15.4 Reactivity and Power Distribution Anomalies

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Reactivity insertion can also occur as a result of failure or misoperation of the reactor reactivity control system. These events include servo failure, outer shim control cylinder or neck shim drive failures, operator error, etc.

The loop scrams are a part of the loop control system intended to limit conditions with the purpose of assuring the investment in data and equipment is protected (Durney 1994b). The effect of the control system was considered in establishing the frequency of occurrence for events leading to a decrease in coolant density in the IPT (a positive reactivity insertion occurs for this change). As is noted below, the events are considered both with and without the loop-initiated scrams. Neither sequence results in a power transient that is damaging to the fuel elements. These loop scrams are not required to prevent damage to the fuel elements.

A number of faults which result in a positive reactivity insertion are postulated. A discussion of these events is included in this section. The identified faults are as follows:

A. IPT decompression with subsequent voiding due to failure of the 1/2-in. piping

B. Rapid regulating rod withdrawal

C. Withdrawal of all outer shims from 10^{-10} N_F

D. Withdrawal of all outer shims and neck shims from 10^{-5} N_F

E. Withdrawal of all outer shims and one neck shim from N_L (one percent of N_F)

F. Cold water injection

G. IPT voiding due to rupture disc or relief valve failure or to opening of an accessible valve

H. Slow power increase or lobe power balance shift due to shim or operator failure

I. Loop flow coastdown or loop loss of temperature control with a loop instrumentation initiated reactor trip

J. PALM drive system failures

   1. Simultaneous movement of two cycling tests

   2. Uncontrolled movement of a positioner test

   3. Simultaneous movement of Three PALM Tests

K. Withdrawal of all outer shims and neck shims from 10^{-10} N_F

L. IPT voiding due to opening of a normally inaccessible valve

M. Pressure tube flow bypass due to a MUCH test train failure with loop scram

N. Loss of loop temperature control due to heat exchanger failure or line heaters sticking on without loop scram

O. Slow lobe power balance shift due to shim, lobe power indicating system or operator failure with operator compensation

P. 0.50$ step insertion which envelopes a number of events such as:

1. Limited perched fuel element (Condition 2)
2. Loop experiment hardware failure (Condition 3)
3. Loss of reflector coolant (Condition 3)
4. Reflector movement toward core (Condition 3)
5. Perched fuel drop of two fuel elements from within measurement accuracy (Condition 3)
6. Cold water addition from opening of primary pump discharge valve (Condition 3)
7. Gas release into the core (Condition 3)
8. Perched fuel element (greater than the measurement uncertainty) (Condition 4)
9. Movement of two reflector blocks (Condition 4).

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T. Withdrawal of all outer shims and neck shims from 10^{-5} N_F with failure of the WRS

U. Rapid regulating rod withdrawal with failure of the WRS

V. Withdrawal of all outer shims and neck shims from 10^{-5} N_F coupled with rapid regulating-rod-withdrawal at 10^{-3} N_F

W. Driven test loop blowdown with experiment hardware failure

X. Voiding in all IPTs due to a simultaneous flow coastdown in all loops

Y. Loss of loop temperature control in the AHTL due to heat exchanger failure or line heaters stuck on without a loop scram

Z. Loop flow coastdown without a loop scram

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Based on analysis in the DBR (SAR-39), events A-J are considered to be Condition 2 occurrences, events K-O Condition 3, and events Q-BB Condition 4. Faults in this section were previously evaluated in the DBR. Based on the existing analysis, event A (IPT decompression and voiding due to failure of 1/2-in. piping) which is the limiting Condition 2 transient event for power operation was selected for reanalysis. Event D, uncontrolled withdrawal of all outer shims and neck shims from 10^{-5} N_F, which is a limiting Condition 2 startup event for the reactor was selected for detailed discussion. The limiting step event (event P) selected for reanalysis is a 0.50$ step which envelops several events which are near steps that are difficult to characterize with time responses. The results for this event would meet the damage criteria established for Condition 2 faults. The analysis is used to envelop the Condition 2, 3, and 4 faults shown in event P above. One Condition 4 event was identified for reanalysis: event R, greater than 1/2-in. diameter pipe rupture in pressurized water loop out-of-pile piping, which represents the bounding reactivity insertion event for experiments.

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15.4.1 Flux Traps

The ATR has a cloverleaf arrangement of aluminum clad, plate type fuel elements which result in flux traps for experiment irradiations. The ATR also has a combination of rotating shim drums outside the fueled area with small diameter shim rods inside the fueled area which allow operation with different power (flux) levels in different segments of the core. Figure 15.4-1 shows the configuration with pressurized water loop facilities located in five of the flux traps and capsule irradiation facilities in the remaining four flux traps. This is the configuration of the reactor following the 1994 Core Internals Changeout and is the reference configuration for the UFSAR. Safety rods are contained in selected flux traps. The safety rods are the only reactivity control modeled for reactor shutdown. The core may be described as consisting of 5 lobes (four corner lobes of the cloverleaf and the center area). The coolant through the core is divided into quadrants with the corner lobes in separate quadrants and the center lobe divided equally into each of the quadrants.

The ATR has a strong negative temperature reactivity coefficient in the fuel elements. The flux traps have a positive temperature reactivity coefficient. Evaluations for events which consider reactivity changes from the reactor core as a mitigating effect consider the influence of the flux traps as well as the fuel elements.

15.4.1.1 Modeling Assumptions. The flux traps provide irradiation space for experiments which are cooled by the reactor primary coolant or for experiments which have their own coolant in self-contained loops. The materials and their configuration in the flux trap influence the power distribution in the fuel elements adjacent to the flux trap. The contents of the flux trap also influence the reactivity worth associated with density changes of the coolant.

The analysis of reactivity and power distribution anomaly events was completed by the use of RELAP5, ATR-SINDA, and SINDA-SAMPLE. The power transients resulting from the reactivity insertions must be generated from the RELAP5 reactor model and input to ATR-SINDA. The power transient is sensitive to the strong feedback from the core region. This feedback is sensitive to the power distribution within the core (SAR-39; Terry 1994). The previous analysis (SAR-39) demonstrated that the most conservative approach for obtaining the power transient for the ATR-SINDA calculation is to assume the event occurs with a balanced power division; this approach was verified by Terry (1994).

The feedback from the fuel is developed from physics calculations using the PDQ computer program. Further conservatisms are obtained by assuming outer shims are fully withdrawn to minimize the feedback. Since the kinetics model in RELAP5 assumes the core is a point, an adjustment for the effects of power shifts during the transient is included by considering the local effect of voiding in the flux trap.

The balanced power division necessarily consists of lobe powers that do not exceed 50 MW each. However, the analysis is completed for higher lobe powers as needed by initializing the ATR-SINDA model to correspond to the desired power level. The power transient information from RELAP5 is normalized to the initial condition chosen for ATR-SINDA. The approach allows a very conservative power transient to be developed for the reactivity insertion sequences.

The power transient will cause some heating within the remainder of the core which will result in density changes. These changes within the flux traps are relatively small but positive reactivity effects. In order to consider the effect of this cascading, an additional reactivity input is included in RELAP5. This effect is only significant for relatively severe transients. The cascade reactivity developed for the previous analysis of a 0.75$ step (SAR-39) is used as the basis for considering cascading in the analysis of the loop blowdowns. The assumed cascading becomes an envelope considered routinely in the operation.

The pressurized water loop facilities with an IPT in the flux traps can initiate a reactivity insertion as a result of various failures such as pipe rupture (loss of pressure control), loss of flow, loss of temperature control, or test failure. The reactivity insertion rate from thermal-hydraulic upsets in the IPT was calculated with the RELAP5 models of the loop facilities with an assumed total reactivity insertion for complete voiding of the IPT (0.80$ for standard and AHTL IPT and 1.0$ for LIPT). The loop facility initial conditions were selected to maximize the rate of reactivity insertion. The actual total reactivity and rate insertion associated with a failure are dependent on the operating conditions and physical configuration. Variables that can influence the magnitude of the reactivity perturbation are:

- Operating parameters (pressure, temperature, and flow rate)
- Water fraction inside the flow shroud
- Relative lobe power level (fraction of core power generated by the lobe)
- Power generation in the test and structural materials

- Test materials and their configuration
- Flow direction through the test train
- Volume of the inpile tube (large or standard)
- Heat transfer area
- Break location and size
- Direct moderator heating
- Reactivity weighting within the flux trap
- Heat transfer mode.

The UFSAR analysis used four basic configurations for the high pressure water loops. The configurations were selected based on previous experience and the current test program (SAR-39) and provide a basis for operation of the standard, AHTL, and LIPT loops. The pressurized water loop facilities are described in Chapter 10 (Experiment and Irradiation Facilities).

Those configurations are:

A. Standard loop facility with an IPT flow tube and a standard test train including fission power (i.e., experiment fuel) in the test train

B. Standard loop facility with a MUCH test train with no fission power in the test train

C. AHTL facility with a MUCH test train and no fission power in the test train

D. LIPT loop facility with a LIPT test train with fission power.

While the loop facilities maintain a fixed geometry where they interface with the core components (i.e., reflector or safety rod assembly), there is flexibility for variations within the irradiation facility (e.g., IPT). There are many variables with a wide range of possible values and numerable combinations of the values. No predetermined operating envelope covering all possible operating conditions could be provided. Therefore the analysis used initial conditions shown in Table 15.4.1 that resulted in severe but acceptable (i.e., results meet the ATR Plant Protective Criteria) reactivity transients. In some cases, existing sensitivity analyses have been used to provide a range of parameters in Table 15.4-1. Where no existing sensitivity data are available, the single value used in the analysis is listed. The loop facility experiment safety analysis and CSAP for each operating cycle demonstrate that the cycle specific loading and operating conditions also result in reactivity insertion events with consequences that meet the ATR Plant Protective Criteria. The cycle-specific analyses can use direct comparison with the initial conditions described in this Section, analyses of specific operating conditions or parameter sensitivity analyses using the methodology described in this Section, or other approved methodology. Acceptability can be shown by comparison of the reactivity transient or reactor power transient with those described in Sections 15.4.4, 15.4.5, and 15.4.6 or comparison of resulting statistical margins with the appropriate ATR Plant Protection Criteria. Additional information on the experiment safety analysis process is provided in Chapter 10 (Experiment and Irradiation Facilities).

15.4.2 Withdrawal of All Outer Shims and Neck Shims From 10^{-5} N_F

15.4.2.1 Identification of Causes and Accident Description. This event is not a limiting event in the Condition 2 category, but a discussion is included since it is the limiting startup reactivity insertion event and demonstrates the adequacy of the WRS.

The deterrents to continuous withdrawal of all outer shims and all 22 neck shims above 10^{-5} N_F are normal withdrawal procedures and interlocks that allow selection of only a single neck shim. The evaluation conservatively assumes all outer shims and 22 neck shims are withdrawn simultaneously. The determination of the reactivity insertion rate assumes an extreme 80/20 lobe power split and all control elements moving at the point of maximum differential reactivity insertion to maximize the reactivity insertion rate. The rate of reactivity insertion as derived in Chapter 4 (Reactor) is 0.079$/sec (0.053$/sec from the outer shim cylinders and 0.026$/sec from the 22 neck shim rods).

The necessary protection for this event is provided by the WRS with diverse protection provided by the neutron level subsystem.

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15.4.2.3 Conclusions. The WRS limits the consequences of the shim withdrawal event consistent with the Condition 2 criteria. The response requirements for the WRS are shown in Table 15.4-2.

15.4.3 Slow Power Increase or Lobe Power Balance Shift Due to Shim or Operator Failure

15.4.3.1 Identification of Causes and Accident Description. The power distribution in the reactor core (i.e., lobe-to-lobe power level) is controlled by the positioning of outer shim control cylinders and the neck shim rods. If an operator or the control system were to rotate the control cylinders out adjacent to the high power lobe, the regulating rod would maintain the total reactor power level constant while the power would increase in the high power lobe. The regulating rod would maintain the total reactor power until the regulating rod reached the inner limit of travel. Then the total reactor power would increase.

The neutron level and the WRS subsystem channels share thimbles and cable routings. This evaluation conservatively assumes that the channel of the neutron level and WRS subsystems in the high power lobe does not respond to the power increase due to some common fault or that the high-power lobe is not adjacent the PPS chambers.

15.4.3.2 Analysis of Effects and Consequences

15.4.3.2.1 Method of Analysis—The response of the reactor was evaluated for a range of lobe powers between 40 and 75 MW. It was assumed that the regulating rod would maintain the total reactor power constant and that no reactivity feedback would occur due to the changing power in the high power lobe. Two pairs of shim drums in one quadrant were rotated at 20 degrees/min with an effect on the lobe power of 10 MW/$. After the insertion of 0.57$, the maximum worth of a regulating rod, the total reactor power was allowed to increase (Atkinson 1976c).

15.4.3.2.2 Results—Evaluation of this event indicates that the maximum lobe power will be 1.34 N_F (80.4 MW lobe power when the initial lobe power is 60 MW) when the reactor is tripped by either the neutron level or the WRS subsystem channels not adjacent the high-power lobe (Atkinson 1976b). Analysis of steady-state operation shows the protective margins at 88.5 MW lobe power are 3.04σ for CHF and 4.05σ for FI when two pumps are operating. With three-pump operation at 96.0 MW lobe power, the margins are 3.25σ for CHF and 4.44σ for FI (Polkinghorne 1994a).

