CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be ≤ 3.2 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. T_{avg} ≥ 525°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal of the reactor vessel head,

b. For specifically affected individuals CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

DESCRIPTION OF AMENDMENT REQUEST

ANO-2 Technical Specification (TS) 3.1.3.4 presently requires individual full length Control Element Assembly (CEA) drop time from a fully withdrawn position to be less than or equal to 3.0 seconds from the time the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position, with the Reactor Coolant System (RCS) average temperature greater than or equal to 525°F and all reactor coolant pumps operating. The maximum CEA drop time limit assures that the CEA drop time, and therefore the rate of negative reactivity insertion, is maintained consistent with that used in the ANO-2 Safety Analysis Report (SAR) accident analyses. The temperature and reactor coolant pump operating conditions specified assure that the measured drop times will be representative of insertion times experienced during operating conditions at power .

CEA drop times are measured in accordance with TS 4.1.3.4 requirements. A change in the measurement methodology used at ANO-2 has revealed that the indicated drop time for certain CEAs exceeds the 3.0 seconds specified by TS 3.1.3.4. The method used previously for measuring CEA drop time involved interrupting the power to the Control Element Drive Mechanisms (CEDMs) from each individual CEDM breaker. The new test method implemented during the sixth refueling outage (2R6) involves interrupting the power to all the CEDMs simultaneously via the Reactor Trip Breakers (RTBs). The CEAs and CEDMs are described in the ANO-2 SAR Section 4.2.3, and the reactor trip system is described in SAR Section 7.2.1 and Figures 7.2-5 and 7.2-7A.

Testing utilizing the new test method revealed an additional time delay factor due to the circuit configuration established when utilizing the RTBs. A detailed comparison of the testing methods and an explanation of this circuit phenomenon were provided in our letter dated May 5, 1988 (2CANØ588Ø1). It is important to note that the actual physical drop time of the CEAs has not increased, as demonstrated in the above referenced letter.

The proposed change would increase the specified maximum CEA drop time in TS 3.1.3.4 from 3.0 to 3.2 seconds to allow for the additional time delay factor in the new measurement methodology.

For those events that are potentially impacted by the proposed increase in allowable CEA drop times, most are in fact bounded by existing analyses as demonstrated below. Two events are specifically addressed by application of a penalty on the available reactor overpower margin (ROPM). This penalty will be implemented by increasing the Core Protection Calculator System (CPCS) addressable constant "BERR1" (DNBR power uncertainty multiplier) in accordance with Technical Specification 6.8.1.g. This increase will ensure that a CPCS DNBR trip will occur in sufficient time so that the conclusions of the affected analyses remain unchanged.

DETERMINATION OF SIGNIFICANT HAZARDS

AP&L has reviewed the ANO-2 SAR Design Basis Accident Analysis to determine the impact, if any, of an increased Control Element Assembly (CEA) drop time on the thirty-five (35) Chapter 15 events considered in the SAR. The results of this review are presented in two categories; those events that are unaffected by the proposed change in CEA drop time and those that are potentially impacted by the increased CEA drop time. In addition, two potentially affected analyses from Chapter 6 of the SAR, and one analyses from the ANO-2 Cycle 2 Reload Analysis are considered.

The accidents have been re-evaluated considering the currently approved analyses of record as defined by the Safety Analysis Report and cycle specific reload reports. Although not credited in the following evaluations it is noteworthy that significant additional conservatisms are available to support the final conclusions. First, as explained in AP&L letter dated May 5, 1988, (2CAN058801), the analyses model reactivity insertion assuming all rods are inserted at the Technical Specification limit, or slowest acceptable times. In reality, significant additional reactivity is inserted by virtue of the majority of CEAs which insert quicker than the Technical Specification limit. Second, many of the existing analyses utilize overly conservative inputs. For example, Beginning of Cycle full power events assume a positive moderator temperature coefficient (MTC) which is prohibited by the current Technical Specifications. A revised analysis crediting the proper MTC value would provide significantly more favorable results. Also most analyses assume a higher initial thermal power (2900 MW thermal) than allowed. Nonetheless, the following evaluations credit none of these conditions, demonstrating the significant conservatisms in the analyses.

Unaffected Design Basis Accidents

The review of the 35 Chapter 15 Design Basis Accidents determined that 23 accidents were unaffected by increase in CEA drop time. The bases for this determination are presented below.

Ten of the accident analyses do not predict a reactor trip or do not credit a reactor trip; therefore, the conclusions of the SAR analysis for these accidents are independent of the CEA drop time. The accidents which do not involve a reactor trip are listed below with the corresponding SAR Chapter 15 subsection number.

Idle Loop Startup (15.1.6)

Idle loop startup is defined as the startup of a reactor coolant pump, without observance of prescribed operating procedures, assuming that both reactor coolant pumps in that loop were idle. Note: the worst case for this event is at power with less than four RCP operating, which is outside the allowed ANO-2 operating conditions.

Major Rupture Of Pipes Containing Reactor Coolant Up To And Including Double-Ended Rupture Of Largest Pipe In The Reactor Coolant System (Loss of Coolant Accident) (15.1.13)

This analysis applies only to the dose consequences associated with LOCA's. These doses are based on preset accident source terms and are independent of reactor trip CEA drop time considerations. The more pertinent LOCA analysis for the purpose of demonstrating compliance with 10CFR50.46 (SAR Section 6.3.3) is described later.

Inadvertent Loading Of A Fuel Assembly Into The Improper Position (15.1.15)

Two accidents are considered in this section: (1) The erroneous loading of fuel pellets or fuel rods of different enrichment in a fuel assembly; and (2) The erroneous placement or orientation of fuel assemblies. The accidents are analyzed independent of any reactor trip considerations.

Waste Gas Decay Tank Leakage Or Rupture (15.1.16)

A rupture of a waste gas decay tank is analyzed to determine the limiting result from any malfunction in the gaseous waste system and is not affected by reactor trip considerations.

Break In Instrument Line Or Other Lines From Reactor Coolant Pressure Boundary That Penetrate Containment (15.1.22)

This analysis describes potential breaks in lines from the reactor coolant pressure boundary that penetrate containment. There are no instrument lines from the RCS that penetrate containment. The effects of increased CEA drop times for breaks in other lines (i.e. sample lines) are bounded by the LOCA Analysis addressed below.

Fuel Handling Accident (15.1.23)

This event assumes that a spent fuel assembly is dropped during fuel handling and the subsequent dose effects from the fission products released are analyzed; consequently, reactor trip considerations are not affected.

Small Spills Or Leaks Of Radioactive Material Outside Containment (15.1.24)

This analysis describes the effects of spills or leaks as they are postulated to occur, assuming transportation to ground water. It is apparent that reactor trip considerations are not involved.

Fuel Cladding Failure Combined With Steam Generator Leak (15.1.25)

This analysis describes the effects of the release of activity due to a secondary steam release. The effects are based on steam generator tube leaks, 1% fuel failure, and relief valve releases and are related to normal operations; consequently, reactor trip considerations are not involved.

Loss Of Service Water System (15.1.30)

This analysis describes the effects of postulated partial losses of the Service Water System (SWS). The analysis conclusions are that no single failure of the SWS can lead to a LOCA. Reactor trip considerations are not involved in the analysis of this event.

Inadvertent Operation Of ECCS During Power Operation (15.2.32)

This analysis describes the effects of inadvertent ECCS actuation while operating at power. There is no effect (and also no reactor trip) because RCS pressure exceeds the HPSI pump shutoff head by a significant margin.

Nine of the accidents involve or potentially involve a reactor trip, where the accident conditions are either slow to develop (i.e., specified acceptable fuel design limits (SAFDL) are not approached rapidly prior to reactor trip) or approach a SAFDL well after a trip, such that an increased CEA drop time of less than one-half of a second will not significantly change the conclusions of the analysis. The accidents that are not significantly affected are:

CEA Misoperation (15.1.3)

The effects of a full length CEA drop, a part length CEA drop, and a part length CEA subgroup drop are analyzed. The full length CEA drop event was recently reevaluated as part of the Core Protection Calculator (CPC) System Improvement Program (reference Technical Specification Amendment No. 70). As a part of this program, the CPC calculated penalty factors (PFs) for inward CEA deviations were eliminated (i.e., set equal to 1.0) in an effort to avoid unnecessary reactor trips. Appropriate operating margins and Technical Specification (TS) requirements for a power reduction following CEA drop were implemented to compensate for the PF reduction. Therefore, this event is insensitive to an increased CEA drop time since a reactor trip does not occur.

The single part length CEA insertion and the part length CEA subgroup insertion ANO-2 TSs have been revised (TS Amendment No. 37) to restrict the insertion of the part length CEAs. This change was made in order to reduce the analytical complexity of the CPC analyses. This change restricted insertion of the part length CEAs such that, at power levels above 50%, the reactivity inserted due to a single or subgroup part length CEA drop is always negative. For single part length CEA drops below 50% full power, sufficient margin to operating limits or the DNBR limit are maintained such that a reactor trip is not necessary. For part length CEA subgroup drops, a DNBR reactor trip does occur. Sufficient operating margin is available such that the SAFDLs are not challenged so that the proposed increase in the CEA drop time has no effect upon the conclusions of this analysis.

Uncontrolled Boron Dilution Incident (15.1.4)

The effect of a gradual reduction in the reactor coolant boron concentration is analyzed. This event is slow enough to allow the operator to be alerted to the effects of the dilution by a reactor trip and take corrective action before significant shutdown margin is lost. Therefore, an additional time of less than one-half of a second in the CEA drop time will not significantly affect the analysis conclusions.

