ATTACHMENT

TO

1CAN039604

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT ONE

DOCKET NO. 50-313

April 23, 1996

1CAN049606

U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Station P1-137
Washington, DC 20555

Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Proposed Technical Specification Change Revising The Pressurizer Code Safety As-Found Setpoint Tolerance

Gentlemen:

Attached for your review and approval are proposed Technical Specification (TS) changes to allow Arkansas Nuclear One - Unit 1 (ANO-1) to revise the as-found setpoint tolerance for the pressurizer code safeties described by the Bases associated with Specifications 2.2 and 3.1.1 from +1/-3% to ±3%. The changes also increase the relief flowrate of the pressurizer code safeties described in the Bases associated with Specification 3.1.1 from 300,000 lb/hr to 324,000 lb/hr, reword the Bases associated with Specification 3.1.7 to describe the actual value of moderator temperature coefficient used as an input to the startup accident analysis, and revise the values for minimum and maximum pressurizer water level specified by Specification 3.1.3.4 to refer to a figure that will be incorporated in this change. These changes are supported by revised startup accident and rod withdrawal accident analyses. Proposed changes to the ANO-1 Safety Analysis Report incorporating the new analysis results have also been included for your use in reviewing this change request.

The new startup and rod withdrawal accident analyses were performed using the RELAP5/MOD2-B&W computer code to justify an increase in pressurizer code safety valve as-found tolerance to +3%. The analyses verified, using conservative assumptions, that a +3% tolerance is acceptable for two pressurizer code safety valves. The analyses also showed that a maximum pressurizer water level of 259 inches below 15% Rated Power and a maximum level of 320 inches when at or above 15% Rated Power produces acceptable results.

Currently, when a pressurizer code safety valve setpoint is found to be outside of the +1/-3% setpoint tolerance, the other pressurizer code safety valve must be tested and the occurrence must be tracked under the ANO 10CFR50 Appendix B corrective action program. With this

change, those occurrences when a pressurizer code safety valve setpoint is found outside of a +1% setpoint tolerance, but within the proposed +3% setpoint tolerance, would not require testing of the other pressurizer code safety valve and would not require tracking of the corrective action. The change still requires any valve setpoint found to be outside of a ±1% tolerance be returned to within the ±1% as-left tolerance as currently described in the Bases associated with Specification 2.2.

Entergy operations currently utilizes the 1980 Edition of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code at ANO-1. Subsection IWV-3512 of this Edition of the Code endorses ASME Performance Test Code (PTC) 25.3-1976 for the testing of safety and relief valves. The pressurizer code safety valves are currently tested in accordance with this standard.

In accordance with 10CFR50.55a(f)(4)(iv), Entergy Operations requests approval to use the 1989 Edition of Section XI of the ASME Code to test the ANO-1 pressurizer code safety valves beginning with testing to be conducted during our next refueling outage which is currently scheduled to commence on September 17, 1996. This Edition, which has been incorporated by reference in 10CFR50.55a(b)(2), endorses ASME/American National Standards Institute (ANSI) Operations and Maintenance (OM) Code, Part 10 [OMa-1988 Addenda to the OM-1987 Edition per 10CFR50.55a(b)(2)(viii)]. This Edition of OM Part 10 endorses OM Part 1 (1987), and allows a ±3% tolerance for as-found testing of safety valves. In adopting the 1989 ASME Code for pressurizer code safety valve testing, Entergy Operations commits to adopt all the related requirements of OM Part 1. The ANO-1 safety analysis was reviewed and determined to be unaffected by this change in testing requirements.

The proposed TS change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

Entergy Operations requests that the effective date for this TS change be within 30 days of approval. Although this request is neither exigent nor emergency, your prompt review is requested prior to our next refueling outage.

Very truly yours,

JWY/cws
Attachments

To the best of my knowledge and belief, the statements contained in this submittal are true.

SUBSCRIBED AND SWORN TO before me, a Notary Public in and for __________ County and the State of Mississippi, this _____ day of _____________, 1996.

____

Notary Public
My Commission Expires _____________

cc: Mr. Leonard J. Callan
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector
Arkansas Nuclear One
P.O. Box 310
London, AR 72847

Mr. George Kalman
NRR Project Manager Region IV/ANO-1 & 2
U. S. Nuclear Regulatory Commission
NRR Mail Stop 13-H-3
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Mr. Bernard Bevill
Acting Director, Division of Radiation
Control and Emergency Management
Arkansas Department of Health
4815 West Markham Street
Little Rock, AR 72205

ATTACHMENT

TO

1CAN039604

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT ONE

DOCKET NO. 50-313

DESCRIPTION OF PROPOSED CHANGES

The proposed changes to the Arkansas Nuclear One - Unit 1 (ANO-1) Technical Specifications (TSs) are as follows:

- The Bases associated with Specification 2.2 were revised to reflect a new pressurizer code safety setpoint as-found tolerance of ±3%.
- The Bases associated with Specification 3.1.1 were revised to reflect a new pressurizer code safety setpoint as-found tolerance of ±3% and pressurizer code safety valve relief flowrate of 324,000 lb/hr.
- The pressurizer water level requirements of Specification 3.1.3.4 have been revised to refer to Figure 3.1.3-1, Pressurizer Level Acceptable Range.
- A new page has been inserted to allow incorporation of a new figure. Figure 3.1.3-1 shows the acceptable ranges for pressurizer water level as a function of reactor power, as required by the revised Specification 3.1.3.4.
- The Bases associated with Specification 3.1.7 were revised to indicate the actual value of moderator temperature coefficient used as an input in the startup accident analysis instead of the currently specified range of values reference.

BACKGROUND

The reactor coolant system (RCS) serves as a barrier which prevents the release of radionuclides contained in the reactor coolant to the reactor building atmosphere. A pressure safety limit of 2750 psig (110% of design pressure) has been established and is specified by TS 2.2.1. The RCS is protected against overpressure by two pressurizer code safety valves mounted on top of the pressurizer. The ANO-1 pressurizer code safety valves are Dresser model 31759A Δp-to-open, spring-to-close pressure relief valves. The required capacity of these valves is determined from considerations of: (1) the reactor protection system, (2) pressure drop (static and dynamic) between the point of highest pressure in the RCS and the pressurizer, and (3) accident or transient overpressure conditions. The pressurizer code safety valves are described in ANO-1 Safety Analysis Report (SAR) Section 4.2.4.1.

TS Table 4.1-2 requires testing of one pressurizer code safety valve setpoint every 18 months. Currently, the Bases associated with TS 2.2 state that the as-found lift setpoint may be 2500 psig +1/-3%. If the setpoint is found to be outside of a ±1% tolerance band, it must be reset to 2500 psig ±1%. If the setpoint is found to be outside of the +1/-3% tolerance band, the remaining pressurizer code safety valve setpoint must be tested in accordance with Section III of the ASME Code (PTC 25.3). ASME/ANSI OM Part 1 (1987) allows a ±3% tolerance band for the as-found testing of code safety valves.

Testing results since 1M89 are summarized in Figure #1 attached to this submittal) for all three ANO-1 pressurizer code safety valves (two valves are in service and one is a spare).

As shown in Figure #1, two of the twelve tests conducted since 1M89 were not within the proposed setpoint tolerance of +3/-3%. The high setpoints both occurred during 1R11 and were attributed to the practice of "jack and lap" after setpoint testing. This process allowed the valve to be partially disassembled, leaving the spring intact, in order to lap the seats to eliminate post testing leakage. After the process was completed, the valve was re-assembled without further setpoint testing. Based on recent information, this repair method can not be used on the Dresser valves without re-verifying the setpoint because the valves utilize four spiral wound gaskets between the body to bonnet interface. Since the gaskets may not compress to the same degree after re-assembly, the spring compression could change thus affecting the setpoint. After both valves lifted out of tolerance during 1R11, the spare valve which had been in storage since 1R10 was tested, and also lifted out-of-tolerance. Since this valve had also been "jack and lapped" without re-verifying the setpoint in 1R10, the practice of "jack and lap" without subsequent setpoint verification was determined to be questionable. Now, if a valve is "jack and lapped," its setpoint must be re-verified.

During 1R12, PSV-1001 lifted 1.5% above setpoint. Because the valve lifted out of the current setpoint tolerance of +1/-3%, PSV-1002 also had to be tested to meet code requirements. PSV-1002 was found 0.64% above it's setpoint. Expanding the setpoint tolerance range to ±3% would reduce the likelihood of a valve being found out of tolerance. This in turn would reduce the probability of subsequent valve testing during each outage.

DISCUSSION OF CHANGE

The two limiting accidents identified in the TS 3.1.3.4 Bases with respect to pressurizer code safety valve response are the startup accident (SAR Section 14.1.2.2) and the rod withdrawal accident (SAR Section 14.1.2.3). Analyses have been performed to demonstrate the acceptability of a +3% pressurizer code safety valve setpoint tolerance in the event of a startup accident or a rod withdrawal accident. The methodology for analyzing these accidents is identical to that employed in the ANO-1 Safety Analysis Report using an improved computer code - RELAP5/MOD2-B&W. The acceptance criteria for these analyses are: (1) Peak RCS pressure must remain below the safety limit of 2750 psig, and (2) peak reactor thermal power must remain below 112% Rated Power. All computer analyses were performed using the RELAP5/MOD2-B&W computer code. The RELAP5\MOD2 code has been previously submitted to the NRC for review in B&W Topical Report BAW-10193P, "RELAP5/MOD2-B&W For Safety Analysis of B&W-Designed Pressurized Water Reactors," dated August 14, 1995. A comparison of the RELAP5/MOD2 - B&W prediction of the startup accident with that of CADDS, an approved code for analyzing this event for B&W-designed pressurized water reactors, was provided in BAW-10193P.

The analysis demonstrates that a startup accident from hot zero power with a pressurizer code safety valve setpoint tolerance of 3% above the pressurizer code safety valve setpoint of 2500 psig will not result in a peak RCS pressure greater than 2750 psig or a reactor thermal power greater than 112% Rated Power. This analysis included additional sensitivity studies that demonstrated acceptable results in the event of a startup accident assuming one pressurizer code safety valve lifted at 5% above the pressurizer code safety valve setpoint of 2500 psig

and the other pressurizer code safety valve failing to actuate to relieve RCS pressure. A bounding value for moderator temperature coefficient, $+0.9 \times 10^{-4}\ \Delta K/K/°F$, was assumed in the analysis in place of the range of coefficients referred to in the Bases associated with TS 3.1.7.

The Bases associated with Specification 3.1.3.4 indicate that the specified pressurizer levels assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a startup accident and that the water level is above the minimum detectable level. The Bases do not, however, specifically state which pressurizer levels are analytically justified for any specific power levels. In other words, the Bases do not indicate what initial pressurizer level was assumed in either the startup accident or control rod withdrawal accident. The original analyses did, however, employ conservative methods and setpoints while utilizing nominal values for the operational parameters.

It was recognized that a more appropriate requirement for pressurizer level was necessary to accommodate the thermal expansion associated with the reactivity addition and the conservative assumptions used in the startup and rod withdrawal event analyses. The operational range for pressurizer level is approximately 140 inches at 0% Rated Power, and approximately 220 inches at 100% Rated Power. The startup accident design analysis, using conservative input assumptions, justified a maximum pressurizer level of 259 inches. Since postulated rod withdrawal events at higher power levels are considered to have less severe consequences due to the effects of the assumed power level on the input assumptions, this limit was considered unnecessarily restrictive for operation above 15% Rated Power.

A control rod withdrawal analysis was performed at a power level of 15% Rated Power to support the proposed setpoint tolerance change. This analysis is considered to be bounding from 15% Rated Power to 100% Rated Power due to the ramping of the moderator temperature coefficient from a value of $+0.9 \times 10^{-4}\ \Delta k/k/°F$ at 0% Rated Power to a value of $+0.0 \times 10^{-4}\ \Delta k/k/°F$ at 95% Rated Power. The analysis provided acceptable results, assuming an initial pressurizer level of 320 inches when the unit is above 15% Rated Power.

Proposed changes to the ANO-1 SAR incorporating the new analysis results have been included for your use in reviewing the proposed TS changes. Based on these analyses, ANO-1 proposes to revise the as-found pressurizer code safety valve setpoint tolerance to ±3%. If found outside of a ±1% tolerance band, the pressurizer code safety valve setpoint will continue to be reset to 2500 psig ±1%, as required by Section III of the ASME Code and as described in the Bases associated with Specification 2.2. The Bases associated with Specification 3.1.1 have been revised to reflect the change in as-found tolerance, and to reflect the pressurizer code safety valve relief flowrate of 324,000 lb/hr used in the reanalysis of the startup and rod withdrawal accidents. The pressurizer code safety valve relief flowrate was revised from 300,000 lbm/hr to 324,000 lbm/hr to reflect the actual relief capacity of the pressurizer code safety relief valve and to remove excess conservatism from the analyses. The Bases associated with Specification 3.1.7 have been revised to describe the value of moderator temperature coefficient used in the startup accident analysis as a bounding value.

A new figure, Figure 3.1.3-1, has been added on inserted page 21a specifying the acceptable range for pressurizer level as a function of reactor power. The minimum pressurizer level for all power levels remains at the currently specified 45 inches. From 0% to 15% Rated Power, the pressurizer maximum water level is 259 inches. From 15% to 100% Rated Power, the pressurizer maximum water level is 320 inches.

Figure 3.1.3-1 also contains a note to clarify that the specified pressurizer levels and reactor power levels do not contain an allowance for instrument error. The previous pressurizer level requirements were specified as "indicated" levels. No reference was made in the associated Bases to indicate whether instrument error was included in these limits. Since the values for pressurizer level and reactor power used as inputs to the startup and rod withdrawal analyses were not corrected by the inclusion of instrument error, this note indicates that values used for controls in the plant procedures should be corrected for the instrument error allowance.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of these standards as they relate to this amendment request follows:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The startup accident and the rod withdrawal accident have been reanalyzed to justify the proposed increase in pressurizer code safety valve as-found tolerance. The analyses establish more appropriate boundaries and re-analyze the same initiators as are currently found in the ANO-1 Safety Analysis Report. Changing the as-found setpoint tolerance does not change how the pressurizer code safety valve operates as it will continue to be reset to 2500 psig ±1% prior to reactor startup.

The acceptance criteria for these analyses are that the reactor coolant system (RCS) pressure shall not exceed the safety limit of 2750 psig (110% of design pressure) and that the reactor thermal power remains below 112% Rated Power. The analyses using the proposed setpoint tolerance have shown that the acceptance criteria were met and that the consequences of the events were essentially the same as those in the ANO-1 SAR. Analyses were performed to determine the pressurizer maximum water level that would prevent the RCS from exceeding the safety limit of 2750 psig in the event of either a startup accident or a rod withdrawal accident. More appropriate pressurizer level requirements have been incorporated in accordance with these analyses.

Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed changes introduce no new mode of plant operation. The reanalysis of the startup accident and the rod withdrawal accident were performed using methodologies identical to that employed in the ANO-1 SAR and an improved computer code (RELAP5/MOD2). The pressurizer code safety valve setpoint will continue to be reset at 2500 psig ±1% prior to reactor startup and will continue to function to maintain RCS pressure below the safety limit of 2750 psig. Analyses were performed to determine the pressurizer maximum water level that would prevent the RCS from exceeding the safety limit of 2750 psig in the event of either a startup accident or a rod withdrawal accident. More appropriate pressurizer level requirements have been incorporated in accordance with these analyses.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

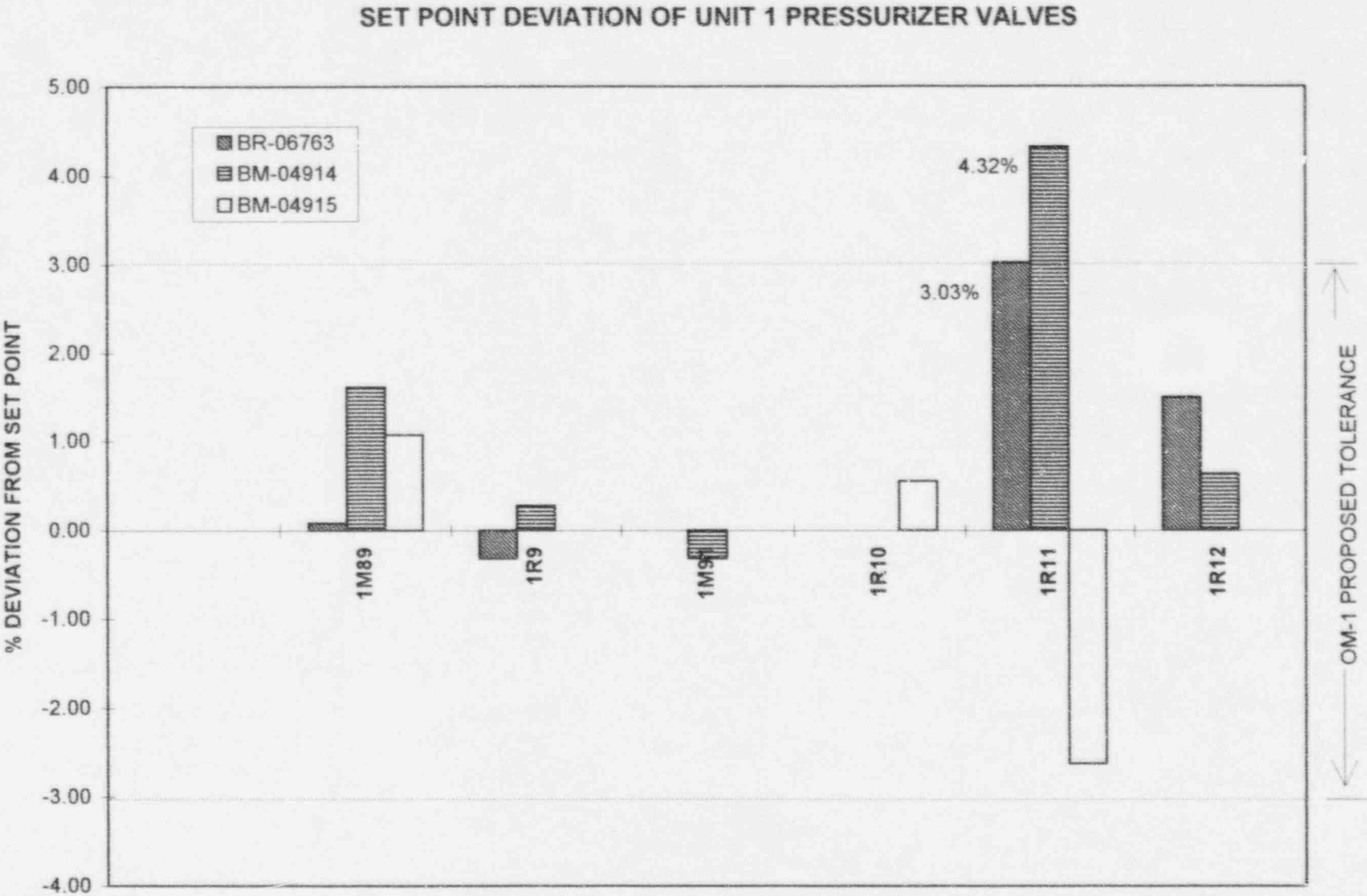
Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

The safety function of the pressurizer code safety valves is not altered as a result of the proposed change in setpoint tolerance. The reanalysis of the startup accident and rod withdrawal accident have shown that with a ±3% setpoint tolerance, the pressurizer code safety valves will function to limit RCS pressure below the safety limit of 2750 psig. The sensitivity studies for the startup accident showed the acceptance criteria would still be met even if one pressurizer code safety valve lifted at 5% above 2500 psig at startup conditions. Additional analyses were performed to determine the pressurizer maximum water level that would prevent the RCS from exceeding the safety limit of 2750 psig in the event of either a startup accident or a rod withdrawal accident.

Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve a significant hazards consideration.

Figure 1 of
1CAN049606



PROPOSED TECHNICAL SPECIFICATION CHANGES

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

2.2.2 The setpoint of the pressurizer code safety valves shall be in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1968.

Bases

The reactor coolant system (1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110 percent of design pressure.(2) The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110 percent of design pressure. Thus, the safety limit of 2750 psig (110 percent of the 2500 psig design pressure) has been established. (2) The settings for the reactor high pressure trip (2355 psig) and the pressurizer code safety valves (2500 psig ±1%)(3) have been established to assure that the reactor coolant system pressure safety limit is not exceeded. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig ±3%. However, if found outside of a ±1% tolerance band, they shall be reset to 2500 psig ±1%. The initial hydrostatic test is conducted at 3125 psig (125 percent of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig.(4)

REFERENCES

(1) FSAR, Section 4

(2) FSAR, Section 4.3.11.1

(3) FSAR, Section 4.2.4

(4) FSAR, Table 4-1

BASES:

The plant is designed to operate with both reactor coolant loops and at least one reactor coolant pump per loop in operation, and maintain DNBR above 1.30 (for the BAW-2 correlation) and 1.18 (for the BWC correlation) during all normal operations and anticipated transients.(1)

Whenever the reactor coolant average temperature is above 280°F, single failure considerations require that two loops be operable.

The decay heat removal system suction piping is designed for 300°F thus, the system can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident. (5) The pressurizer code safety valve lift setpoint shall be 2,500 psig ±1 percent allowance for error and each valve shall be capable of relieving 324,000 lb/h of saturated steam at a pressure not greater than 3 percent above the set pressure. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig ±3 percent. However, if found outside the ±1 percent tolerance band, they shall be reset to 2500 psig ±1 percent.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internal vent valves (1) ensure operability, (2) ensure that the valves are not open during normal operation, and (3) demonstrate that the valves begin to open and are fully open at the forces equivalent to the differential pressures assumed in the safety analysis.

The reactor coolant vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The operability of at least one reactor coolant system vent path from the reactor vessel head, the reactor coolant system highpoints, and the pressurizer steam space ensures the capability exists to perform this function. The valve redundancy of the vent paths serves to minimize the probability of inadvertent actuation and breach of reactor coolant pressure boundary while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path. Testing requirements are covered in Section 4.0 for the class 2 valves and Table 4.1-2 for the vent paths. These are consistent with ASME Section XI and Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," 11/80.

REFERENCES

(1) FSAR, Tables 9-10 and 4-3 through 4-7
(2) FSAR, Section 4.2.5.1 and 9.5.2.3
(3) FSAR, Section 4.2.5.4
(4) FSAR, Section 4.3.10.4 and 4.2.4
(5) FSAR, Section 4.3.7

3.1.3 Minimum Conditions for Criticality

Specification

3.1.3.1 The reactor coolant temperature shall be above 525F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply.

3.1.3.2 Reactor coolant temperature shall be to the right of the criticality limit of Figure 3.1.2-2.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1 percent Δk/k until a steam bubble is formed and a pressurizer water level within the limits of Figure 3.1.3-1 is established.

3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality.

3.1.3.6 The reactor shall not be made critical until at least 2 of the 3 emergency-powered pressurizer heater groups are operable. With less than 2 of the 3 required heater groups operable, restore the required heater groups to operable status within 72 hours. If the required heater groups are not restored to operable status within 72 hours, be in hot shutdown within the following 12 hours.

3.1.3.7 With any of the above limits violated, restore the reactor to within the limit in 15 minutes or be in at least Hot Shutdown within the next 15 minutes.

Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent Δk/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated Δk/k would limit the magnitude of power excursion resulting from a reduction of moderator density.

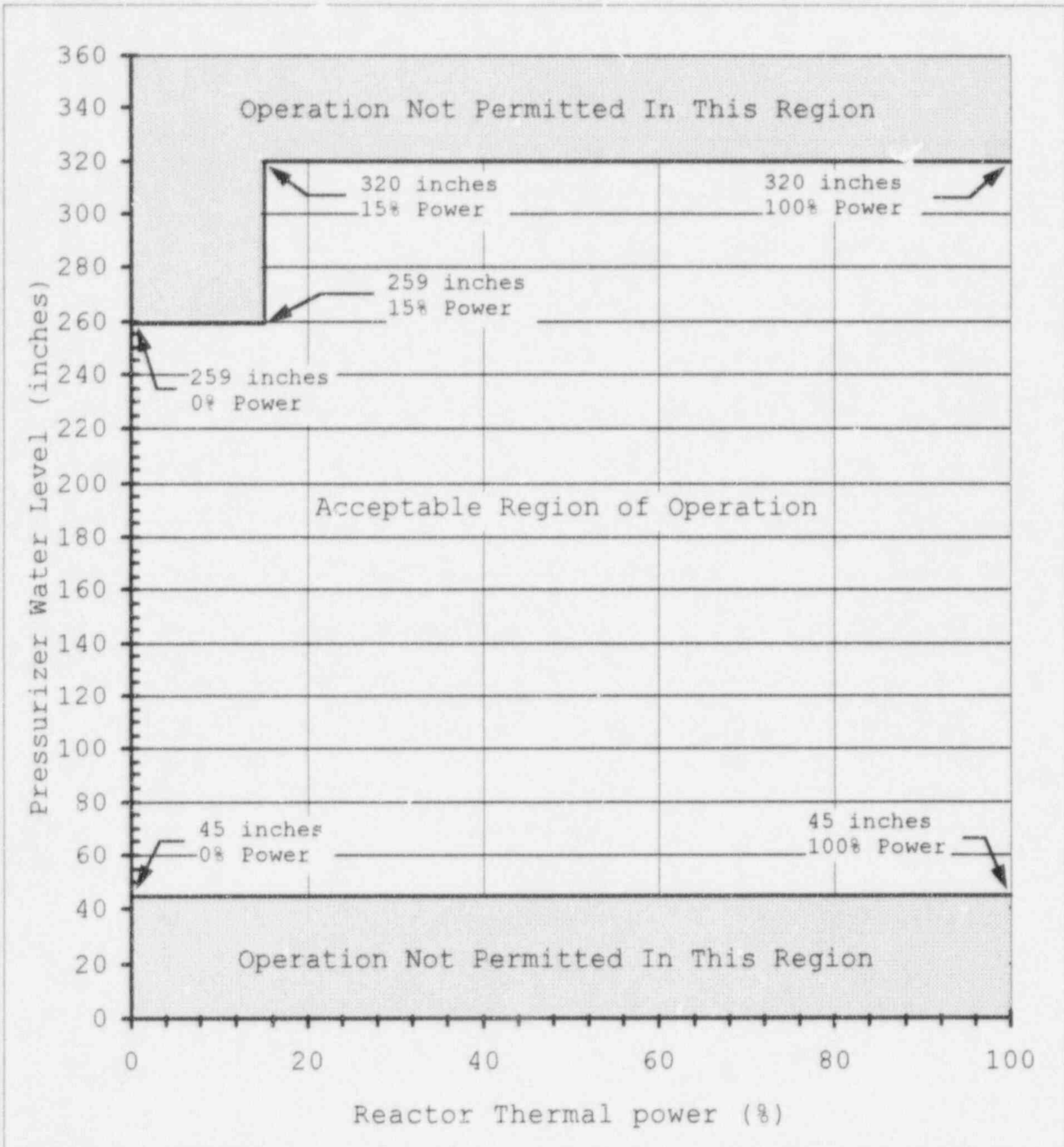


Figure 3.1.3-1

ANO-1 Pressurizer Level Acceptable Region of Operation

NOTE: The values specified for pressurizer level and reactor power do not contain an allowance for instrument error.

3.1.7 Moderator Temperature Coefficient of Reactivity
Specification

3.1.7.1 The moderator temperature coefficient (MTC) shall be non-positive whenever thermal power is ≥95% of rated thermal power and shall be less positive than 0.9×10^{-4} $\Delta k/k/°F$ whenever thermal power is <95% of rated thermal power and the reactor is not shutdown.

3.1.7.2 The MTC shall be determined to be within its limits by confirmatory measurements prior to initial operation above 5% of rated thermal power after each fuel loading. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the limits in 3.1.7.1 above.

3.1.7.3 With the MTC outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

Bases

A non-positive moderator coefficient at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power, the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.9 \times 10^{-4}$ $\Delta k/k/°F$ corrected to 95% of rated power. The most limiting event for positive MTC, the Startup Accident, has been analyzed for a bounding moderator temperature coefficient of $+0.9 \times 10^{-4}$ $\Delta k/k/°F$.

MARKUP OF CURRENT ANO-1 TECHNICAL SPECIFICATIONS

(FOR INFO ONLY)

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

2.2.2 The setpoint of the pressurizer code safety valves shall be in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1968.

Bases

The reactor coolant system [1] serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110 percent of design pressure.[2] The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110 percent of design pressure. Thus, the safety limit of 2750 psig (110 percent of the 2500 psig design pressure) has been established. [2] The settings for the reactor high pressure trip (2355 psig) and the pressurizer code safety valves (2500 psig ±1%)[3] have been established to assure that the reactor coolant system pressure safety limit is not exceeded. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig ~~+1, -3~~±3%. However, if found outside of a ±1% tolerance band, they shall be reset to 2500 psig ±1%. The initial hydrostatic test is conducted at 3125 psig (125 percent of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig.[4]

REFERENCES

(1) FSAR, Section 4

(2) FSAR, Section 4.3.11.1

(3) FSAR, Section 4.2.4

(4) FSAR, Table 4-1

BASES:

The plant is designed to operate with both reactor coolant loops and at least one reactor coolant pump per loop in operation, and maintain DNBR above 1.30 (for the BAW-2 correlation) and 1.18 (for the BWC correlation) during all normal operations and anticipated transients.(1)

Whenever the reactor coolant average temperature is above 280°F, single failure considerations require that two loops be operable.

The decay heat removal system suction piping is designed for 300°F thus, the system can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident. (5) The pressurizer code safety valve lift setpoint shall be 2,500 psig ±1 percent allowance for error and each valve shall be capable of relieving ~~300,000~~ 324,000 lb/h of saturated steam at a pressure not greater than 3 percent above the set pressure. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig ~~+1, -3~~ ±3 percent. However, if found outside the ±1 percent tolerance band, they shall be reset to 2500 psig ±1 percent.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internal vent valves (1) ensure operability, (2) ensure that the valves are not open during normal operation, and (3) demonstrate that the valves begin to open and are fully open at the forces equivalent to the differential pressures assumed in the safety analysis.

The reactor coolant vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The operability of at least one reactor coolant system vent path from the reactor vessel head, the reactor coolant system highpoints, and the pressurizer steam space ensures the capability exists to perform this function. The valve redundancy of the vent paths serves to minimize the probability of inadvertent actuation and breach of reactor coolant pressure boundary while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path. Testing requirements are covered in Section 4.0 for the class 2 valves and Table 4.1-2 for the vent paths. These are consistent with ASME Section XI and Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," 11/80.

REFERENCES

(1) FSAR, Tables 9-10 and 4-3 through 4-7
(2) FSAR, Section 4.2.5.1 and 9.5.2.3
(3) FSAR, Section 4.2.5.4
(4) FSAR, Section 4.3.10.4 and 4.2.4
(5) FSAR, Section 4.3.7

3.1.3 Minimum Conditions for Criticality

Specification

3.1.3.1 The reactor coolant temperature shall be above 525F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply.

3.1.3.2 Reactor coolant temperature shall be to the right of the criticality limit of Figure 3.1.2-2.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1 percent $\Delta k/k$ until a steam bubble is formed and ~~an indicated water level between 45 and 305 inches is established in the~~ pressurizer water level within the limits of Figure 3.1.3-1 is established.

3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality.

3.1.3.6 The reactor shall not be made critical until at least 2 of the 3 emergency-powered pressurizer heater groups are operable. With less than 2 of the 3 required heater groups operable, restore the required heater groups to operable status within 72 hours. If the required heater groups are not restored to operable status within 72 hours, be in hot shutdown within the following 12 hours.

3.1.3.7 With any of the above limits violated, restore the reactor to within the limit in 15 minutes or be in at least Hot Shutdown within the next 15 minutes.

Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent $\Delta k/k$.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated $\Delta k/k$ would limit the magnitude of power excursion resulting from a reduction of moderator density.

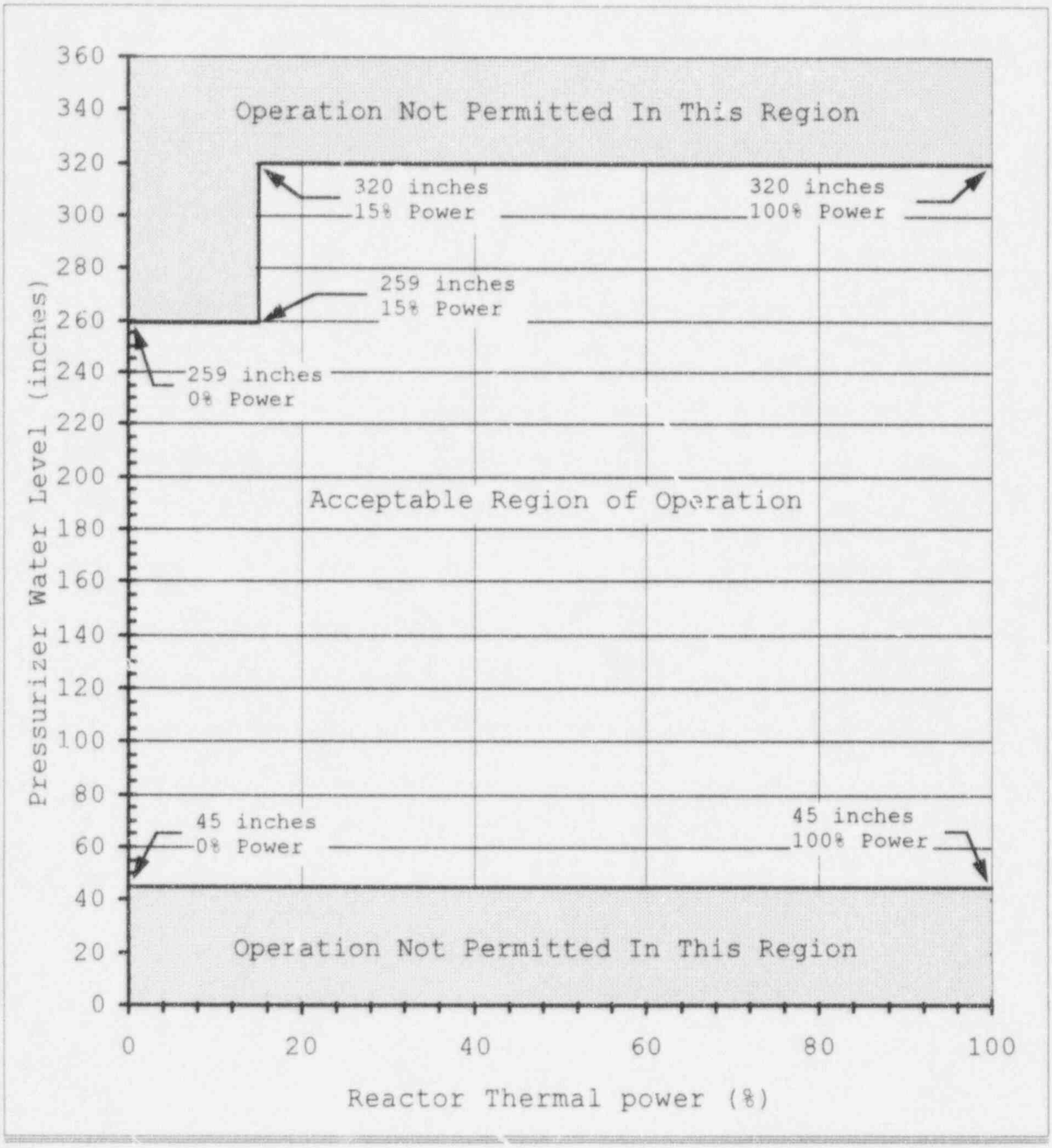


Figure 3.1.3-1

ANO-1 Pressurizer Level Acceptable Region of Operation

NOTE: The values specified for pressurizer level and reactor power do not contain an allowance for instrument error.

3.1.7 Moderator Temperature Coefficient of Reactivity Specification

3.1.7.1 The moderator temperature coefficient (MTC) shall be non-positive whenever thermal power is ≥95% of rated thermal power and shall be less positive than 0.9×10^{-4} Δk/k/°F whenever thermal power is <95% of rated thermal power and the reactor is not shutdown.

3.1.7.2 The MTC shall be determined to be within its limits by confirmatory measurements prior to initial operation above 5% of rated thermal power after each fuel loading. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the limits in 3.1.7.1 above.

3.1.7.3 With the MTC outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

Bases

A non-positive moderator coefficient at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power, the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.9 \times 10^{-4}$ Δk/k/°F corrected to 95% of rated power. The most limiting event for positive MTC, the Startup Accident, has been analyzed for a bounding~~range of~~ moderator temperature coefficient~~s including~~of $+0.9 \times 10^{-4}$ Δk/k/°F.