15.4.3.2.3 Conclusions—The neutron level and WRS subsystems will prevent the lobe power from exceeding 1.34 N_F during a power imbalance with concurrent failure of the channel in the lobe with the power imbalance. The quadrant differential temperature subsystem with a setpoint of 1.24 times the normal quadrant ΔT will limit the overpower to 1.45 N_F. The neutron level, the WRS, and the quadrant differential temperature subsystems limit the consequences of the slow power increase and lobe power balance shift events consistent with Condition 2 criteria.

15.4.4 Inpile Tube Decompression with Subsequent Voiding Due to Failure of the 1/2-Inch Piping

15.4.4.1 Identification of Causes and Accident Description. The decompression and subsequent voiding of an IPT as a Condition 2 event can occur as a result of the following:

A. Mechanical failure of any ½-in. or smaller piping

B. Inadvertent opening of:

1. any valve isolated line
2. any pressure relief device.

This can occur as a result of operator error or mechanical failure of components. Piping failures considered are 1/2-in. or less for this probability classification. The evaluation envelops low pressures resulting from control system failures as well. The analysis can be shown to envelop other failures in large diameter pipes that are isolated from the IPT by substantial piping lengths. Such evaluations may be completed on a case basis to support other activities (e.g., the application of the Inservice Inspection (ISI) plan to the loops).

The limiting failure is a shear of a 1/2-in. pipe in the drain manifold attached to the loop piping at the heater legs. This event results in IPT voiding and a positive reactivity insertion. A scram occurs on high power and is initiated by the neutron level subsystem. The dominant shutdown mechanism is feedback from the fuel elements. The safety rod insertion provides some mitigation of the event and ensures the reactor remains subcritical. Diverse protection is provided by the WRS.

15.4.4.2 Analysis of Effects and Consequences

15.4.4.2.1 Method of Analysis—The RELAP5/MOD3 code was used to calculate the reactivity transient from loop voiding for this analysis (Nielsen 1994). The four configurations evaluated and the applicable operating conditions are described in Section 15.4.1.1. The pipe break was assumed to occur instantaneously resulting in a double-ended-offset shear (DEOS).

The reactivity history from Nielsen was input to the detailed ATR RELAP5 model (Terry 1994) as described in Section 15.4.1.1 to obtain initial conditions and power history for input to ATR-SINDA. A scram with a response time of 0.025 s was assumed to occur at 1.2 N_F. An additional positive reactivity insertion was added as a ramp beginning at zero time and inserting 0.05$ in 0.13 s (it is constant after 0.13 s) to conservatively represent cascading from other loops. The cascade feedback for this event was estimated using results from other analyses. The cascade reactivity as a function of time for the more severe transient from a 0.75$ step is reported in the DBR (SAR-39). This curve was used to determine the cascade feedback reactivity insertion for the ½-in. pipe break transient. To obtain a reasonable estimate of the cascade feedback reactivity, the time of the peak power was obtained from the 0.5$ step in Section 15.4.5. The cascade reactivity insertion at the time of the peak power for the 0.5$ step was used as an approximation for the cascade reactivity insertion for the ½-in. pipe break transient. The amount of cascading reactivity inserted at that time was picked from the DBR curve. The resulting value was about 0.025$ and was conservatively addressed by adding 0.05$ as a ramp over 0.13 s.

The approach to the CHF was evaluated using ATR-SINDA and SINDA-SAMPLE.

15.4.4.2.2 Results—The loop blowdown analyses showed that the bounding reactivity insertion occurred for the standard loop with the MUCH configuration. The insertion rate was initially about 6$/second and reached 0.61$ total insertion at 0.2 seconds for an average rate of about 3$/second. The insertion history is listed in Table 15.4-3.

The RELAP5 results for reactor power show a maximum of 428 MW at 0.12 seconds for the two-pump case and 365 MW at 0.13 seconds for the three-pump case. These responses are shown in Figures 15.4-3 and 15.4-4 respectively.

The ATR-SINDA and SINDA-SAMPLE calculations (Polkinghorne 1994a) show that the event meets the acceptance criteria for a Condition 2 event (margins to CHF and FI exceed 3σ) as well as the criteria for a Condition 3 event (2σ to buckling). Results of the calculations are as follows:

Parameter	2-pump	3-pump
Peak fuel plate centerline temperature (Figures 15.4-5 and -6)	635°F	660°F
CHF margin (Figures 15.4-7 and -8)	3.40σ	3.3σ
Flow instability margin (Figures 15.4-7 and -8)	4.3σ	4.3σ
Fuel plate buckling margin (Figures 15.4-7 and -8)	3.3σ	2.7σ

15.4.4.2.3 Conclusion—The protective criteria margins will be preserved with maximum effective Plate 15 powers of 443 MW and 417 MW for the three-pump operation and the two-pump operation, respectively. A neutron level subsystem, which initiates reactor trip at 1.2 N_F with a response time of 0.025 seconds is also required by the analytical assumptions. However, as is evident from the time scale involved, the dominant effect in terminating the event is inherent feedback. The scram does contribute to the termination of the event and ensures the reactor remains subcritical.

15.4.5 0.50$ Step Insertion

15.4.5.1 Identification of Causes and Accident Description. The 0.50$ step is a hypothetical fault which is used as a bounding event for the following several near step events:

A. Limited perched fuel element (Condition 2)

B. Loop experiment hardware failure (Condition 3)

C. Loss of reflector coolant (Condition 3)

D. Reflector movement toward core (Condition 3)

E. Perched fuel drop of two fuel elements from within measurement accuracy (Condition 3)

F. Cold water addition from opening of primary pump discharge valve (Condition 3)

G. Gas release into the core (Condition 3)

H. Perched fuel element (greater than the measurement uncertainty) (Condition 4)

I. Movement of two reflector blocks (Condition 4).

The fuel elements are positioned and seated using a fuel handling tool. The elevations are measured and recorded, and the measurements are repeated with quality verification. The Condition 2 fault assumes a fuel element could be perched (i.e., not fully seated) not more than one in. after primary coolant flow is established and reactor criticality is attained. Advanced Test Reactor Critical Facility (ATRC) measurements indicate a maximum insertion of 0.062$ for this fault (SAR-39).

Failure of the loop test hardware during reactor operation is considered at the Condition 3 frequency of occurrence (Durney 1989). The reactivity insertion from hardware failure is administratively limited via the CSAP to 0.10$ for consistency with the assumption in the loop blowdown analysis in Section 15.4.6.

Loss of cooling to the two reflector blocks surrounding the high power lobe, gas leakage, or void progression through the reflector coolant holes are Condition 3 events that would result in a maximum calculated reactivity insertion of 0.38$ (SAR-39).

Two beryllium blocks surround each of the four corner lobes. Specific failures of the beryllium or its supports could allow the block to reposition toward the fuel elements. The Condition 3 failure of one block adjacent to a high power lobe (relative) can cause a reactivity insertion not greater than 0.25$ (Durney 1972).

The fuel elements are seated by an operator with the fuel handling tool. The elevation of each fuel element is then independently measured and recorded. However, seating of two fuel elements perched as much as 1-in. is assumed to be a Condition 3 event. Conservative measurements indicate a maximum insertion worth of 0.062$ for a 1-in. insertion of the highest worth element, or 0.124$ for two elements (SAR-39).

Cold water addition by opening a PCP discharge valve can result in a positive reactivity addition at the Condition 3 frequency of occurrence. The maximum change and rate of change in water temperature is limited by the physical arrangement of the PCS. The PCS consists of one coolant loop with five heat exchangers and four PCPs. Two or three of the PCPs are normally in operation with the remaining PCP in standby. Only the PCPs have isolation valves so the only location in the PCS where a volume of cold water can accumulate is at the standby PCP. Because of the arrangement of the PCS, the accumulation of cold water for rapid insertion into the core corresponds to less than a 0.10$ reactivity insertion (SAR-39).

Gas release into the core which can result as a consequence of a failure in the envelope tube and is a Condition 3 fault. The gas can potentially enter the fuel annulus and degrade heat transfer (also cause a negative reactivity insertion) or flow through the core external to the fuel annulus where voiding is positive. The significant reactivity effect is voiding of the flux trap around the IPT (the effects on the fuel element are described in Section 15.10). The failure of the envelope tube is assumed to be a Condition 3 event since the jacket is not inspected for flaws (ultrasonic exams are not performed during service). The Condition 3 failure of the envelope tube is assumed to be progressive rather than abrupt. Therefore, the enveloping gas flow is assumed to be the delivery rate to the envelope tube. Surveillance of the gas flow and the tank activities would be expected to detect the failure before it progresses to this maximum flow rate. The gas delivery system is regulated and has a pressure relief system; consequently, a failure of the envelope tube with higher than normal gas pressure is not considered credible (Atkinson 1994). The maximum gas flow occurs for the AHTL and the LIPT; however, the limiting effect is in the AHTL where the void coefficient is positive in the annulus outside the envelope tube.

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The Condition 4 perched fuel element drop is assumed to be a drop from a height greater than the measurement accuracy. The limiting element perch hypothesized is 9 in. based on an evaluation that indicates the element could potentially be perched by 7.25 in. (Durney 1994a). To be of concern, the insertion of the fuel element would have to occur after the reactor is critical. The perched element would have to be missed in the tank closure activities and not be forced into the core by the differential pressure associated with full flow. Analyses indicate the total insertion for an element falling 9 in. is not more than 0.5$ (Durney 1994a).

Movement of a single reflector block results in a maximum of 0.25$ (Durney 1972). Simultaneous movement of two blocks could result in a reactivity insertion of 0.50$.

The necessary protection for this event is provided by the neutron level PPS subsystem. Diverse protection is provided by the WRS.

15.4.5.2 Analysis of Effects and Consequences

15.4.5.2.1 Method of Analysis—The step reactivity insertion was input to the detailed ATR RELAP5 model (Terry 1994) as described in Section 15.4.1.1 to obtain initial conditions and power history for input to ATR-SINDA. The reactivity insertion of 0.50$ was inserted at time zero as an instantaneous step. Cascading at 0.04 s is only 0.03$ for the 0.75$ step which reached 756 MW. Since power is substantially less for the 0.5$ step, the feedback will be less than 0.03$. Cascading for this event is considered not significant; therefore, no cascade feedback from the flux traps was assumed. A reactor trip occurred from the neutron level subsystem at 1.2 N_F with a response time of 0.025 seconds.

The approach to CHF was evaluated using the ATR SINDA and SINDA-SAMPLE (Polkinghorne 1994a).

15.4.5.2.2 Results—The RELAP5 results show a peak reactor power of 435 MW occurs at 0.04 seconds for the two pumps and 369 MW at 0.04 seconds for three pumps. The transient power response is illustrated in Figures 15.4-9 and -10 for the two-pump and three-pump cases, respectively.

The ATR-SINDA and SINDA-SAMPLE calculations show that the event meets the acceptance criteria for a Condition 2 event (margins to CHF and FI exceed 3σ) as well as the criteria for a Condition 3 event (2σ to buckling). Results of the calculations are as follows:

Parameter	2-pump	3-pump
Peak fuel plate centerline temperature (Figures 15.4-11 and -12)	613°F	637°F
CHF margin (Figures 15.4-13 and -14)	3.76σ	3.76σ
Flow Instability Margin (Figures 15.4-13 and -14)	4.95σ	5.02σ
Fuel Plate Buckling Margin (Figures 15.4-13 and -14)	3.95σ	3.34σ

15.4.5.2.3 Conclusion—The protective criteria margins are maintained for both two pump and three pump operation during this event. A neutron level PPS subsystem which initiates reactor trip at 1.2 N_F with a response time of 0.025 seconds is required by the analytical assumptions.

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15.4.7 Bounded Events - Reactivity and Power Distribution Anomalies

15.4.7.1 Rapid Regulating Rod Withdrawal. A component failure can cause the regulating rod to drive out. This event can occur at power levels between N_L and N_F. The reactivity insertion for this event is 1.05$/sec with a maximum insertion of 0.57$ as derived in Chapter 4 (Reactor). This insertion is not as severe as the 1/2-in. break evaluated in Section 15.4.4 which results in a ramp with a maximum rate of nearly 6$/sec and a maximum insertion of 0.61$. The consequences of regulating rod event are bounded by the limiting small pipe break event discussed in Section 15.4.4.

15.4.7.2 Withdrawal of all Outer Shims from 10^{-10} N_F. Failure of a timer interlock can occur due to a single failure in the Log N system. As described in 15.4.2, the maximum insertion rate is 5.3¢/second. The ramp rate is less than the rate of 30¢/sec evaluated in Section 15.4.2, and the consequences are bounded.

15.4.7.3 Withdrawal of all Outer Shims and One Neck Shim from N_L. This fault can occur as a result of operator error. The maximum addition rate for withdrawal of a single neck shim is 0.2¢/sec [Chapter 4 (Reactor)]. Combination with the outer shim rate results in an insertion rate of 5.5¢/sec, which is less than the 30¢/sec ramp discussed in Section 15.4.2.

15.4.7.4 Cold Water Injection. This event is discussed in Section 15.1.

15.4.7.5 Inpile Tube Voiding Due to a Rupture Disk or Relief Valve Failure or an Open and Accessible Valve. Faults which result in high loop pressure, such as excess makeup, high temperature, or pressurizer heaters stuck on, will result in rupture disc and/or relief valve opening. Debris can result in failure to reseat. Drain valve opening can result due to operator error. Drain valves are located on one-half in. piping.

