failure

E. A closure of the PCS butterfly valve from operating position (approximately 45% open) to 38% open

F. Slow closing primary pump discharge check valve

G. Complete closure of the flow control butterfly valve

H. Closure of a 24-in. reactor inlet isolation valve

I. Rupture of a 16-in. outlet pipe inside the ATR vessel

J. A long-term complete loss of flow

K. Failure of the first actuating PPS subsystem during a Condition 2 event.

Based on analysis in the DBR (EG&G Idaho 1994) and Brower (1993), items A-E are considered to be Condition 2 events, Item F a Condition 3 event, and items G-K Condition 4 events.

The limiting Condition 2 event is a loss of commercial power to the site. For Condition 3, the limiting event is the slow closing of a primary pump discharge check valve. The limiting Condition 4 event is a long-term, complete loss of flow. These events are discussed in Sections 15.3.1 through 15.3.4. Section 15.3.5 contains a discussion of the other events enveloped by the analysis. Maintaining the accident frequencies above requires limiting probability of fires that can result in loss of flow (see Section 15.16.1).

15.3.1 Loss of Commercial Power

15.3.1.1 Identification of Causes and Accident Description. Loss of commercial power, a Condition 2 event, will result in a coastdown of the operating PCPs. The loss can be of any duration, dependent on the cause. For example, the loss can be caused by weather, line failure, breaker failures, or dispatcher failure. The operability of the emergency coolant pumps is not influenced by this event. The flow will coast down to emergency coolant flow (the reactor is scrammed on low inlet pressure), and decay heat will be removed by the SCS (circulation provided by the UCW).

The necessary protection for this event is provided by a diverse protective system. Initial protection for the reactor fuel is provided by a reactor trip initiated by the inlet pressure PPS subsystem. Diverse protection is provided by the vessel differential pressure PPS subsystem.

The pressure taps for the protective channels are located in the inlet and outlet piping in the pipe tunnel. The RELAP5 model located these taps at the location of the pressure taps for control of the inlet pressure (the taps for PIT-1-1). These taps are at a lower elevation than the PPS taps. Therefore, the modelling results in a conservative result for the low-pressure subsystem (Pafford 1994c) because of the elevation head difference. For the differential pressure subsystem, the elevation head is not an issue (elevation effects are removed by taking the difference in two pressures). Although flow losses are different at the two locations, the difference is small and the evaluation by Pafford (1994c) shows that the modelling is conservative.

15.3.1.2 Analysis of Effects and Consequences

15.3.1.2.1 Method of Analysis—The response of the PCS was analyzed with RELAP5/MOD 2.5 (Terry 1994). The reactor core was modeled as four quadrant channels and one "hot" fuel element channel. The two-pump and three-pump operating modes were evaluated.

The analysis included the core feedback reactivity including the flux traps. Two reactivity tables were used to represent feedback from the flux traps including three flux traps with capsule irradiation facilities and six flux traps with IPTs. This was the core configuration between the 1993 process control upgrade and the 1994 core internals changeout. The input tables were generated from two separate RELAP5 analyses for the two flux trap types. These two calculations used the reactor power history obtained for a flow coastdown transient without feedback (Nielsen 1991 and 1993). The RELAP5 void fractions were coupled with PDQ calculations of reactivity versus flux trap void fraction to obtain functions of reactivity versus time for the two flux trap types. The reactivity feedback from the IPTs results from loss of half the operating loop pumps when commercial power is lost. The reactivity feedback from the flux traps without IPTs and for the flux traps with IPTs is shown in Tables 15.3-1 and -2, respectively. The flux trap feedback as a result of the coastdown is positive and is represented by

the sum of the data in these two tables. The overall core feedback is negative. The flux trap feedback does not become significant by the time minimum margins are reached; as shown below the minimum margins are reached in approximately one second. The flux trap facilities were changed to five flux traps with IPTs and four flux traps with capsule irradiation facilities during the 1994 core internals changeout. As shown in Tables 15.3-1 and -2 and discussed in Terry (1994), the IPT flux traps insert more positive reactivity than the capsule irradiation facility flux traps during the time period of concern for this event (i.e., ~ 1 second). Therefore, the six IPT/three capsule facility configuration used in the analysis envelops the baseline five IPT/four capsule facility configuration. The positive flux trap feedback for the 5/4 configuration would be smaller than that for the 6/3 configuration.

Due to the nature of ATR operation new experiments are occasionally inserted into the reactor. When new experiments are placed into the reactor, additional analysis is performed to provide assurance that the reactor response with new experiments meets the established safety envelope.

The secondary coolant pumps were assumed to coast down due to the loss of commercial power.

Reactor trip was from the low vessel inlet pressure subsystem with a setpoint of 324.8 psia and a response time of 0.1 sec. The same setpoint was used for both two-pump and three-pump operation.

15.3.1.2.2 Results—The sequence of major events for the RELAP5 transients is tabulated in Table 15.3-3. The temperatures for the fuel plates from the RELAP5 calculations are shown for information although they are not used for the analysis of the plates; the ATR-SINDA model is used to establish the plate temperatures and limits. The time histories for the core differential pressure are depicted in Figures 15.3-1 and -2 for two-pump and three-pump operation, respectively. The loss of pumping power results in a low-pressure scram in approximately one second.

The peak fuel plate temperatures as calculated with ATR-SINDA (Polkinghorne 1994) are 515°F with two pump (Figure 15.3-3) and 555°F for three pumps (Figure 15.3-4). These fuel temperatures, 515°F and 555°F, are higher than the fuel temperatures for the RELAP5 results shown in Table 15.3-3 because ATR-SINDA models the core hot spot while RELAP5 models the hot element. The RELAP5 models do not accurately reflect the detailed power distribution within the fuel plates of the hot fuel element. The corresponding two-pump CHF and FI margins were 4.36σ and 5.80σ, respectively. The three-pump minimum margins to CHF and FI were 4.07σ and 5.44σ, respectively. The margins to CHF and FI for two-pump and three-pump operation are plotted in Figures 15.3-5 and -6, respectively.

15.3.1.2.3 Conclusions—A low vessel inlet pressure PPS subsystem which initiates reactor trip at 324.8 psia (312.6 psig) will preserve the reactor protection criteria for CHF and FI for both two and three pump operation. The vessel differential pressure (ΔP) PPS subsystem is diverse to the inlet pressure subsystem (see Section 15.3.5.6).

15.3.2 Slow Closing Primary Pump Discharge Check Valve

15.3.2.1 Identification of Causes and Accident Description. The discharge from each PCP has a check valve which normally closes as the flow coasts down and directs the remaining coolant flow (usually emergency coolant flow) through the piping system to the reactor vessel (see Chapter 5 (Primary Coolant System) for a piping diagram). This event assumes failure of a check valve to close which allows emergency coolant flow to circulate through the primary pump back to the emergency coolant pump suction and partially bypass the reactor core.

The loss of flow from the PCPs combined with a slow closing check valve will result in a reactor trip initiated by the reactor inlet pressure PPS subsystem with diverse protection provided by the vessel differential pressure PPS subsystem.

15.3.2.2 Analysis of Effects and Consequences

15.3.2.2.1 Method of Analysis—The method of analysis with RELAP5 was the same as discussed in Section 15.3.1.2.1 with the following exception. The check valves at the discharge of the PCPs were modeled as motor valves with a closure rate that resulted in flow rate histories equivalent to the calculated rate in the base model. For this analysis, it was assumed that one of the check valves remained open.

15.3.2.2.2 Results—After the check valve that isolates the emergency coolant pump recirculation loop opens, about 30% of the emergency coolant pump output (with a failure of the one the PCP check valves) circulates through the PCP to the emergency coolant pump inlet (Terry 1994). (Tests performed with a blocked open PCP check valve (Magleby 1968) showed that less than 29% of the emergency pump flow would actually bypass the reactor.) The peak coolant temperatures calculated with RELAP5 remain below 200°F. ATR-SINDA and SINDA-SAMPLE analyses were not performed for this event since it progresses much the same as the loss of commercial power event. After the scram, the flow will decrease to 70% of that considered in Section 15.3.1.2.1 (Terry 1994); however, because of the scram, there will be little difference in the margins or the peak fuel temperature (minimum margins occur before flow reduces to the emergency coolant pump output). Additionally, experimental data indicate that an emergency coolant flow of 2000 gpm will maintain a margin of three standard deviations or more from FI immediately following a primary coolant flow coastdown (Atkinson 1994). Since the nominal emergency coolant pump flow is 4,700 gpm, the flow through the core is 3290 gpm with an open check valve which is substantially greater than the 2,000 gpm required from the experimental data.

15.3.2.2.3 Conclusions—The low inlet pressure PPS subsystem, having a setpoint and response time as defined in Section 15.3.1.2.3 in conjunction with design flow resistances in the reactor PCS, result in acceptable system performance for this fault.

15.3.3 Long-Term Complete Loss of Flow

15.3.3.1 Identification of Causes and Accident Description. Loss of commercial power in conjunction with a loss of diesel power will result in coastdown of the operating PCPs and failure of one emergency coolant pump; however, the dc-motor emergency coolant pump will operate on battery power for at least 30 minutes. (The dc-motor pump starts automatically if in standby mode and continues to operate if in service.) Failure of the ac-motor emergency coolant pump (or any of the PCPs) to start due to loss of power or mechanical failure will result in complete loss of flow after 30 minutes. Following the loss of forced flow, natural convection becomes established in the reactor vessel.

The probability that both diesel and commercial power will fail simultaneously is relatively high. However, the probability of restoring one or both sources within 30 minutes is also relatively high (Thatcher 1997). If the battery-backed power is available to the dc-motor emergency coolant pump, a complete loss of flow in less than 30 minutes is beyond design basis; the complete loss of flow after 30 minutes is estimated to be a Condition 4 event (Thatcher 1997).