Loss Of Normal Feedwater Flow (15.1.8)

The effects of an instantaneous complete loss of main feedwater flow to both steam generators and the subsequent low steam generator water level reactor trip are analyzed. The dynamic response of the primary and secondary systems to a loss of main feedwater flow event is such that the event is considered slow to develop in comparison to CEA rod insertion time considerations. The reactor trip occurs at approximately 36 seconds into the event and emergency feedwater is not credited for 65 seconds. The analysis demonstrates that if emergency feedwater is delayed for an additional 25 seconds, the consequences of the event are still acceptable. Therefore, it is apparent that a delay in CEA drop time of less than one-half of a second has virtually no effect upon the conclusions of the analysis.

Failure Of The Regulating Instrumentation (15.1.11)

Malfunction or failure of regulating systems could result in a reactor trip in the event a core SAFDL is approached. These events do not challenge the safety limits to the extent that an increased CEA drop time of less than one-half of a second would significantly change the analysis conclusions.

Internal And External Events Including Major And Minor Fires, Floods, Storms, And Earthquakes (15.1.12)

This analysis describes improbable naturally occurring events and events caused by mechanical or electrical failure of plant components. The effects of the events are various in nature. The worst case effect is a plant trip. For these events the plant safety limits are not challenged, and the conclusions of the analyses are unaffected by a increase of up to one-half second in CEA drop time.

Control Room Uninhabitability (15.1.26)

This event postulates conditions requiring evacuation of the control room and analyzes the subsequent effects. During control room evacuation, the operator is credited with tripping the reactor. An increased CEA drop time of less than one-half of a second has a negligible effect on the analysis conclusions due to the much larger times assumed for operator action.

Failure Or Overpressurization Of Low Pressure Residual Heat Removal System (15.1.27)

Significant failures of the shutdown cooling system would typically occur well after reactor trip and therefore would not be affected by an increased CEA drop time of less than one-half of a second. An overpressurization of the system during normal operations could cause a rupture of the piping and subsequent reactor trip; however, this event is bounded by the LOCA analyses discussed below.

Loss Of One DC System (15.1.31)

This analysis considers the effects of a loss of one DC System. The conclusions are that, due to the redundant nature of the system, a safe plant shutdown will occur. The conclusions of the analysis are not affected by increases in the CEA drop time.

Loss Of Instrument Air System (15.1.34)

This analysis describes the effects of a loss of instrument air due to events such as failure of air compressors or component rupture. The worst case event would result in a plant trip. However, during this transient, the SAFDLs are not challenged and an increase of up to one-half second in CEA drop time does not affect the conclusions of the analysis.

Three of the accidents are applicable to boiling water reactors only. Since ANO-2 is a pressurized water reactor, no accident analysis was performed in the SAR. The accidents are:

Failure Of Air Ejector Lines (BWR). (15.1.17)

Failure Of Charcoal Of Cryogenic System (BWR). (15.1.19)

Spectrum Of Rod Drop Accidents (BWR). (15.1.21)

Potentially Impacted Design Basis Accidents

The following eleven Chapter 15 events (plus the Asymmetric Steam Generator Transient) involve a rapid approach to a safety limit during the same time frame as the scram. The review of these analyses involves the same analysis technique used for the low power events described in AP&L letter to the NRC dated May 5, 1988 (2CAN058801). Briefly, this involves comparing the design "scram reactivity versus time" data used in the docketed analyses to the revised "scram reactivity versus time" which incorporates the increased CEA drop time (3.2 seconds to 90% inserted) and comparisons to space-time neutronics methods. The space-time neutronics methods are discussed in CE Topical Reports "HERMITE Space Time Kinetics", CENPD-188-A, March 1976 and "FIESTA One Dimensional Two Group Space Time Kinetics Code for Calculating PWR Scram Reactivities", CEN-122, November 1979. The detailed review shows that for these events the revised scram reactivity versus time data is conservative relative to the design reactivity versus time data at the crucial time in the transient, during the closest approach to a safety limit.

Uncontrolled CEA Withdrawal From A Subcritical Condition (15.1.1)

The withdrawal of CEAs from subcritical conditions adds reactivity to the reactor core and in conjunction with a positive moderator reactivity coefficient causes both the core power level and the core heat flux to increase.

This event approaches a SAFDL (minimum DNBR) at approximately 2.2 seconds after the trip breakers open (Table 7.1.6-2 of ANO-2 Cycle 2 Reload Analysis Report (RAR)) As shown in Table 1, and illustrated in Figure 1, the revised scram reactivity data for a total scram insertion of 5.0 %Δρ and a +0.6 ASI (Table 7.1.6-1 of the Cycle 2 RAR) at the time of approach to a SAFDL, is more conservative than the design data. Therefore, the conclusions of this event are unchanged. This event was previously evaluated for the effects of increased CEA drop times (using CEA drop times of 3.18 seconds) by AP&L letter 2CAN058801 dated May 5, 1988.

Uncontrolled CEA Withdrawal From Critical Conditions, 1% Power (15.1.2)

This event approaches a SAFDL (minimum DNBR) at approximately 2.2 seconds after the trip breakers open (Table 7.1.6-4 of Cycle 2 RAR). As shown in Table 1, and illustrated in Figure 1, the revised scram reactivity data for a total scram insertion of 5.0 %Δρ and a +0.6 ASI (Table 7.1.6-3 of Cycle 2 RAR), at the time of approach to a SAFDL is more conservative than the design data. Therefore, the conclusions of this event are unchanged. This event was previously evaluated for the effects of increased CEA drop time (using CEA drop times of 3.18 seconds) by AP&L letter 2CAN058801 dated May 5, 1988.

Uncontrolled CEA Withdrawal From Critical Conditions, 100% Power (15.1.2)

The use of space-time scram reactivity data is not sufficient to completely offset the increase in the CEA drop time. This event is discussed further in the next section.

Total And Partial Loss Of Reactor Coolant Forced Flow (15.1.5)

The loss of forced reactor coolant flow event from 100 percent power produces a rapid decrease in core heat removal from the fuel due to the decrease in the core coolant flow and consequentially an approach to the DNBR limit.

a. Four Pump Coastdown

This event is initiated from the simultaneous loss of power to all four reactor coolant pumps resulting in the coastdown of the forced reactor coolant flow. This event approaches a SAFDL (minimum DNBR) at approximately 2.15 seconds after the trip breakers open (Table 7.1.8-2 of Cycle 2 RAR). The results of this transient, COLSS required overpower margin, are handled parametrically on Axial Shape Index (ASI) from within the Technical Specification limiting condition for operation (LCO) of +0.3 to -0.3 ASI. As shown in Tables 3 through 5, and illustrated in Figures 3 through 5, the revised scram reactivity data at the time of approach to a SAFDL is more conservative than the design data. The data represents a total scram insertion of 8.0 $\%\Delta\rho$ and the full range of the ASI LCO. Therefore, the conclusions of this event are unchanged.

b. Seized Rotor

This event is initiated by the rapid stop (seizure) of one reactor coolant pump rotor and results in the rapid coastdown of core flow to the asymptotic 3-pump flow rate. The currently approved analysis predicts that the event will exceed a SAFDL (minimum DNBR) and fail a small amount of fuel. The failed fuel calculation is done at asymptotic flow conditions and is therefore insensitive to a small delay in the CEA drop time. However, as shown in Table 6, and illustrated in Figure 6, the revised scram reactivity data for a total scram insertion of 8.0 $\%\Delta\rho$ and a 0.0 ASI (Table 7.2.5-2 of Cycle 2 RAR), at the time of approach to a SAFDL is more conservative than the design data. Therefore, the conclusions of this event are unchanged.

Loss Of External Load And/Or Turbine Trip (15.1.7)

The effects of increased CEA drop times upon this event are the same as the Loss of Condenser Vacuum event presented below (15.1.28).

Loss Of All Normal And Preferred AC Power To The Station Auxiliaries (15.1.9)

The effects of increase CEA drop times upon this event are the same as the Loss of Flow event (four pump coastdown) presented above (15.1.5).

Major Secondary System Pipe Breaks With Or Without A Concurrent Loss Of AC Power (15.1.14)

A break in the main feedwater system piping results in a rapid heatup of the primary system due to the loss of the secondary heat sink. This heatup will cause a rapid increase in the primary system pressure.

A break in the main steam system piping results in a cooldown of the primary system. This cooldown, in conjunction with a negative moderator temperature coefficient of reactivity, results in a positive reactivity addition and causes reactor power to increase.

a. Feedwater Line Break

This event approaches the upset pressure limit (i.e. the RCS pressure safety limit). At approximately 2.7 seconds after the trip breakers open, the reactor power has decreased enough to terminate the event. The time of peak RCS pressure is 3.2 seconds after the trip breakers open. As shown in Table 7, and illustrated in Figure 7, the revised scram reactivity data for a total scram insertion of 7.0 $\%\Delta\rho$ and a +0.3 ASI (Table 7.2.3-1 of Cycle 2 RAR), at the time of interest is more conservative than the design data. Therefore, the conclusions of this event are unchanged.

b. Steam Line Break

This event approaches a SAFDL (minimum DNBR) well after a trip such that an increase in CEA drop time of one-half of a second will not significantly change the conclusions of the Chapter 15 analysis. The analyses also demonstrate that the critical times for the "return to power" concern occur well into the transient significantly after reactor trip has occurred; therefore, the effects of increased CEA drop time of less than one-half of a second would not significantly affect the conclusions of the return to power analyses.

c. Steam Line Break With Loss of AC

With the loss of AC, the transient behavior of this event prior to reactor trip is covered by the Loss of Reactor Coolant Flow event presented above (15.1.5). The post trip return to power considerations for the Steam Line Break with loss of AC event are the same as for the previous case (no loss of AC).