The size of the opening in the relief valve is less than the inside diameter of the one-half in. pipe (Pace 1987). Therefore, the relief valve opening is enveloped by the analysis for the limiting small pipe break in Section 15.4.4 (note that the relief valve is listed in Section 15.4.4). Opening a drain valve requires a finite amount of time to complete. Since the valve is located on one-half in. piping, the assumed instantaneous failure of an inlet pipe discussed in Section 15.4.4 will also bound this event.

15.4.7.6 Loop Flow Coastdown or Loop Loss of Temperature Control With a Loop Instrumentation Initiated Reactor Trip. This fault is initiated by a loss of commercial power (or other events affecting the loop pumps) or failure of the temperature control due to controller failure or operator error.

Analyses for a flow coastdown in an ATR inpile loop, or a loss of the heat exchangers in the loop, determined that with loop low flow or high inlet temperature trips, the reactivity addition due to these accidents is within the control ability of the ATR regulating rod (Atkinson 1976a; Durney 1980a, 1980b, 1980c, 1980d, 1980e, 1980f; McCracken 1994, Polkinghorne 2006, Stites 2008). Therefore, no power increase would occur in the ATR for either of these accidents before the reactor would be tripped by the loop instrumentation.

15.4.7.7 Powered Axial Locating Mechanism Drive System Failures

15.4.7.7.1 Simultaneous Movement of Two Cycling Tests—Simultaneous movement of two PALM tests can occur from several sources, e.g., a high-temperature updrive that occurs in one PALM while the other is cycling normally, or a control system failure. As shown in Durney (1982), the limitation on the reactivity effect of the PALM test movement (the reg rod must control for normal cycling) results in the test worth of a single PALM being 0.49ρ where ρ is the total regulating rod worth. With this limitation, the worth of two tests is very nearly the worth of the regulating rod. The speed of the drive may approach 12 in/sec (Durney 1982) resulting in the minimum time for full stroke (36 in.) being 3 sec. Because the reactivity insertion is essentially the same but occurs over a longer time, the event is enveloped by the abrupt withdrawal of the regulating rod (Section 15.4.7.1) and meets the Condition 2 protection criteria.

15.4.7.7.2 Uncontrolled Movement of a Positioner Test—The PALM can be used for positioning a test train during reactor operation. Normal operation of the PALM for test positioning requires small movements resulting in reactivity changes within the range of the regulating rod (Durney 1984). However, failures of the hardware or mechanism or operator failures could result in large movement with reactivity insertions that are outside of the control range of the regulating rod. Additionally, if the test is withdrawn to the upper limit, the configuration could represent an unacceptable potential void reactivity insertion. These two considerations are addressed in the analysis for use of the PALM for test positioning (Durney 1984). Administrative controls on use of the PALM in positioner mode limit the potential for a significant reactivity insertion to a low frequency. Any positive reactivity insertion would be compensated first by the regulating rod and then by the automatic insertion of the outer shim cylinders. The PPS would limit any power transient to $\leq 1.2\ N_F$ thus meeting the Condition 2 protection criteria.

The potential void analysis in the CSAP and experiment safety analysis considers the full axial travel of PALM tests. Any restrictions necessary to limit the potential void reactivity insertion for driven tests are imposed by the CSAP. The administrative controls mentioned above are further discussed in Chapter 10 (Experiment and Irradiation Facilities).

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15.4.7.8 Withdrawal of All Outer Shims and Neck Shims from 10^{-10} N_F. Failure of the control rod withdrawal timer interlock, along with operator error, can result in withdrawal of all outer shims and neck shims from 10^{-10} N_F. This results in a reactivity addition rate of 7.9¢/sec as discussed in Section 15.4.2 Since the evaluation in Section 15.4.2 is based on calculations for enveloping initial powers, the results in Section 15.4.2 are enveloping.

The consequences for this fault are bounded by the outer shim and neck shim withdrawal described in Section 15.4.2.

15.4.7.9 Inpile Tube Voiding Due to Opening of a Normally Inaccessible Valve. This event is identical to the event in Section 15.4.7.5 except that the valve is normally inaccessible. The consequences for this fault are bounded by the event discussed in Section 15.4.4.

15.4.7.10 Pressure Tube Flow Bypass Due to a Maximum Useful Capacity Holder Test Train Failure with Loop Scram. The MUCH configuration utilizes the experiment shroud as the flow tube. A failure of the shroud or test train would result in a bypass of flow from the inlet to the outlet. This will result in void formation and a positive reactivity insertion. Analyses have shown that the peak insertion for this fault is 0.39$ (for a total IPT voiding of 0.80$) when the loop experiment flow scram occurs at 1.45 seconds (McCracken 1994).

This insertion is less than a 0.50$ step, and the consequences are bounded by the event discussed in Section 15.4.5. Although the 0.5$ step insertion is used to bound this event, it is shown separately from the near step events because a specific analysis of the potential reactivity insertion was performed (McCracken 1994).

15.4.7.11 Loss of Loop Temperature Control Due to Heat Exchanger Failure or Line Heaters Sticking on Without Loop Scram. This fault is initiated by temperature control valve failure, controller failure for heat exchanger flow or line heaters, or loss of secondary flow. This causes a gradual temperature rise (3°F per second maximum) in the loop coolant with a resultant decrease in density. The modest reactivity insertion due to the density change would increase the power slowly to the neutron level PPS subsystem setpoint and cause a scram. However, the regulating rod can compensate the slow ramp up to a point slightly beyond saturation temperature at the IPT inlet. Voiding will progress relatively rapidly upon reaching this condition. However, the rate of reactivity addition is expected to be less than the rate observed for the one-half in. pipe break for two reasons. First, the total reactivity has been reduced by the amount of regulating rod movement. Second, the rate of addition for the heatup should be influenced by the rate at which the heat of vaporization is added which is slower than the propagation of the depressurization from the pipe break. Based on the calculations for MUCH (McCracken 1994), it may also be expected that the loop relief valve will lift due to the added heat. Lifting the loop relief valve is enveloped by the one-half in. pipe break. The event is enveloped by the blowdown analysis in Section 15.4.4.

15.4.7.12 Slow Lobe Power Balance Shift Due to Shim, Lobe Power Indicating System or Operator Failure with Operator Compensation. The regulating rod is generally maintained within the operating range by rotation of shim drum pairs. Erroneous movement of shims in or adjacent to a high power lobe could occur as a result of shim failures or a failure of the lobe power indicating system. If an additional operator error resulted in compensation in other lobes by inserting shims, lobe powers exceeding the PPS neutron level subsystem or WRS scram setpoints may be reached. This assumes the PPS neutron level subsystem and WRS channels adjacent to the high power lobe are not functional.

Analyses for this fault have demonstrated that the quadrant differential temperature PPS subsystem will trip the reactor before exceeding a power level of 1.45 N_F (Atkinson 1976b; Atkinson 1976c). The maximum lobe power (145% of 60 MW) is 87 MW. A more detailed discussion of the event is provided in Section 15.11.13.2.

The consequences of this event are bounded by the limited overpower discussed in Section 15.4.3.

15.4.7.13 Withdrawal of all Shims and Safety Rods from 10^{-10} N_F. Withdrawal of the outer shims and neck shims is discussed in Section 15.4.2. As with the analysis described in Section 15.4.2, the determination of the reactivity insertion rate assumes an extreme 80/20 lobe power split and all control elements moving at the point of maximum differential reactivity insertion to maximize the insertion rate. The rate of reactivity insertion from the safety rods, as derived in Chapter 4 (Reactor), is 0.14$/sec. Combining this with the 0.079$/sec from the outer shims and neck shims results in a total insertion rate of 0.22$/sec. As shown in Figure 15.4-2, the peak power is less than 321.6 MW for a ramp rate of 0.30$/sec. The event is acceptable as described in Section 15.4.2.2.

15.4.7.14 Withdrawal of all Outer Shims from 10^{-10} N_F with Failure of the Wide Range Subsystem. As discussed in Section 15.4.2, the maximum ramp rate for this fault is 0.053$/sec. Analysis in Polkinghorne (1994b) for an unlimited ramp insertion of 0.079$/sec showed that the PPS neutron level subsystem with a reactor trip at 1.2 N_F would limit power to 312 MW which is 4% above the trip setting. As described in Section 15.4.2, the margins to CHF are greater than 3.0σ for more than 7.2% above the trip setting. This event is acceptable as a Condition 4 event for both two and three-pump operation since CHF does not occur (no melting).

15.4.7.15 Withdrawal of All Outer Shims and Neck Shims from 10^{-5} N_F with Failure of the Wide Range Subsystem. This fault is the same as the fault discussed in Section 15.4.2 with concurrent failure of the WRS. The maximum ramp rate is 0.079$/sec. The analysis of a ramp of 0.079$/sec from 10^{-10} N_F (Polkinghorne 1994b) envelops this event. Beginning the ramp at 10^{-10} N_F inserts more reactivity into the system than beginning at 10^{-5} N_F. Therefore as noted in Section 15.4.7.14, the event meets Condition 4 criteria.

15.4.7.16 Rapid Regulating Rod Withdrawal with Failure of the Wide Range Subsystem. The failure of the regulating rod is discussed in Section 15.4.7.1. The envelope discussed in 15.4.7.1 assumes that the event is terminated by the feedback and the high power scram. There is no consideration of a scram on rate of change (WRS function); therefore, the evaluation for the high probability event (Condition 2) is applicable to the event with failure of the WRS.

15.4.7.17 Withdrawal of All Outer Shims and Neck Shims from 10^{-5} N_F Coupled with Rapid Regulating-Rod-Withdrawal at 10^{-3} N_F. This fault is a combination of the faults described in Sections 15.4.7.1 and 15.4.2, with concurrent failure of the shim withdrawal permit interlock when the reg-rod is below 20 in. withdrawn. The combined accident is a ramp of 0.079$/sec from 10^{-5} N_F to 10^{-3} N_F, then 1.129$/sec (0.079$/sec from the outer shim cylinders plus 1.05$/sec from the regulating rod) for 0.543 seconds (total reg-rod insertion is 0.57$), then an unlimited ramp at 0.079$/sec. An analysis of this event (Polkinghorne 1994b) demonstrates that the WRS terminates the event before the regulating rod withdrawal begins. The power level does not reach 10^{-3} N_F.

15.4.7.18 Driven Test Loop Blowdown with Experiment Hardware Failure. The driven test has the potential for being located at various axial positions dependent on desired operating conditions. The variable position results in a variable drop distance for the test and a potentially variable reactivity effect if there is a test failure. As described in Section 15.4.6, the blowdown of the loop due to a large pipe break was assumed to cause failure of the test train. Driven tests are limited such that this event is enveloped by the analysis in Section 15.4.6. The evaluation of individual test trains is discussed in Section 15.4.1.

Harris (1981) evaluated the impact of the test on the test stop. The test stop is supported by the flow tube. There is significant strain for the flow tube due to the drop, but failure is not predicted.

15.4.7.19 Voiding in All Inpile Tubes Due to a Simultaneous Flow Coastdown or Loss of Temperature Control in All Loops. Failure of the commercial and diesel power sources will result in simultaneous coastdown of all loop pumps. If the loop scrams fail, voiding in all IPTs could occur. Loss of heat sink (i.e., failure of the HDW system) to the experiment loops or loop heater failing on without a loop scram could also cause voiding in all IPTs. The loss of temperature control (event titled "Voiding in all IPTs due to a simultaneous loss of temperature control in all loops") progresses slower than the coastdown and is enveloped by the coastdown analysis. Several passes are required for heating in the loss of temperature control; the flow coastdown occurs in less than 3.5 seconds. Concurrent failure of the loop protective system is assumed, so no reactor scram will occur until the neutron level PPS subsystems trip at 1.2 N_F.

An evaluation of the flow coastdown (Durney 1993) indicates the reactivity insertion for nine loops (seven standard IPT and two LIPT) begins with a modest rate of 0.45$/s to a total of about 1.0$ and then becomes more rapid with a rate of nearly 3.0$/s to a total of about 4$. The PPS will initiate a scram during the first modest insertion and terminate the event. A scram effectively stops heating in the IPT, and the effect of density change is nullified. The peak power for the event is 310 MW and occurs at 0.52 s (Terry 1994). As noted in 15.4.2.2 the overpower (24% in this case) is enveloped and acceptable since the margins to CHF are greater than 3σ. (The evaluation for nine loops is a conservative estimate for configurations with fewer IPTs.) (Polkinghorne 2006, Stites 2008)

15.4.7.20 Loss of Loop Temperature Control in the ATR High Temperature Loop Due to Heat Exchanger Failure or Line Heaters Stuck Without a Loop Scram. Failure of the temperature control system will result in a slow heatup of the loop. Analysis for an AHTL with a MUCH structural test shows the reactivity insertion rate is less than 0.008$/sec prior to opening of the relief valves (McCracken 1994). After relief valve opening, the blowdown is conservatively bounded by the inlet pipe DEOS.

Consequences for this fault are bounded by the event discussed in Section 15.4.6.

15.4.7.21 Loop Flow Coastdown Without a Loop Scram. This event initiator is the same as discussed in Section 15.4.7.19 but for a single loop. The reactivity insertion is about a ninth of that considered in Section 15.4.7.19. As discussed in that section, the initial reactivity insertion rate is low followed by a higher reactivity insertion rate. The initial ramp is 0.05$/sec followed by a 0.33$/sec ramp (Durney 1993) which is enveloped by the same event for all loops discussed in Section 15.4.7.19. (Polkinghorne 2006, Stites 2008)

15.4.7.22 Pressure Tube Cooling Flow Bypass Due to a Maximum Useful Capacity Holder Test Train Failure Without a Loop Scram. Initiation of flow bypass is discussed in Section 15.4.7.10. This fault assumes concurrent failure of the loop protective system scram. With failure of the loop scram, the event can progress to voiding in the IPT. The progression of the voiding was developed in McCracken (1994); it progresses no faster than the DEOS of the inlet piping. Therefore, the event is enveloped by the loop piping failure discussed in Section 15.4.6. The analyses show that the peak insertion for this fault is less than 0.5$ (for a total IPT voiding of 0.80$) so the event is also bounded by the 0.5$ step insertion consequences.