PROPOSED ANO-1 SAFETY ANALYSIS REPORT CHANGES

(For Use in Review of Proposed TS Changes)

3A.7.2.1 Startup Event

Delete

~~The beginning-of-cycle moderator temperature coefficient at hot zero power (HZP) for Cycle 13 is given as $+0.36 \times 10^{-4}$ Δk/k/°F in Table 3A-8. This parameter is used in the startup event analysis. A sensitivity study was performed in the SAR that varied the moderator temperature coefficient up to $+0.90 \times 10^{-4}$ Δk/k/°F, but the remaining startup event analyses in the SAR considered an MTC of zero. An evaluation has been done to verify that the results of the analyses in the SAR do validate the use of a moderator temperature coefficient of $+0.90 \times 10^{-4}$ Δk/k/°F at hot zero power. The analysis, therefore, bounds the Cycle 13 parameters.~~

3A.7.2.2 Steam Line Failure

The steam line break (SLB) accident was evaluated based on the reactivity feedback, termed the reactivity deficit, at conditions below HZP (532F and 2200 psia). The reactivity deficit for the steam line break analysis is 0.93702 %Δk/k. This value includes the effects of both fuel and moderator temperature changes. The reactivity deficit predicted for Cycle 13 using the same SLB system conditions is 1.07 %Δk/k (Table 3A-8). The Cycle 13 value, calculated by NEMO, is larger than the SLB analysis value, indicating a greater reactivity feedback for the Cycle 13 core. The cross section library used by NEMO to calculate the Cycle 13 reactivity deficit has not been benchmarked to the final SLB temperature and pressure of the moderator and temperature of the fuel. For conservatism, an uncertainty of 0.2 %Δk/k has been applied to the above Cycle 13 NEMO reactivity deficit calculation to bound the cross section data uncertainties. The rod insertion limits have been verified to accommodate the difference between the NEMO reactivity deficit for Cycle 13 and the TRAP2 reactivity deficit used for the MSLB analysis.

3A.7.2.3 Non-LOCA Safety Analysis Conclusions

The key cycle-specific parameters for each of the events in chapter 14 of the ANO-1 SAR were reviewed. It has been concluded that the non-LOCA safety analyses remain bounding for Cycle 13 operation.

3A.7.3 LOCA EVALUATION

The emergency core cooling system (ECCS) evaluation model (EM) reported in BAW-10103A, Rev. 3 (reference 12) has been approved for the analysis of large break loss-of-coolant accidents (LOCA) for the B&W-designed plants. The EM has been upgraded with the B&W-modified version of FLECSET (reference 13). The application of the EM to the B&W-designed, 177-fuel assembly, lowered-loop nuclear steam supply (NSS) system is reported in BAW-10104PA, Rev. 5 (reference 14). The fuel performance data input to the EM have been provided by TACO2 and current TACO3 computer codes (references 15 and 4).

The analyses are performed generically, using the limiting values of key parameters for all of the operating B&W-designed 177-fuel assembly lowered-loop plants. The LOCA linear heat rate (LHR) limits include the combined effects of the NUREG-0630 cladding swell and rupture model, the BWC CHF correlation, reduced fuel rod pre-pressure, and the B&W-modified version of FLECSET.

ARKANSAS NUCLEAR ONE
Unit 1

TABLE 3A-8

COMPARISON OF KEY PARAMETERS FOR ACCIDENT ANALYSIS

Parameter	Safety Analysis Value	Cycle 13 Value
BOC (a) Doppler coefficient, 10-5, Δk/k/°F	-1.17(e)	-1.61
EOC (b) Doppler coefficient, 10-5, Δk/k/°F	-1.30	-1.80
BOC moderator coefficient (HFP), 10-4, Δk/k/°F	0.0	-0.22
EOC moderator coefficient (HFP), 10-4, Δk/k/°F	-4.0	-3.23
BOC moderator coefficient (HZP), 10-4, Δk/k/°F	+0.9	+0.36
SLB reactivity deficit, %Δk/k	0.93702(c,d)	1.07(d)
All rod bank worth (HZP), %Δk/k	12.90	7.56
Maximum single group worth (HZP), %Δk/k	Nominal 3.0	2.59
Inverse boron worth (HFP), ppm/%Δk/k	140	152
Maximum ejected rod worth (HFP), %Δk/k	0.65	≤0.65
Maximum dropped rod worth (HFP), %Δk/k	0.65	≤0.20
Initial boron concentration (HFP), ppm	2270	2042

(a) BOC denotes beginning of cycle.

(b) EOC denotes end of cycle.

(c) Used in the steam line break analysis.

(d) Calculated over a moderator temperature range of 532F to 477.51F, a fuel temperature range of 532F to 650.7F, and a core pressure range of 2200 psia to 735.87 psia.

(e) Doppler coefficient used for Startup Event was -1.3×10^{-5} Δk/k/°F.

ARKANSAS NUCLEAR ONE
Unit 1

TABLE 3A-9

ANALYSIS STATUS OF NON-LOCA SAFETY ANALYSIS

Event	Effective Cycle For Analysis Of Record	Cycle-Specific Parameters Bounded?
Startup Event	~~1~~ 13	~~Section 3A.7.2~~ Yes
Rod Withdrawal at Power Event	1	Yes
Moderator Dilution Event		
At Power	12	Yes
During Refueling	12	Yes
Cold Water Event	1	Yes
Loss of Coolant Flow System Response[a]		
Locked Rotor Event	1	Yes
Four-Pump Coastdown Event	1	Yes
Four-to-Two Pump Coastdown Event	1	Yes
Dropped Rod Event	1	Yes
Loss of Electric Power Events		
Loss of Load Event	1	Yes
Complete Loss of AC Power Event	1	Yes
Turbine Overspeed Event	1	Yes
Fuel Handling Accident	1	Section 3A.7.1
Steam Line Failure Event	12	Section 3A.7.2
Steam Generator Tube Failure Event[b]	1	Yes
Rod Ejection Event	1	Yes
Loss-of-Coolant Event	Section 3A.7.3	Section 3A.7.3
Maximum Hypothetical Accident	Section 3A.7.1	Section 3A.7.1
Waste Gas Decay Tank Rupture Event	Section 3A.7.1	Section 3A.7.1

(a) The plant system response (including power, RCS flow, core inlet temperature, and system pressure) has been shown to be bounding for cycle 13. The DNB analysis is discussed separately in section 3A.6.

(b) For dose consequences of the steam generator tube rupture event, refer to section 3A.7.1.

D. A short-period withdrawal stop and alarm are provided in the intermediate range.

E. A high flux level and a high pressure trip are provided in the power range.

14.1.2.2.2 Reactor Protection Criteria

The criteria for reactor protection for this accident are:

A. Reactor thermal power shall not exceed 112 percent of rated power.

B. RCS pressure shall not exceed code pressure limits.

14.1.2.2.3 Methods of Analysis

102.67% of design
A conservative (less negative)

A B&W digital computer model of the reactor core and RCS was used to determine the characteristics of this accident. This model used ~~[illegible]~~ reactor coolant flow but no heat transfer out of the system and no sprays in the pressurizer. ~~[illegible]~~ Doppler coefficient was used such that ~~although~~ the Doppler coefficient is much larger (more negative) than this ~~for the principal part of~~ for actual plant operation ~~the transient~~. The rods were assumed to be moving out along the steepest part of the rod worth versus rod travel curve. The values of the principal parameters used in this analysis are listed in Table 14-3.

In addition, the criterion for minimum movable control rod worth is that a shutdown margin of one percent Δk/k at the hot standby condition is required (Section 3.1.2.2). The startup accident has been analyzed using the minimum tripped rod worth with the maximum worth stuck rod as part of the analysis. The startup accident was analyzed from 0.5% Δk/k subcritical at the hot, pressurized condition.

14.1.2.2.4 Results of Analysis

the reactivity addition rate that results in the peak pressure and thermal power.

Figure 14-1 shows the results of ~~withdrawing the maximum worth control rod group at the maximum rod speed from 0.5% Δk/k subcritical. This rod velocity and worth result in the maximum reactivity addition rate.~~ The Doppler effect terminates the neutron power (neutron power is defined as the total energy release from fission) rise, but the heat input to the reactor coolant increases the pressure past the trip point and the transient is terminated by the high pressure trip.

10^{-9} rated power

Figure 14-2 shows the results of withdrawing all Control Rod Assemblies (CRAs) at the maximum speed from ~~0.5% Δk/k subcritical~~. This results in a maximum possible reactivity addition rate. The total rod worth used in this analysis is slightly greater than the calculated worth (Table 3-5). The power rise is terminated by the negative Doppler effect. The high neutron flux trip takes effect after the peak power is reached and terminates the transient. The peak thermal heat flux is significantly less than the rated power heat flux.

neutron

A sensitivity analysis was performed ~~[illegible]~~ for the peak pressure and thermal power reached during a startup accident ~~transient~~ (protected by the high pressure trip) to determine the effect of varying several key parameters. Variation of the high pressure trip delay time from ~~0.1~~ 0.55 to 0.7 second resulted in a change in peak thermal power of less than ~~five~~ one percent and a change in the peak pressure of 6 psi.

Figure~~s~~ 14-3 ~~[illegible]~~ shows the effect of varying the reactivity addition rate on the peak thermal power ~~[illegible]~~. This reactivity rate was varied from more than an order of magnitude below the single rod group rate ~~used for the analysis~~ to a rate slightly above that for simultaneous withdrawal of all rods. The slower rates will result in the pressure trip being actuated. Only the very fast rates actuate the high neutron flux level trip.

Insert A

Figure~~s~~ ~~[illegible]~~ 14-6 shows the peak thermal power and peak pressure variation as a function of ~~a wide range of~~ ~~[illegible]~~ moderator coefficients for ~~[illegible]~~ a range of reactivity addition rates that result in worst case peak pressure and thermal power.

Insert B

~~Figures 14-7 and 14-8 are the corresponding results for the withdrawal of all rods.~~ Table 14-4 summarizes the results of the postulated startup accidents.

It is concluded that the reactor is completely protected against any startup accident involving the withdrawal of any or all control rods, since in no case does the thermal power approach the design overpower condition and the peak pressure never exceeds code allowable limits.

14.1.2.3 Rod Withdrawal Accident at Rated Power Operation

14.1.2.3.1 Identification of Cause

A rod withdrawal accident pre-supposes an operator error or equipment failure resulting in accidental withdrawal of a control rod group while the reactor is at rated power. As a result, the power level increases, the reactor coolant and fuel rod temperatures increase, and, if the withdrawal is not terminated by the operator or the protection system, core damage would eventually occur.

The following provisions are made in the design for the indication and termination of this accident.

A. High reactor coolant outlet temperature alarms.

B. High RCS pressure alarms.

C. High pressurizer level alarms.

D. High reactor coolant outlet temperature trip.

E. High RCS pressure trip.

F. High power level, i.e., neutron flux level, trip.

Insert A

The high pressure trip setpoint was varied for the peak pressure and thermal power case resulting from a reactivity addition rate of 1.73 E-4 (ΔK/K)/sec. An increase of the high pressure trip setpoint by 5 psi resulted in the peak pressure increasing by less than two psi and the peak thermal power increasing by less than one percent.

Variation of the assumed effective delayed neutron fraction (βeff) changes the reactivity addition rate which results in the peak pressure and thermal power. A decrease in the βeff from 0.007 to 0.006 resulted in a reduction of the peak pressure by two psi and an increase in the peak thermal power by less than one percent based on reactivity addition rates that result in peak pressure and thermal power.

Variation of the assumed axial peaking factor changes the reactivity addition rate which results in the peak pressure and thermal power. Analysis of the results with axial peaking factors of 1.0, 1.7, and 2.0 showed the axial peaking factor of 1.5 used for the analyses discussed for this event results in the peak RCS pressure. Although different axial peaking factors result in different peak thermal powers the margin available for thermal power is less limiting than the margin available for peak pressure.

The effect of varying the initial power level has shown that lower initial power in conjunction with high reactivity addition rates can result in higher peak thermal powers. These same studies show there is still margin to the rated thermal power even if all rods are simultaneously withdrawn at the maximum rate of withdrawal from an initial power of 1 E-9 watts. The power rise is terminated by the negative Doppler effect. The high neutron flux trip terminates the event. The pressure increases slowly until the PSV lifts. The resultant peak pressure in the RCS will be dictated by the PSV lift pressure plus any pressure differential between the PSV and the peak RCS pressure location.

The effect of varying the number of RCPs operating at the onset of the event show that the reactivity addition rate that results in the peak pressure and thermal power will change due to the different initial conditions. The resultant change in peak pressure of initiating the event with 3 RCPs versus 4 RCPs operating is an increase in the peak pressure by about 6 psi, while the peak thermal power remains approximately the same or slightly lower than results initiated from 4 RCP initial conditions.

Figure 14-4 shows the effect of varying the pressurizer safety valve (PSV) lift setpoint tolerance (accumulation) from 3% to 5% (assuming all other inputs remain constant).

Figure 14-5 shows the effect of varying the pressurizer safety valve flow rate from a single PSV flow rate of 300,000 lbm/hr to 2 PSVs with a flow rate of 324,000 lbm/hr/valve.

Insert B

The peak RCS pressure was found to be dependent on the initial pressurizer level. Higher initial pressurizer levels result in less volume to accommodate the expansion of the RCS volume due to the heat input caused during the startup event. Figure 14-7 shows the resultant peak pressure corresponding to the reactivity addition rate that results in peak pressure.

Figure 14-8 shows the effect of varying the reactivity addition rate on peak pressure.

ARKANSAS NUCLEAR ONE
Unit 1

Table 14-3

STARTUP ACCIDENT PARAMETERS
~~(Initial Fuel Load)~~

Maximum Rod Speed, in./min	30
Maximum Number of CRAs	~~61~~ 60
Maximum Rod Worth, All Rods, % Δk/k	12.9
Maximum Reactivity Addition Rate, All ~~61~~ 60 Rods at Max Speed, (Δk/k)/s	9.27×10^{-4}
Maximum Rod Worth of Single Group When Reactor is Critical, % Δk/k	3.0
Maximum Reactivity Addition Rate for Single Rod Group, (Δk/k)/s	2.15×10^{-4}
Doppler Coefficient [redacted], (Δk/k)/°F	~~1.17×10^{-5}~~ -1.3×10^{-5}
Moderator Coefficient [redacted], (ΔK/K)/°F	~~Zero~~ $+0.9 \times 10^{-4}$
Peak Thermal Power Permitted (Design Overpower), % rated power	112
Trip Parameters	
High Pressure Trip Setpoint, psia	2400
Delay for High Pressure Trip, s	~~0.5~~ 0.6
Delay for High Flux Trip, s	~~0.3~~ 0.4
Control Rod Travel Time to 2/3 Insertion, s	1.4
Delayed Neutron Fraction (β_{eff})	0.007
Number of PSV's	2
PSV Lift Tolerance (Accumulation)	+3% (75 psi)
PSV Flow Rate, lbm/hr/valve	324,000
Initial Power, watts	2.568 (1×10^{-9} rated power)
Initial Pressurizer Level, inches	180
Number of RCPs in operation	4
Core Flux Axial Peaking Factor	1.5

ARKANSAS NUCLEAR ONE
Unit 1

Table 14-4

SUMMARY OF STARTUP ACCIDENT ANALYSIS

1. Peak thermal power for withdrawal rates less than that corresponding to the withdrawal of all rods is always less than rated power.

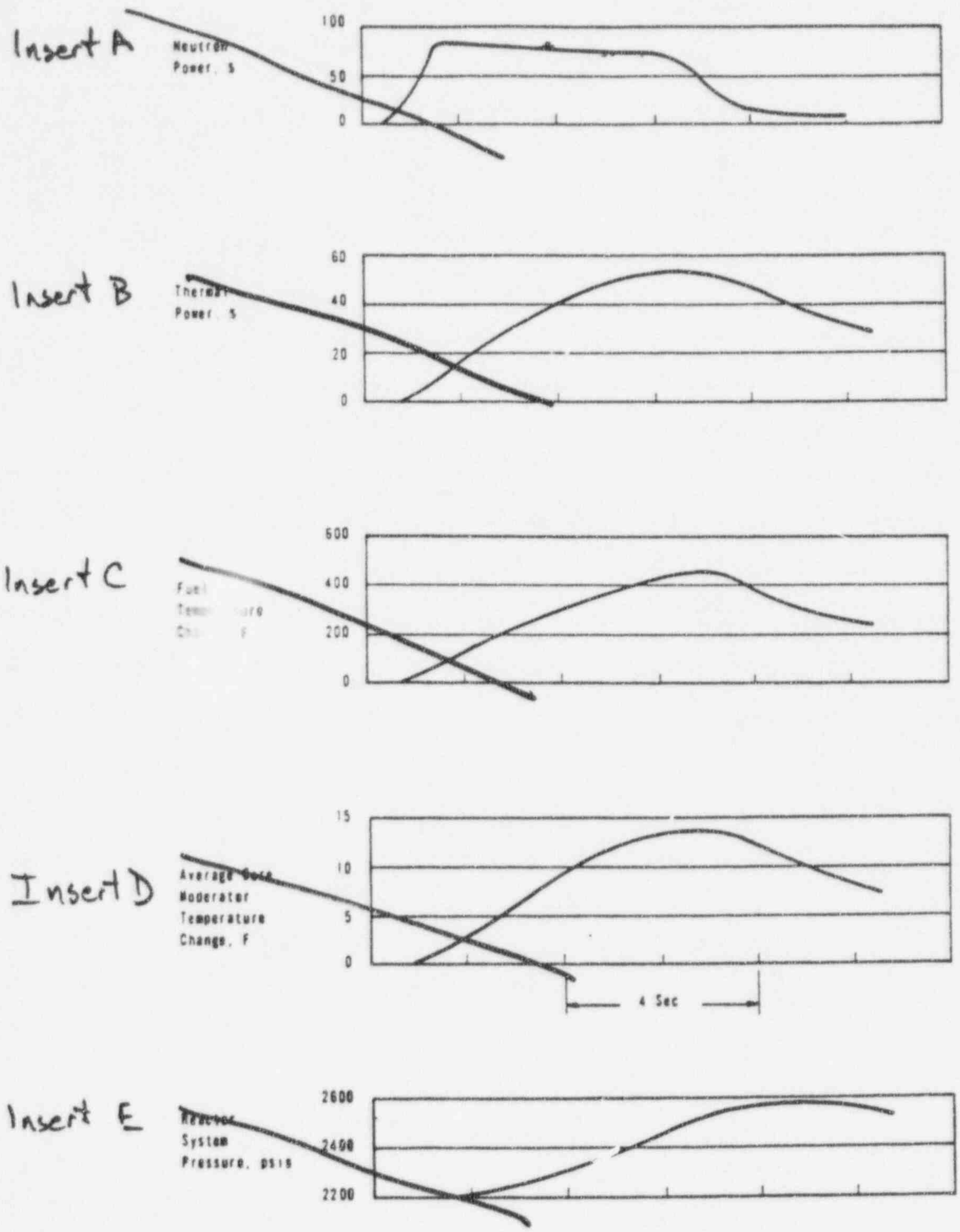
2. Average fuel temperature in the average fuel rod never exceeds ~~1,030~~ °F.
1,000

3. ~~The nominal single-group rod withdrawal causes a peak pressurizer pressure high enough to actuate the relief valves (at 2,515 psia). These valves have sufficient capacity to handle the resultant coolant expansion.~~