The PCS design does not provide a path for natural convection flow around the primary coolant piping system. There is a natural convection coolant path within the vessel. (During the natural convection process in the vessel, cooler water in the flow distribution tank will rise heating as it passes up through the core. From the core, the water will mix with cooler water in the upper plenum and vessel. Cooler water in the vessel will pass downward through openings in the reflector and outside of the reflector tank and return to the flow distribution tank. There are ambient losses from the vessel which account for the cooler water in the vessel and upper plenum region.) However, heat losses through the vessel walls are not sufficient to dissipate the total decay heat, and the coolant in the reactor vessel will heat up. At this point, the transient is similar to the complete loss of secondary (Section 15.2.1) in that pressure will rise, the relief valves will cycle, and depressurization will be required if forced circulation is not restored before the vessel coolant temperature reaches approximately 200°F.

Initial protection for the reactor fuel is provided by the vessel inlet pressure PPS subsystem reactor trip. Diverse protection is provided by the vessel differential pressure subsystem.

15.3.3.2 Analysis of Effects and Consequences

15.3.3.2.1 Method of Analysis—The analysis methodology for this fault was the same as described in Section 15.3.1.2.1 except that the emergency coolant pump was secured at t = 30 minutes rather than continuing to operate. The secondary coolant pumps were secured concurrent with the PCPs, and secondary flow was linearly ramped to zero in ten seconds (i.e., no credit was taken for decay heat removal by the UCW system).

15.3.3.2.2 Results—Adequate thermal margins are maintained for the first 30 minutes following the loss of commercial power (Terry 1994). The results of the complete loss of flow after 30 minutes are presented and discussed starting 30 minutes after reactor scram; that is, time equal zero in the figures and the discussion is 30 minutes after scram. Fuel plate temperatures at midplane are shown in Figures 15.3--7 and -8 for two pump and three pump operation, respectively. Channel 19 coolant velocities are shown in Figures 15.3--9 and -10. Figures 15.3-7 through 15.3--10 were obtained from SINDA results. For two-pump operation, the flow reversed 26 s after the emergency coolant pump started to coast down; the flow reversed 24 s after emergency coolant pump started to coast down for three-pump operation (Terry 1994; Polkinghorn 1994). The margins to CHF and FI are shown in Figures 15.3-11 and -12. The margins for two-pump operation are 2.54σ (CHF) and 7.52σ (FI); the margins for three-pump operation are 2.43σ (CHF) and 5.98σ (FI). The minimum margin to fuel plate buckling is about 6σ, and the minimum margin to AWIT is greater than 37σ (Polkinghorne 1994).

15.3.3.2.3 Conclusions—The required margins to AWIT are maintained for the complete loss of flow assuming that the battery-powered emergency coolant pump operates until the batteries are depleted (at least 30 minutes). This assumption requires battery-backed power to be available to the dc-motor emergency coolant pump. The long-term success requires depressurization of the PCS and replacement of coolant inventory as it boils off.

15.3.4 Complete Closure of the Flow Control Butterfly Valve

15.3.4.1 Identification of Causes and Accident Description. The butterfly valve is designed to provide fine adjustment of the primary coolant flow. The valve is constructed with an external stop to keep the valve from closing completely. Since there is not an internal stop, certain shaft failures, those "downstream" of the external stop, would allow the valve to rotate freely. The external stop keeps the valve at least 20% open.

Three mechanical failures could potentially lead to rapid closing of the butterfly FCV:

- Gear box failure
- Key (shaft) failure
- Shaft failure.

Gear box failures and most shaft failures would allow the valve to close to the external stop. However, some failures of the shaft, and any key failure, would bypass the external stop and allow the disk to rotate without being limited by the stop.

The consequences of such butterfly valve failures are a pressure pulse upstream from the valve and a sudden reduction in PCS flow. Since the valve disk is smaller than the valve throat, a complete stoppage of flow cannot occur even with the valve fully closed. The consequences of the reduction in flow are discussed in Section 15.3.4.2; the consequences of the pressure increase are discussed in Section 15.13.2. If the failure bypassed the external stop, the valve disk would rotate through the fully-closed position and oscillate about the fully closed position before finally reaching the fully-closed position (Durney 1994). Both failure to the stop and failure with the stop ineffective are considered.

The reactor fuel is protected by the low inlet pressure and low differential pressure PPS subsystems as well as the design of the butterfly FCV disk and location of the stop.

15.3.4.2 Analysis of Effects and Consequences

15.3.4.2.1 Method of Analysis—The mechanical failures were considered by Brower (1993) and were shown to be low probability events (consistent with Condition 4). Qualitative treatment of torques and energy conservation (Durney 1994) indicate that the valve disk will rotate through the closed position, reverse direction of rotation, and oscillate to the closed position if the stop is ineffective.

This sequence was analyzed using RELAP5/MOD 2.5 with the base model which has a simplified five channel core (four quadrant channels and one "hot" fuel element channel) (Pafford 1994a, 1994b). The reactivity scram curve was an enveloping curve for two or three PCP operations. Feedback from the core or flux traps was not included.

The sequence was modeled by closure of the butterfly valve to the stop and by rotation through the stop and the closed position. The valve closing was modeled with a linear ramp for motion. The time to rotate to the stop was 0.016 s for the two-pump operation and 0.0149 s for the three-pump operation. When fully closed, the valve flow area was 1.66 square ft. The flow loss coefficient for the closed valve was obtained from benchmark calculations of test data for operation of one and two PCPs.

The reactor trip was on low vessel inlet pressure. For the analyses, the trip setting was 324.8 psia with a 0.1 sec response time for three-pump operation and 308.2 psia for two-pump operation with a 0.1 sec response time.

The reactor protection criteria were evaluated using ATR-SINDA and SINDA-SAMPLE with boundary conditions from the RELAP5 results (Polkinghorne 1994).

15.3.4.2.2 Results—The sequence of major events as calculated with RELAP5 are listed in Tables 15.3-4 and 15.3-5. The temperatures for the fuel plates from the RELAP5 calculation are shown for information although they are not used for the analysis of the fuel plate; the ATR-SINDA model is used to establish the plate temperatures and limits. A pressure pulse was generated as the valve closed, and it propagated upstream, dampening as it progressed. The peak pressure occurred directly upstream of the valve. The discussion of the pressure pulse is presented in Section 15.13.2.

The more significant flow perturbation occurs for closure to the stop. The minimum margins were developed for this result. (If the stop is not effective, the disk moves through the fully-closed position resulting in a relatively minor flow perturbation.)

The peak fuel plate temperatures (for the case where the valve closes to the stop) calculated with ATR-SINDA are shown in Figure 15.3-13 and -14; the results demonstrate that only a small temperature increase occurs. The SINDA-SAMPLE calculated minimum CHF margin was 5.26σ and the minimum FI margin was 7.80σ for two-pump operation and 4.27σ (CHF) and 5.91σ (FI) for three-pump operation.

15.3.4.2.3 Conclusions—The flow through the valve with the disk at 22.5° is sufficient to maintain the minimum margins to CHF for three-pump operation without a scram. A scram does occur for two-pump operation; however, the margins to CHF are ample, and a scram may not be necessary. The disk design and the PPS settings are sufficient to protect the fuel.

If the disk rotates through the stop, a scram occurs for both two and three-pump operation. The margins to CHF are greater than those for closure to the stop.

The event results in minimum margins that are greater than 3σ to CHF. Since CHF does not occur (no melt), the event meets Condition 4 criteria.

15.3.5 Bounded Events - Decrease in Reactor Primary Coolant Flow Rate

15.3.5.1 Trip of One (Two-Pump Operation) or Two (Three-Pump Operation) of the Operating Primary Coolant Pumps or Failure of One Operating Primary Coolant Pump due to Mechanical Failure. This fault results in one PCP in operation. This can result as a consequence of individual power supply failure (line, breaker, etc.), mechanical failure (event titled "Loss of one operating PCP due to any mechanical failure), or actuation of the PPS subsystems for high primary coolant flow.

This is only a partial failure of the PCS flow. The consequences are bounded by consequences for the flow reduction described in Section 15.3.1.

15.3.5.2 Sleeve Failure in a 16-In. Outlet Pipe. The outlet pipes have a sliding joint design inside the reactor vessel. A study for this configuration determined that the maximum joint opening due to wear and sleeve failures would be equivalent to a 2-in. hole in the outlet pipe (Atkinson 1995). This hole provides a flow path that would bypass the core.

Calculations for the 2-in. holes show that there is an insignificant decrease in flow through the core (Merrill and Haroldsen 1976). The consequences of this event are bounded by the consequences for the flow reduction described in Section 15.3.1.

15.3.5.3 Flow Control Butterfly Valve Closure from Operating Position. This slight closure of the flow control butterfly valve (approximately 45% to 38% open) could be initiated by operator error. The valve is manually adjusted during reactor operation to provide the required vessel differential pressure. The minor misadjustment of the valve would reduce flow on the order of 10% (EG&G Idaho 1994). However, since the flow reduction would be much less than the flow reduction to cause a scram, the margins to CHF are well above 3.0σ, and the event is bounded by the event discussed in Section 15.3.1.

15.3.5.4 Closure of a 24-in. Reactor Inlet Isolation Valve. The reactor primary coolant piping system has two 24-in. inlet lines to the reactor vessel. Each line has an isolation valve. Sudden closure of one of these valves was assumed, and analyses were completed to assess the resulting perturbation to the primary coolant flow. The results show that the effect on reactor flow was minimal (Varacalle 1975, 1976). Polkinghorne (1994) shows that the margins to CHF are not less than three standard deviations for a three-pump differential pressure of 42.0 psid, which is a reduction in flow to less than 60%. Since the flow is only minimally affected by the valve closure, the margin to CHF can be assumed to be significantly greater than three for this event, and it is acceptable as a low probability event.

15.3.5.5 Rupture of a 16-in. Outlet Pipe Inside the ATR Vessel. The primary coolant flow exits from each core quadrant through a 16-in. pipe, which is in the inlet flow annulus. A rupture in this pipe will allow coolant flow to bypass the reactor core. Analysis has shown that a hole 12 in. in diameter will result in a flow decrease to 67.5% of normal in the quadrant affected (Merrill and Haroldsen 1976). Analysis (Atkinson 1995) has shown that the largest, extremely unlikely break in the in-vessel piping is smaller than 7-in equivalent diameter. Therefore, the maximum credible break would show a much smaller flow decrease than 67%. From Polkinghorne (1994), the margins to CHF for two-pump operation are at least three standard deviations for a differential pressure of 23.7 psid (for three-PCP the differential pressure is 42.0 psid). Since the flow would have to be reduced to 60% or less to approach three standard deviations to CHF, the event is acceptable as a low probability occurrence.