15.4.8 Failure of Protective Subsystems

The neutron level and wide range PPS subsystems each generate a scram signal on power level (analysis of reactivity events assumes 1.2 N_F). Therefore, identical protection for power level is provided by these subsystems; for events terminated on high power, either is primary and the other diverse. The WRS provides protection for startup events and terminates them on rate of change (scram signal generated because the signal changes too rapidly to allow a range change). For these events (Sections 15.4.7.14 and 15.4.7.15 the neutron level subsystem is shown to be diverse.

Table 15.4-1. Range[a] of loop analysis parameters used to establish maximum core power response.

Parameter	Standard (Fueled)	Standard (MUCH)	LIPT	AHTL MUCH
Lobe Power (MW)	60	60	34	60
Test Fission Power (kw)	0-200	0	0-200	0
Loop Flow (gpm)	20-30	29.4-44.3	200-300	29.4-44.3
Venturi Flow (gpm)	20-30	20-30	200-300	20-30
IPT Exit Temperature (°F)	560-590	590-605	640	680
Inlet Pressure (Psia)	2200-2400	2200-2400	2200-2400	3275
Water Fraction	0.3-0.5	0.3-0.5	0.44	0.3-0.5
Pressure Tube Void ($)	0.0-0.8	0.0-0.8	0.0-1.0	0.0-0.8
Test Failure Reactivity ($)	0.0-0.10	0.0-0.10	0.0-0.10	0.0-0.10
Test flow direction[b]	up	up	up	up

a. In some cases, existing sensitivity analyses have been used to provide a range of parameters in table 15.4-1. Where no existing sensitivity data area available, the single value used in the analysis is listed.

b. Flow through the experiment is up, previous analysis (SAR-39) indicates that the configuration is enveloping for MUCH.

Table 15.4-2. Response requirements for the WRS (pressurized operation).

	Range (% of N_F)				
Downward Re-range Power (% of N_F)	Low Power	Full Power	1.2 N_F Trip Power (% of N_F)	1.15 N_F Trip Power (% of N_F)	Scram Delay Time (msec)
27.0	30.0	100.0	120.0	115.0	25
9.0	10.0	30.0	36.0	34.5	25
2.70	3.0	10.0	12.0	11.5	25
0.90	1.0	3.0	3.60	3.45	25
0.270	0.30	1.0	1.20	1.15	35
0.090	0.10	0.30	0.36	0.345	35
0.00270	0.030	0.10	0.120	0.115	35
	0.010	0.030	0.0360	0.0345	35
	0.0030	0.10	0.0120	0.0115	35
	0.0010	0.0030	0.00360	0.00345	35
(For N_F = 250 MW.)					
	Range (MW)				
Re-range Power (MW)	Low Power	Full Power	1.2 N_F Trip Power (MW)	1.15 N_F Trip Power (MW)	Scram Delay Time (msec)
67.5	75.0	250.0	300.0	287.5	25
22.5	25.0	75.0	90.0	86.25	25
6.75	7.5	25.0	30.0	28.75	25
2.25	2.5	7.5	9.0	8.62	25
0.675	0.75	2.5	3.0	2.87	35
0.225	0.25	0.75	0.90	0.862	35
0.0675	0.075	0.25	0.30	0.287	35
0.0225	0.025	0.075	0.090	0.086	35
0.00675	0.0075	0.025	0.030	0.029	35
	0.0025	0.0075	0.0090	0.0086	35

Table 15.4-3. Reactivity insertion for 1/2-in. drain line (MUCH).

Time(s)	Reactivity($)
0.0	0.000
0.032	0.123
0.064	0.335
0.100	0.468
0.132	0.532
0.164	0.584
0.200	0.608
0.232	0.612
0.264	0.601
0.300	0.570
0.332	0.551
0.364	0.533
0.400	0.556
0.500	0.660
0.600	0.685
0.700	0.701
0.800	0.720
0.900	0.700
1.000	0.565
1.200	0.512
1.500	0.503
1.900	0.441

Table 15.4-4. Reactivity insertion for pipe break (greater than 1/2-in.).

Time(s)	Reactivity($)
0.000	0.000
0.020	0.170
0.040	0.307
0.080	0.443
0.120	0.486
0.160	0.516
0.200	0.539
0.280	0.565
0.300	0.570
0.360	0.584
0.400	0.597
0.500	0.630
0.600	0.584
0.700	0.483
0.800	0.450
0.900	0.443
1.000	0.446
1.200	0.448
1.400	0.424
1.600	0.407
1.940	0.401

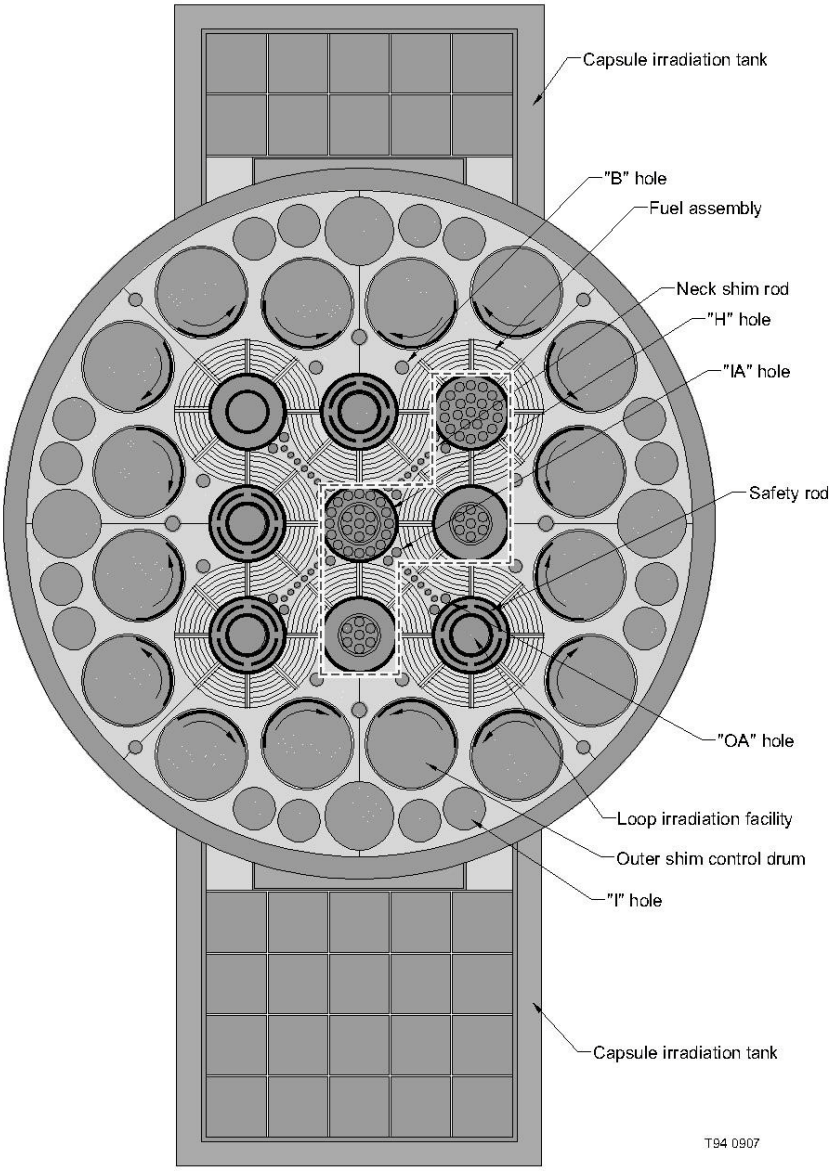

Figure 15.4-1. ATR core cross-sectional diagram. **(For Information Only)**

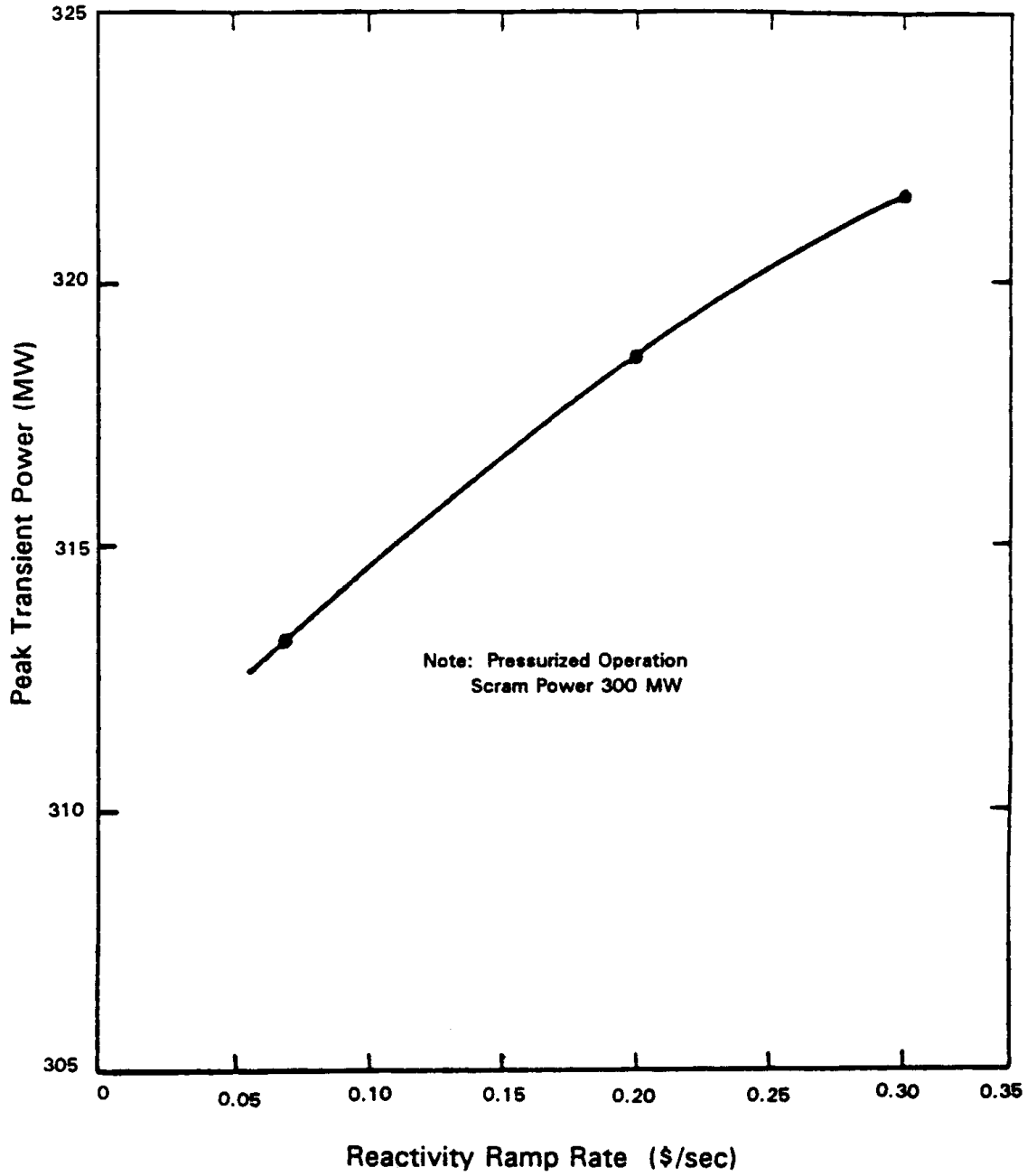

Figure 15.4-2. Maximum expected power versus constant ramp rate for pressurized operation with uniform power distribution. **(For Information Only)**