See Attached Insert C

Insert C

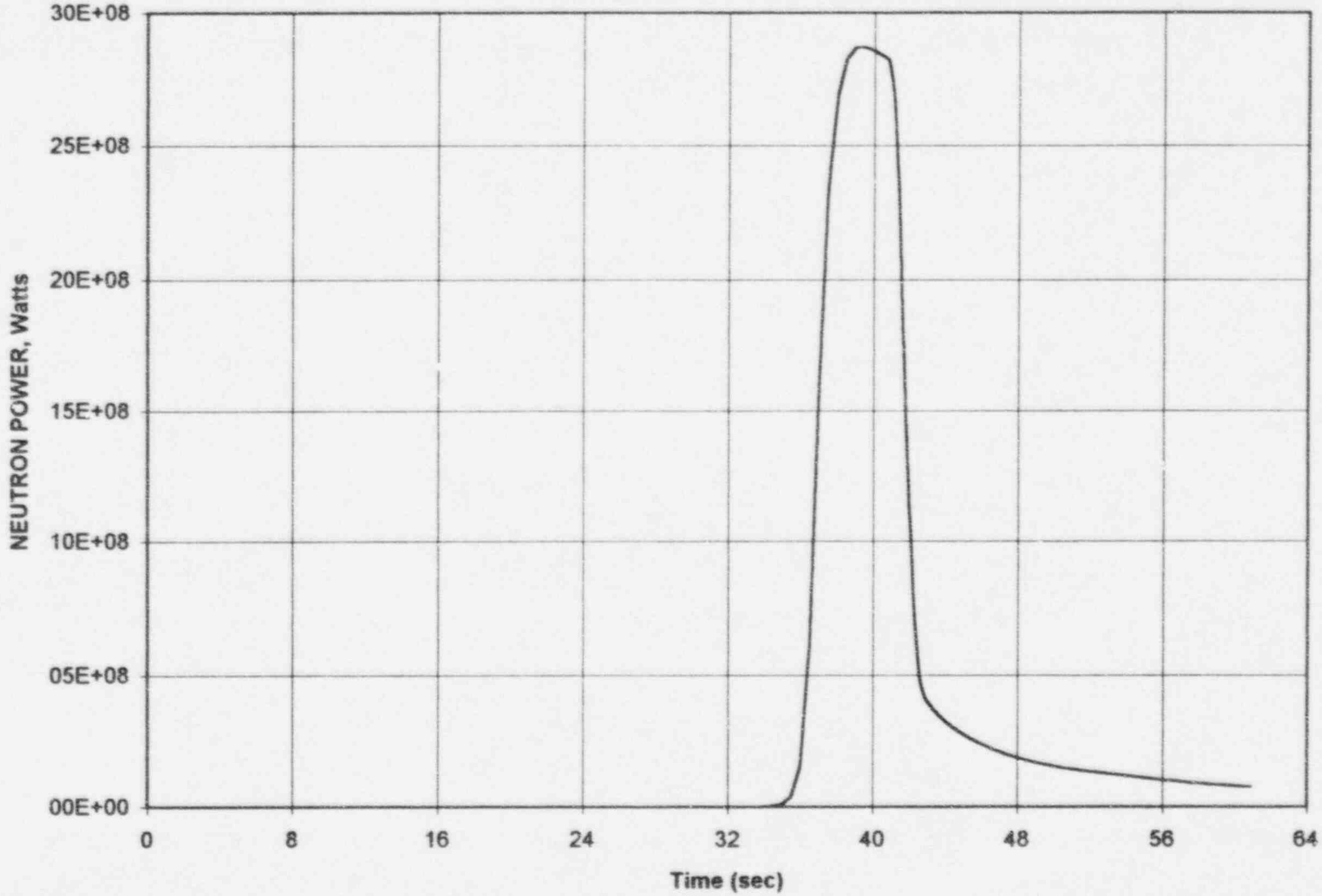
The peak RCS pressure was assured to be less than 2750 psig using a pressurizer level of 180 inches (minus any applicable uncertainty) with two pressurizer safety valves (PSVs) relieving at a 2590 psia setpoint and a flow rate of 324,000 lbm/hr/valve. The peak RCS pressure was also assured to be less than 2750 psig with only a single PSV relieving at a 2640 psia setpoint and a flow rate of 300,000 lbm/hr at a pressurizer level of 180 inches (minus any applicable uncertainty).



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	STARTUP ACCIDENT FROM 10^{-9} RATED POWER USING A [redacted]; HIGH PRESSURE REACTOR TRIP IS ACTUATED	FIG. NO. 14-1

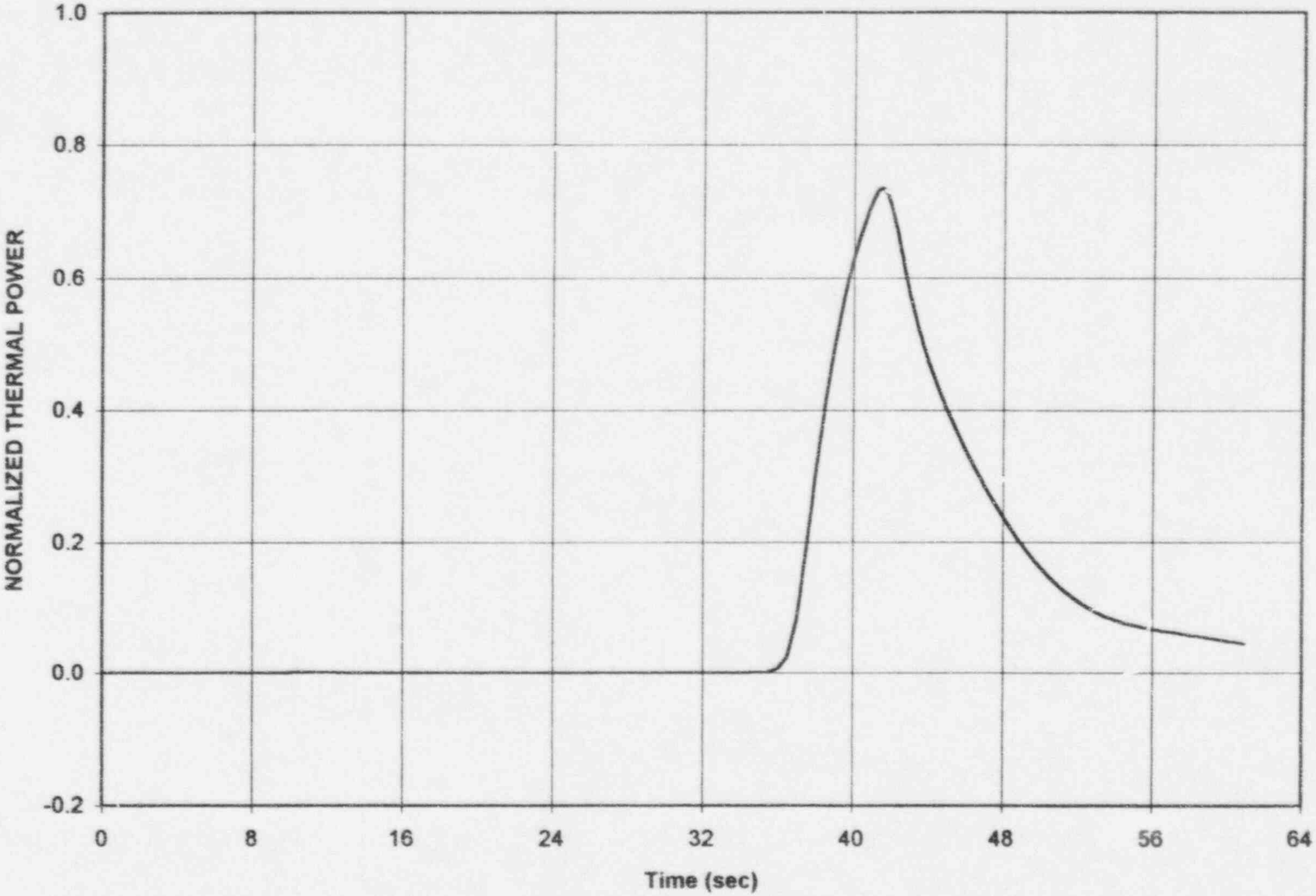
Reactivity Addition Rate of $0.173 \times 10^{-4} \frac{\Delta K/K}{sec}$

Neutron Power Versus Time for Startup Accident From 1E-09 Rated Power Using A Reactivity Addition Rate of 1.73 E-04 (DK/K)/sec; High Pressure Reactor Trip Is Actuated



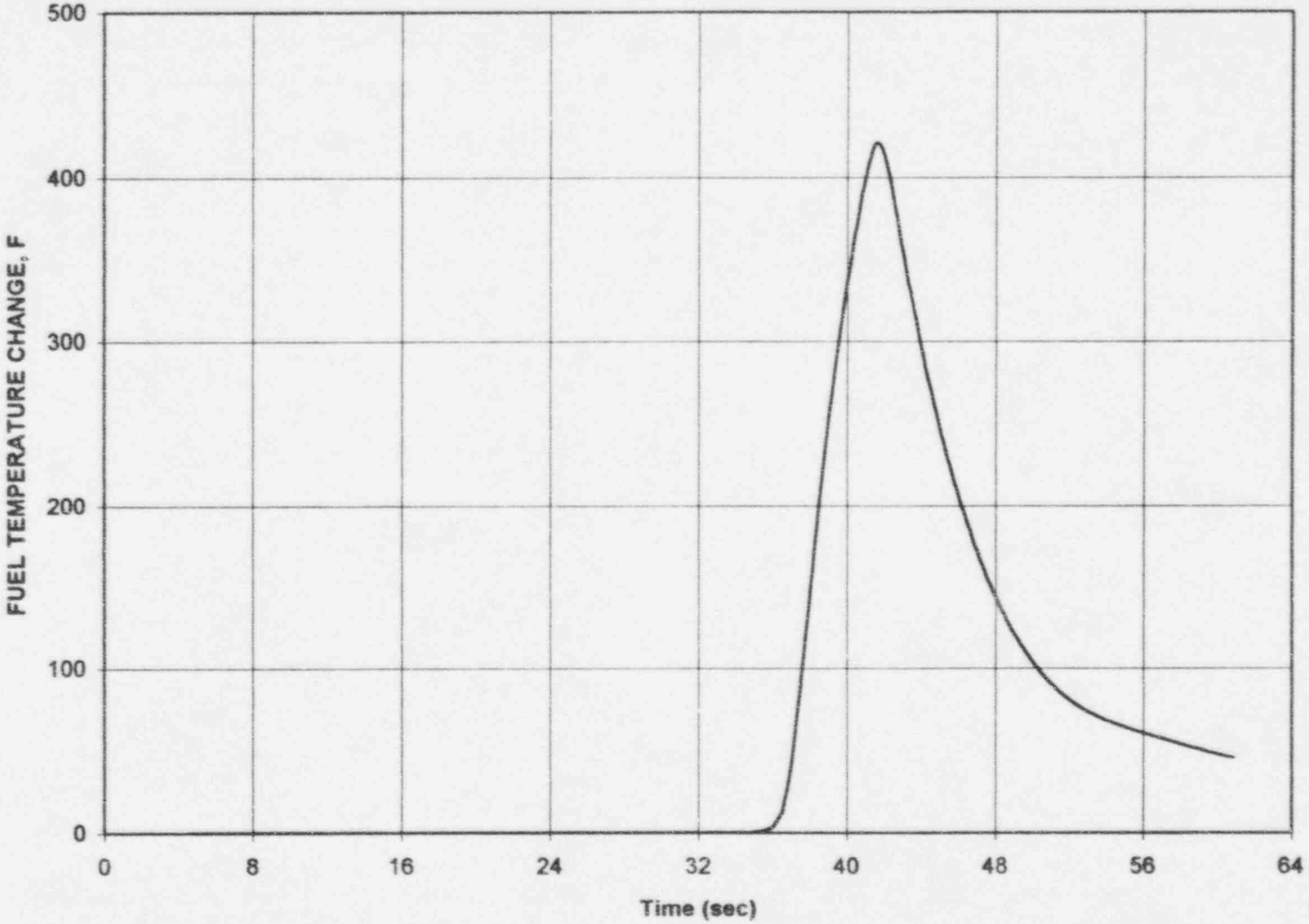
insert A

Thermal Power Versus Time for Startup Accident From 1E-09 Rated Power Using A Reactivity Addition Rate of 1.73 E-04 (DK/K)/sec; High Pressure Reactor Trip Is Actuated



Insert B

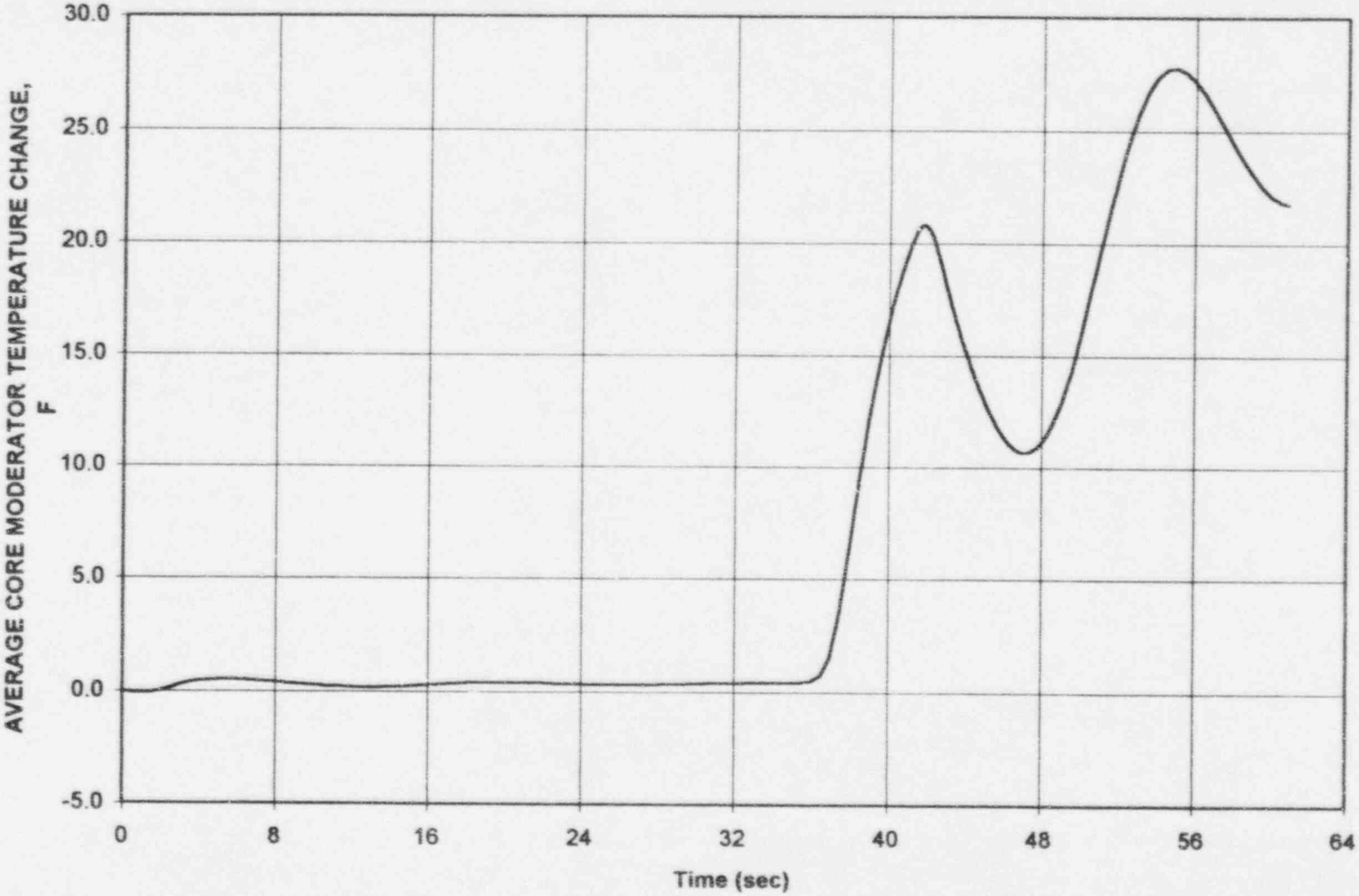
Fuel Temperature Change Versus Time for Startup Accident From 1E-09 Rated Power Using A Reactivity Addition Rate of 1.73 E-04 (DK/K)/sec; High Pressure Reactor Trip Is Actuated