Steam Generator Tube Rupture With Or Without Loss Of AC Power (15.1.18)

The penetration of the barrier between the RCS and the main steam system due to a steam generator (SG) tube rupture is analyzed. For a SG tube rupture without a loss of AC power (i.e., AC power available), a low DNBR trip prevents the DNB safety limit from being exceeded during this transient. Due to the relatively slow decrease in DNBR (~1/75 DNBR unit/sec.), the additional CEA drop time of less than one-half of a second would have negligible effect on this analysis.

For a SG tube rupture with a loss of AC power the effects of an increased CEA drop time upon this event are the same as the "Loss of Flow" event (four pump coastdown) presented above.

CEA Ejection (15.1.20)

The rapid ejection of a CEA from the core causes the reactor power to rapidly increase for a brief period before the power rise is terminated by Doppler feedback. A reactor trip limits the maximum enthalpy in the fuel during the transient.

a. From 0 Percent Power

   The time of maximum deposited energy is at approximately 2.5 seconds after the trip breakers open (Figure 7.2.1-2 of Cycle 2 RAR). As shown in Table 1, and illustrated in Figure 1, the revised scram reactivity data for a total scram insertion of 2.4 $\%\Delta\rho$ and a +0.6 ASI (Table 7.2.1-1 of Cycle 2 RAR), at the time of interest, is more conservative than the design data. Therefore, the conclusions of this event are unchanged.

b. From 100 Percent Power

   The time of maximum deposited energy occurs approximately 2.2 seconds after the trip breakers open (Figure 7.2.1-1 of Cycle 2 RAR). As shown in Table 8, and illustrated in Figure 8, the revised scram reactivity data for a total scram insertion of 5.4 $\%\Delta\rho$ and a -0.3 ASI (Table 7.2.1-1 of Cycle 2 RAR), at the time of interest, is more conservative than the design data. Therefore, the conclusions of this event are unchanged.

Loss Of Condenser Vacuum (15.1.28)

This event is initiated by a turbine trip due to a loss of condenser vacuum without a simultaneous reactor trip, and assumes that the main feedwater pump steam turbines trip at the same time. The loss of load causes steam generator pressure to increase to the opening pressure of the main steam safety valves. The reduction of the secondary heat sink leads to a heat up of the RCS and in the presence of a positive MTC, an increase in core power. This event approaches the upset pressure limit (maximum RCS pressure design safety limit) at approximately 2.6 seconds after the trip breakers open, (Table 7.1.4-2 of Cycle 2 RAR). As shown in Table 2, and illustrated in Figure 2, the revised scram reactivity data for a total scram insertion of 5.4 $\%\Delta\rho$ and a +0.3 ASI, (Table 7.1.4-1 of Cycle 2 RAR), at the time of interest is more conservative than the design data. Therefore, the conclusions of this event are unchanged.

Turbine Trip With Coincident Failure Of Turbine Bypass Valves To Open (15.1.29)

This event is bounded by the Loss of External Load and/or Turbine Trip (15.1.7) and is covered by the description above for the Loss of Condenser Vacuum (15.1.28)

Turbine Trip With Failure Of Generator Breaker To Open (15.1.33)

Due to the redundant 500 KV breaker failure relaying scheme provided as part of the switchyard protection scheme, this event would not differ significantly from the turbine trip considered in Item 7 (Loss of External Load and/or Turbine Trip). In any case, the differences are unrelated to considerations of the associated reactor trip; therefore, the conclusions of the analysis are unaffected.

Malfunction Of Turbine Gland Sealing System (15.1.35)

This analysis describes the effects of a malfunction of the turbine gland sealing system. The worst case assumes that a loss of condenser vacuum occurs, resulting in a turbine trip and subsequent reactor trip. The event is already discussed as item 15.1.28. In any case, for this transient, a delay of less than one-half second in CEA drop time does not affect the conclusions of the analysis.

The following two events involve a rapid approach to a SAFDL (minimum DNBR) during the first part of the scram insertion. Even for a top peaked power shape, -0.3 ASI, there is insufficient CEA insertion for space-time neutronic adjustments to totally offset the increased holding coil delay time. Hence, a CPC DNBR power uncertainty penalty will be applied to ensure a trip will occur earlier than in the referenced analyses.

Uncontrolled CEA Withdrawal From Critical Conditions, 100% Power (15.1.2)

This event approaches a SAFDL (minimum DNBR) at approximately 0.9 seconds after the trip breakers open, (Table 7.1.6-6 of Cycle 2 RAR). As shown in Table 2 and illustrated in Figure 2, the revised scram reactivity data, which incorporates the increase CEA drop time of 3.2 seconds and the space-time neutronic adjustment, for a total scram insertion of 5.4 $\%\Delta\rho$ and a +0.3 ASI, (Table 7.1.6-5 of Cycle 2 RAR), at the time of approach to a SAFDL is less conservative than the design data.

If the time of CEA insertion were delayed by 0.3 seconds (i.e., no credit for space-time neutronic adjustments) the maximum increase in the core average heat flux would be less than 0.4% relative to the value in Table 7.1.6-6 of Cycle 2 RAR. Therefore, the CPC DNBR power uncertainty multiplier (BERR1), a CPC addressable constant, will be conservatively increased by a factor of 1.005 in accordance with Technical Specification 6.8.1.g. This will ensure that a CPC DNBR trip will occur at least 0.3 seconds earlier than the trip time presented in Cycle 2 RAR. This is more than sufficient to offset the effect of the increased holding coil decay time so that the conclusions of the event remain unchanged.

Excess Heat Removal Due To Secondary System Malfunction (15.1.10)

Excess heat removal due to a secondary system malfunction causes a decrease in the temperature of the reactor coolant, an increase in reactor power due to the negative moderator temperature coefficient and a decrease in the RCS and steam generator pressures.

The limiting Excess Heat Removal event is the Increased Main Feedwater Event. As shown on Table 15.1.10-3A of the ANO-2 SAR, this event generates a reactor trip on high power well before the DNBR SAFDL is violated so that the increased holding coil decay time does not significantly affect the results of the analysis.

Notwithstanding the justification provided above, the 1.005 penalty which will be applied to the CPC addressable constant BERR1 as discussed above is more than sufficient to ensure that a CPC trip will be generated sufficiently early to totally compensate for the increased holding coil decay time. Therefore, the conclusions of this analysis are unaffected.

Other Analyses

In addition to the Chapter 15 Design Basis Events, several other applicable analyses have been assessed for the effect of the increased CEA drop time and are individually addressed below.

Loss Of Coolant Accidents (SAR section 6.3.3)

In order to demonstrate compliance with 10CFR50.46, extensive analyses were conducted to evaluate the performance of the Emergency Core Cooling System (ECCS) in response to the loss of coolant accidents. The analyses are divided into large break cases and small break cases. For the large breaks, the CEAs are not required to be credited since the reactor is shut down by voids and subsequent borated water injection. For small breaks, the time to reactor trip is reached relatively early in the transient whereas the critical times relate to core uncovery and fuel pin heatup concerns which occur well after reactor trips. Therefore, an increased CEA drop time of less than one-half of a second has virtually no effect upon the analyses or their conclusions.

Asymmetric Steam Generator Transient (ASGT) (Cycle 2 Reload Analysis Report)

This transient results from the instantaneous closure of a single Main Steam Isolation Valve (MSIV). Upon loss of load, pressure and temperature in the affected steam generator increase to the opening pressure of the main steam safety valves. The intact steam generator picks up the lost load, causing its temperature and pressure to decrease. The resulting temperature tilt in the presence of a negative moderator temperature reactivity coefficient causes a core power tilt and an increase in the maximum power peaking factors. This event approaches a SAFDL (minimum DNBR) at approximately 2.3 seconds after the trip breakers open (Table 7.1.10-2 of Cycle 2 RAR). As shown in Table 2, and illustrated in Figure 2, the revised scram reactivity data for a total scram insertion of 5.4 $\%\Delta\rho$ and a +0.3 ASI, at the time of approach to a SAFDL is more conservative than the design data. Therefore, the conclusions of this event are unchanged.

Containment Pressure Analysis (SAR Section 6.2)

The peak pressure analyses address the response of the containment to LOCAs and Main Steam Line Breaks. As indicated in the SAR (section 6.2.1.3.3.3), the LOCA (9.82 sq. ft. double ended pump suction break) is the controlling "Containment DBA" with a peak calculated pressure of 53.4 psig. An increased CEA drop time does not affect the LBLOCA response as described above. The steam line break case is potentially impacted by an increased CEA drop time; however, the increase in mass/energy into the containment has been conservatively estimated and assessed for impact upon the peak containment pressure. The increase in mass/energy release represents approximately .3% of the total used in the peak pressure analyses. Consequently the resultant peak pressure increase has been estimated to be much less than 1 psig, and is still within the existing margin to the containment design pressure limit of 54 psig and less than the limiting LOCA analysis value of 53.4 psig. Therefore, the conclusions of this analysis are not considered to be significantly affected by the proposed increase in CEA drop time.

Arkansas Power & Light has performed an evaluation of the proposed TS change in accordance with 10CFR50.91(a)(1) regarding significant hazards consideration 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 proposed technical specification merely changes the time requirements for insertion of CEA's upon receipt of a reactor trip signal. The increase from 3.0 seconds to 3.2 seconds has been evaluated for impact on the affected analyses for ANO-2 as previously described. Since the change affects only an acceptance criteria for the CEA drop time requirement and involves no material aspect of the plant configuration, the proposed change does not affect the probability of occurrence of any accident previously evaluated.