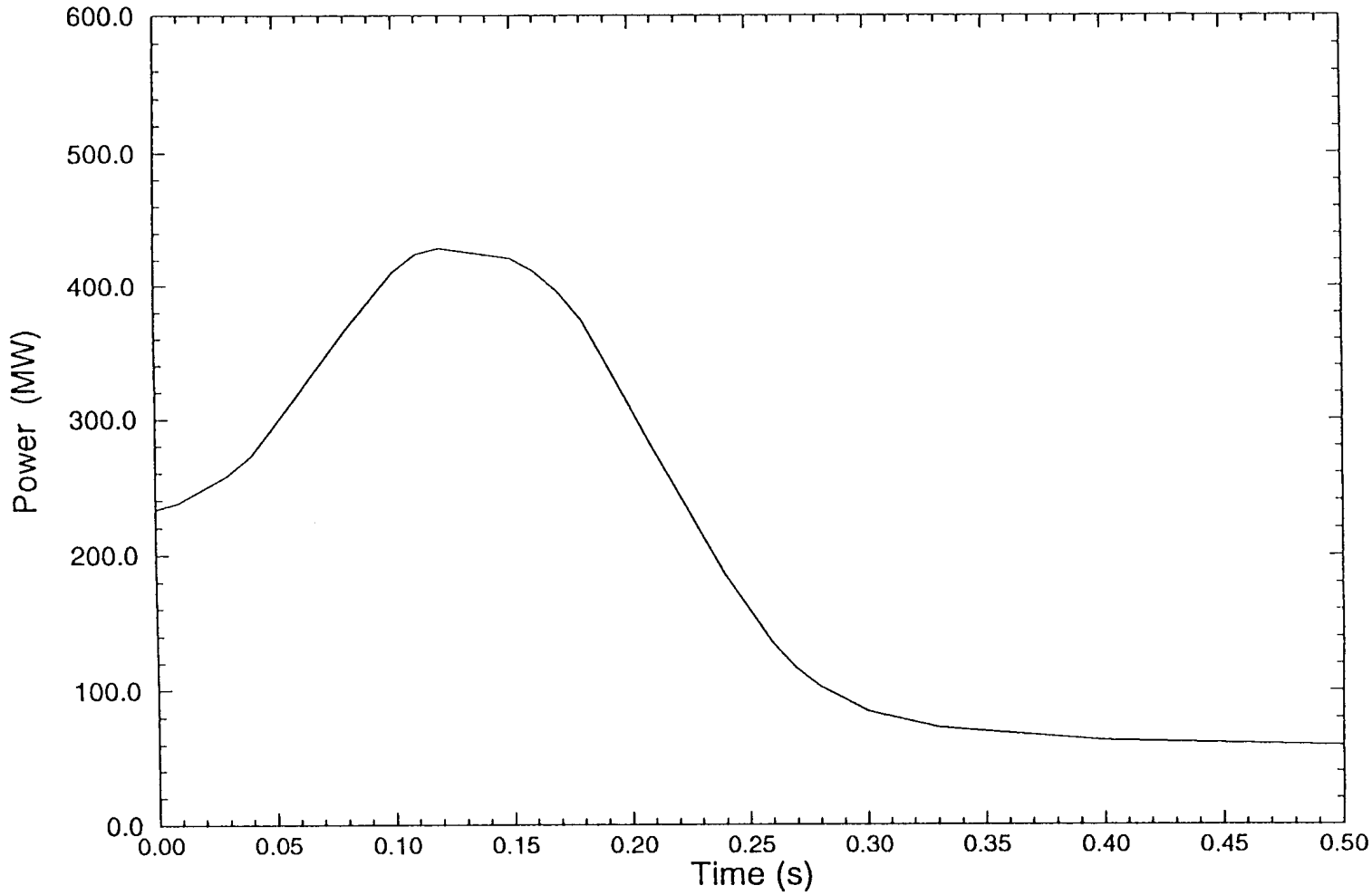

Figure 15.4-3. Reactor power (SIPT 1/2-in. break RIA - two pump operation). **(For Information Only)**

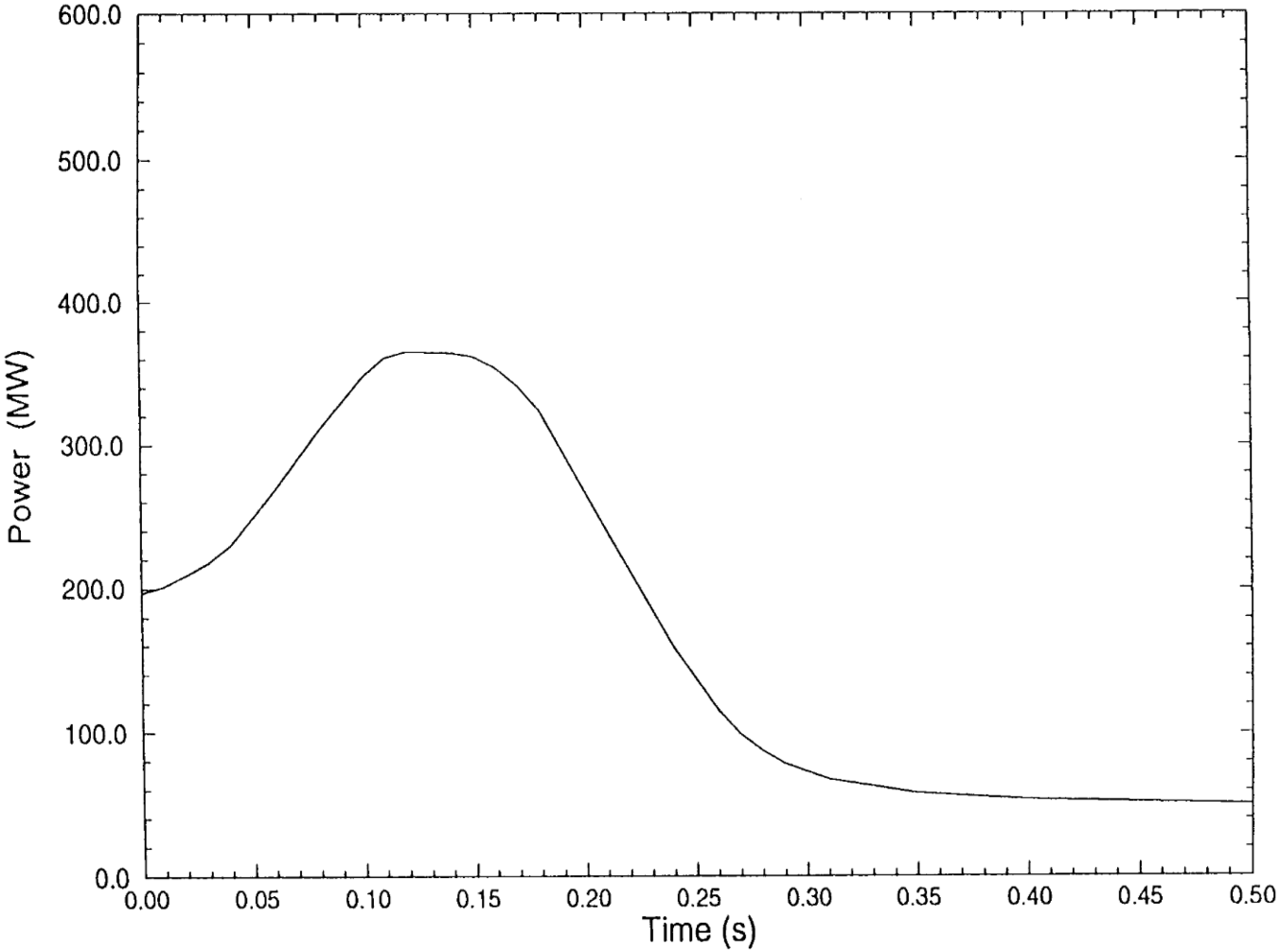

Figure 15.4-4. Reactor power (SIPT 1/2-in. break RIA – three-pump operation). **(For Information Only)**

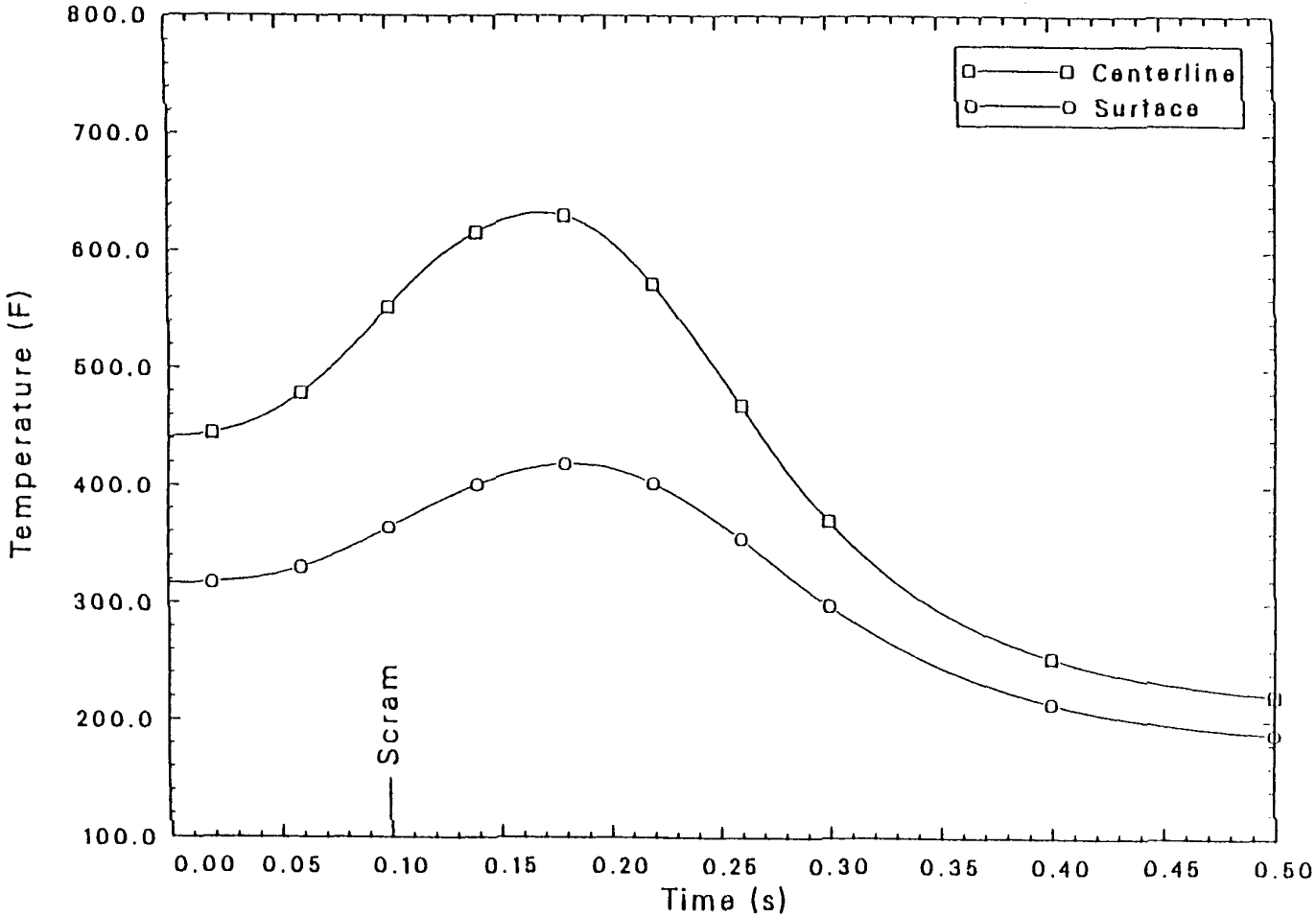

Figure 15.4-5. Maximum fuel plate temperature (SIPT 1/2-in. break RIA – two-pump operation). **(For Information Only)**

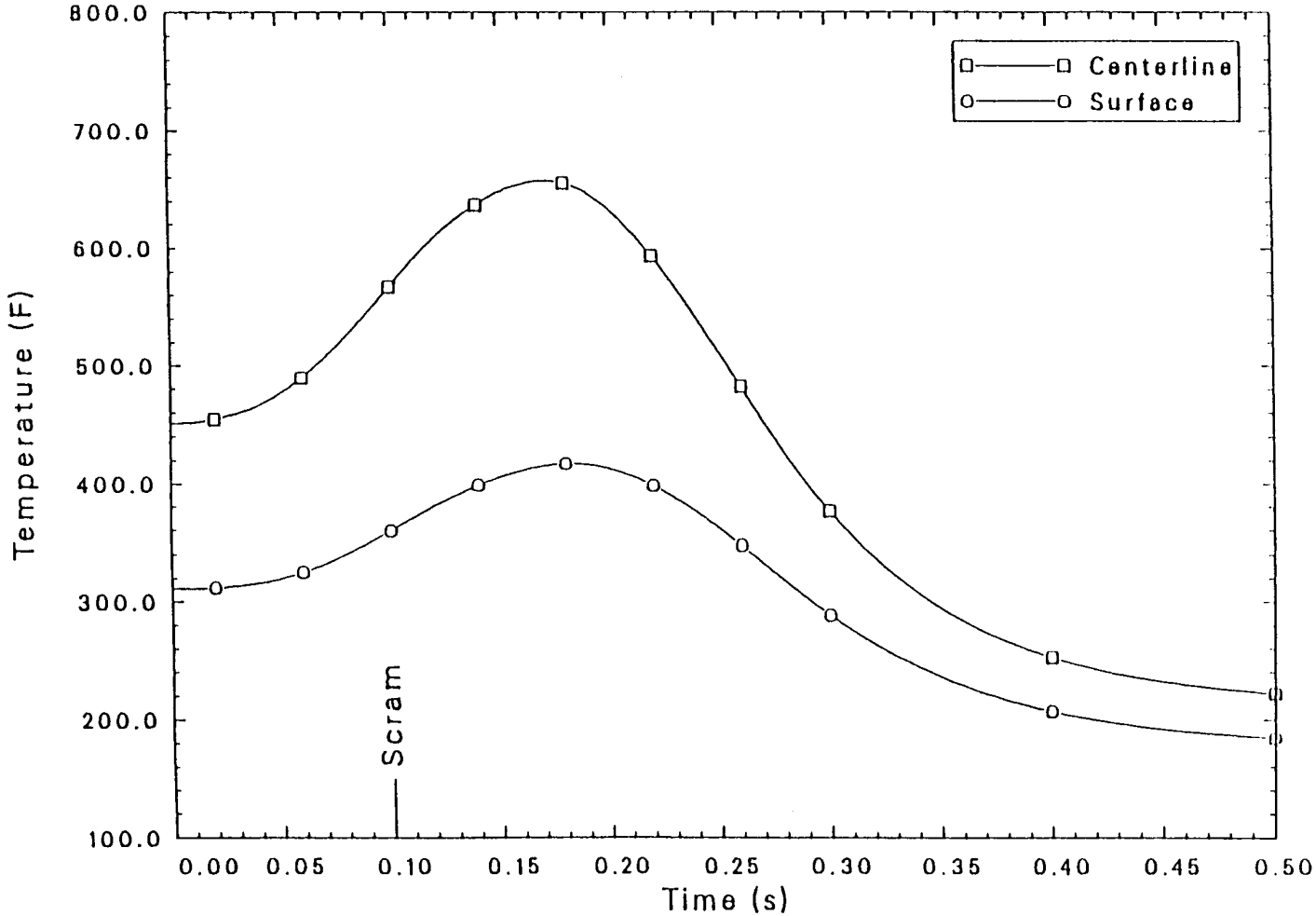

Figure 15.4-6. Maximum fuel plate temperature (SIPT 1/2-in. break RIA – three-pump operation). **(For Information Only)**