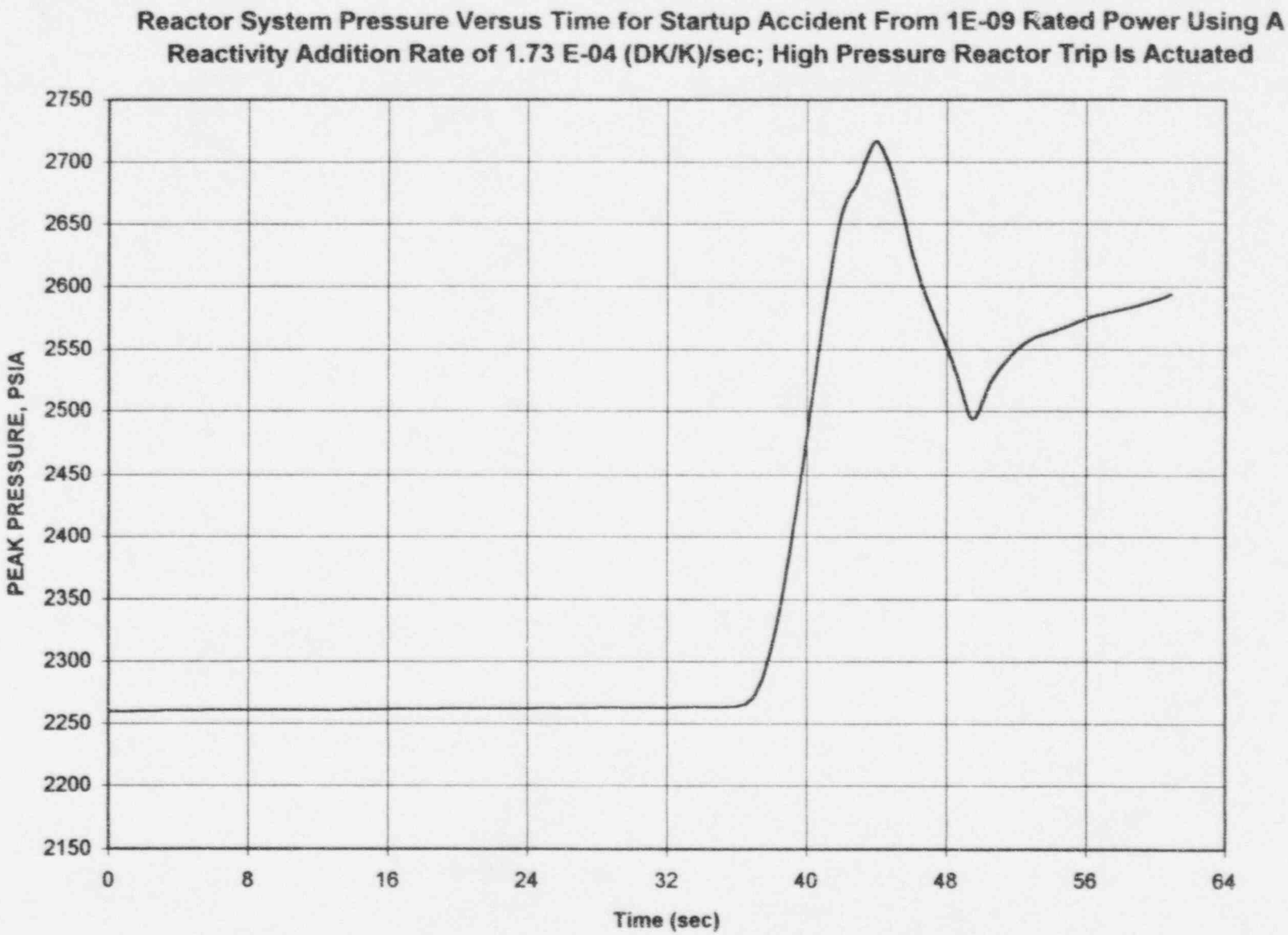


Insert C

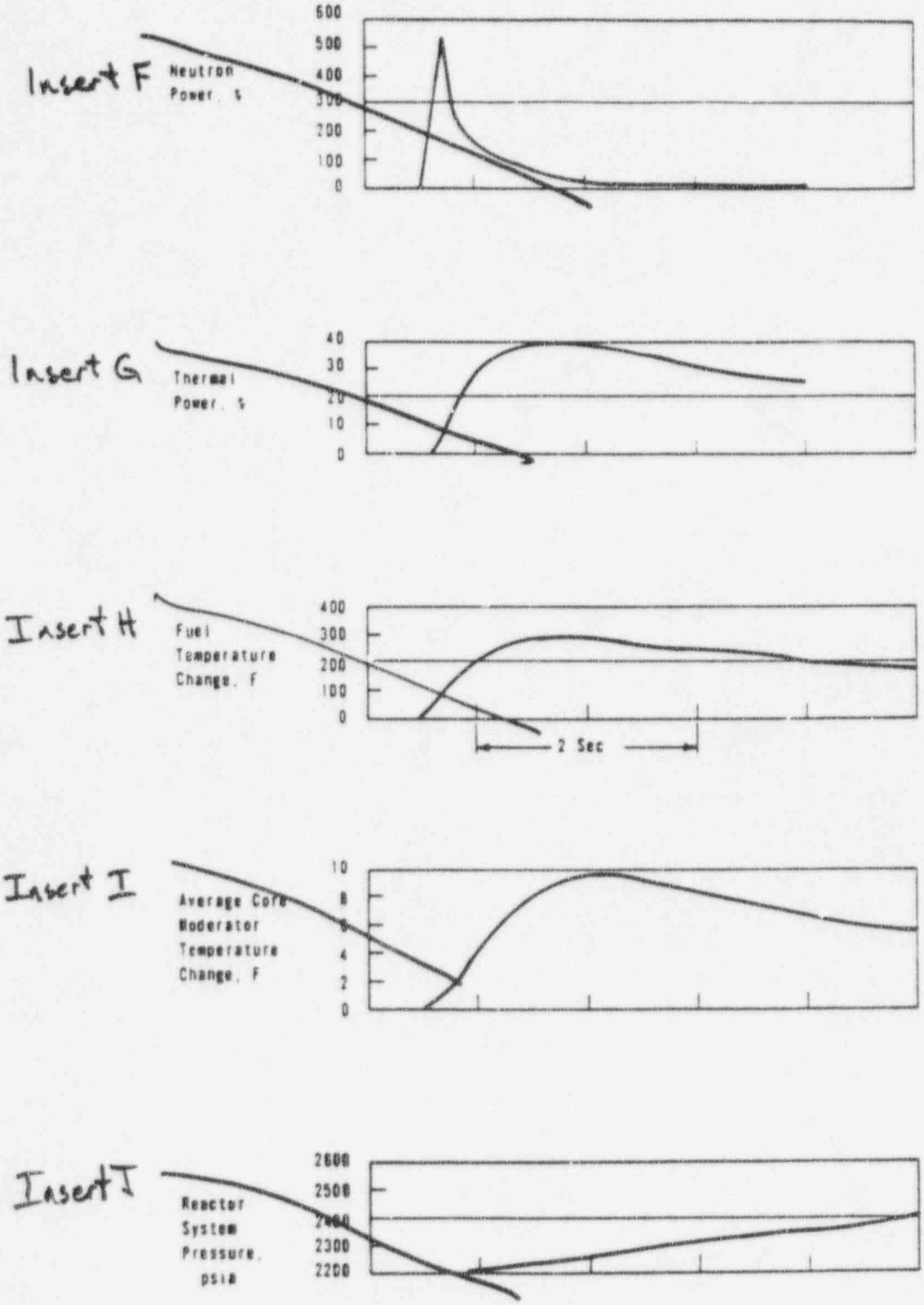
Average Core Moderator Temperature Change Versus Time fcr Startup Accident From 1E-09 Rated Power Using A Reactivity Addition Rate of 1.73 E-04 (DK/K)/sec; High Pressure Reactor Trip Is Actuated

Insert D




Reactor System Pressure Versus Time for Startup Accident From 1E-09 Rated Power Using A Reactivity Addition Rate of 1.73 E-04 (DK/K)/sec; High Pressure Reactor Trip Is Actuated
2750
2700
2650
2600
2550
2500
2450
2400
2350
2300
2250
2200
2150
PEAK PRESSURE, PSIA
0
8
16
24
32
40
48
56
64
Time (sec)

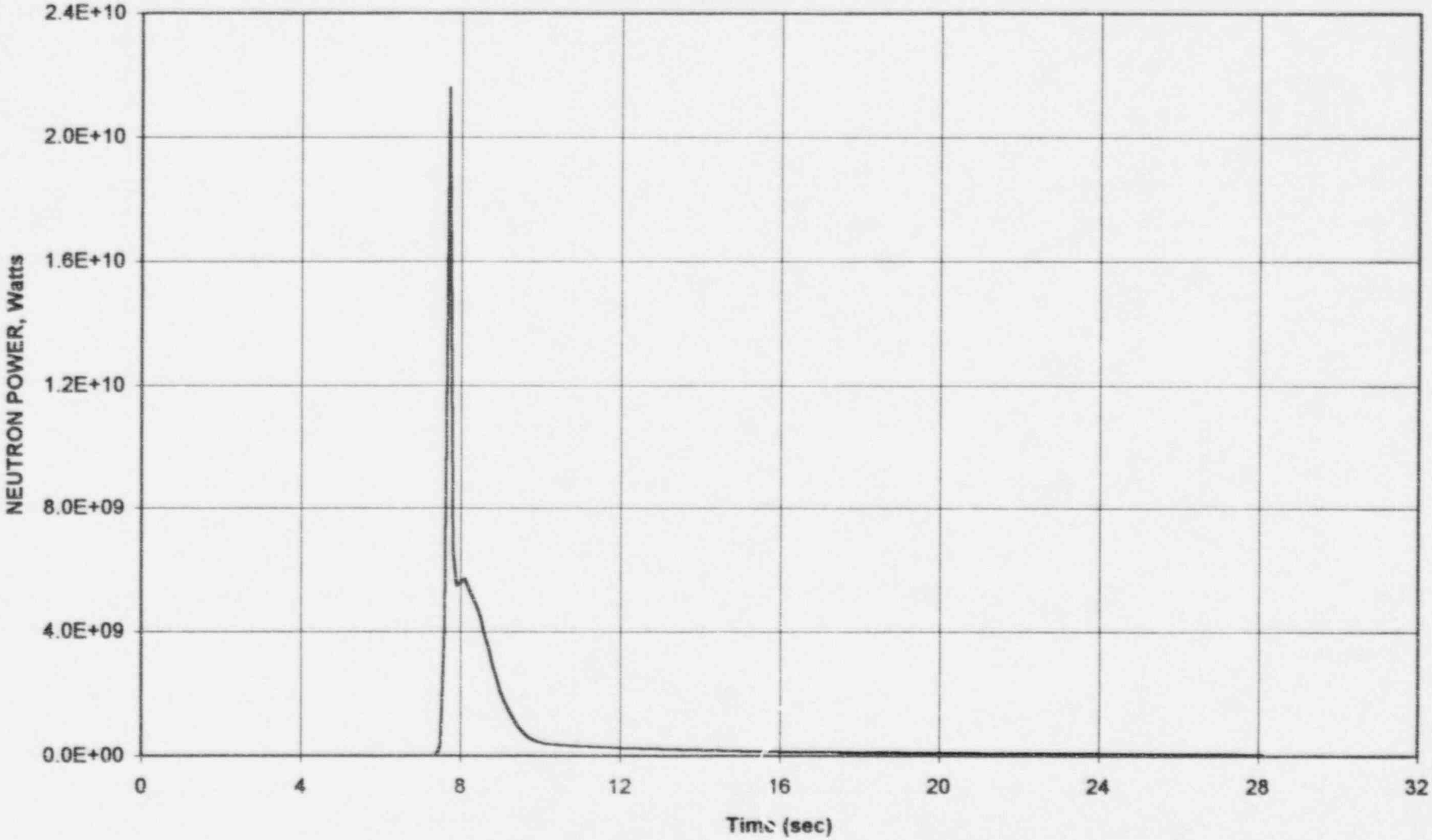
Insert E



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	STARTUP ACCIDENT FROM 10^{-9} RATED POWER USING ALL RODS WITH A [illegible]: HIGH FLUX REACTOR TRIP IS ACTUATED	FIG. NO. 14-2

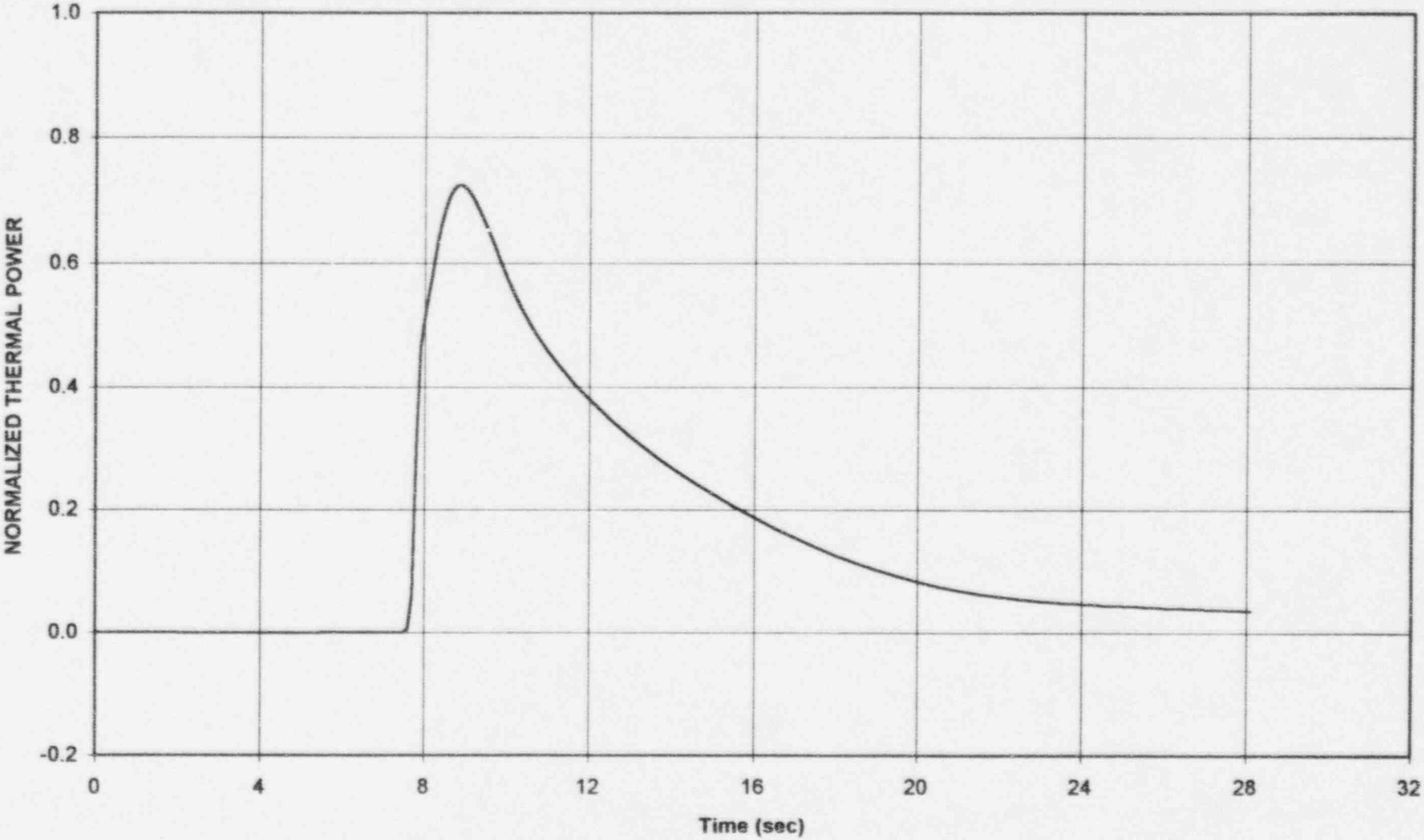
Reactivity Addition Rate of 1×10^{-3} $\Delta K/K/sec$

Neutron Power Versus Time For A Startup Accident From 1E-09 Rated Power Using A Reactivity Addition Rate of 1 E-03 (DK/K)/sec; High Flux Reactor Trip Is Actuated



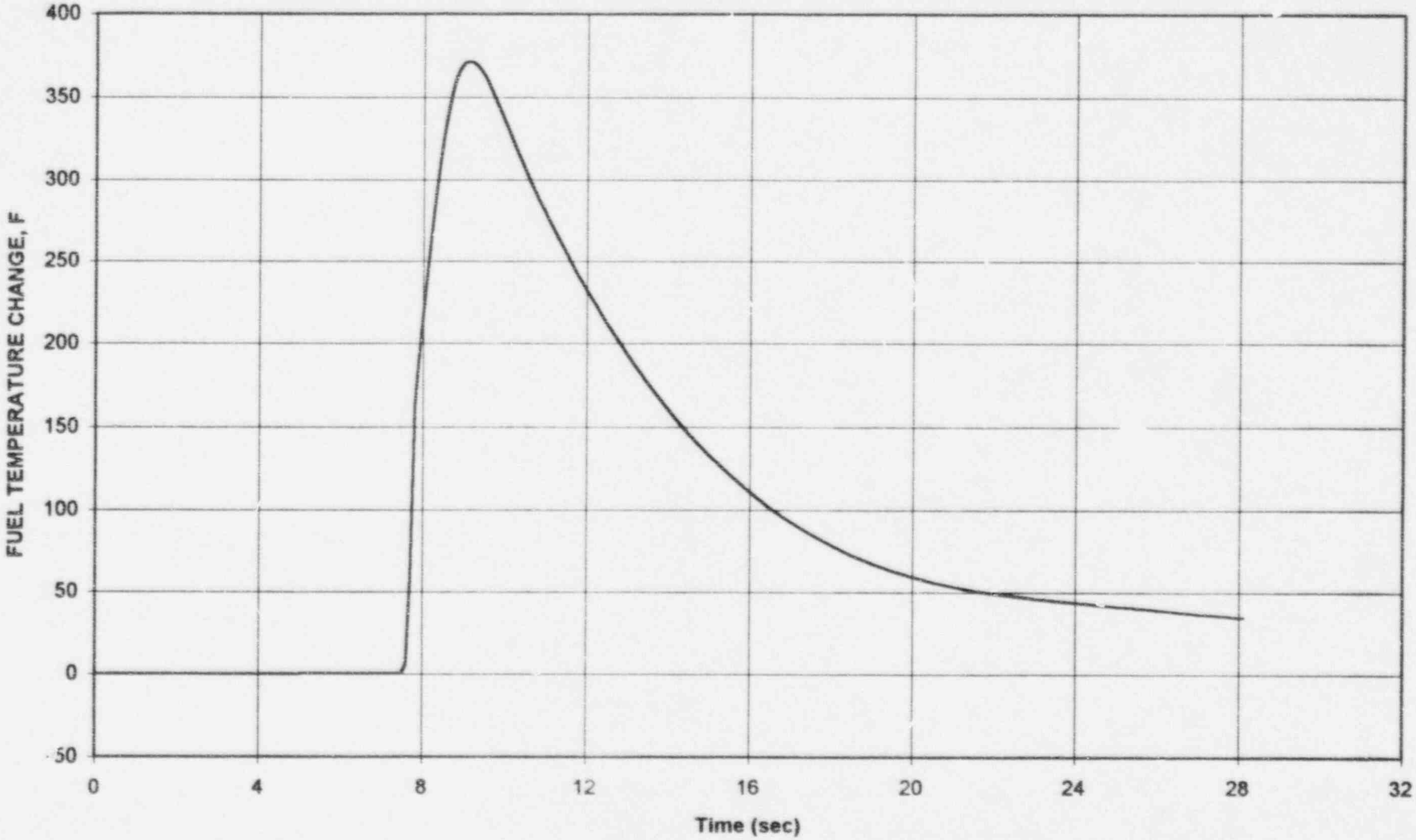
Insert F

Thermal Power Versus Time For A Startup Accident From 1E-09 Rated Power Using A Reactivity Addition Rate of 1 E-03 (DK/K)/sec; High Flux Reactor Trip Is Actuated