The previous discussion of applicable analyses demonstrated that the events are either totally unrelated to CEA drop time considerations or are not significantly impacted. The evaluation demonstrated for each potentially impacted analysis that the consequences of the analysis remain unchanged or are bounded by the existing analysis. The conclusions were based largely on the demonstration of significant conservatism within the analytical inputs such that the effect of the increased CEA drop time was shown to be offset. In one case ("Uncontrolled CEA Withdrawal from a Critical Condition" - 100% Power), the effect of the increased drop time is addressed by an increase of the CPC DNBR power uncertainty multiplier (BERR1) which effectively provides for a quicker reactor trip in response to this event, thus offsetting the longer CEA drop times. Consequently, it has been demonstrated that the proposed change does not involve a significant increase in the 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 change does not involve any new or modified structures, systems, or components; rather, it affects only an acceptance criteria for confirming the required performance of the existing CEA hardware. Therefore, the proposed change would not create the possibility of a new of different kind of accident from any previously evaluated.

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

The margins of safety related to CEA insertion are defined by the analyzed events in the Safety Analysis Report which credit their insertion. As demonstrated in Criterion 1 above, evaluation of each affected analysis confirmed that the previously accepted results were either preserved or not significantly affected. Therefore, it is apparent that the margins of safety reflected in the analytical conclusions are not significantly reduced.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists. This guidance includes examples (51FR7750) of types of amendments that are considered not likely to involve significant hazards considerations. The change proposed in this amendment is not directly comparable to any of the examples identified in 51FR7750.

Based on the above evaluation, AP&L has determined that the proposed change does not involve a significant hazards consideration.

TABLE 1

SCRAM REACTIVITY VERSUS TIME

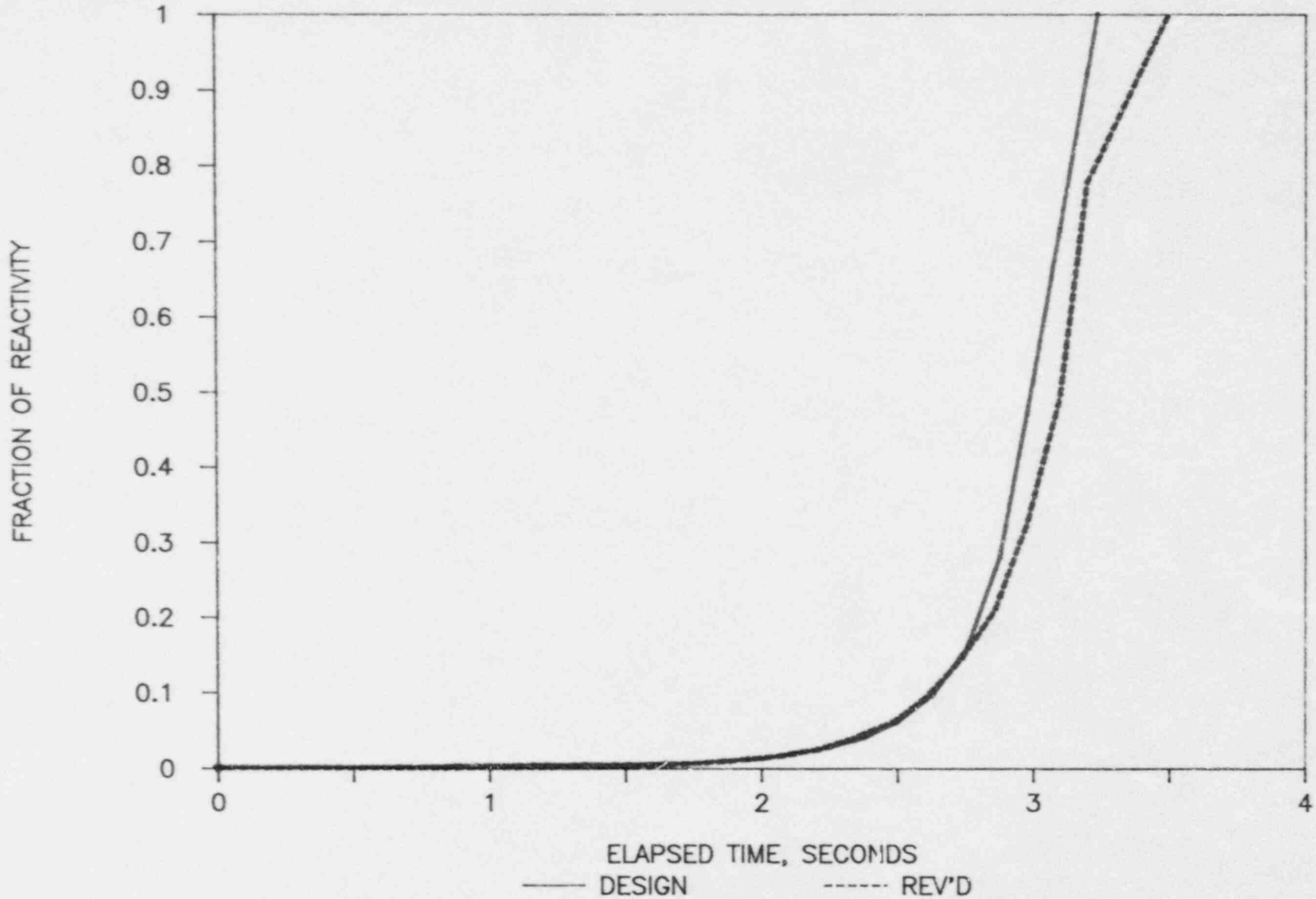
UNCONTROLLED CEA WITHDRAWAL - LOW POWER*

TIME SEC	DESIGN REACTIVITY FRACTION	REVISED REACTIVITY FRACTION
0	0.0000	0.0000
0.30	0.0004	0.0000
0.60	0.0008	0.0000
0.66	0.0009	0.00009
0.84	0.0024	0.00011
0.88	0.0027	0.00012
1.00	0.0035	0.0003
1.05	0.0037	0.0004
1.15	0.0040	0.0007
1.21	0.0041	0.0008
1.31	0.0042	0.0012
1.35	0.0043	0.0014
1.46	0.0045	0.0024
1.51	0.0048	0.0028
1.60	0.0052	0.0038
1.65	0.0057	0.0043
1.72	0.0064	0.0057
1.80	0.0077	0.0072
1.86	0.0086	0.0090
1.95	0.0108	0.0117
2.00	0.0120	0.0136
2.08	0.0164	0.0167
2.11	0.0180	0.0186
2.21	0.0244	0.0250
2.25	0.0270	0.0296
2.34	0.0360	0.0400
2.38	0.0400	0.0458
2.49	0.0593	0.0617
2.50	0.0610	0.0645
2.61	0.0906	0.0950
2.63	0.0960	0.1028
2.73	0.1452	0.1417
2.76	0.1600	0.1571
2.86	0.2600	0.2083
2.88	0.2800	0.2278
2.98	0.4717	0.3250
3.00	0.5100	0.3528
3.10	0.7142	0.4917
3.20	0.9183	0.7767
3.24	1.0000	0.8065
3.50	1.0000	1.0000

*Table 1 & Figure 1 are applicable to both subcritical and 1% power cases for Uncontrolled CEA Withdrawal and to CEA Ejection at 0%.

FIGURE 1

UNCONTROLLED CEA WITHDRAWAL (LOW POWER)



CEA REACTIVITY VS. TIME

TABLE 2

SCRAM REACTIVITY VERSUS TIME

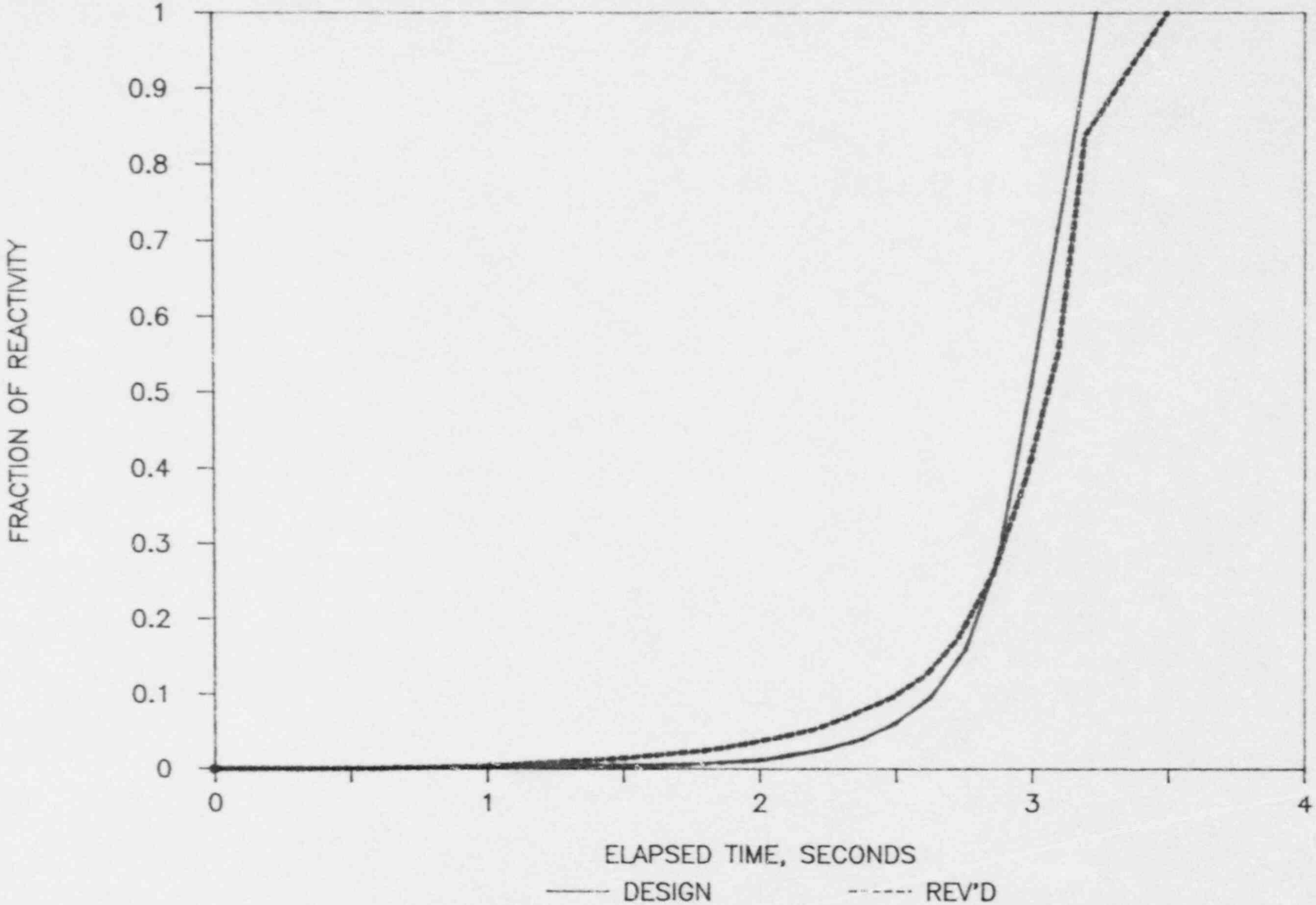
UNCONTROLLED CEA WITHDRAWAL FROM 100% POWER*