15.3.5.6 Failure of First Actuating PPS Subsystem During Condition 2 Event. For the low flow events analyzed, the inlet pressure PPS subsystem, on low pressure trip, initiated release of the safety rods. The vessel differential pressure PPS subsystem is diverse to the inlet pressure subsystem and will trip at approximately the same time, but it has a delay time of 0.4 sec as compared to 0.1 sec for the inlet pressure subsystem. The limiting flow event, the loss of commercial power, was analyzed concurrent with the failure of the inlet pressure PPS subsystem. For this event, the margins to CHF and FI for three-pump operation are 2.71σ and 3.24σ, respectively. The margin to fuel plate buckling is 5.20σ; the minimum margin to AWIT is 37.7σ (Polkinghorn 1994). These margins are acceptable for Condition 4 events. The analysis for loss of flow for two-pump operation indicates that the low differential pressure scram is generated 0.24 s earlier than for three-pump operation (Polkinghorne 1994) which occurred at 1.82 s. An analysis for the flow coastdown (two-pump operation) without a scram from either low pressure or low differential pressure shows that 3σ to CHF is not reached until 2.10 s after initiation of the coastdown (Polkinghorne 1994). Since the differential pressure scram occurs at 1.58 s, the margin to CHF is clearly greater than 3σ. Therefore, the above results envelop the two-pump case, and it is not necessary to complete the analysis for margins to buckling and AWIT for two-pump operation.

Protection of the fuel for low-flow events is provided by a low-pressure setting of 324.8 psia with a response time of 0.1 s for both two and three-pump operation. (The analyses for loss of pumps uses 324.8 psia for both two-pump and three-pump operation; the analyses of the butterfly-valve closure uses 308.2 psia for two-pump operation but 324.8 psia for three-pump operation. The 324.8 psia setting is limiting and the required value for both modes of operation.) The differential pressure (low) setting is diverse for three pumps operation at 45.8 psid and for two-pump operation at 38.0 psid.

The modelling difference noted in Section 15.3.1.1 affects the inlet pressures reported by Polkinghorne (1994) since they were edited at the location of the pressure controller taps rather than at the PPS tap location. In order to provide the safety limits through the PPS instrumentation, the values at the locations of the PPS taps (piping values rather than transmitter values) are provided by Pafford (1994c). The revised differential pressure for 3σ to CHF is 38.6 psid for three-pump operation and 21.3 psid for two-pump operation (Pafford 1994c); these values are selected as safety limits in Chapter 16 (Derivation of the Technical Safety Requirements).

Table 15.3-1. Reactivity feedback from flux traps without inpile tubes during loss of commercial power.

Time (sec)	Reactivity (dollars)	Time (sec)	Reactivity (dollars)
0.00	0.0	5.00	3.112E-2
0.02	6.687E-6	6.00	3.799E-2
0.04	1.346E-5	8.00	5.197E-2
0.10	3.741E-5	10.00	6.658E-2
0.30	3.462E-4	12.00	8.467E-2
0.50	9.816E-4	13.00	9.501E-2
0.76	2.218E-3	14.00	1.061E-1
1.00	3.643E-3	15.00	1.198E-1
1.24	5.206E-3	16.00	1.330E-1
1.50	6.845E-3	17.00	1.417E-1
1.74	8.245E-3	18.00	1.478E-1
2.00	9.840E-3	19.00	1.537E-1
2.50	1.335E-2	20.00	1.555E-1
3.00	1.698E-2	21.00	1.555E-1
3.50	2.042E-2	21.45	1.555E-1
4.00	2.376E-2	1000.00	1.555E-1

Table 15.3-2. Reactivity feedback from inpile tubes during loss of commercial power.

Time (sec)	Reactivity (dollars)
0.0	0.0
0.8	0.01
1.0	0.0128
1.5	0.0175
2.0	0.0207
2.6	0.0233
4.0	0.0267
7.0	0.0317
10.0	0.0357
20.0	0.046
40.0	0.0593
100.0	0.0853

Table 15.3-3. Sequence of events for the loss of commercial power transient calculations (RELAP5).

Events	Time (s) 2 PCP	3 PCP
Transient initiated	0.0	0.0
Reactor scram on low vessel inlet pressure (safety rod insertion begins)	1.145	1.000
Peak hot-spot coolant temperature reached	1.20	1.05
(Peak temperature)	(232.3°F)	(231.1°F)
Peak hot element fuel temperature reached	1.15	1.10
(Peak fuel temperature)	(405.1°F)	(425.8°F)
Transient calculation terminated	98.0	30.0

Table 15.3-4. Sequence of events for the butterfly valve closure transient calculation for three-pump operation (RELAP5).

	Times (s)	
Event	Case 1[a]	Case 2[b]
Transient initiated	0.0	0.0
Butterfly valve completely closed	0.015	0.02
Peak pressure occurs upstream of butterfly valve	0.072	0.024
(Peak pressure)	(506.4 psia)	(510.9 psia)
Reactor scrams on low vessel inlet pressure (safety rod insertion begins)	—[c]	0.154
Peak fuel temperature occurs	1.69	0.158
(Peak temperature)	(457.3°F)	(434.6°F)
Transient calculation terminated	2.0	2.0

a. Valve closes to the stop (22.5 σ)

b. Valve rotates through closed position and continues to the closed position

c. Event did not occur

Table 15.3-5. Sequence of events for the butterfly valve closure transient calculation for two-pump operation (RELAP5).

Event		Times (s)
Transient initiated		0.0
Butterfly valve reaches stop position		0.016
Peak pressure occurs upstream of butterfly valve		0.073
(Peak pressure)		(482.8 psia)
scrams on low vessel differential pressure (safety rod insertion begins)		0.580
Peak fuel temperature occurs		0.585
(Peak temperature)		(421.2°F)
Transient calculation terminated		2.0

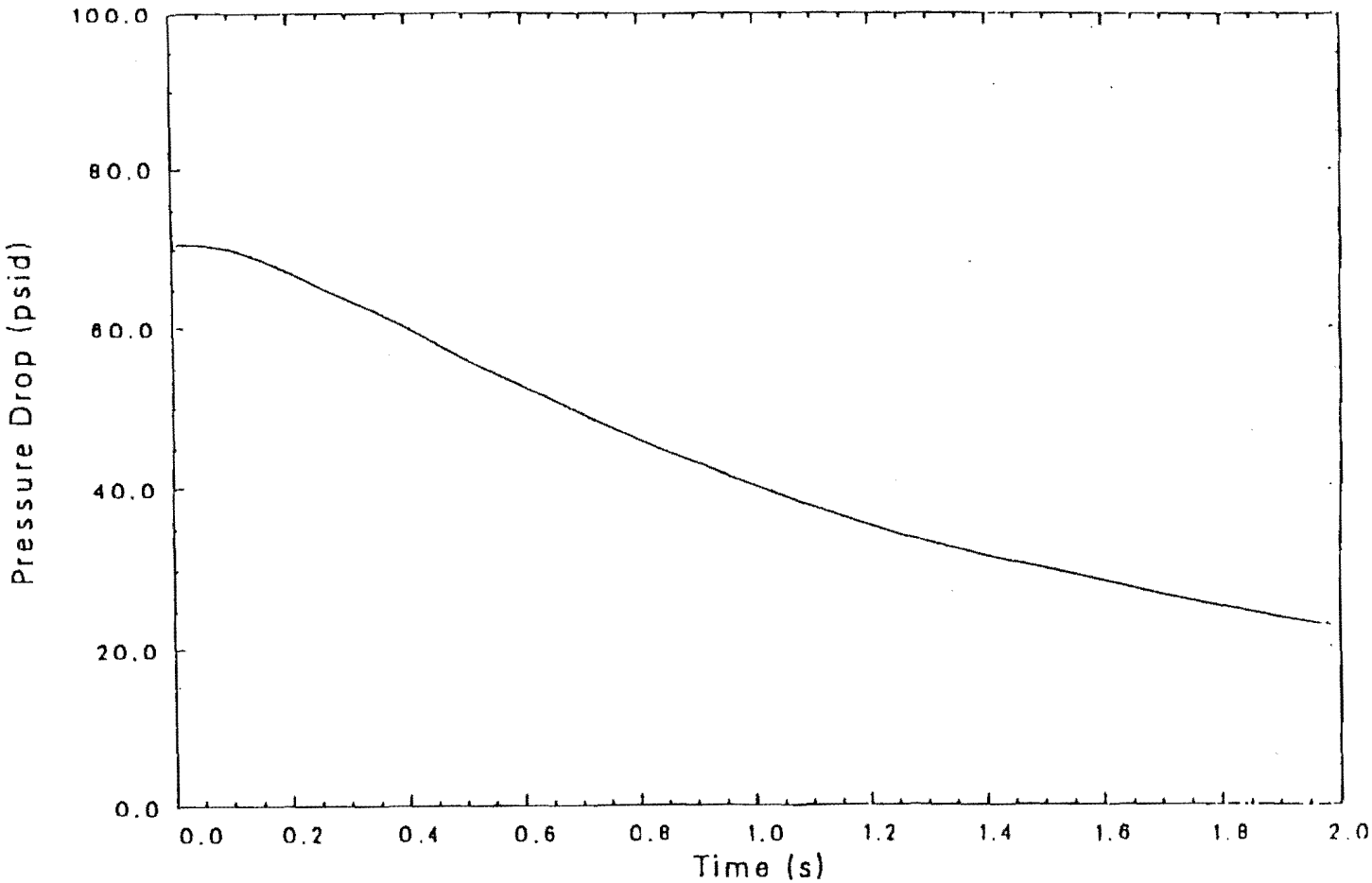

Figure 15.3-1. Core pressure drop (loss-of-commercial power, two-pump operation). **(For Information Only)**