Insert G

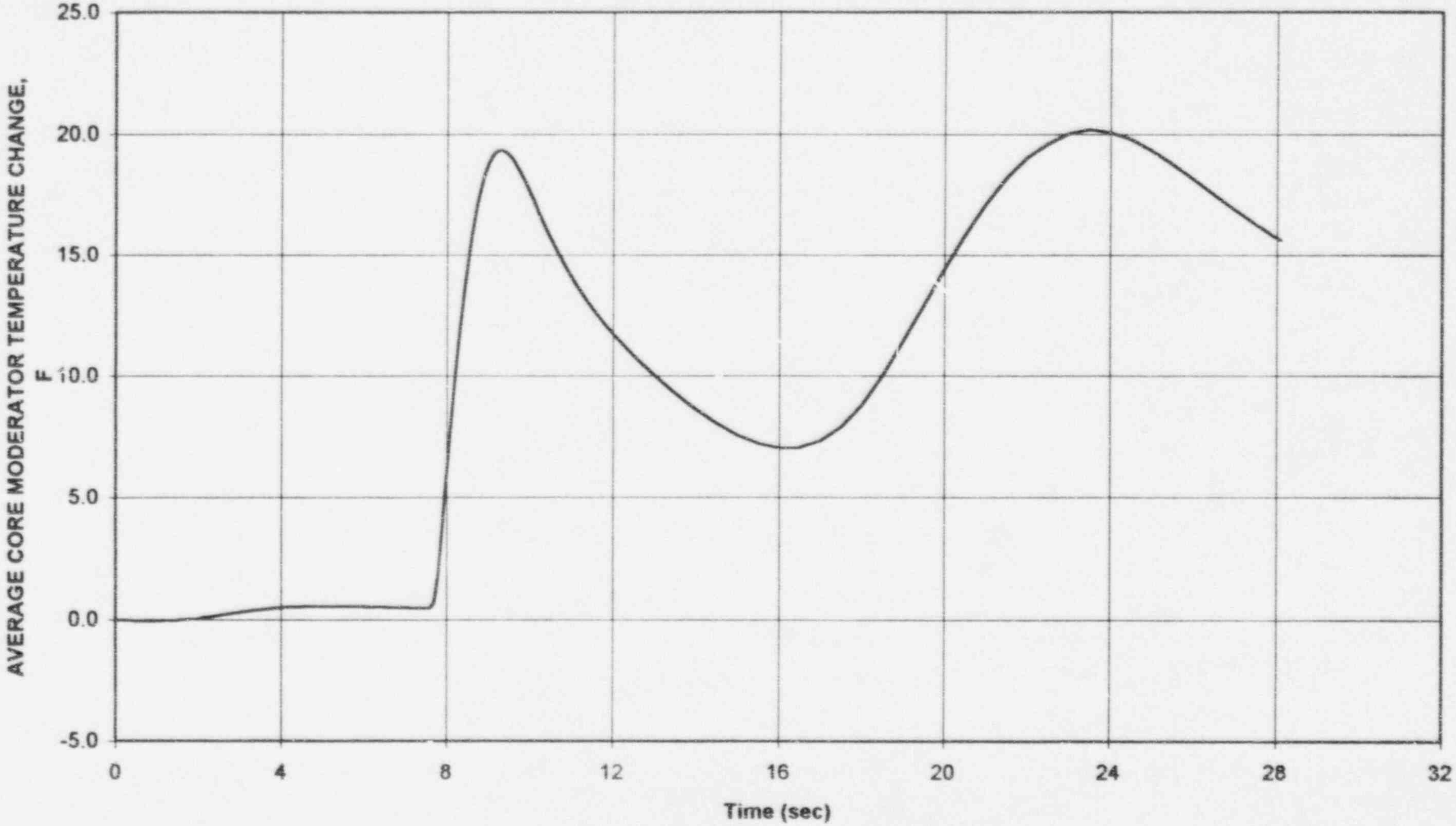
Fuel Temperature Change Versus Time For A Startup Accident From 1E-09 Rated Power Using A Reactivity Addition Rate of 1 E-03 (DK/K)/sec; High Flux Reactor Trip Is Actuated

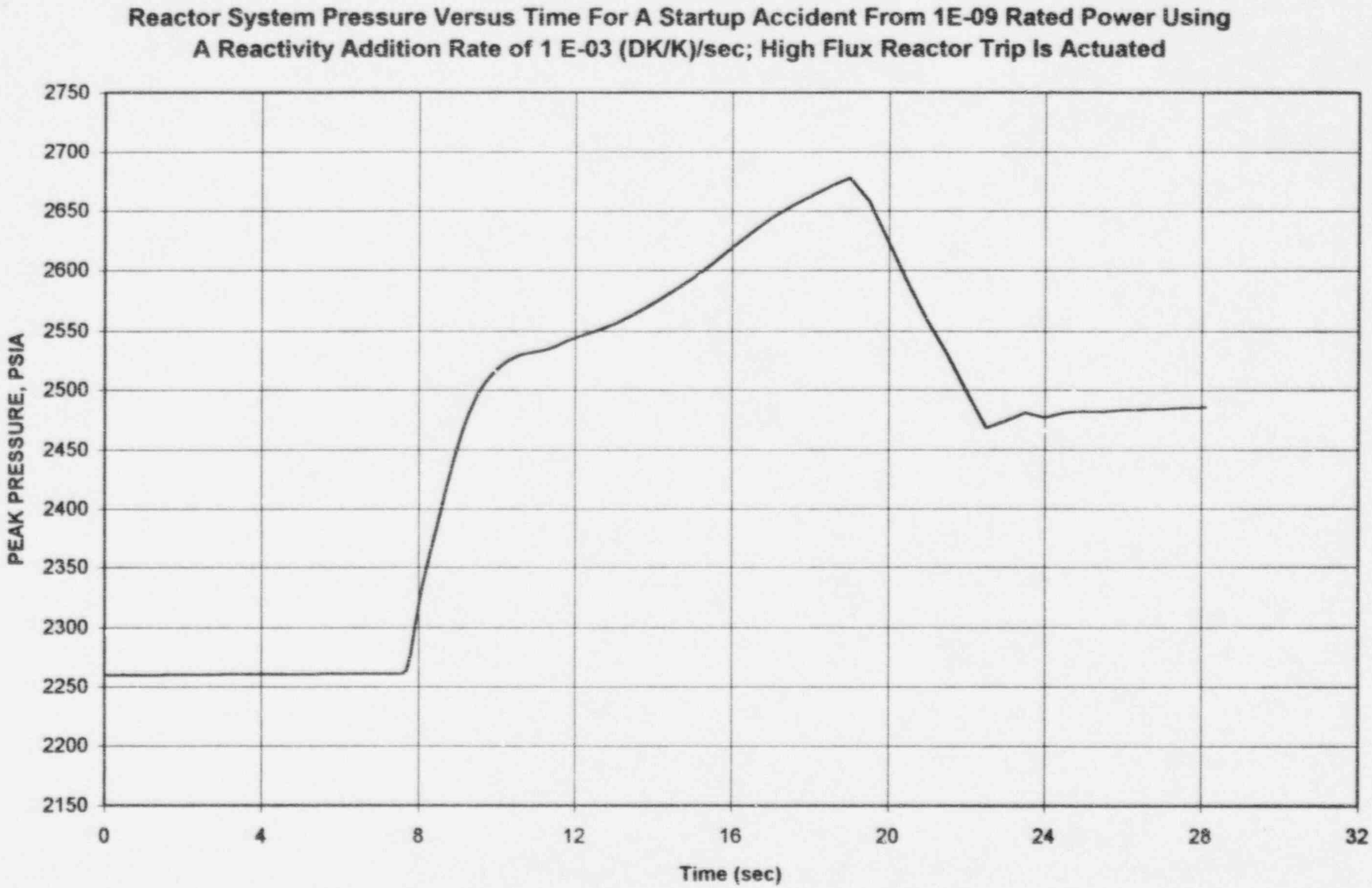


Insert H

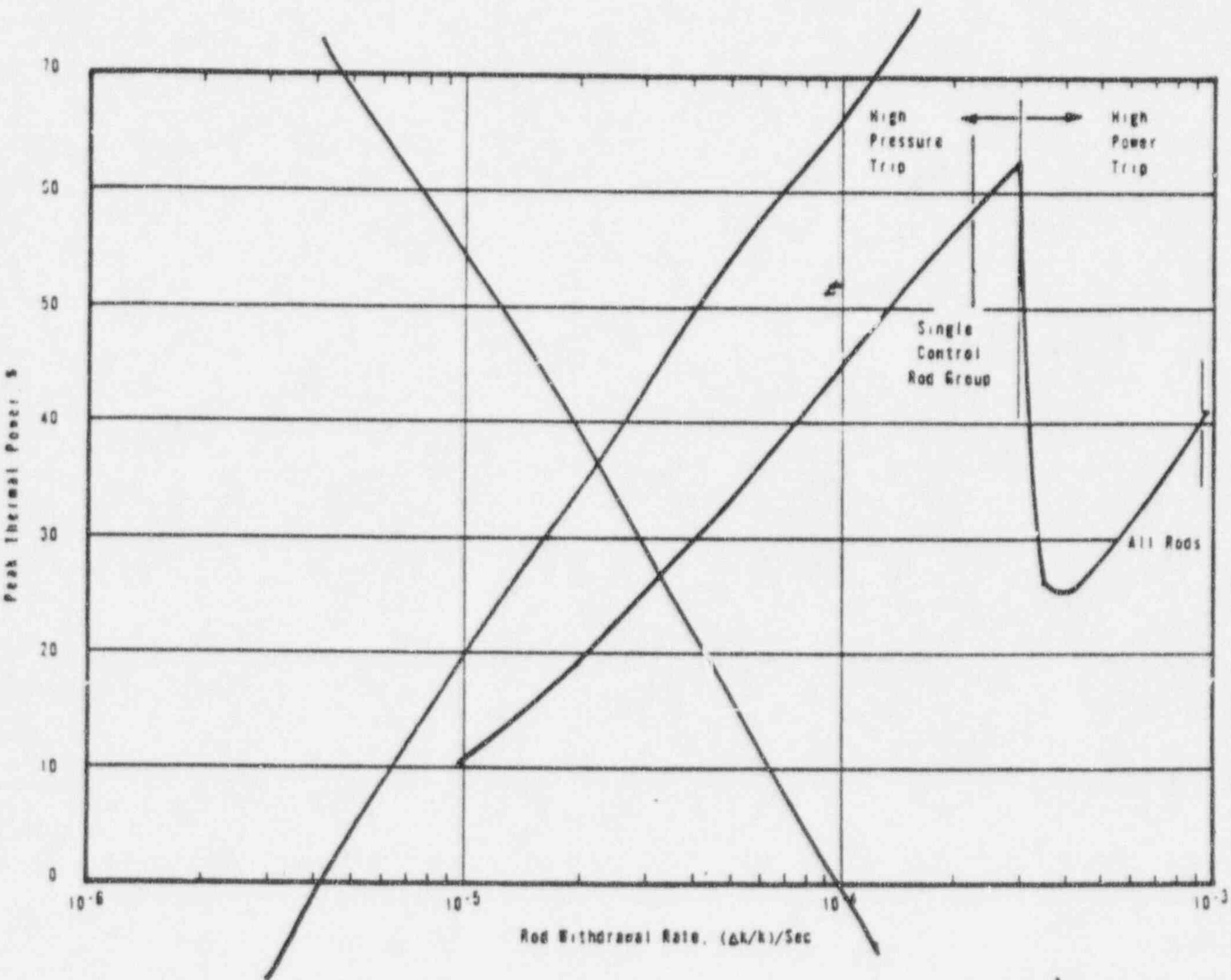
Average Core Moderator Temperature Change Versus Time For A Startup Accident From 1E-09 Rated Power Using A Reactivity Addition Rate of 1 E-03 (DK/K)/sec; High Flux Reactor Trip Is Actuated

Insert I



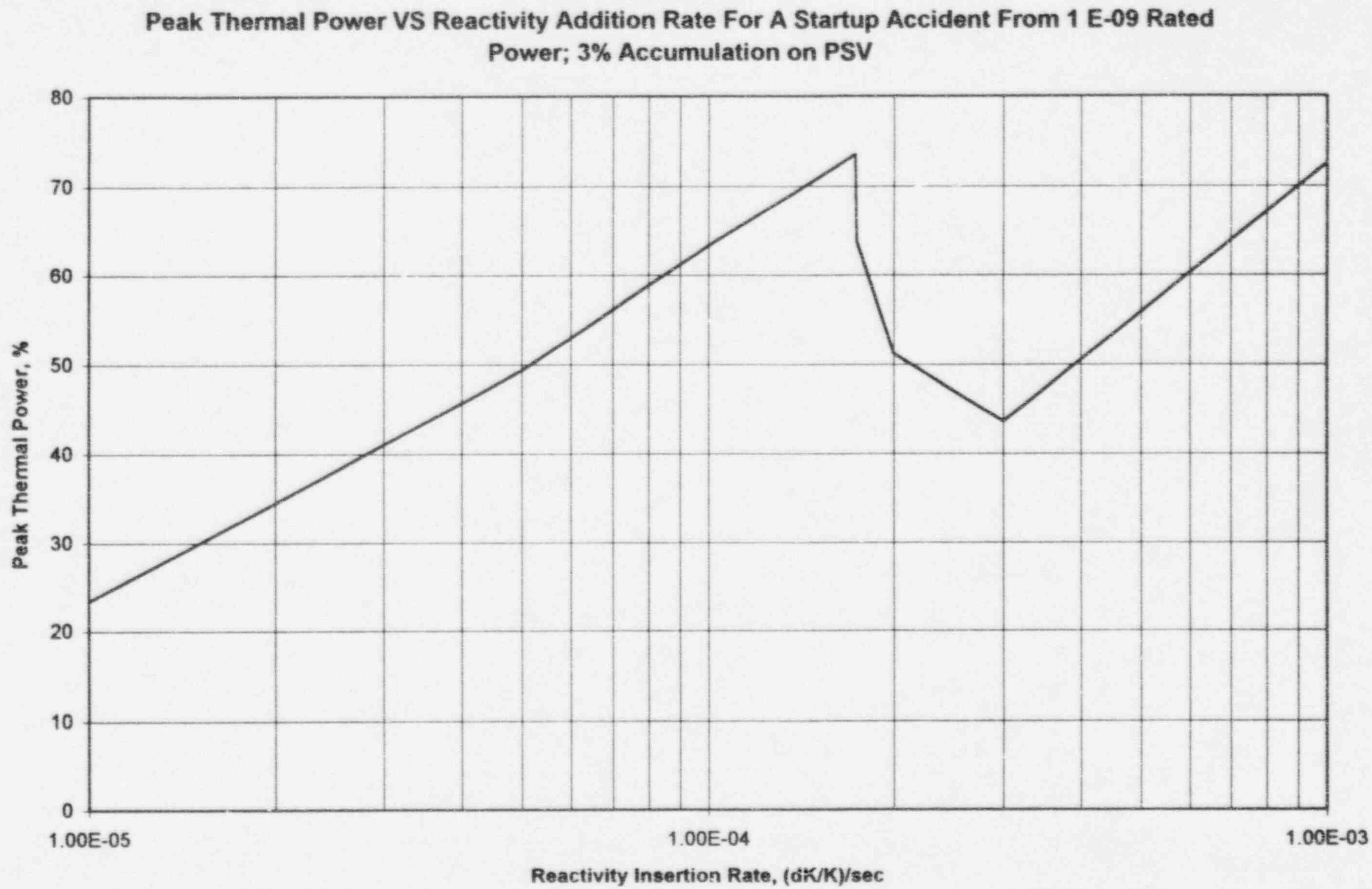

Reactor System Pressure Versus Time For A Startup Accident From 1E-09 Rated Power Using
A Reactivity Addition Rate of 1 E-03 (DK/K)/sec; High Flux Reactor Trip Is Actuated
PEAK PRESSURE, PSIA
2750
2700
2650
2600
2550
2500
2450
2400
2350
2300
2250
2200
2150
0
4
8
12
16
20
24
28
32
Time (sec)

Insert #J

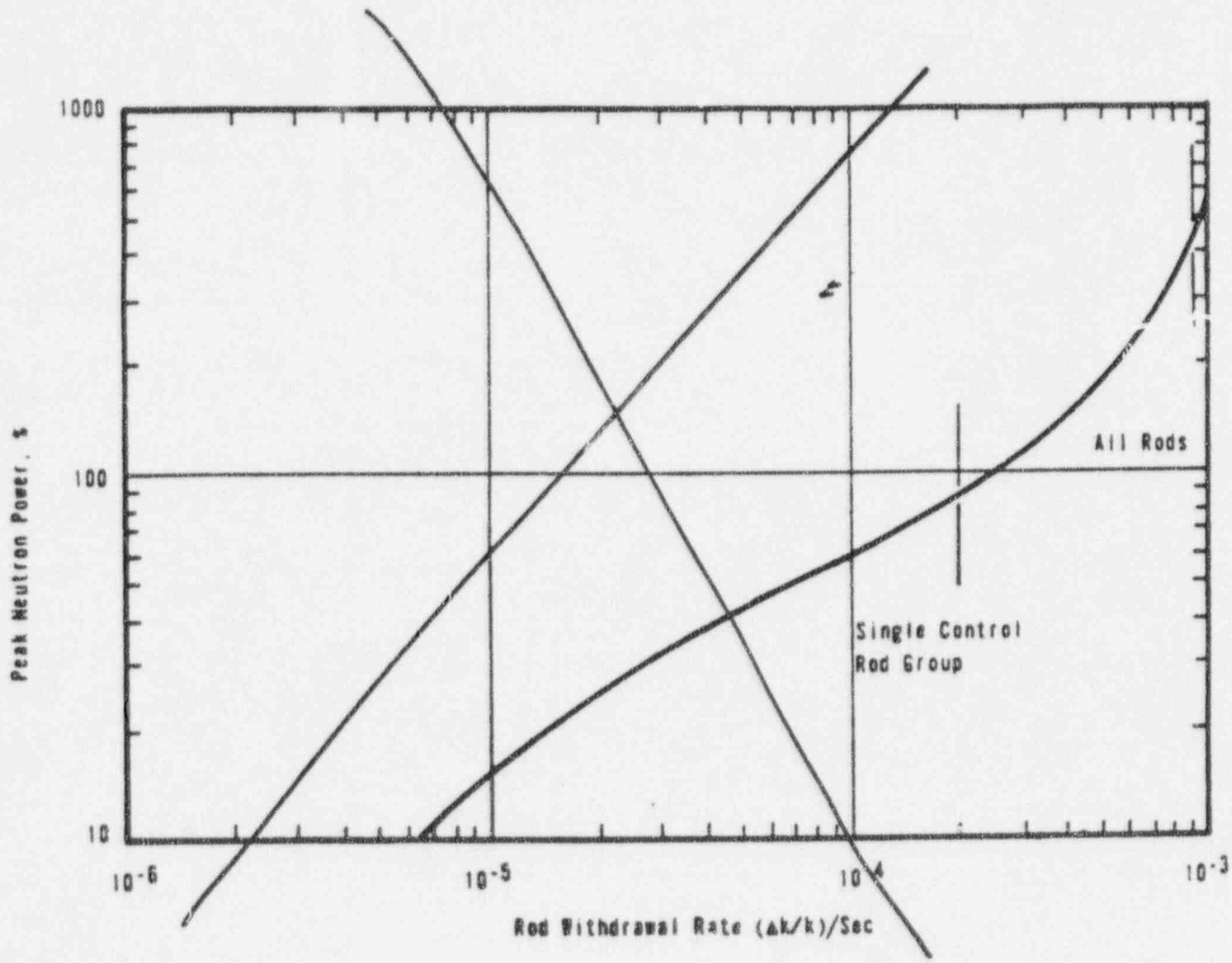


substitute (p 30 of Calc 94-E-0064-01)
Insert K

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	PEAK THERMAL POWER VS ROD WITHDRAWAL RATE FOR A STARTUP ACCIDENT FROM 10^{-9} RATED POWER	FIG. NO. 14-3


Peak Thermal Power VS Reactivity Addition Rate For A Startup Accident From 1 E-09 Rated Power; 3% Accumulation on PSV
Peak Thermal Power, %
Reactivity Insertion Rate, (dK/K)/sec
0
10
20
30
40
50
60
70
80
1.00E-05
1.00E-04
1.00E-03