TIME SEC	DESIGN REACTIVITY FRACTION	REVISED REACTIVITY FRACTION
0	0.0000	0.0000
0.30	0.0000	0.0000
0.60	0.0007	0.0000
0.66	0.0009	0.0012
0.84	0.0024	0.0015
0.88	0.0027	0.0016
1.00	0.0035	0.0039
1.05	0.0037	0.0048
1.16	0.0040	0.0075
1.21	0.0041	0.0087
1.31	0.0042	0.0105
1.35	0.0043	0.0113
1.46	0.0045	0.0136
1.51	0.0048	0.0147
1.60	0.0052	0.0179
1.65	0.0057	0.0196
1.72	0.0064	0.0219
1.80	0.0077	0.0246
1.86	0.0086	0.0281
1.95	0.0108	0.0333
2.00	0.0120	0.0368
2.08	0.0164	0.0425
2.11	0.0180	0.0450
2.21	0.0244	0.0533
2.25	0.0270	0.0590
2.34	0.0360	0.0717
2.38	0.0400	0.0784
2.49	0.0593	0.0967
2.50	0.0610	0.0991
2.61	0.0906	0.1250
2.63	0.0960	0.1331
2.73	0.1452	0.1733
2.76	0.1600	0.1935
2.86	0.2600	0.2608
2.88	0.2800	0.2815
2.98	0.4717	0.3850
3.00	0.5100	0.4131
3.10	0.7142	0.5533
3.20	0.9183	0.8397
3.24	1.0000	0.8611
3.50	1.0000	1.0000

*Table 2 & Figure 2 are also applicable to ASGT and Loss of Condenser Vacuum.

FIGURE 2

UNCONTROLLED CEA WITHDRAWAL – 100% POWER



CEA REACTIVITY VS. TIME

TABLE 3

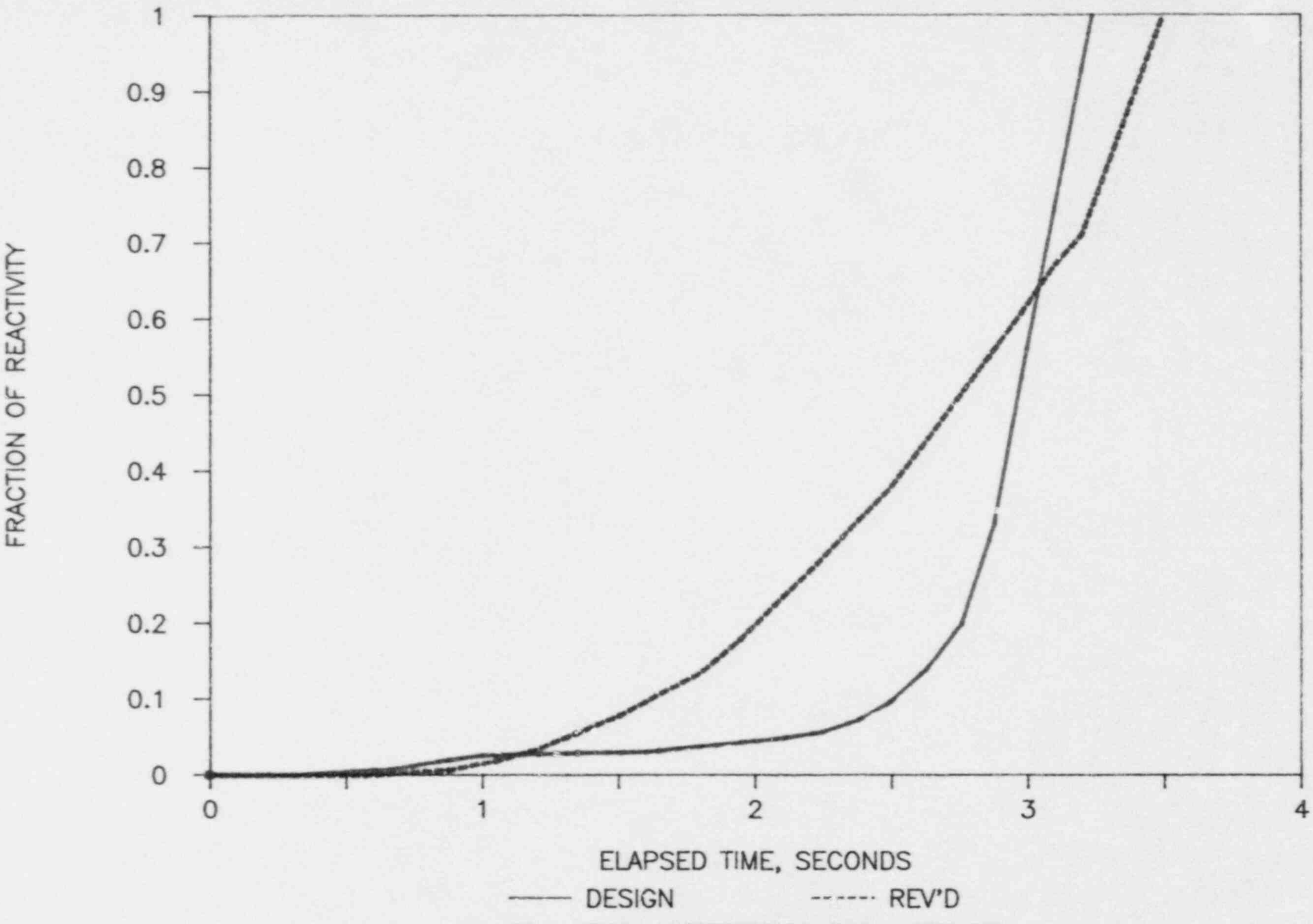
SCRAM REACTIVITY VERSUS TIME

LOSS OF FLOW - FOUR PUMP COASTDOWN (-0.3 ASI)

TIME SEC	DESIGN REACTIVITY FRACTION	REVISED REACTIVITY FRACTION
0	0.0000	0.0000
0.30	0.0000	0.0000
0.60	0.0068	0.0000
0.66	0.0075	0.00152
0.84	0.0180	0.00609
0.88	0.0200	0.00710
1.00	0.0260	0.01536
1.05	0.0268	0.01880
1.16	0.0285	0.02959
1.21	0.0287	0.03450
1.31	0.0290	0.04921
1.35	0.0293	0.05510
1.46	0.0300	0.07139
1.51	0.0304	0.07880
1.60	0.0310	0.09648
1.65	0.0327	0.10630
1.72	0.0350	0.11941
1.80	0.0397	0.13440
1.86	0.0400	0.15240
1.95	0.0432	0.17940
2.00	0.0456	0.19694
2.08	0.0479	0.22500
2.11	0.0490	0.23552
2.21	0.0547	0.27060
2.25	0.0570	0.28562
2.34	0.0674	0.31940
2.38	0.0720	0.33407
2.49	0.0949	0.37440
2.50	0.0970	0.37898
2.61	0.1334	0.42940
2.63	0.1400	0.43908
2.73	0.1862	0.48750
2.76	0.2000	0.50178
2.86	0.3083	0.54940
2.88	0.3300	0.55918
2.98	0.5217	0.60810
3.00	0.5600	0.61832
3.10	0.7433	0.66940
3.20	0.9267	0.70940
3.24	1.0000	0.74815
3.50	1.0000	1.00000

FIGURE 3

LOSS OF FLOW – FOUR PUMP COASTDOWN (–.03 ASI)