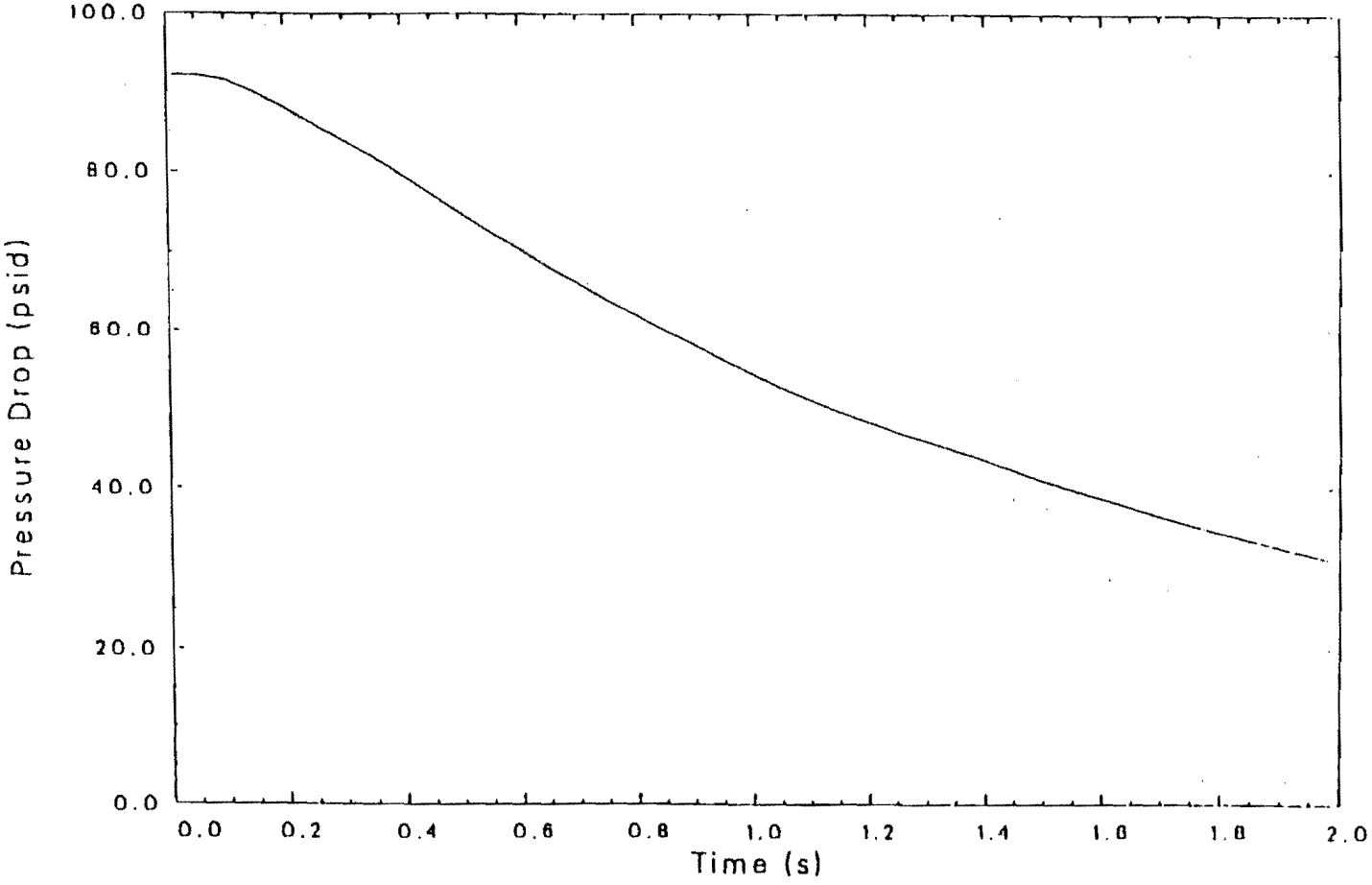

Figure 15.3-2. Core pressure drop (loss-of-commercial power, three-pump operation). **(For Information Only)**

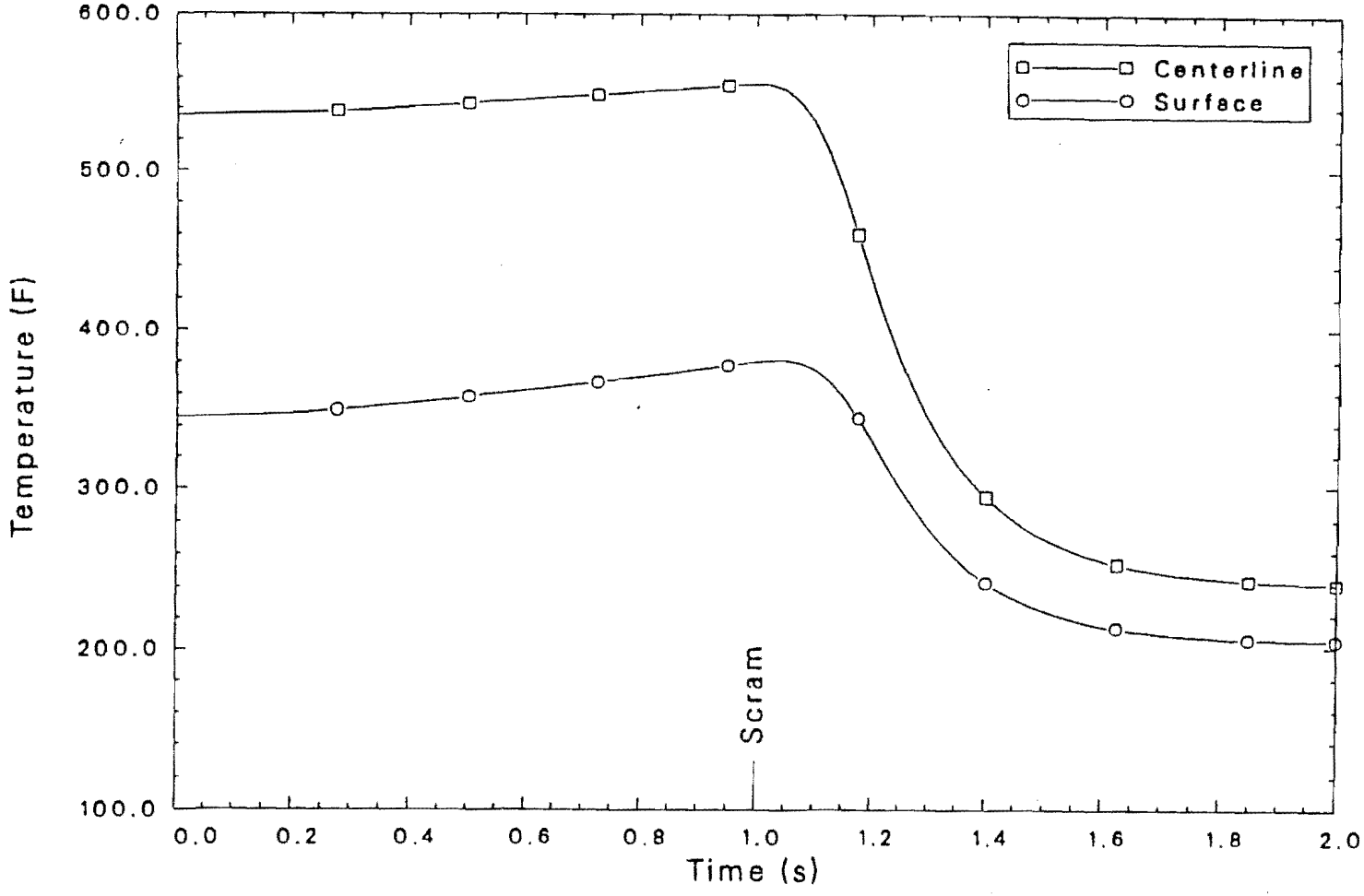

Figure 15.3-3. Maximum fuel plate temperatures (loss-of-commercial power accident, two-pump operation). **(For Information Only)**