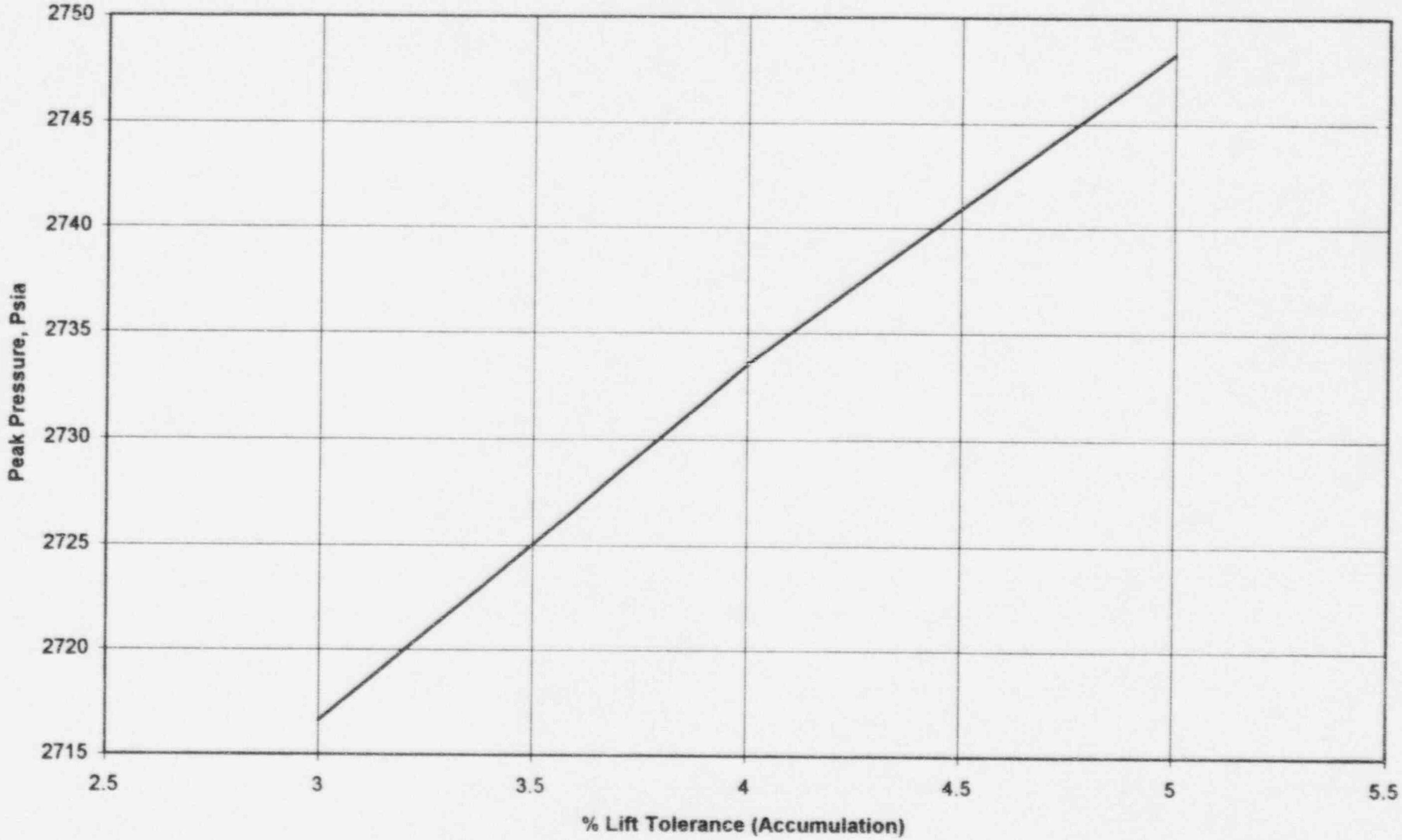
Insert K



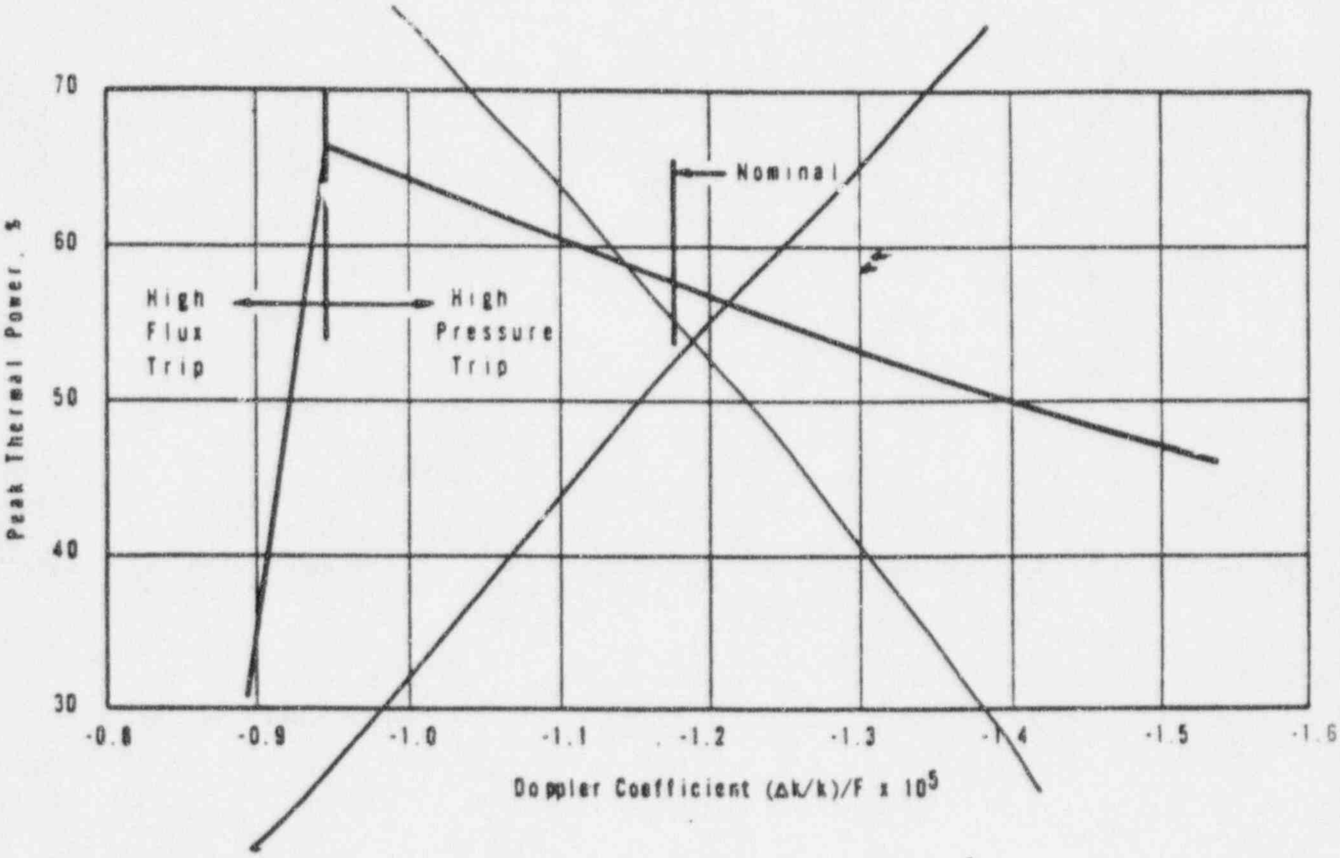
Insert (p 33 of calc 94-E-0064-01)
L

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	Peak Pressure Vs PSV Accumulation ~~PEAK NEUTRON POWER VS ROD WITHDRAWAL RATE~~ FOR A STARTUP ACCIDENT FROM 10^{-9} RATED POWER	FIG. NO. 14-4

Peak Pressure VS PSV Accumulation For A Startup Accident From 1 E-09 Rated Power
Two PSVs



Insert L

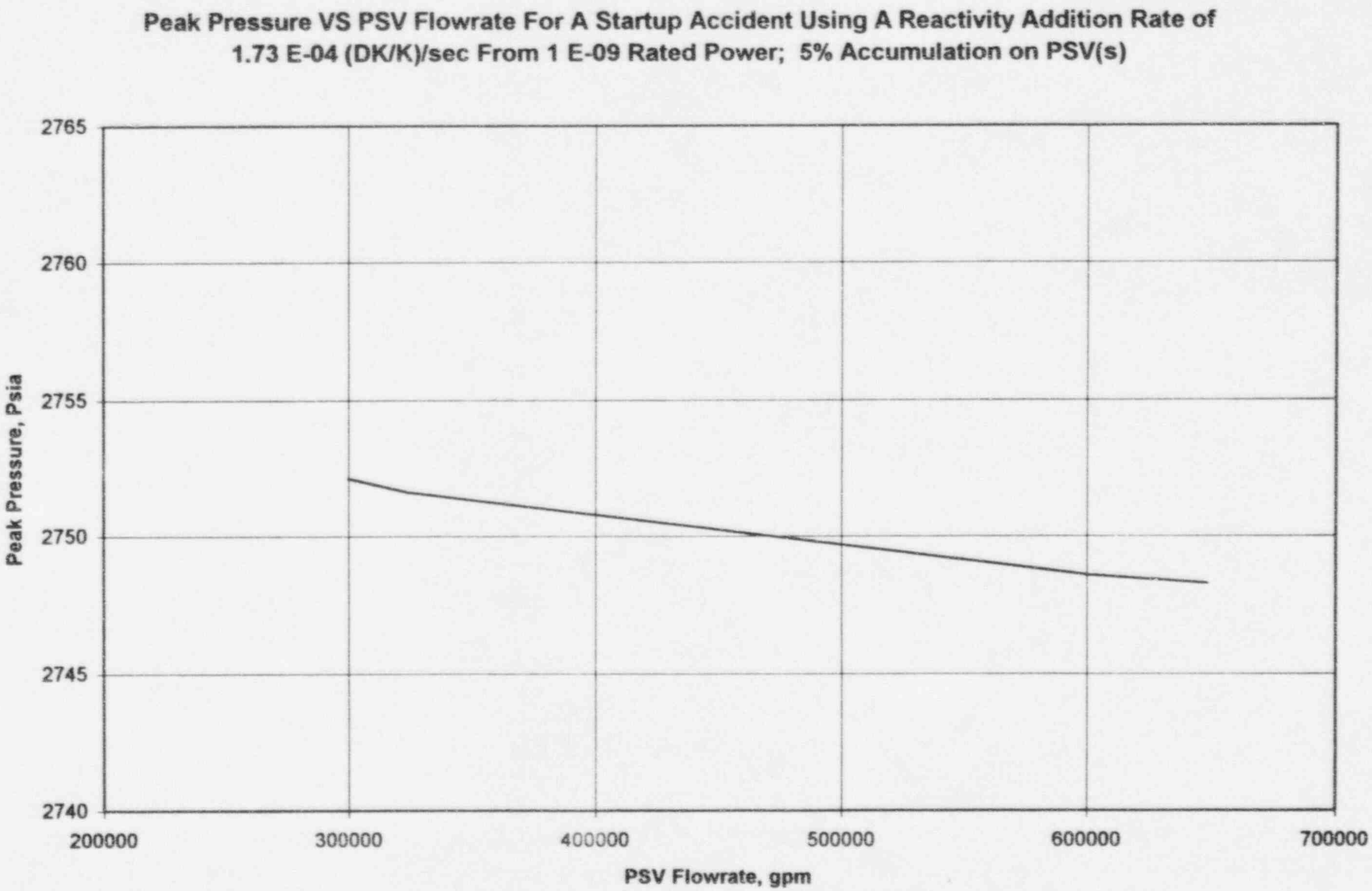


Insert M (P 41 of calc 94-E-0064-01)

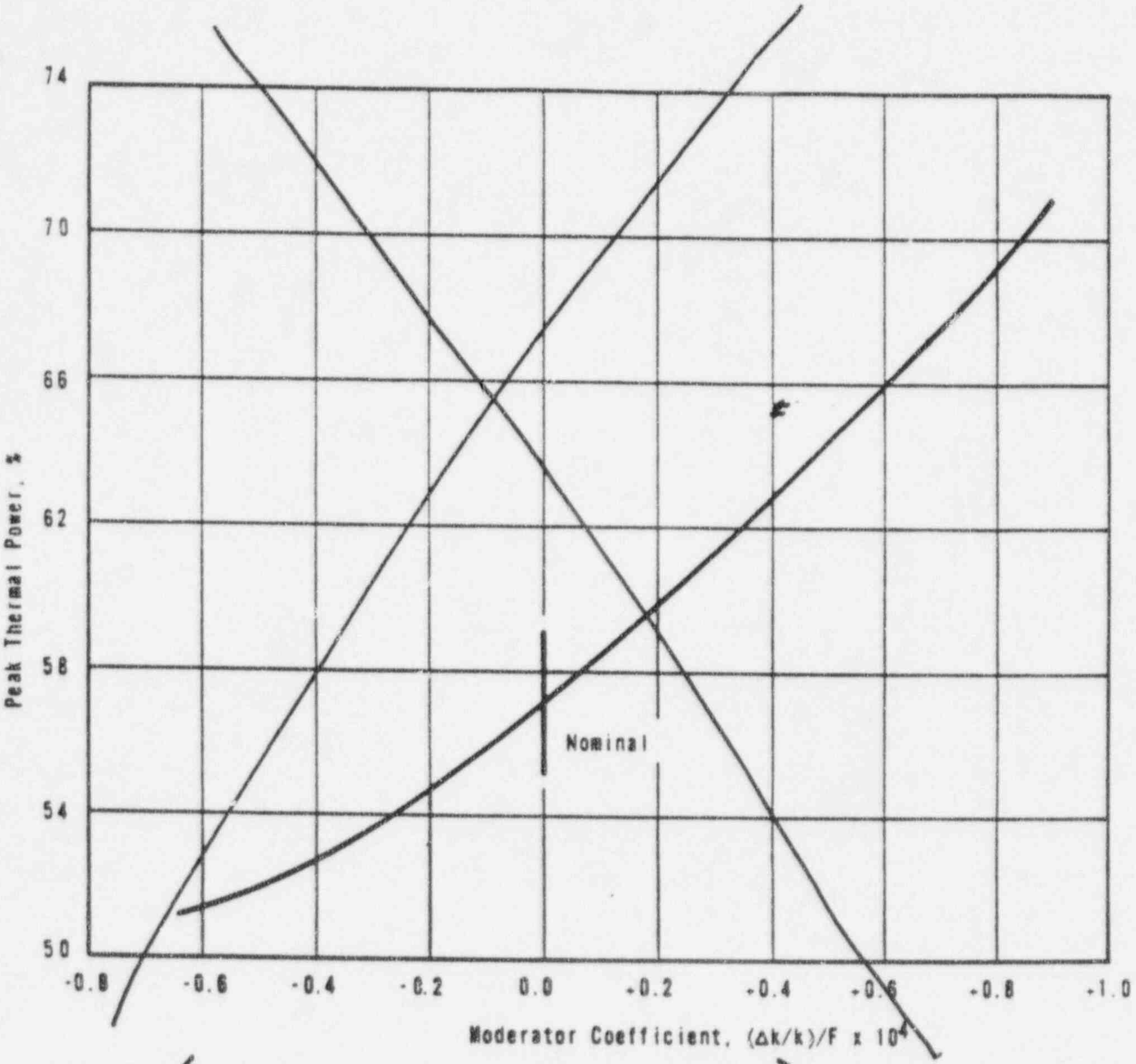
Peak Pressure Vs PSV Flowrate For A Startup Accident Using A Reactivity Addition Rate Of 1.73×10^{-4} ΔK/K/sec From 10^{-9} Rated Power

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	PEAK THERMAL POWER VS DOPPLER COEFFICIENT FOR A STARTUP ACCIDENT USING A 3.0% Δk/k ROD GROUP AT 2.15×10^{-4}(Δk/k)/s FROM 10^{-9} RATED POWER	FIG. NO. 14-5

press vs psv flow



Insert M

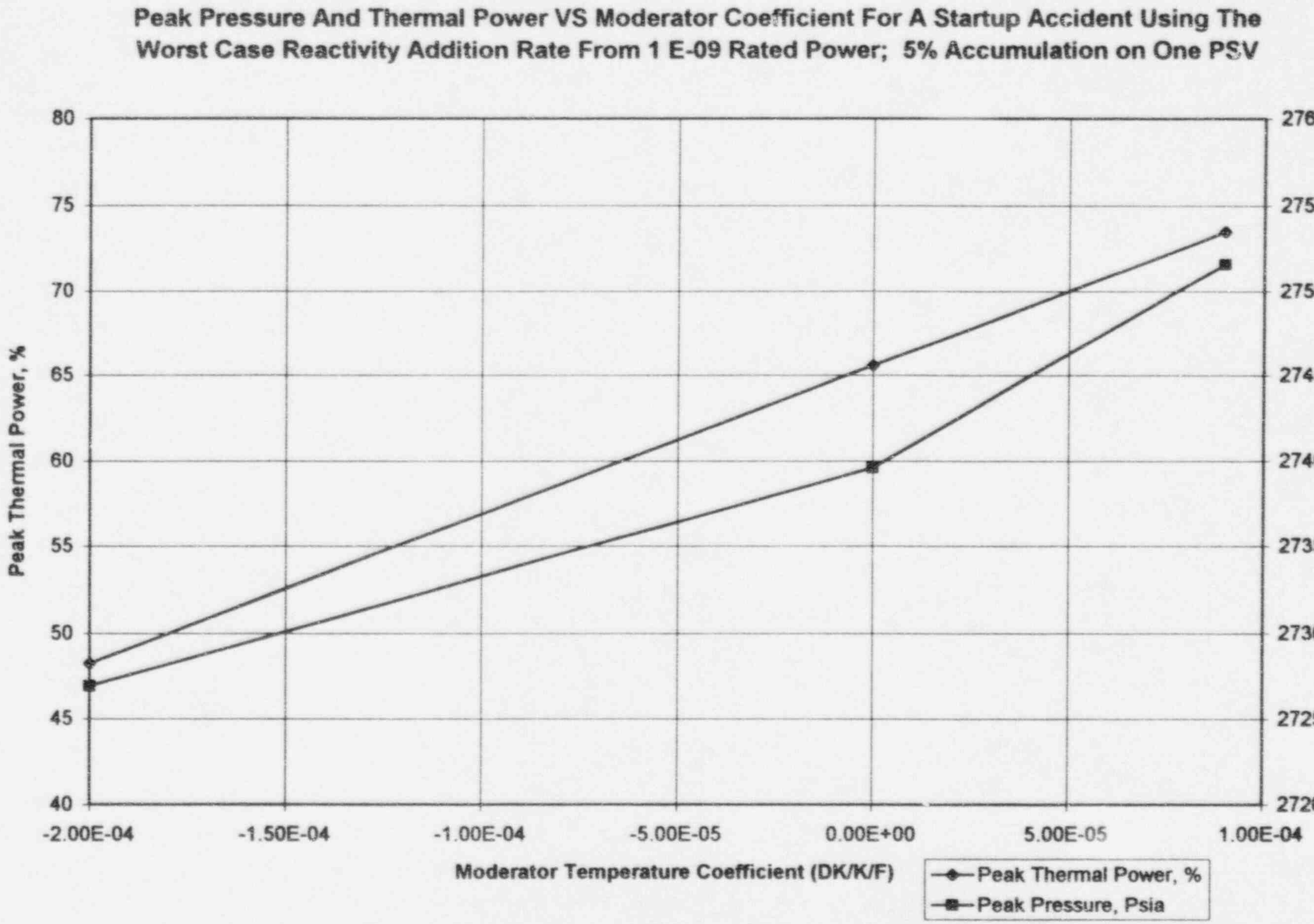


Insert (P 43 of calc 94-E-0064-01)