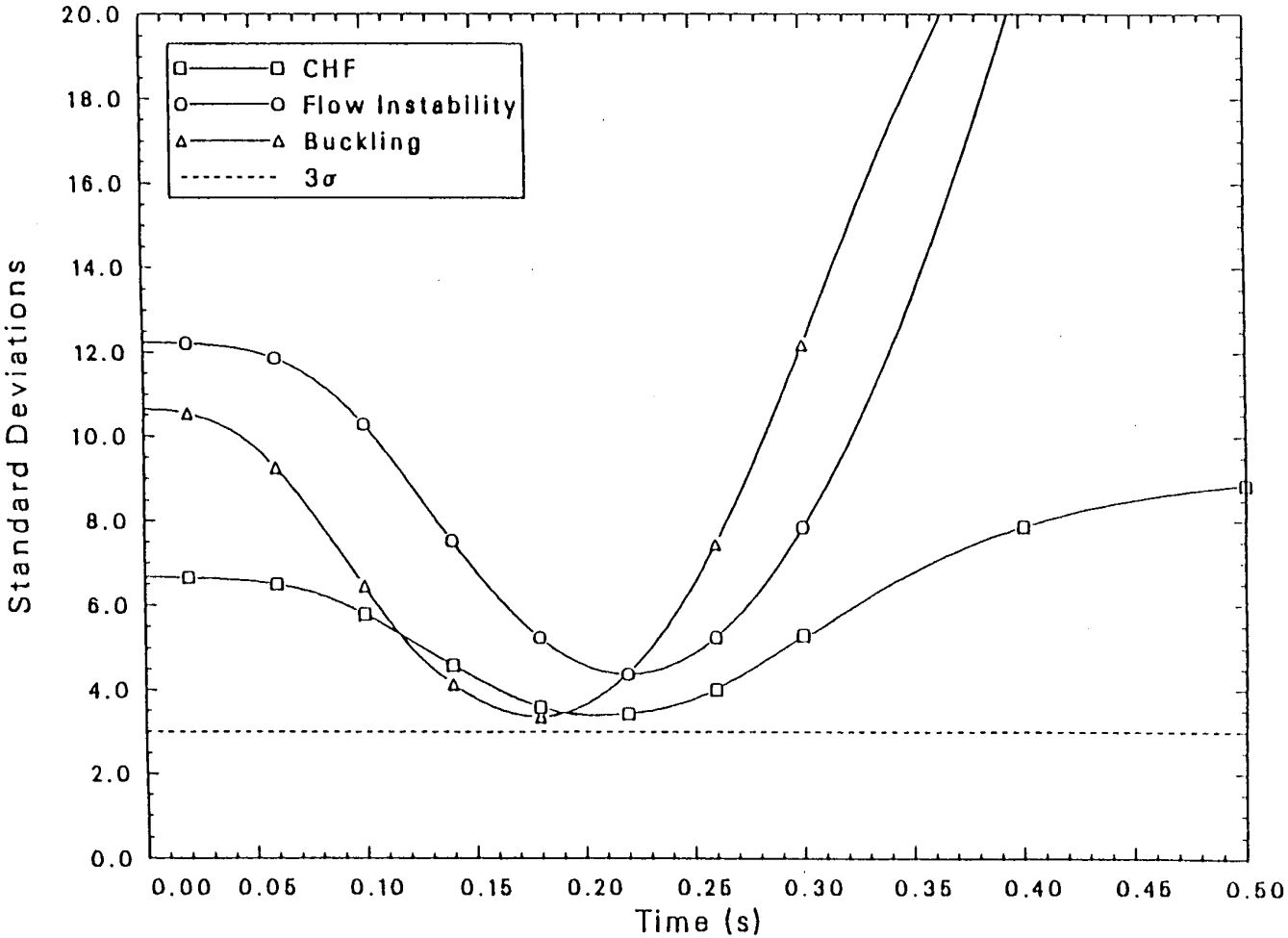

Figure 15.4-7. CHF, FI, and fuel plate buckling margins (SIPT 1/2-in. break - two pump operation). **(For Information Only)**

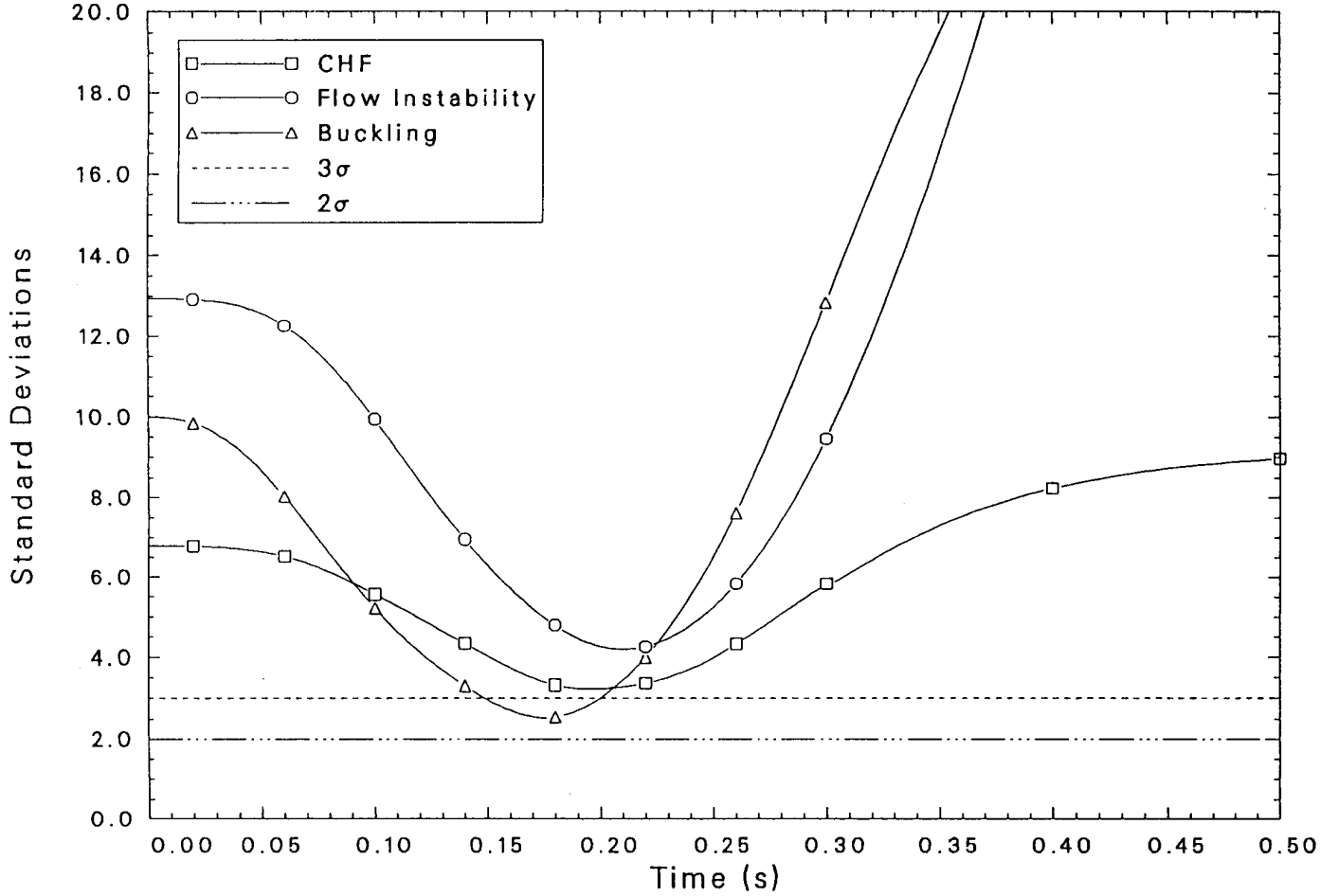

Figure 15.4-8. CHF, FI, and fuel plate buckling margins (SIPT 1/2-in. break RIA, - three-pump operation). **(For Information Only)**

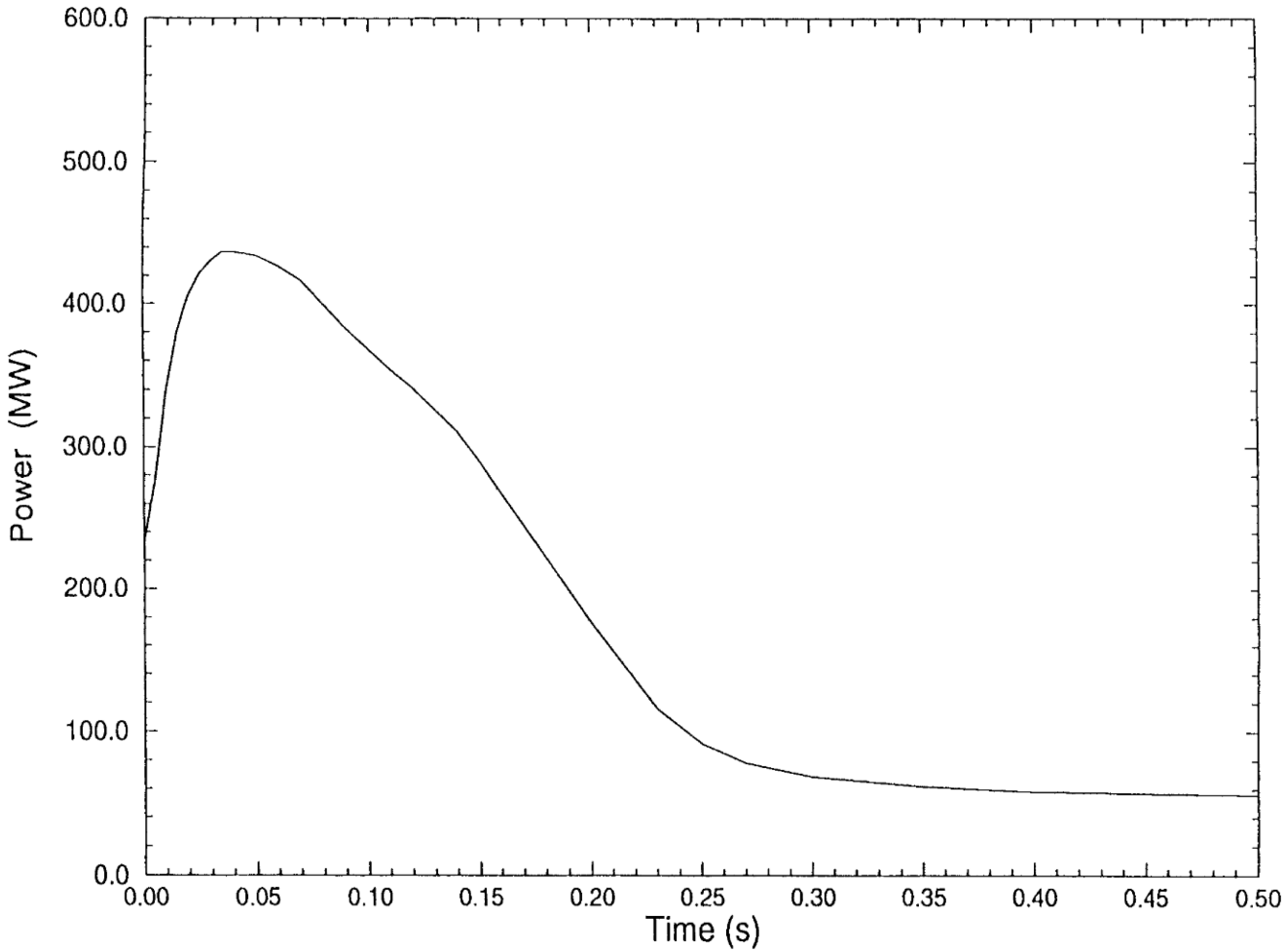

Figure 15.4-9. Reactor power (0.50$ step RIA - two pump operation). **(For Information Only)**