CEA REACTIVITY VS. TIME

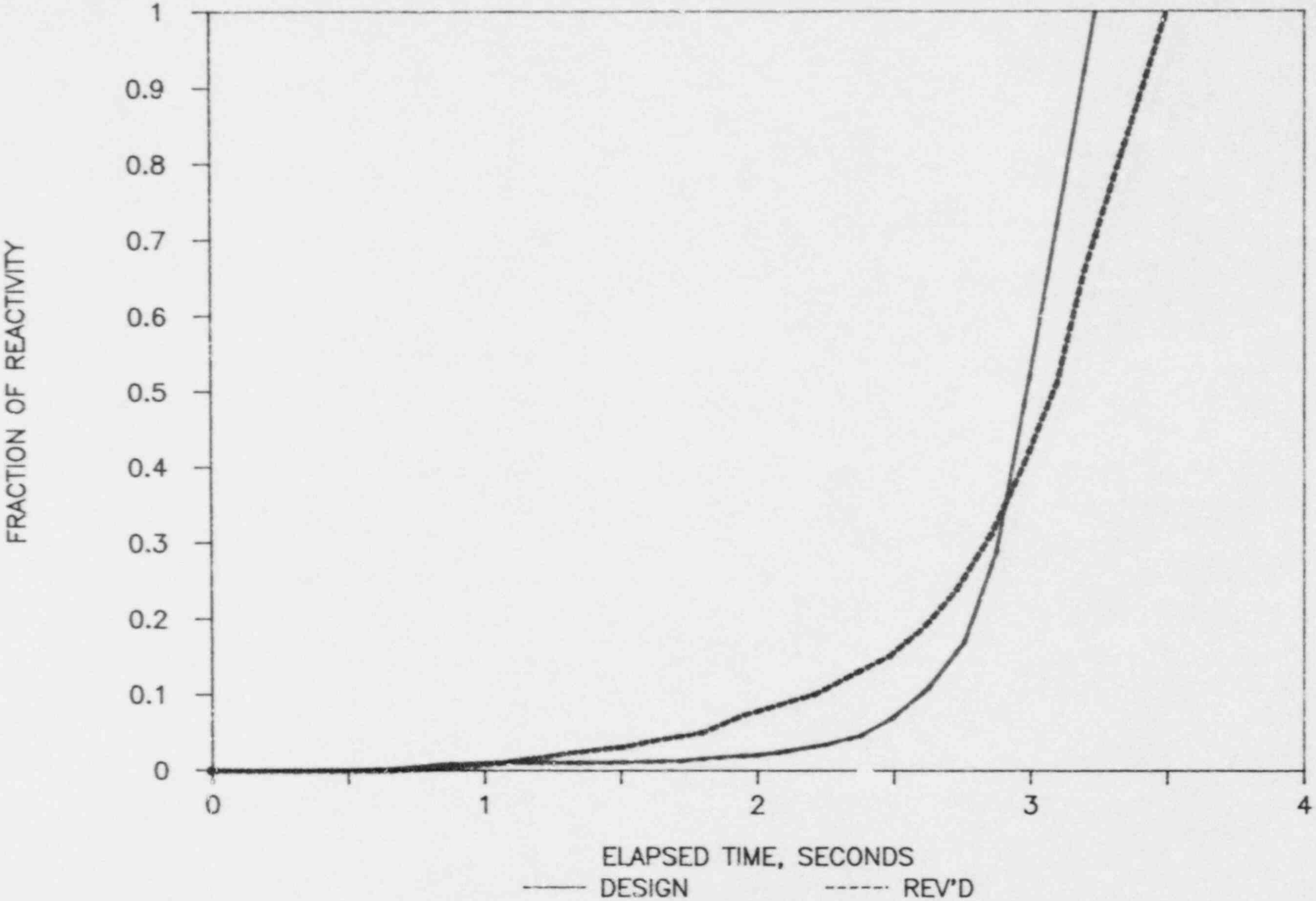
TABLE 4

SCRAM REACTIVITY VERSUS TIME

LOSS OF FLOW - FOUR PUMP COASTDOWN (0.0 ASI)

TIME SEC	DESIGN REACTIVITY FRACTION	REVISED REACTIVITY FRACTION
0	0.0000	0.00000
0.30	0.0000	0.00000
0.60	0.0025	0.00000
0.66	0.0028	0.00084
0.84	0.0090	0.00334
0.88	0.0096	0.00390
1.00	0.0114	0.00891
1.05	0.0115	0.01100
1.16	0.0118	0.01636
1.21	0.0119	0.01880
1.31	0.0120	0.02416
1.35	0.0121	0.02630
1.46	0.0122	0.03056
1.51	0.0125	0.03250
1.60	0.0130	0.03893
1.65	0.0134	0.04250
1.72	0.0140	0.04661
1.80	0.0169	0.05130
1.86	0.0190	0.06030
1.95	0.0209	0.07380
2.00	0.0220	0.07907
2.08	0.0256	0.08750
2.11	0.0270	0.09068
2.21	0.0334	0.10130
2.25	0.0360	0.10859
2.34	0.0436	0.12500
2.38	0.0470	0.13233
2.49	0.0690	0.15250
2.50	0.0710	0.15552
2.61	0.1040	0.18880
2.63	0.1100	0.19713
2.73	0.1562	0.23880
2.76	0.1700	0.25581
2.86	0.2700	0.31250
2.88	0.2900	0.32792
2.98	0.4817	0.40500
3.00	0.5200	0.42292
3.10	0.7200	0.51250
3.20	0.9200	0.66040
3.24	1.0000	0.70568
3.50	1.0000	1.00000

FIGURE 4
LOSS OF FLOW -- FOUR PUMP COASTDOWN (0.0 ASI)



CEA REACTIVITY VS. TIME

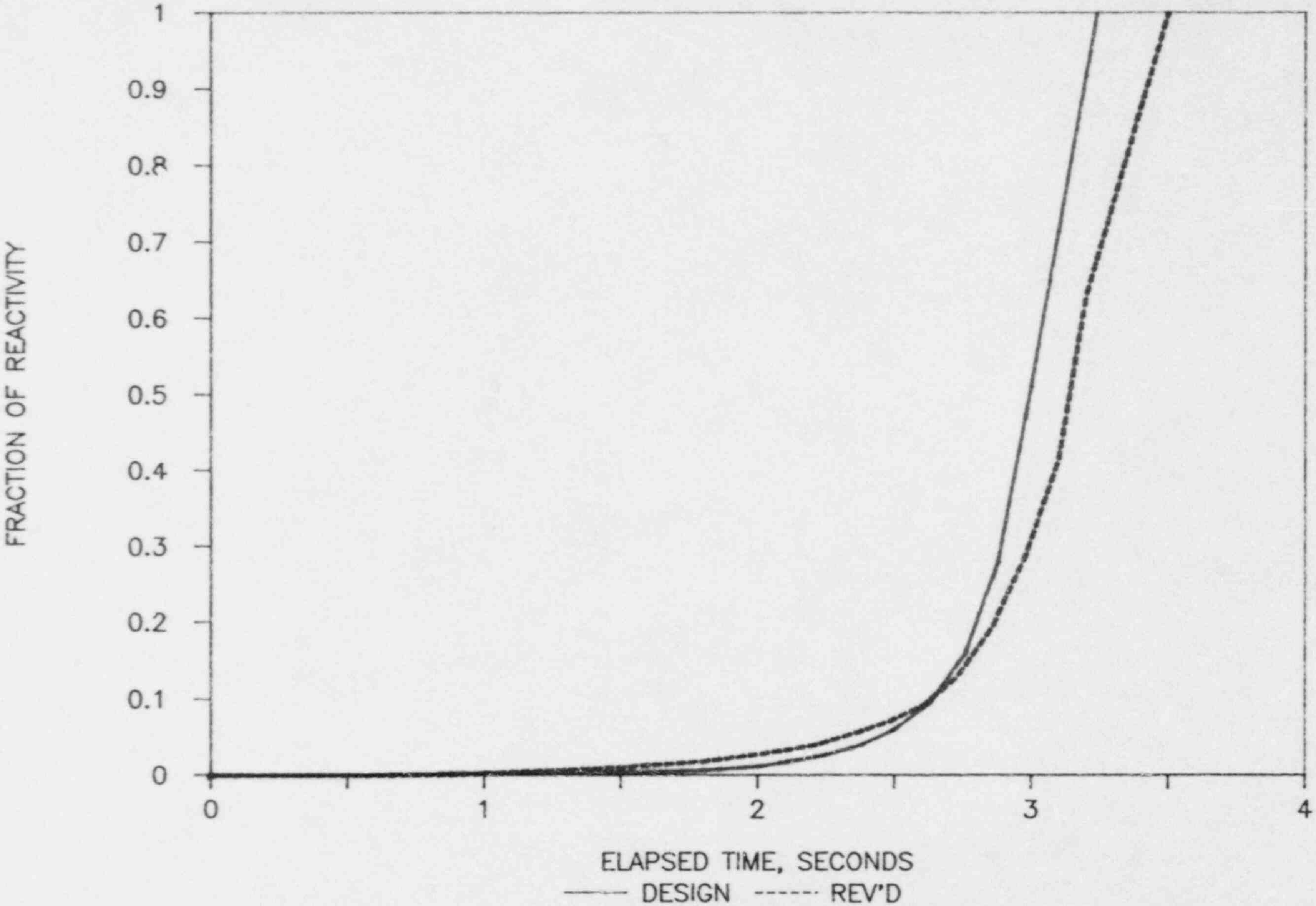
TABLE 5

SCRAM REACTIVITY VERSUS TIME

LOSS OF FLOW - FOUR PUMP COASTDOWN (+0.3 ASI)

TIME SEC	DESIGN REACTIVITY FRACTION	REVISED REACTIVITY FRACTION
0	0.0000	0.00000
0.30	0.0000	0.00000
0.60	0.0008	0.00000
0.66	0.0009	0.00026
0.84	0.0024	0.00103
0.88	0.0027	0.00120
1.00	0.0035	0.00289
1.05	0.0037	0.00360
1.16	0.0040	0.00559
1.21	0.0041	0.00650
1.31	0.0042	0.00793
1.35	0.0043	0.00850
1.46	0.0045	0.01022
1.51	0.0048	0.01100
1.60	0.0052	0.01338
1.65	0.0057	0.01470
1.72	0.0064	0.01643
1.80	0.0077	0.01840
1.86	0.0086	0.02104
1.95	0.0108	0.02500
2.00	0.0120	0.02765
2.08	0.0164	0.03190
2.11	0.0180	0.03377
2.21	0.0244	0.04000
2.25	0.0270	0.04425
2.34	0.0360	0.05380
2.38	0.0400	0.05879
2.49	0.0593	0.07250
2.50	0.0610	0.07427
2.61	0.0906	0.09380
2.63	0.0960	0.09983
2.73	0.1452	0.13000
2.76	0.1600	0.14514
2.86	0.2600	0.19560
2.88	0.2800	0.21113
2.98	0.4717	0.28880
3.00	0.5100	0.30983
3.10	0.7142	0.41500
3.20	0.9183	0.62980
3.24	1.0000	0.67916
3.50	1.0000	1.00000