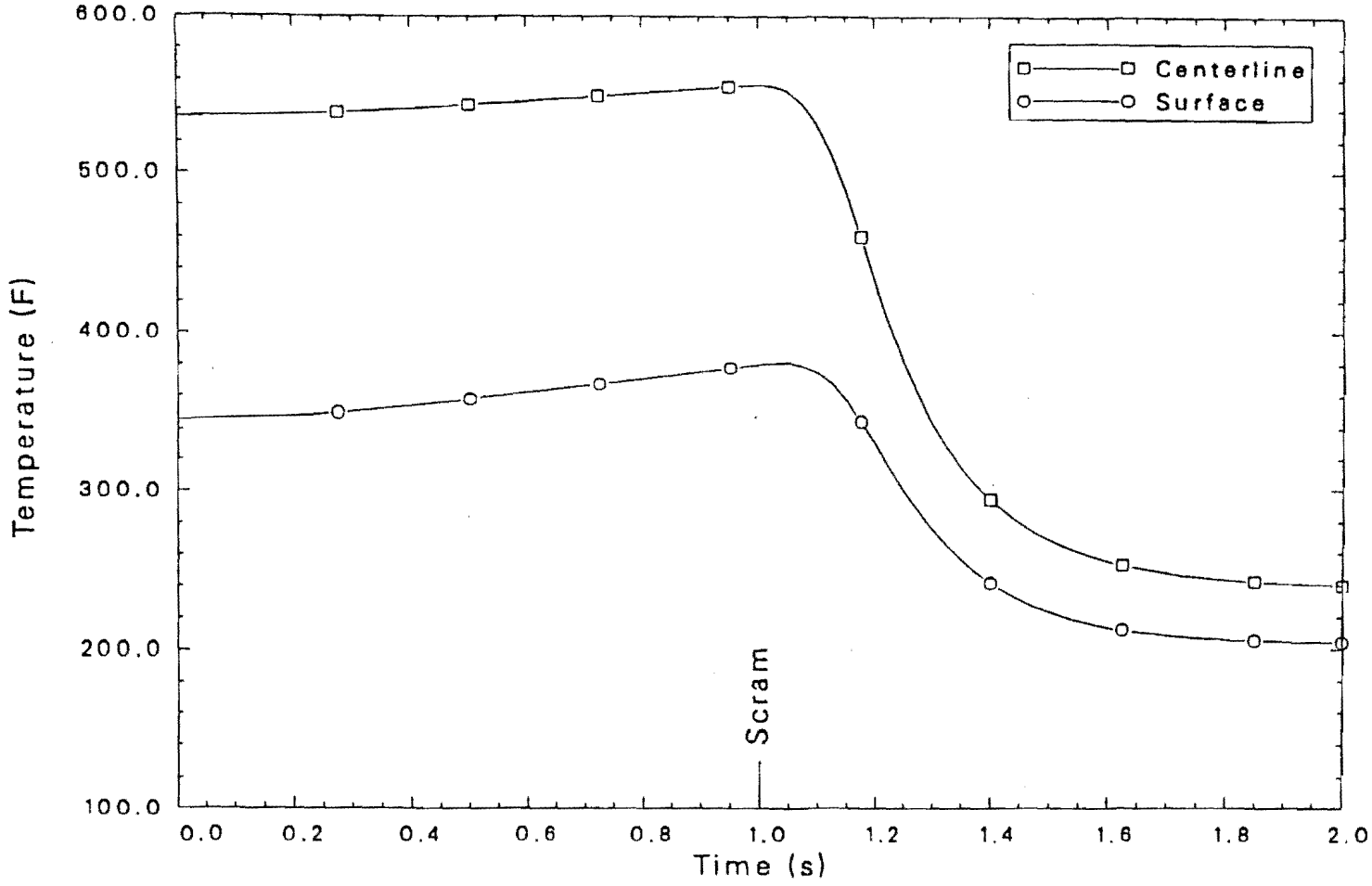

Figure 15.3-4. Maximum fuel plate temperatures (loss-of-commercial power accident, three-pump operation). **(For Information Only)**

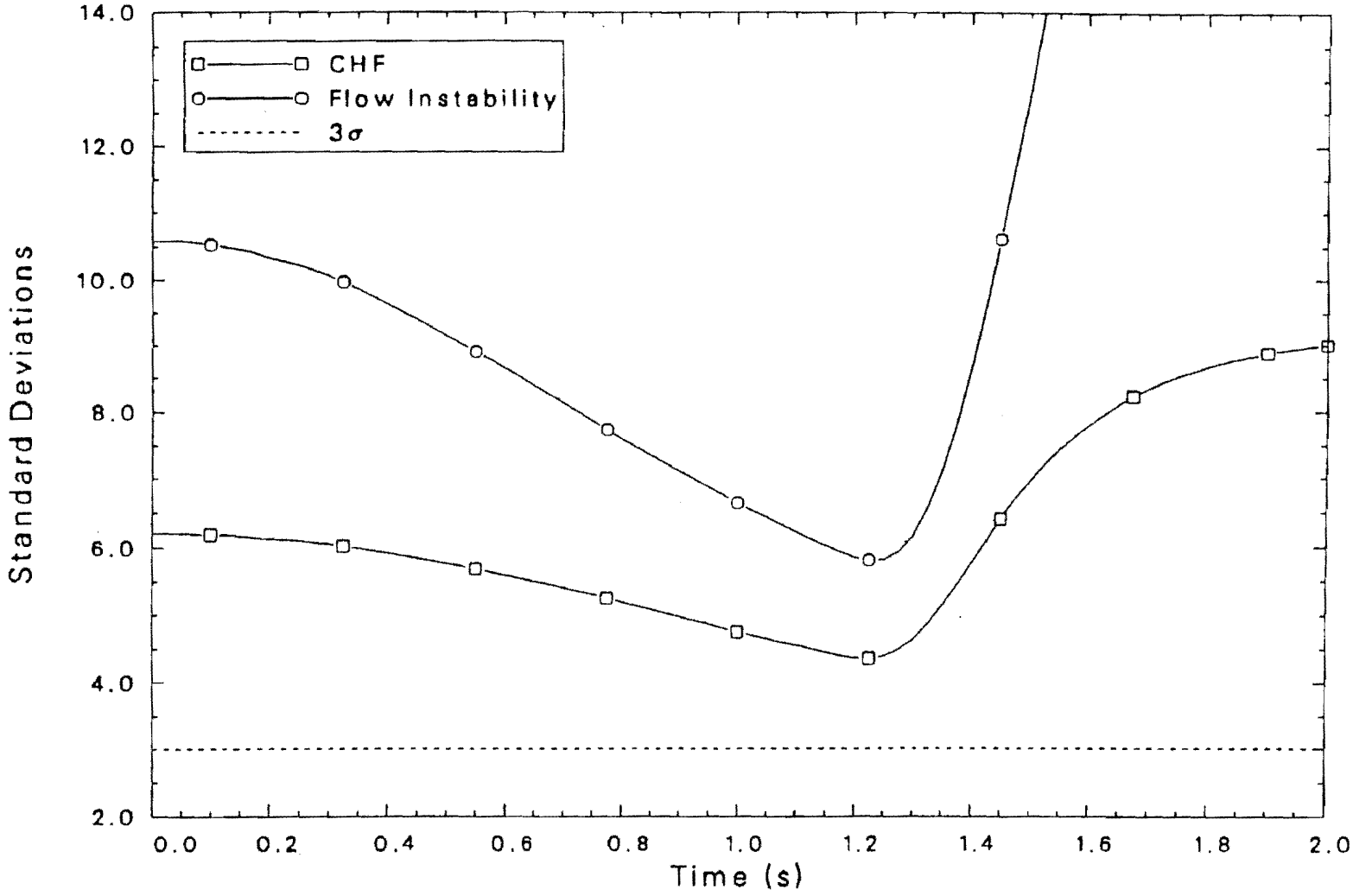

Figure 15.3-5. CHF and FI margins (loss-of-commercial power accident, two-pump operation). **(For Information Only)**

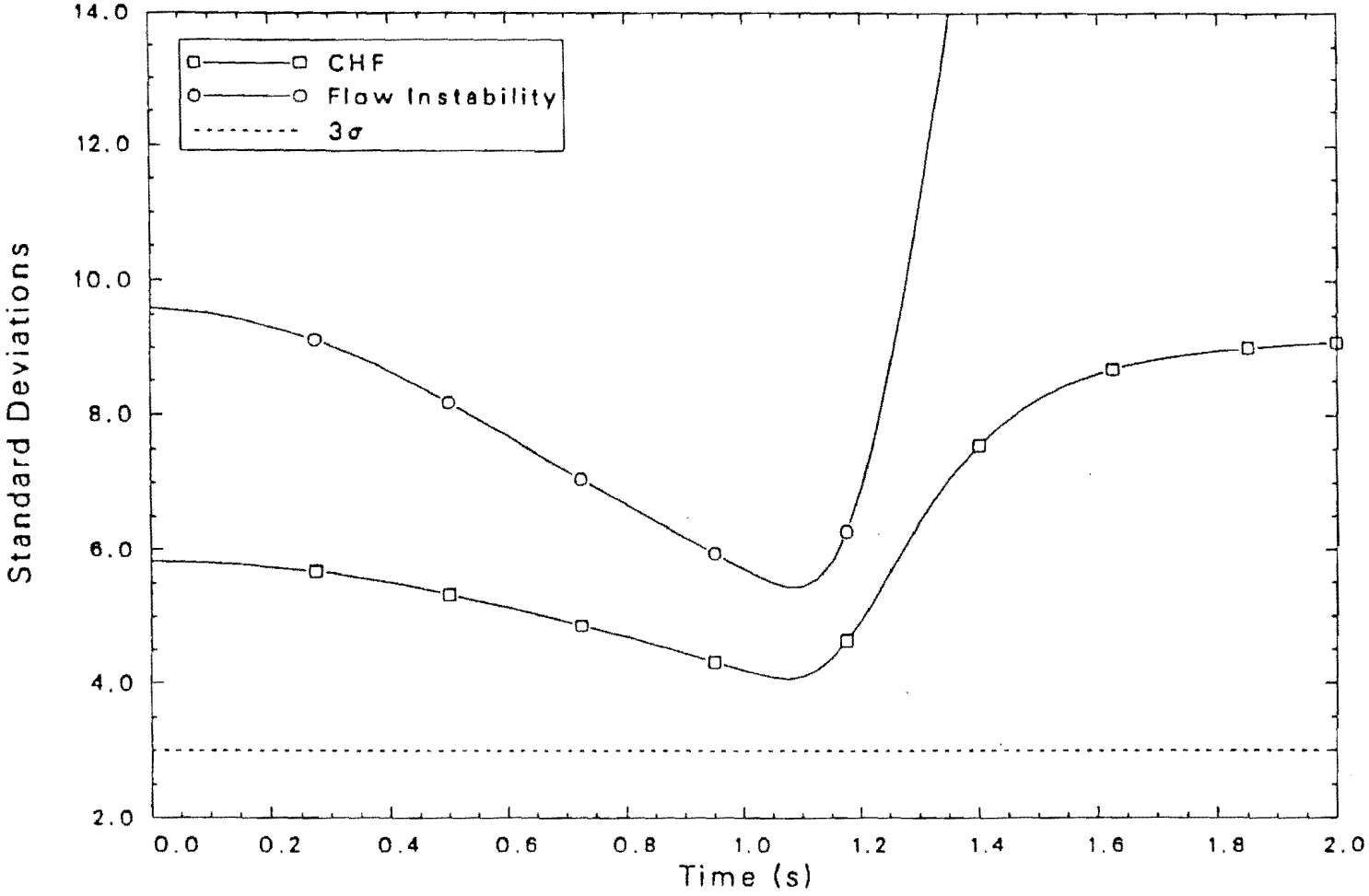

Figure 15.3-6. CHF and FI margins (loss-of-commercial power accident, three-pump operation). **(For Information Only)**