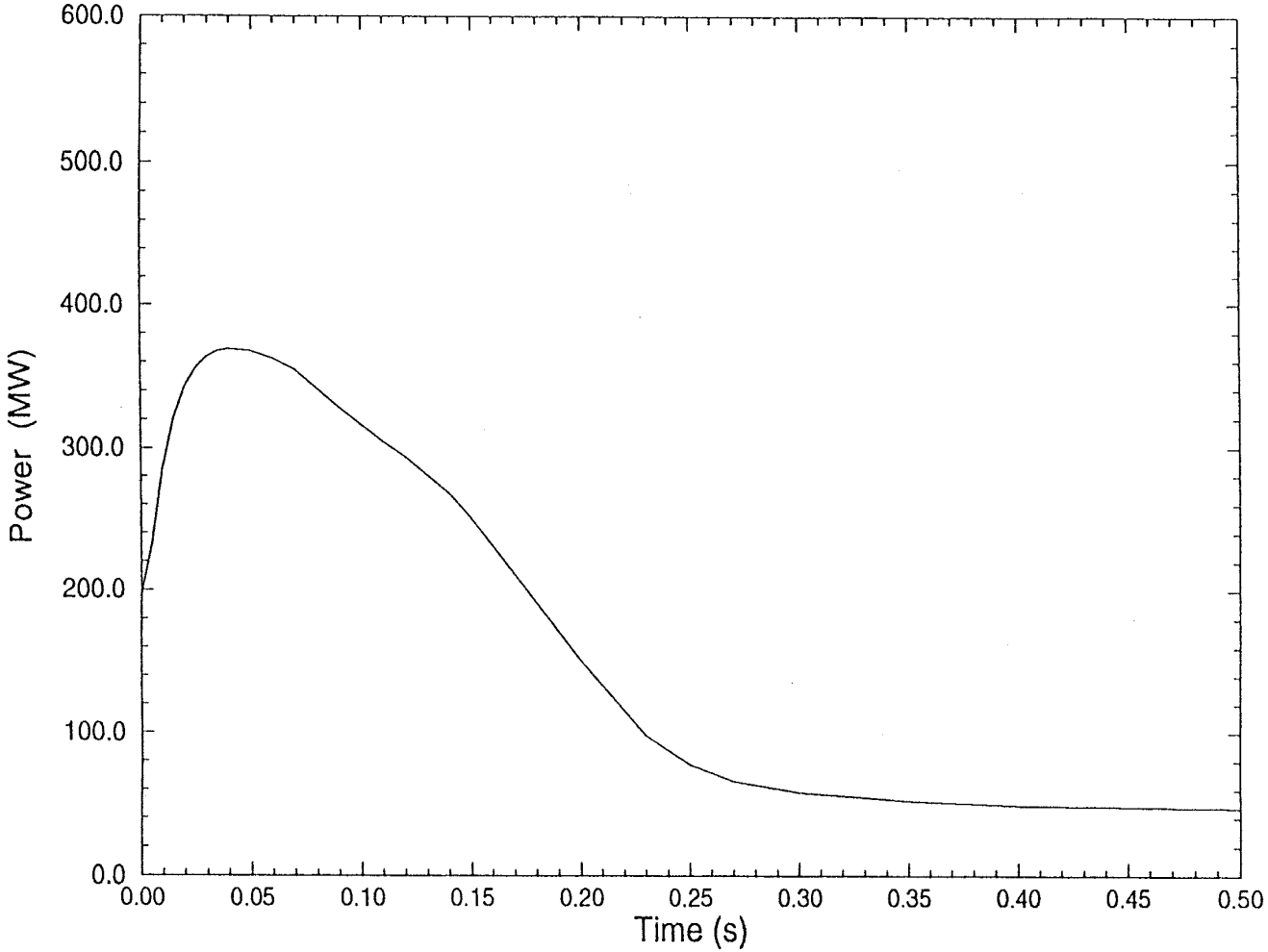

Figure 15.4-10. Reactor power (0.50$ step RIA - three pump operation). **(For Information Only)**

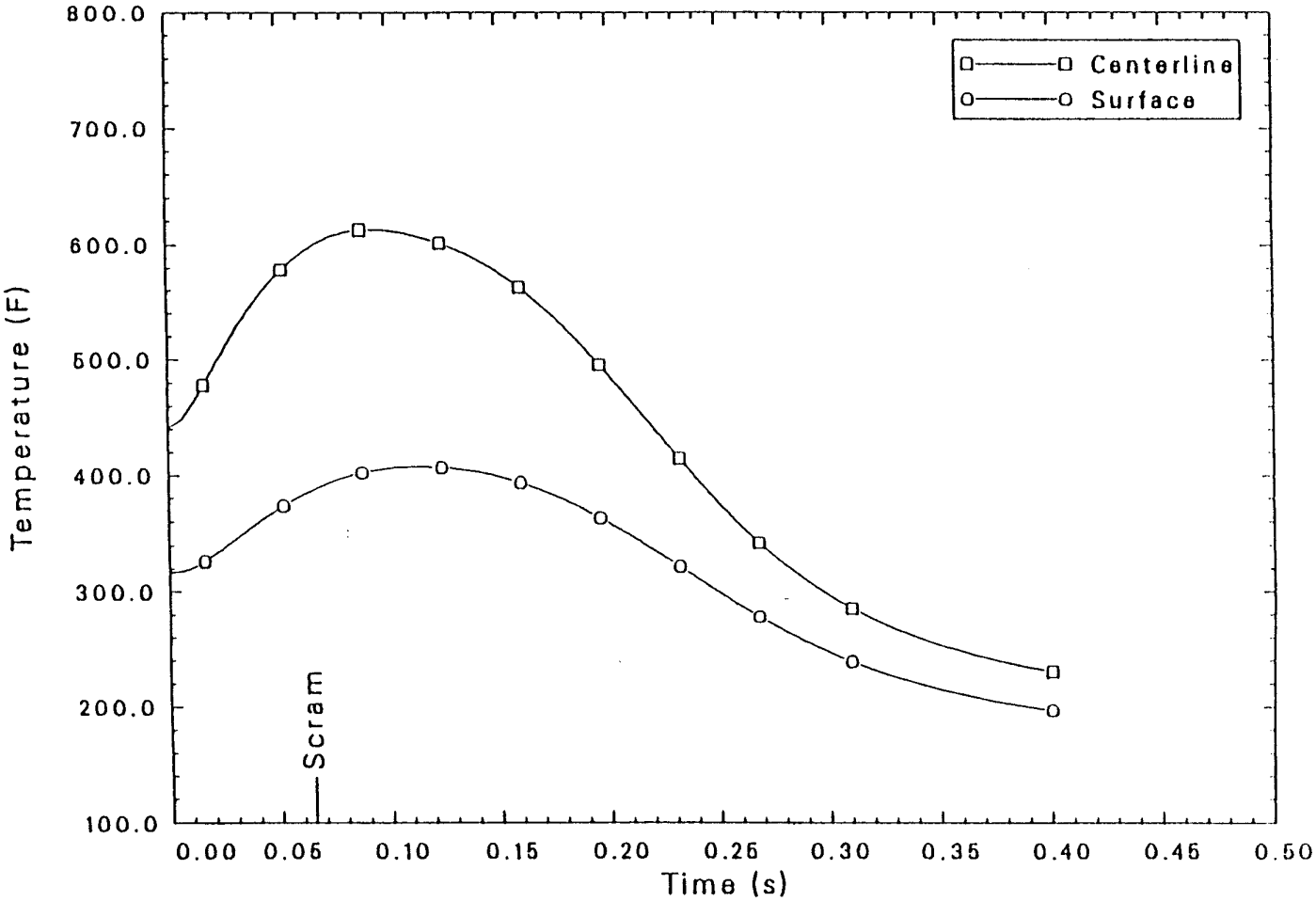

Figure 15.4-11. Maximum fuel plate temperatures (0.50$ step RIA - two pump operation). **(For Information Only)**

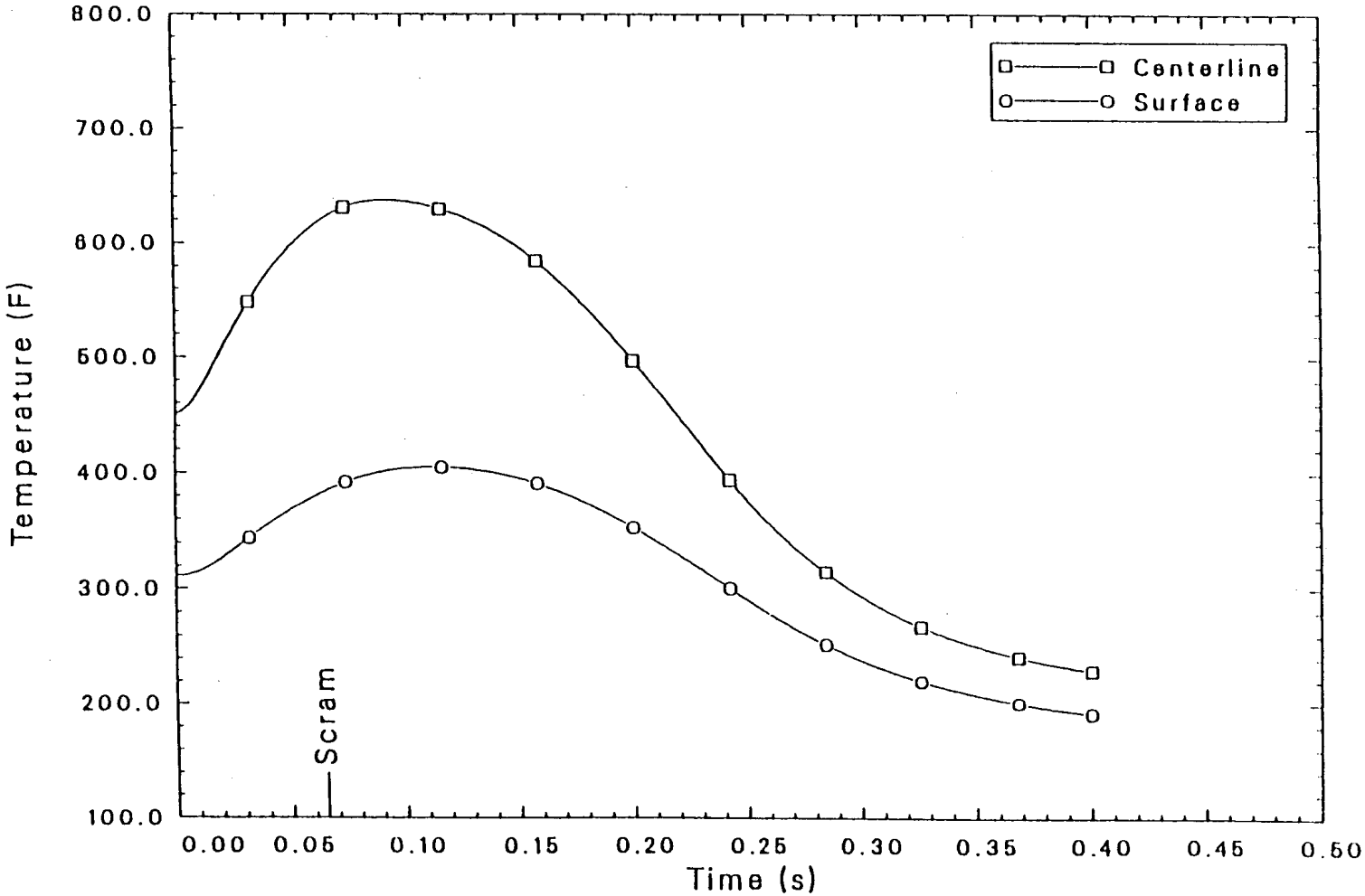

Figure 15.4-12. Maximum fuel plate temperature (0.50$ step RIA - three pump operation). **(For Information Only)**

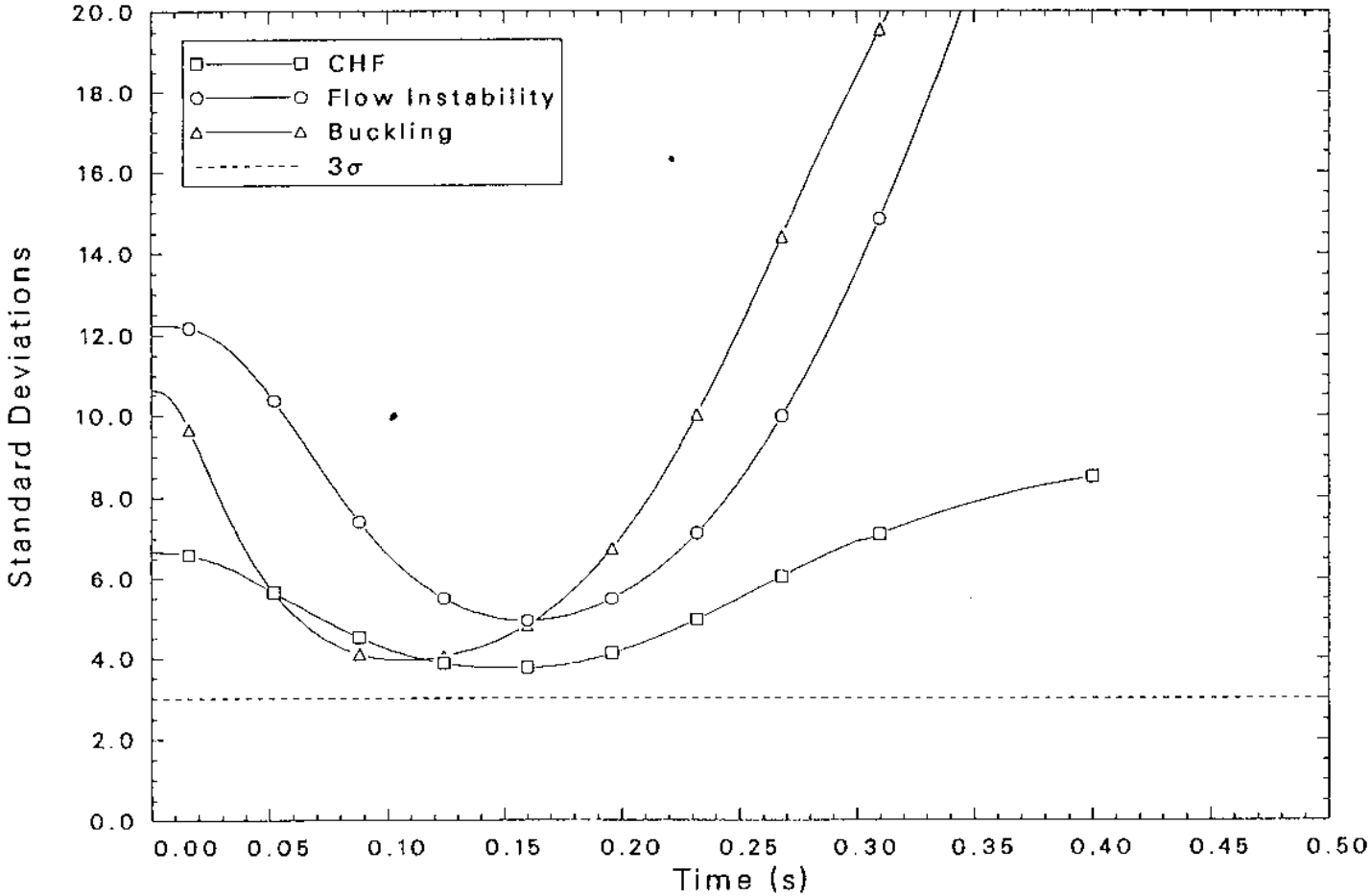

Figure 15.4-13. CHF, FI, and fuel plate buckling margins (0.50$ step RIA – two-pump operation). **(For Information Only)**