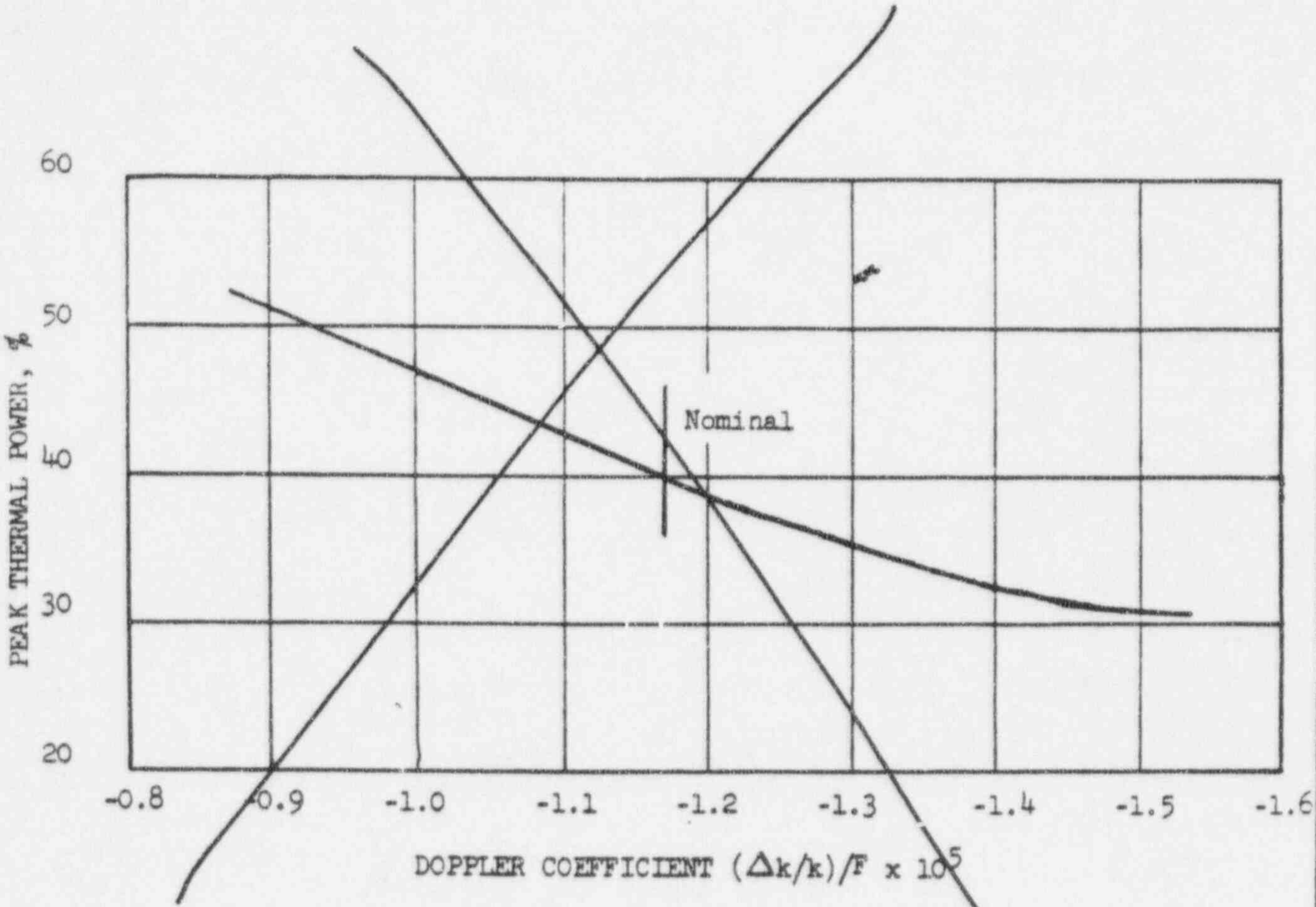
N ~~(add info that x axis units are "x 10^{-4} Δk/k/°F")~~

The Worst Case Reactivity Addition Rate

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	PEAK Pressure and THERMAL POWER VS MODERATOR COEFFICIENT FOR A STARTUP ACCIDENT USING ~~[illegible] ROD GROUP AT [illegible]~~ FROM 10-9 RATED POWER	FIG. NO. 14-6


Peak Pressure And Thermal Power VS Moderator Coefficient For A Startup Accident Using The Worst Case Reactivity Addition Rate From 1 E-09 Rated Power; 5% Accumulation on One PSV
Peak Thermal Power, %
Peak Pressure, Psia
Moderator Temperature Coefficient (DK/K/F)
80
75
70
65
60
55
50
45
40
2760
2755
2750
2745
2740
2735
2730
2725
2720
-2.00E-04
-1.50E-04
-1.00E-04
-5.00E-05
0.00E+00
5.00E-05
1.00E-04
Peak Thermal Power, %
Peak Pressure, Psia

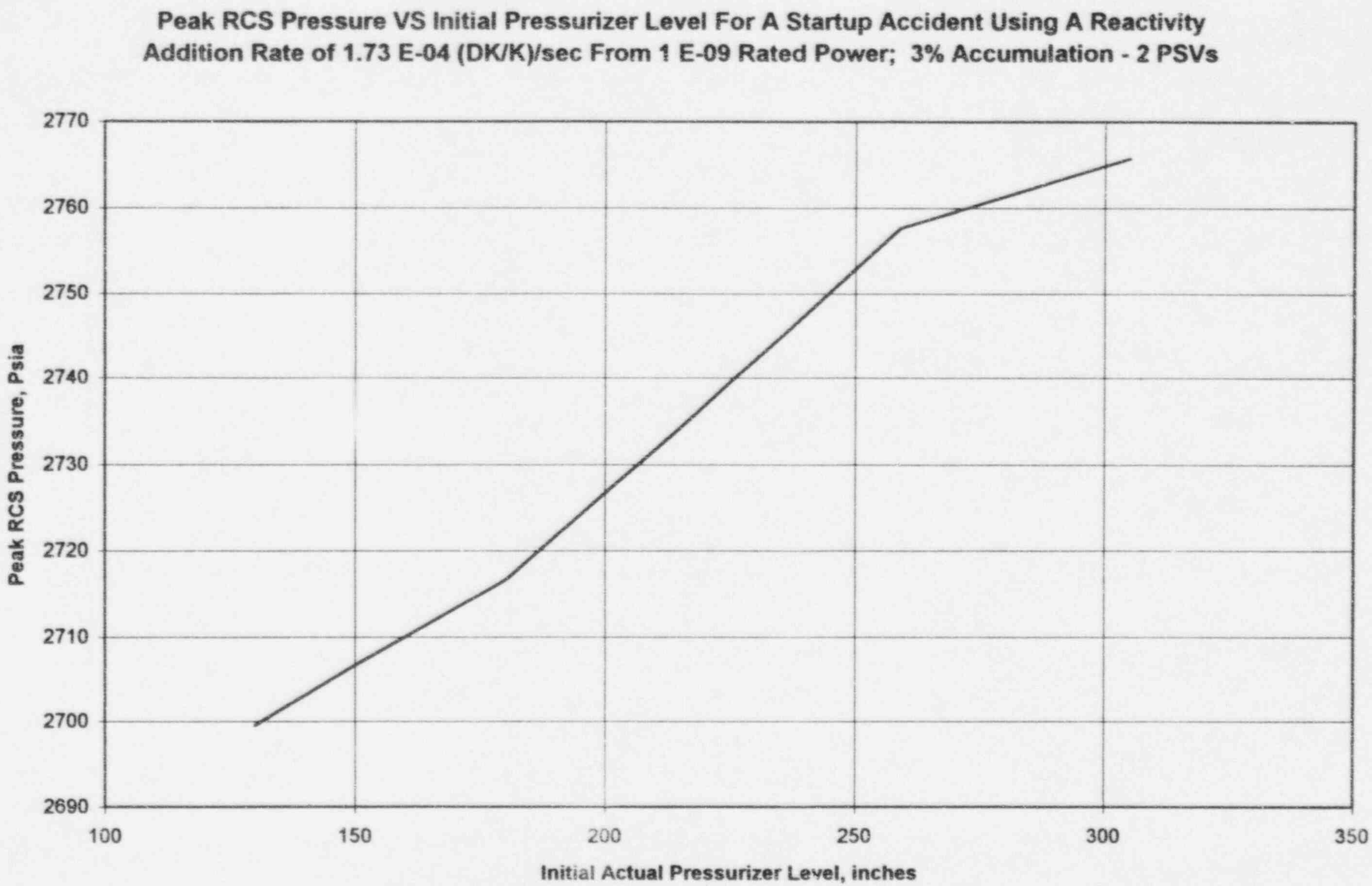
Insert N



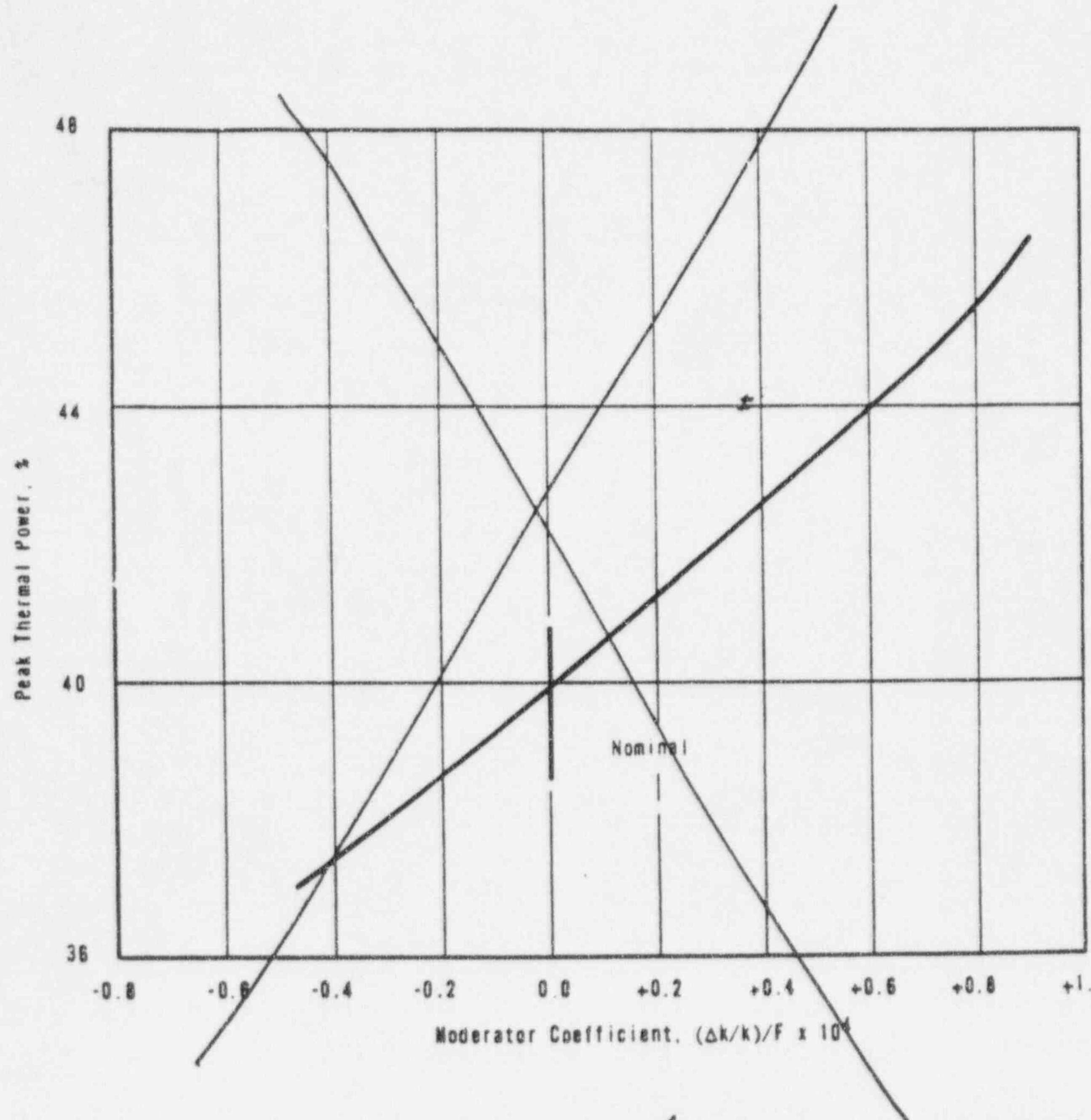
Insert (P17 of calc 94-E-0064-02)

Peak RCS Pressure Vs Initial Pressurizer Level For A Startup Accident Using A Reactivity Addition Rate Of 1.73×10^{-4} ΔK/K/sec from 10^{-9} Rated Power

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	PEAK THERMAL ~~POWER~~ VS DOPPLER COEFFICIENT FOR A STARTUP ACCIDENT USING ALL RODS AT 9.27×10^{-4} ~~(Δk/k)/s~~ FROM 10^{-9} RATED POWER	FIG. NO. 14-7


Peak RCS Pressure VS Initial Pressurizer Level For A Startup Accident Using A Reactivity Addition Rate of 1.73 E-04 (DK/K)/sec From 1 E-09 Rated Power; 3% Accumulation - 2 PSVs
Peak RCS Pressure, Psia
2770
2760
2750
2740
2730
2720
2710
2700
2690
100
150
200
250
300
350
Initial Actual Pressurizer Level, inches

Insert P

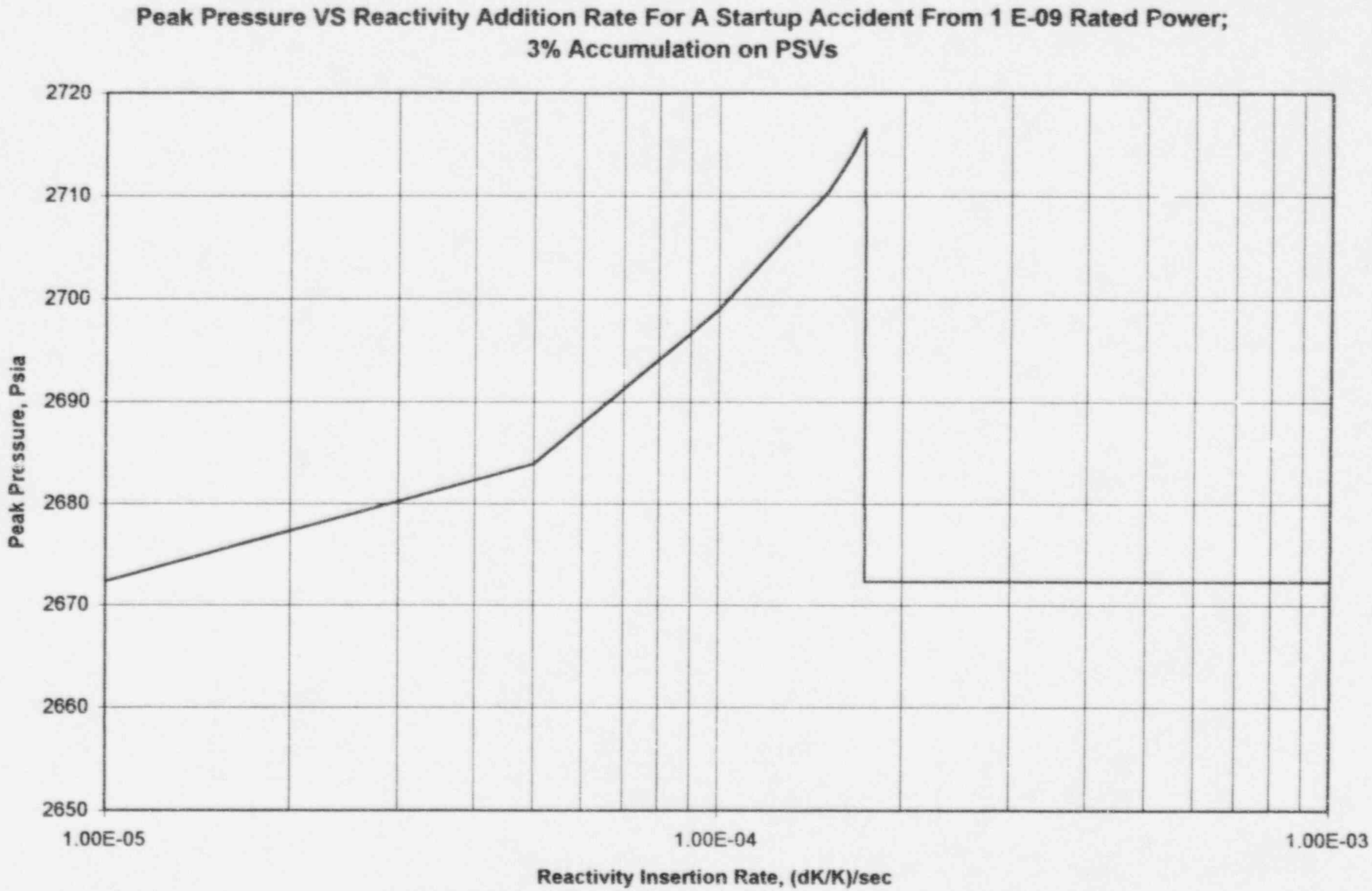


Insert (P 29 of Calc 94-E-0064-01)

~~Delete~~

Peak Pressure Vs Rod Withdrawal Rate for a Startup Accident From 10^{-9} Rated Power

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	~~PEAK THERMAL POWER VS MODERATOR COEFFICIENT FOR A STARTUP ACCIDENT USING ALL RODS AT 9.27×10^{-4} (Δk/k)/s FROM 10^{-9} RATED POWER~~	FIG. NO. 14-8


Peak Pressure VS Reactivity Addition Rate For A Startup Accident From 1 E-09 Rated Power;
3% Accumulation on PSVs
2720
2710
2700
2690
2680
2670
2660
2650
Peak Pressure, Psia
1.00E-05
1.00E-04
1.00E-03
Reactivity Insertion Rate, (dK/K)/sec

Insert Q