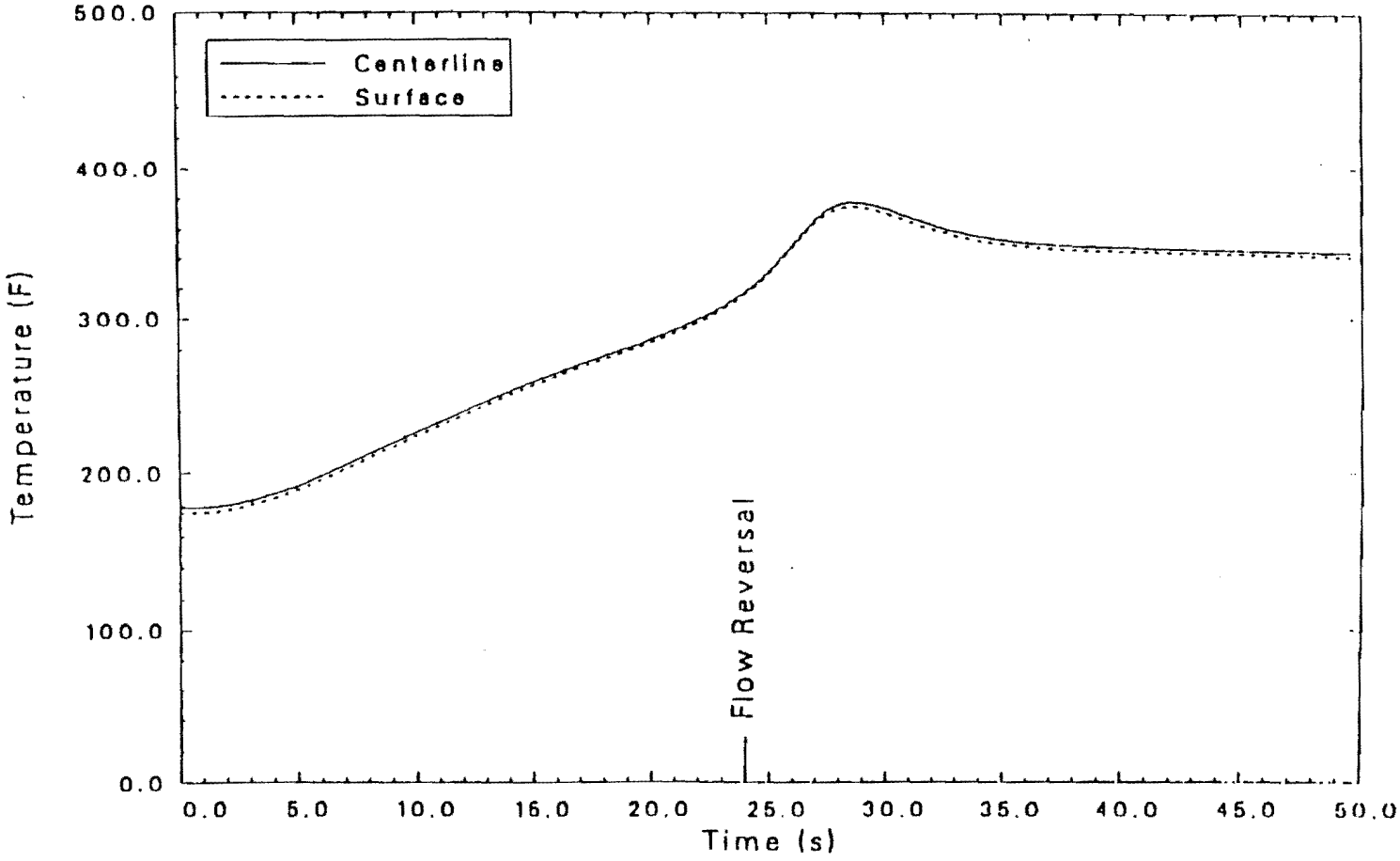

Figure 15.3-7. Mid-core fuel plate temperatures (complete loss of flow accident (CLOFA), two-pump operation). **(For Information Only)**

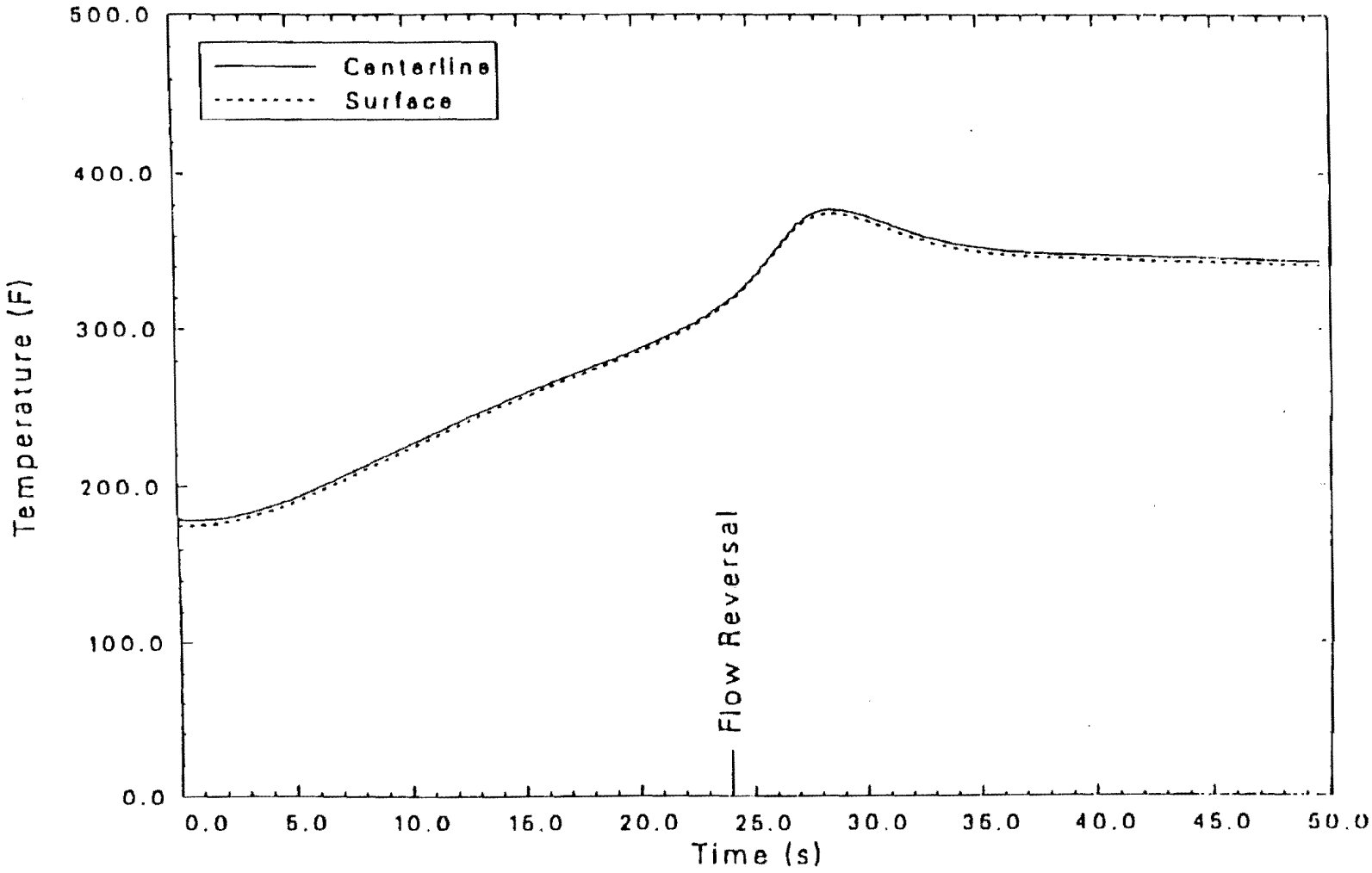

Figure 15.3-8. Mid-core fuel plate temperatures (CLOFA, three-pump operation). **(For Information Only)**