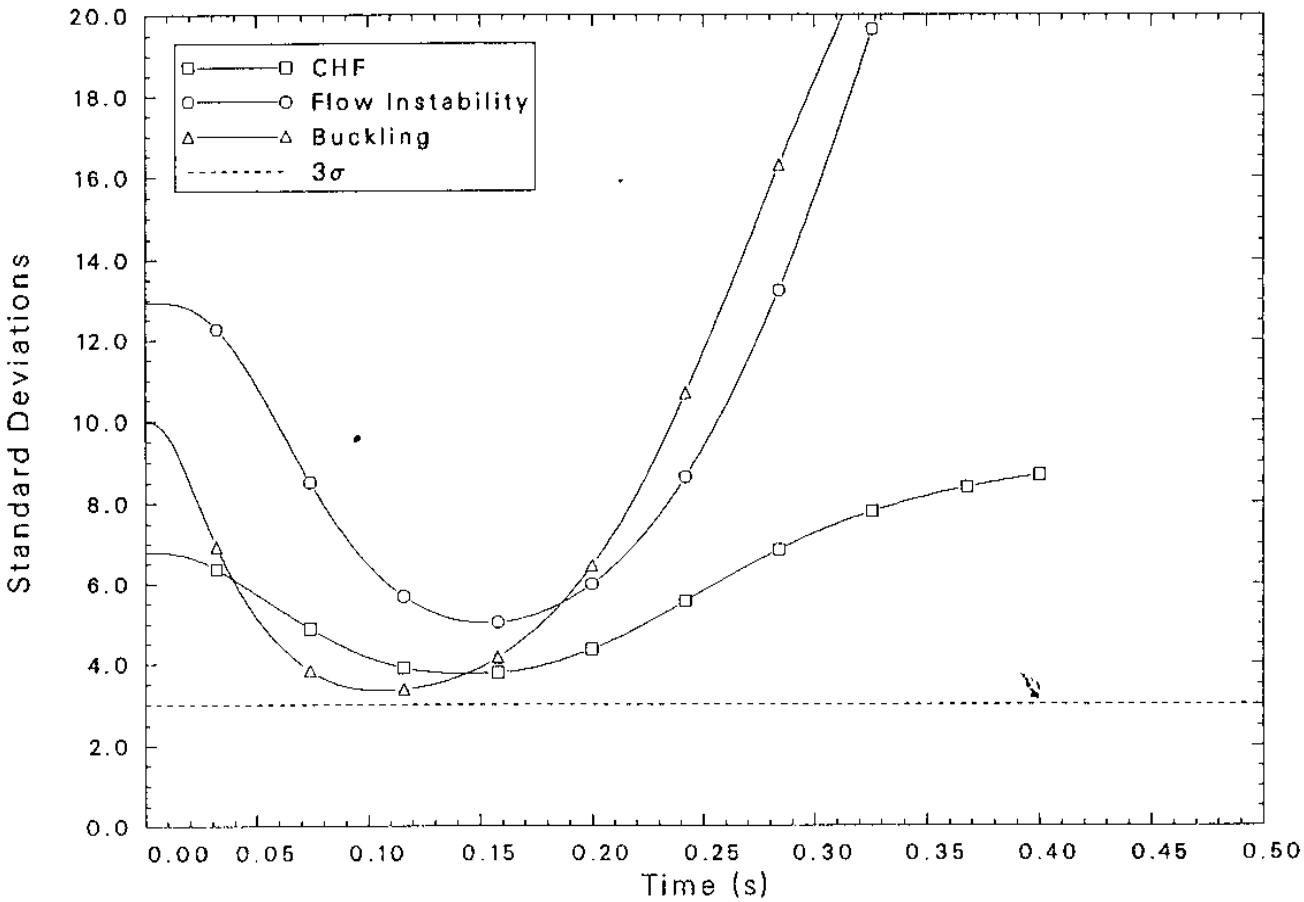

Figure 15.4-14. CHF, FI, and fuel plate buckling margins (0.50$ step RIA – three-pump operation). **(For Information Only)**

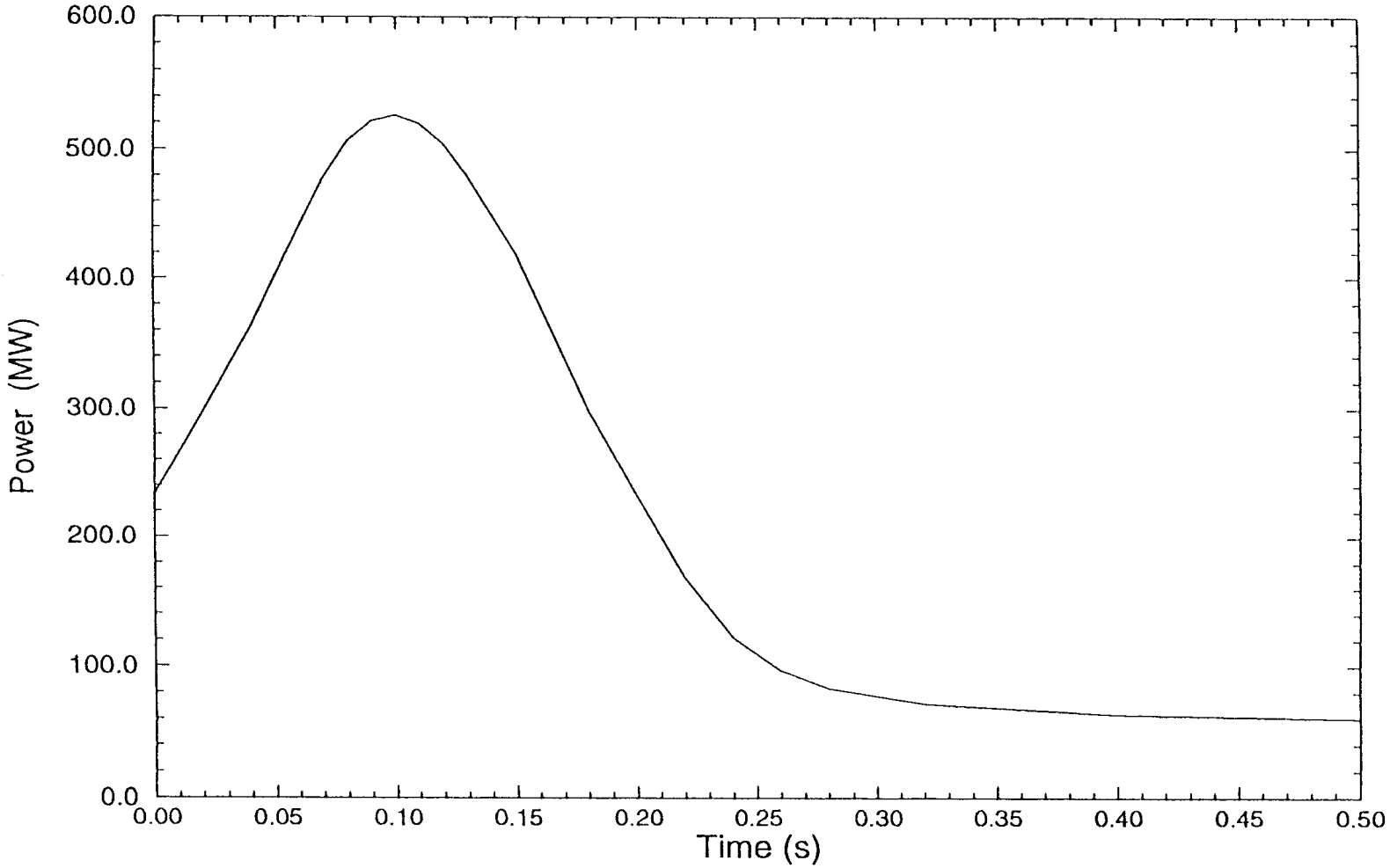

Figure 15.4-15. Reactor power (SIPT pump discharge pipe break RIA - two-pump operation). **(For Information Only)**

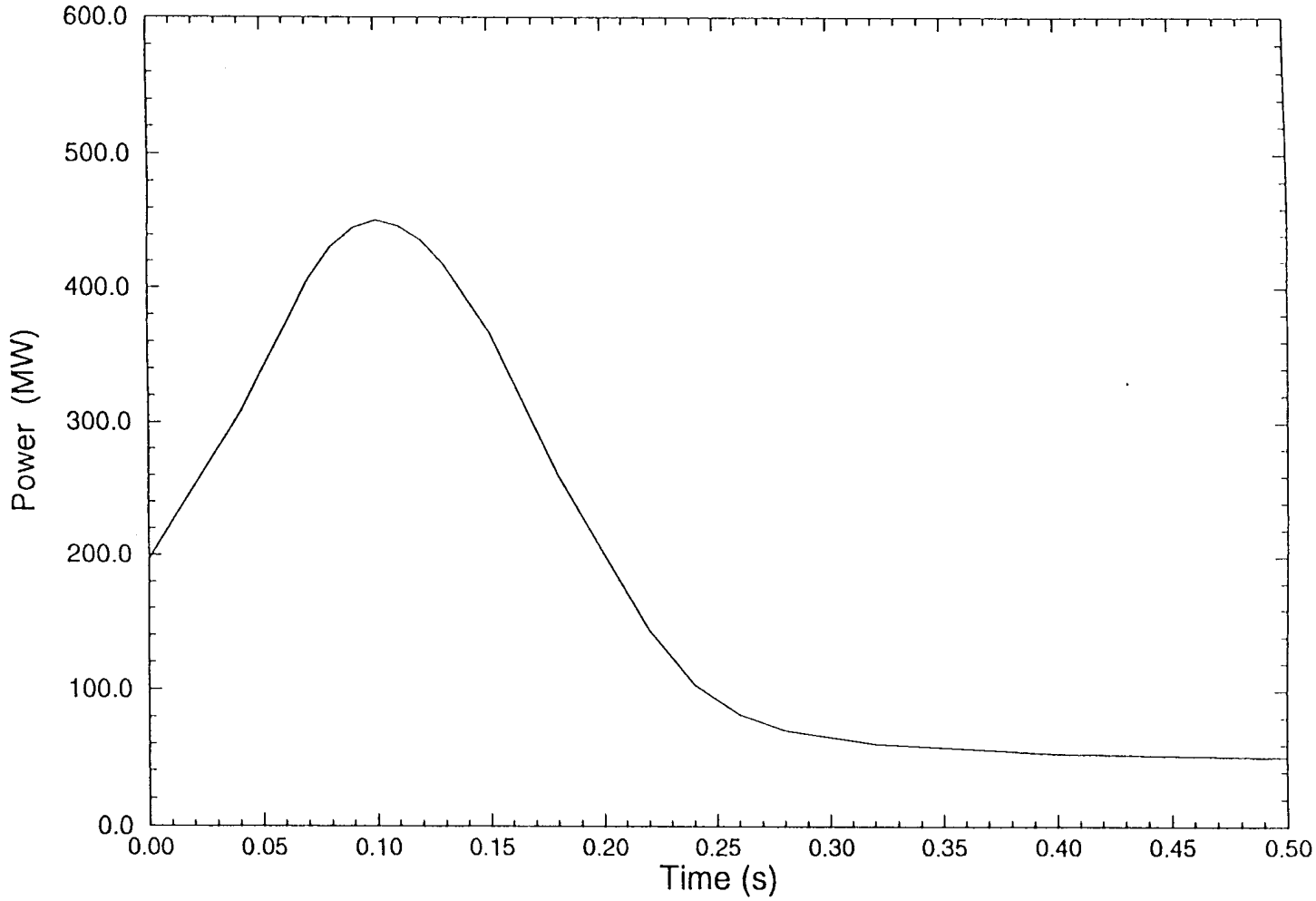

Figure 15.4-16. Reactor power (SIPT pump discharge pipe break RIA - three-pump operation). **(For Information Only)**

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