FIGURE 5
LOSS OF FLOW - FOUR PUMP COASTDOWN (0.3 ASI)



CEA REACTIVITY VS. TIME

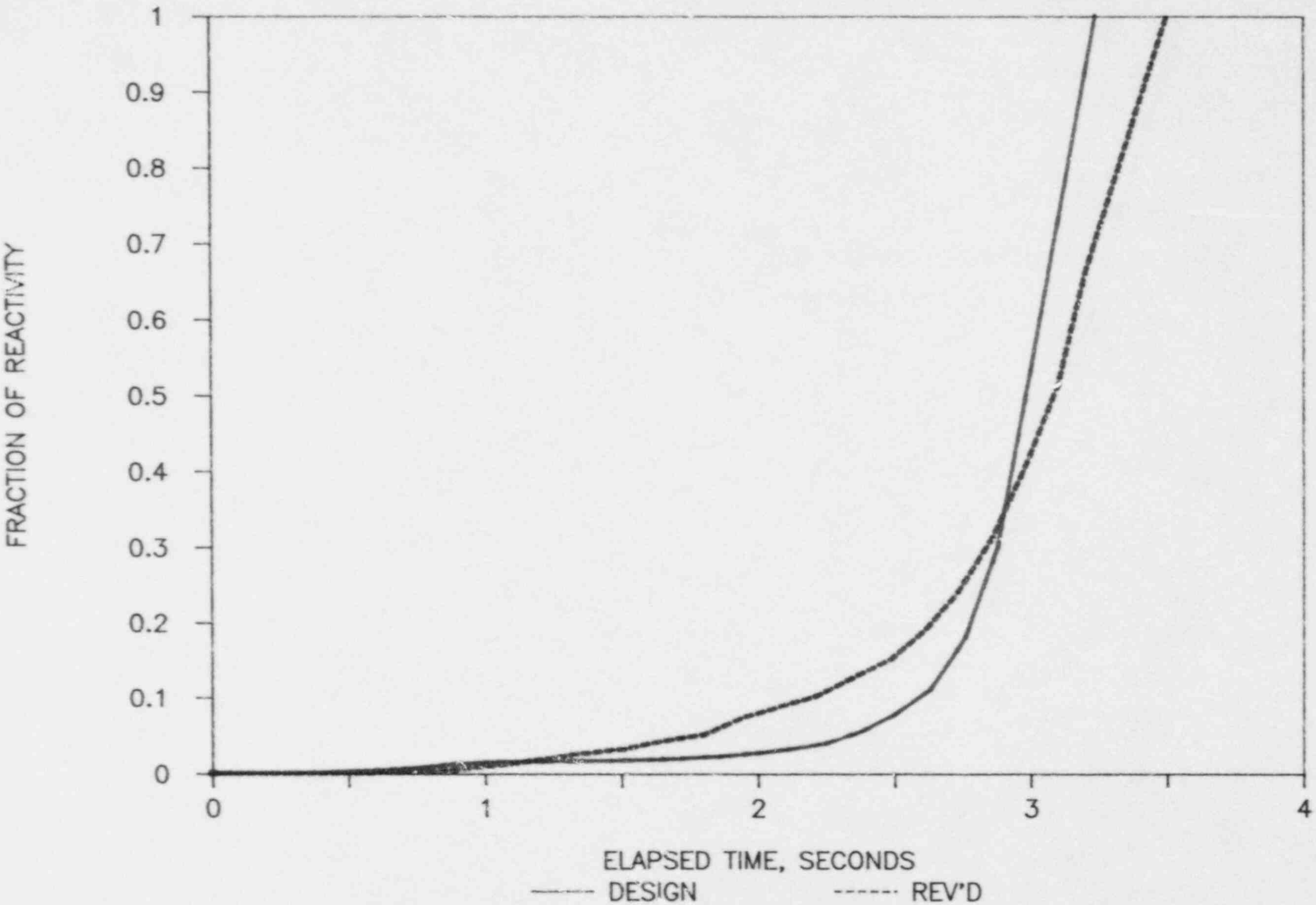
TABLE 6

SCRAM REACTIVITY VERSUS TIME

LOSS OF FLOW - SEIZED ROTOR (0.0 ASI)

TIME SEC	DESIGN REACTIVITY FRACTION	REVISED REACTIVITY FRACTION
0	0.0000	0.0000
0.30	0.0000	0.0000
0.60	0.0041	0.0000
0.66	0.0049	0.00293
0.84	0.0110	0.00372
0.88	0.0120	0.0039
1.00	0.0150	0.0089
1.05	0.0153	0.0110
1.16	0.0160	0.0164
1.21	0.0161	0.0188
1.31	0.0162	0.0242
1.35	0.0164	0.0263
1.46	0.0170	0.0306
1.51	0.0174	0.0325
1.60	0.0180	0.0389
1.65	0.0188	0.0425
1.72	0.0200	0.0466
1.80	0.0217	0.0513
1.86	0.0230	0.0603
1.95	0.0256	0.0738
2.00	0.0270	0.0791
2.08	0.0306	0.0875
2.11	0.0320	0.0907
2.21	0.0377	0.1013
2.25	0.0400	0.1086
2.34	0.0504	0.1250
2.38	0.0550	0.1323
2.49	0.0752	0.1525
2.50	0.0770	0.1555
2.61	0.1049	0.1888
2.63	0.1100	0.1971
2.73	0.1638	0.2388
2.76	0.1800	0.2558
2.86	0.2800	0.3125
2.88	0.3000	0.3279
2.98	0.4917	0.4050
3.00	0.5300	0.4229
3.10	0.7258	0.5125
3.20	0.9217	0.6604
3.24	1.0000	0.7057
3.50	1.0000	1.0000