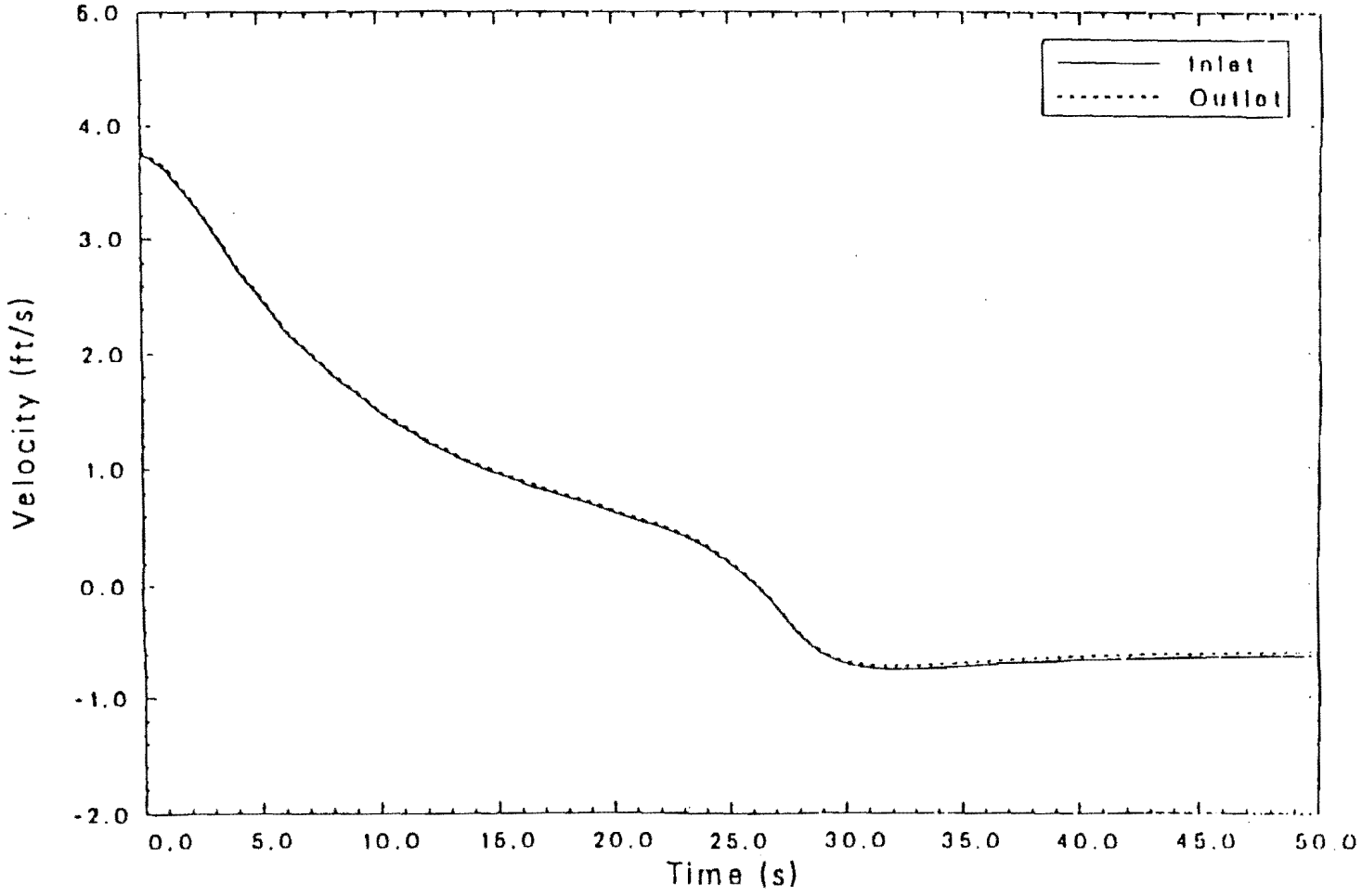

Figure 15.3-9. Hot-stripe inlet and outlet coolant velocities (CLOFA, two-pump operation). **(For Information Only)**

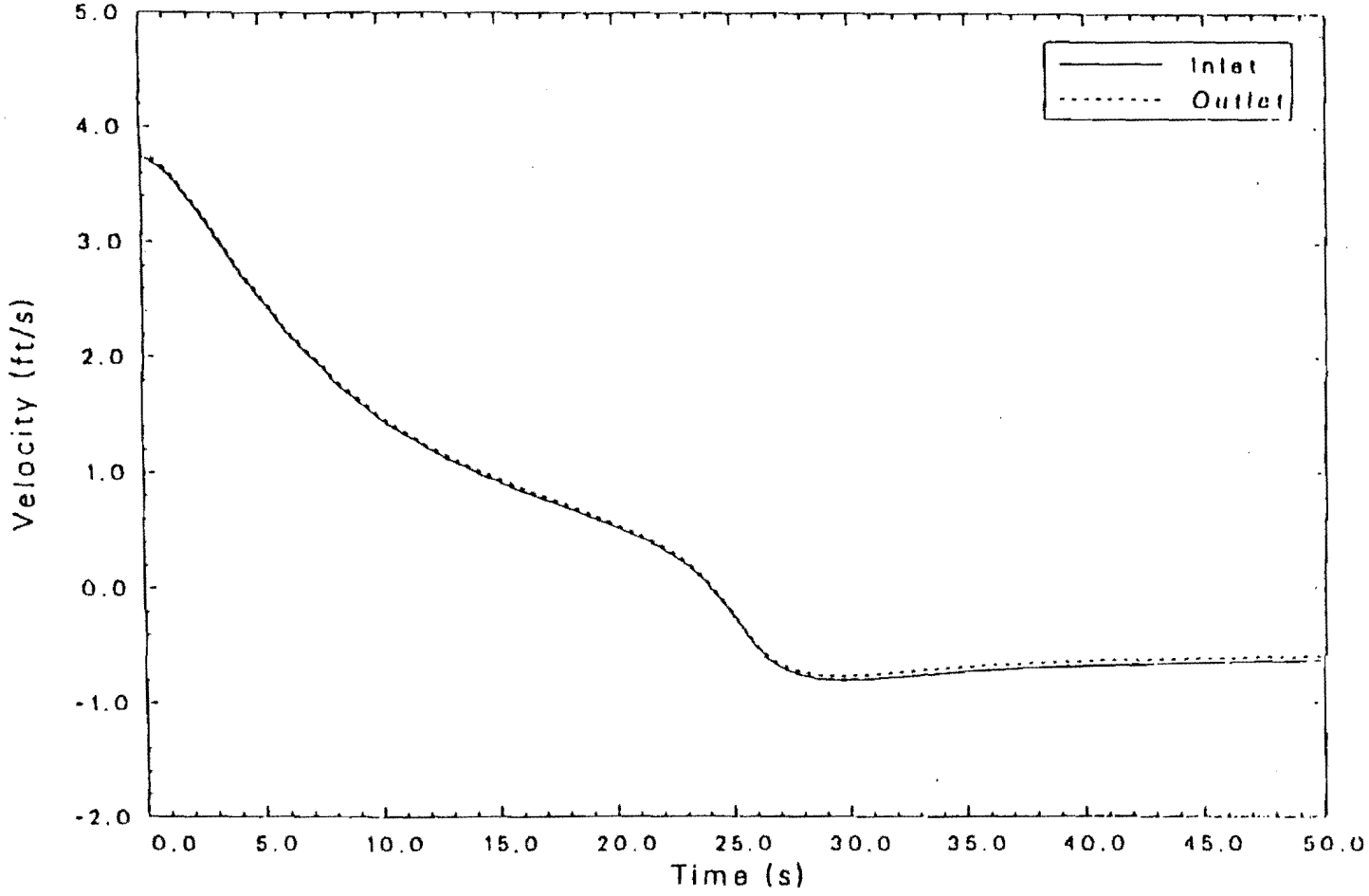

Figure 15.3-10. Hot-stripe inlet and outlet coolant velocities (CLOFA, three-pump operation). **(For Information Only)**

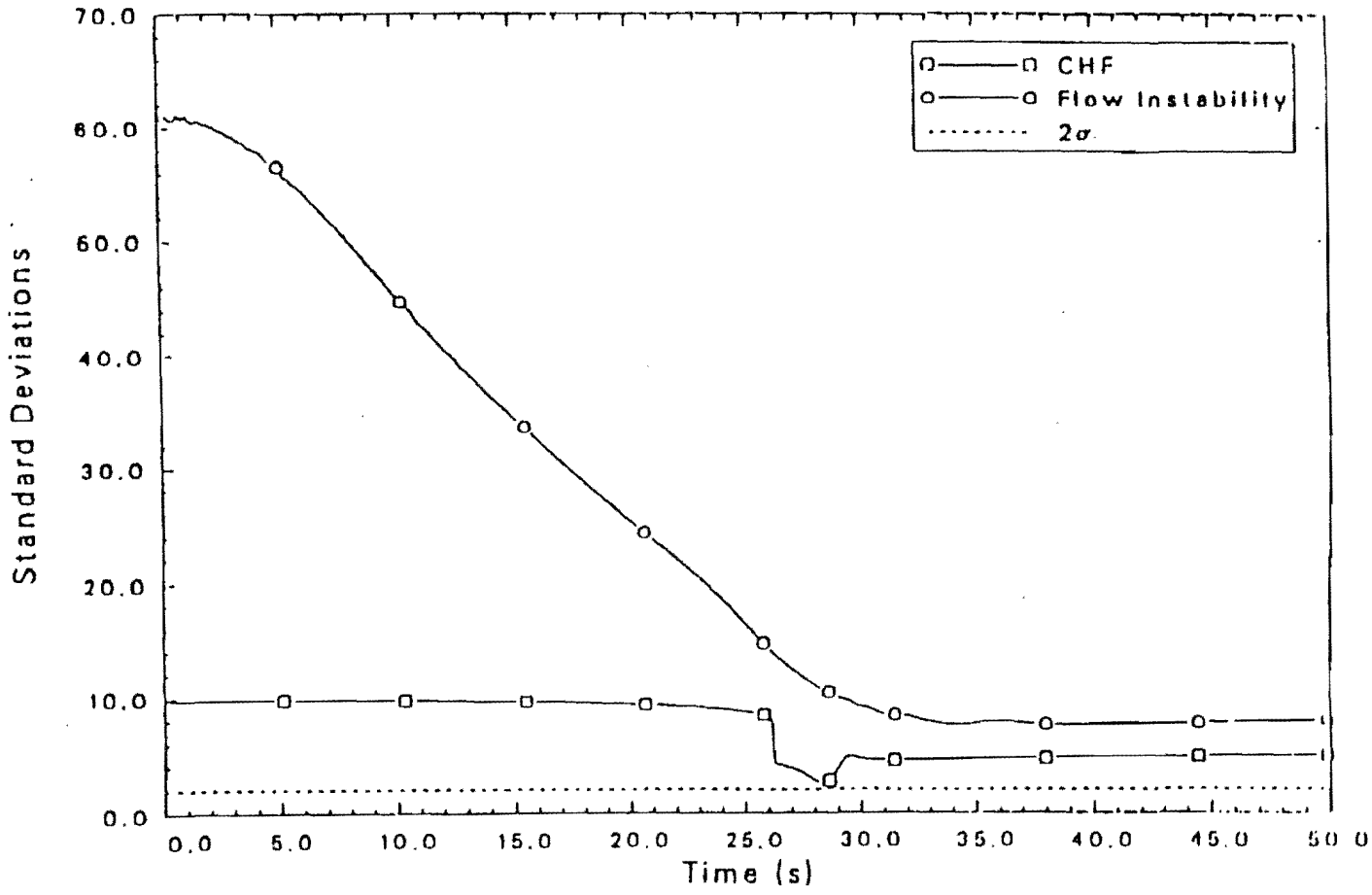

Figure 15.3-11. CHF and FI margins (CLOFA, two-pump operation). **(For Information Only)**