FIGURE 6
LOSS OF FLOW – SEIZED ROTOR



CEA REACTIVITY VS. TIME

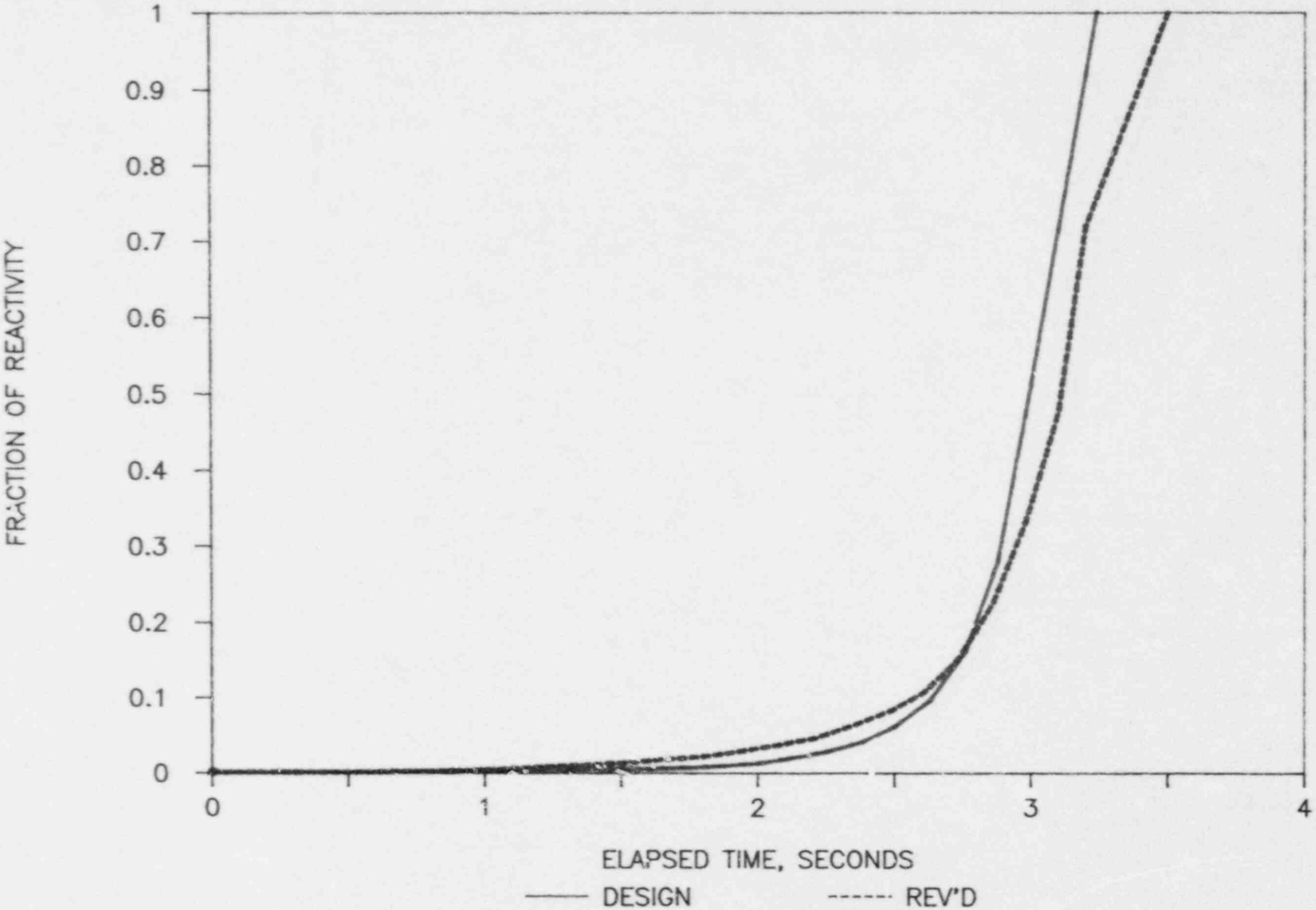
TABLE 7

SCRAM REACTIVITY VERSUS TIME

FEEDWATER LINE BREAK

TIME SEC	DESIGN REACTIVITY FRACTION	REVISED REACTIVITY FRACTION
0	0.0000	0.0000
0.30	0.0000	0.0000
0.60	0.0007	0.0000
0.66	0.0009	0.00105
0.84	0.0024	0.00134
0.88	0.0027	0.0014
1.00	0.0035	0.0033
1.05	0.0037	0.0041
1.16	0.0040	0.0064
1.21	0.0041	0.0074
1.31	0.0042	0.0090
1.35	0.0043	0.0097
1.46	0.0045	0.0117
1.51	0.0048	0.0126
1.60	0.0052	0.0153
1.65	0.0057	0.0168
1.72	0.0064	0.0188
1.80	0.0077	0.0211
1.86	0.0086	0.0241
1.95	0.0108	0.0286
2.00	0.0120	0.0316
2.08	0.0164	0.0364
2.11	0.0180	0.0385
2.21	0.0244	0.0457
2.25	0.0270	0.0505
2.34	0.0360	0.0614
2.38	0.0400	0.0671
2.49	0.0593	0.0829
2.50	0.0610	0.0849
2.61	0.0906	0.1071
2.63	0.0960	0.1140
2.73	0.1452	0.1486
2.76	0.1600	0.1659
2.86	0.2600	0.2236
2.88	0.2800	0.2413
2.98	0.4717	0.3300
3.00	0.5100	0.3541
3.10	0.7142	0.4743
3.20	0.9183	0.7197
3.24	1.0000	0.7571
3.50	1.0000	1.0000

FIGURE 7
FEEDWATER LINE BREAK



CEA REACTIVITY VS. TIME

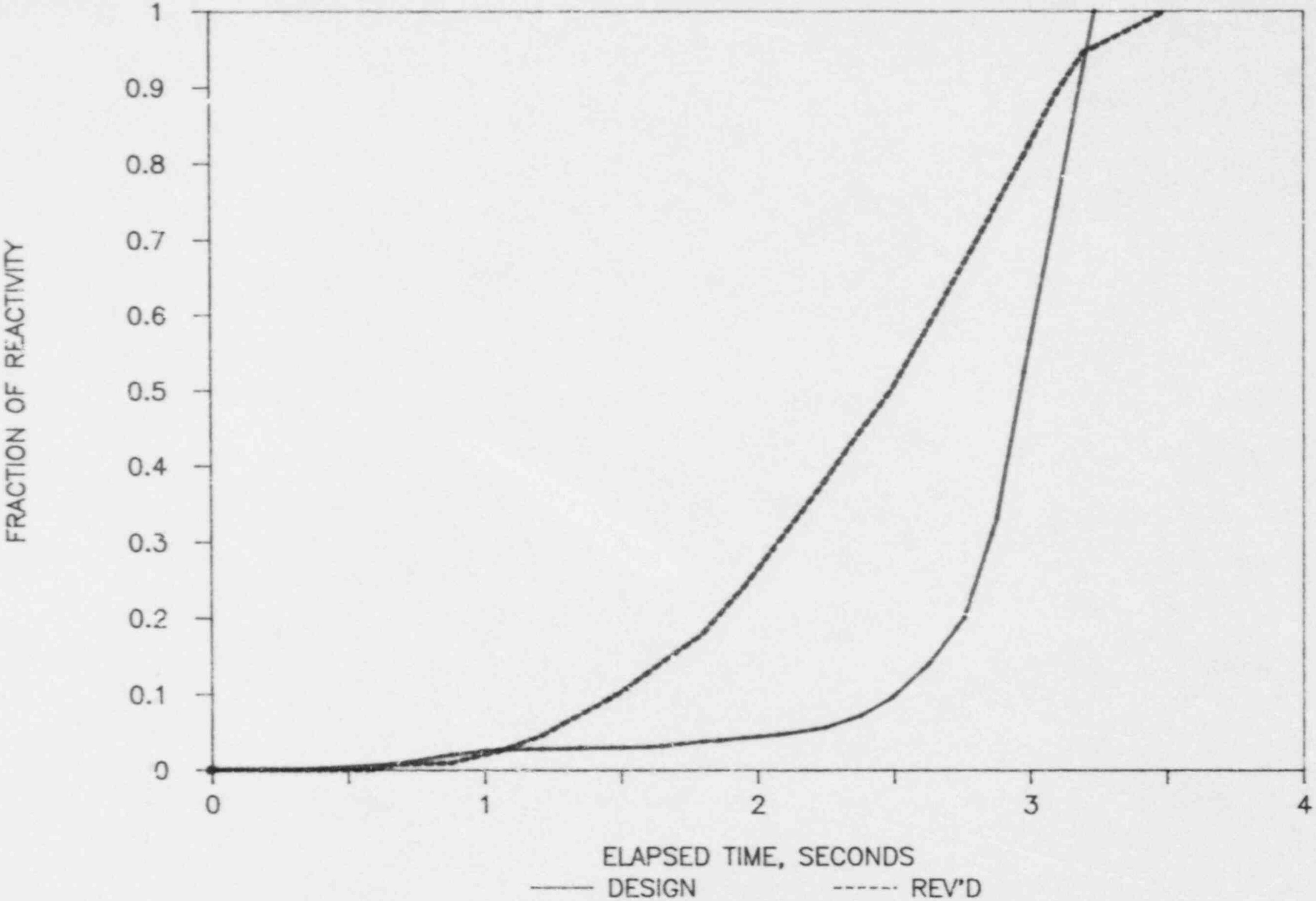
TABLE 8

SCRAM REACTIVITY VERSUS TIME

CEA EJECTION FROM 100% POWER

TIME SEC	DESIGN REACTIVITY FRACTION	REVISED REACTIVITY FRACTION
0	0.0000	0.0000
0.30	0.0000	0.0000
0.60	0.0062	0.0000
0.66	0.0075	0.00705
0.84	0.0180	0.00897
0.88	0.0200	0.0094
1.00	0.0260	0.0205
1.05	0.0268	0.0251
1.16	0.0285	0.0395
1.21	0.0287	0.0460
1.31	0.0290	0.0656
1.35	0.0293	0.0735
1.46	0.0300	0.0952
1.51	0.0304	0.1050
1.60	0.0310	0.1286
1.65	0.0327	0.1417
1.72	0.0350	0.1592
1.80	0.0379	0.1792
1.86	0.0400	0.2032
1.95	0.0432	0.2392
2.00	0.0450	0.2626
2.08	0.0479	0.3000
2.11	0.0490	0.3140
2.21	0.0547	0.3608
2.25	0.0570	0.3808
2.34	0.0674	0.4258
2.38	0.0720	0.4454
2.49	0.0949	0.4992
2.50	0.0970	0.5053
2.61	0.1334	0.5725
2.63	0.1400	0.5854
2.73	0.1862	0.6500
2.76	0.2000	0.6690
2.86	0.3083	0.7325
2.88	0.3300	0.7456
2.98	0.5217	0.8108
3.00	0.5600	0.8244
3.10	0.7433	0.8925
3.20	0.9267	0.9458
3.24	1.0000	0.9530
3.50	1.0000	1.0000

FIGURE 8
CEA EJECTION FROM 100% POWER



CEA REACTIVITY VS. TIME

TABLE 9

EVENTS SENSITIVE TO DELAY IN SCRAM

Event	Time To Closest Approach To A SAFDL
Uncontrolled CEA Withdrawal from a subcritical condition	~ 2.2 Sec. (Figure 1)
Uncontrolled CEA Withdrawal from a critical condition	
(a) 1% Power	~ 2.2 Sec. (Figures 1 & 2)
(b) 100% Power	(Note 1)
Loss of Flow (Note 2)	
(a) Four pump coastdowm	~ 2.15 Sec. (Figures 3, 4 & 5)
(b) Seized rotor	(Note 3) (Figure 6)
Secondary Pipe Breaks	
(a) Feedwater Line Break	~ 3.2 Secs. (Note 4) (Figure 7)
(b) Steam Line Break	Not sensitive to delay in scram
(c) Steam Line Break with Loss of AC	See "Loss of Flow"
CEA Ejection	
a. 0% Power	~ 2.4 Secs. (Figure 1)
b. 100% Power	~ 2.2 Secs. (Figure 8)
Loss of Condenser Vacuum (Note 5)	~ 2.3 Seconds (Figure 2)
Excess Heat Removal	(Note 1)
Asymmetric Steam Generator Transient	~ 2.1 Seconds (Figure 2)

Notes for Table 9:

(1) In accordance with TS 6.8.1.g this event is addressed by incorporating change to a CPC addressable constant (BERR1), resulting in a direct offset of any effects due to increased CEA drop times.

(2) The Loss of AC Power and Steam Generator Tube Rupture with Loss of AC Power are also covered by this item.

(3) The SAFDL (minimum DNBR) is actually exceeded for the seized rotor event. The minimum DNBR is reached relatively early in the transient (~ .95 seconds following the trip) which is driven by the rapid reduction in core flow. The increase in CEA drop time therefore has relatively minor impact upon the analysis.

(4) The closest approach to the SAFDL occurs slightly after the maximum beneficial effect from the revised scram reactivity insertion curves however, the reactor power decrease at 2.7 seconds is sufficient to terminate the event.

(5) The Loss of Load Turbine Trip, Turbine Trip with Failure of Turbine Bypass Valves, Turbine Trip with Failure of Generator Breaker to Open, and Turbine Gland Seal System Malfunction are also covered by this item.