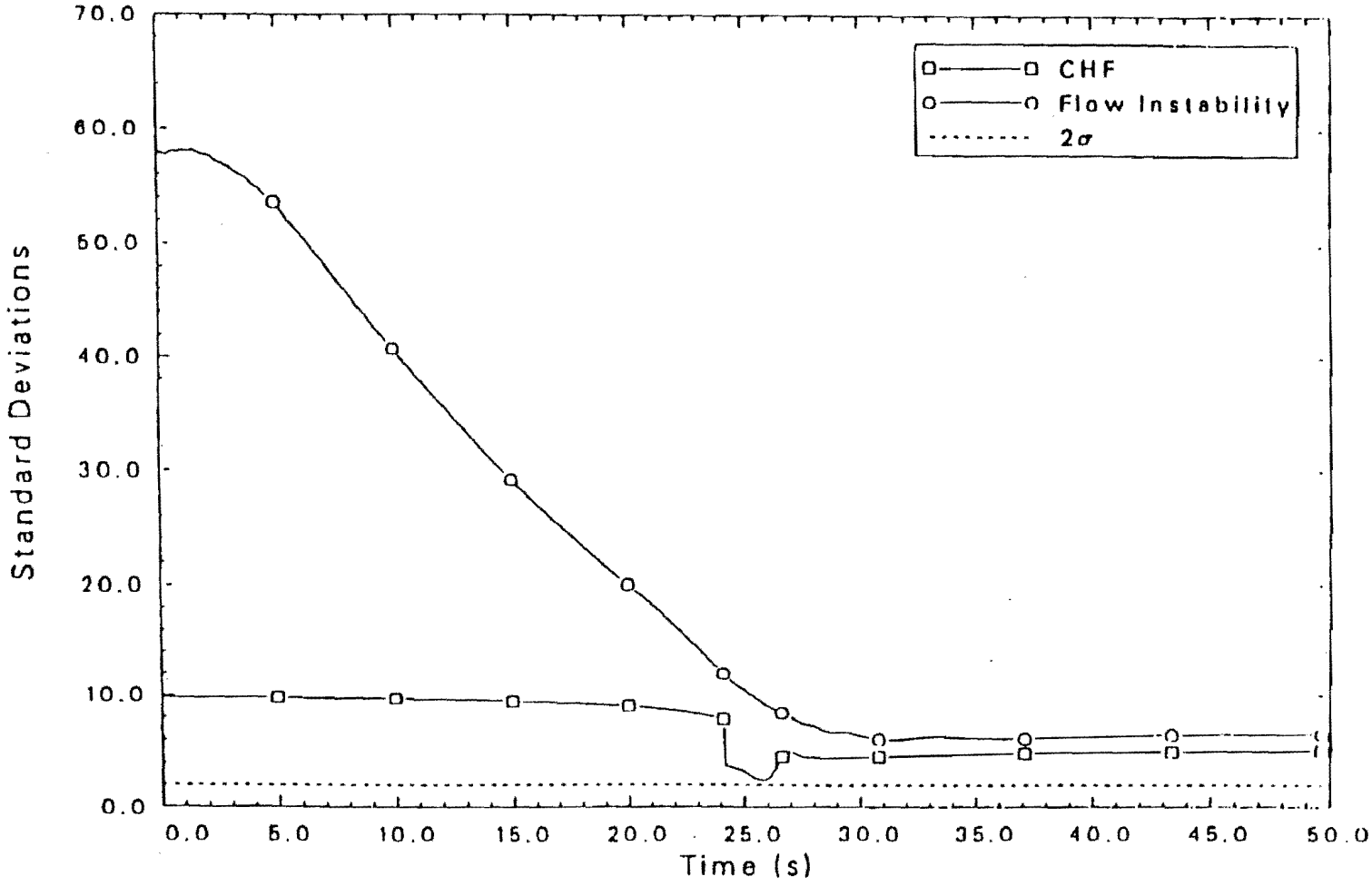

Figure 15.3-12. CHF and FI margins (CLOFA, three-pump operation). **(For Information Only)**

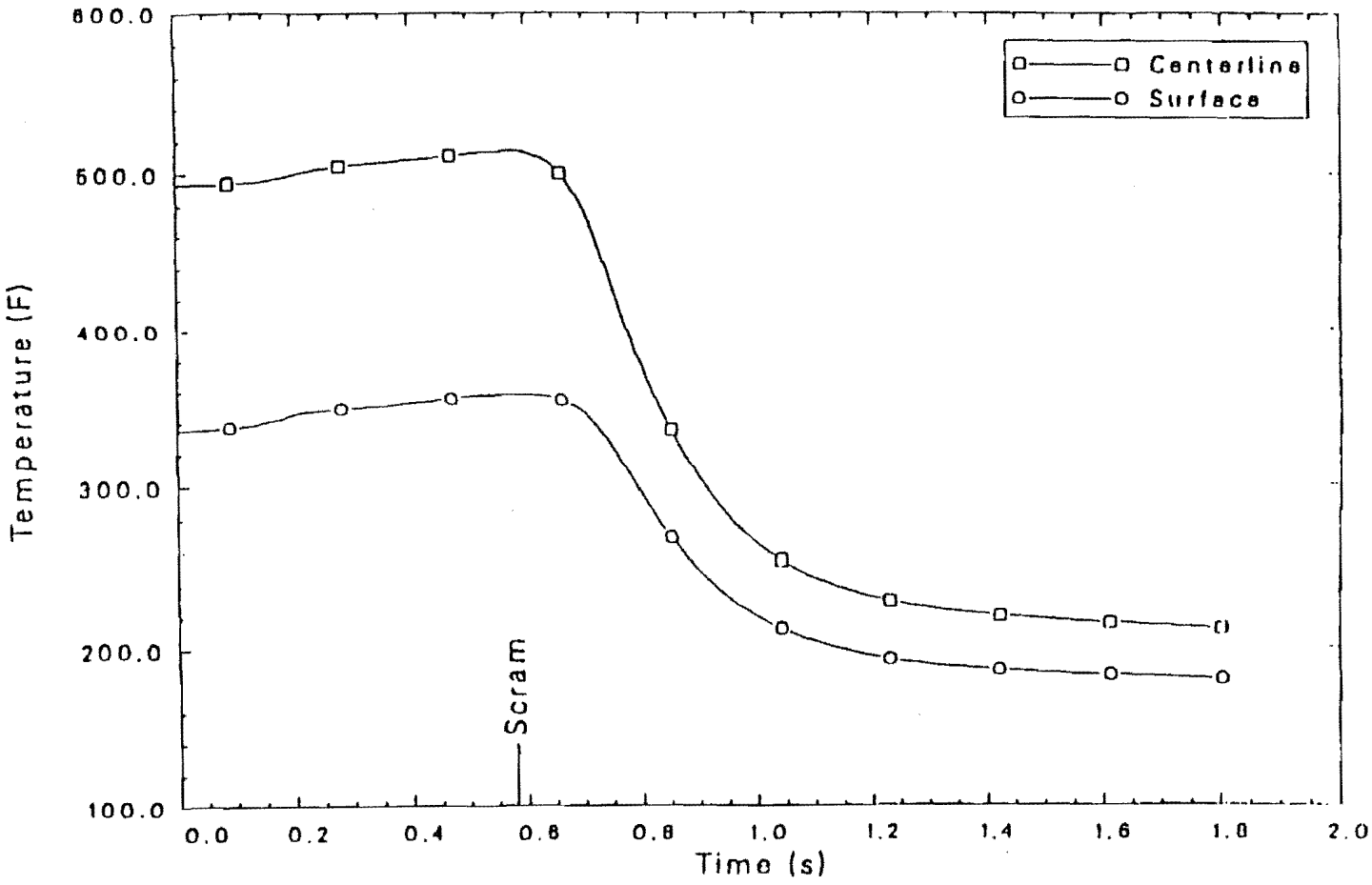

Figure 15.3-13. Maximum fuel plate temperatures (butterfly valve closure (to the stop) accident, two-pump operation). **(For Information Only)**

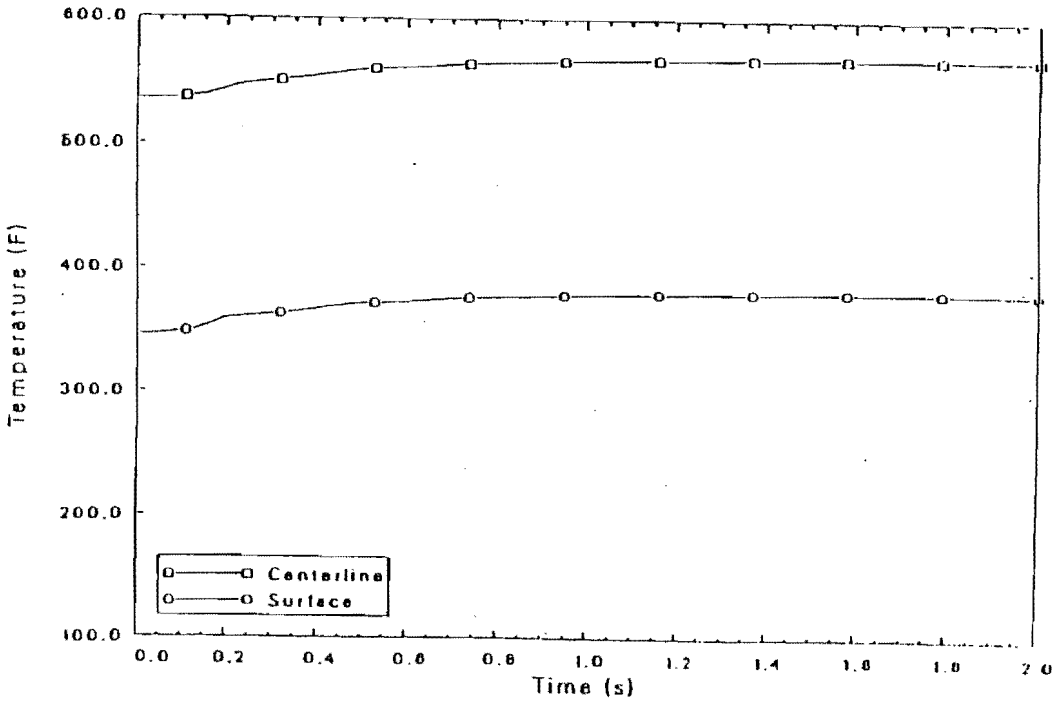

Figure 15.3-14. Maximum fuel plate temperatures (butterfly valve closure (to the stop) accident, three-pump operation). **(For Information Only)**

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