UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. for a Class 103 License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 2 of the San Onofre Nuclear Generating Station	)	DOCKET NO. 50-361 Amendment Application No. 50

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. pursuant to 10 CFR 50.90, hereby submit Amendment Application No. 50.

This amendment application consists of Proposed Change NPF-10-263 to Facility Operating License No. NPF-10. Proposed Change NPF-10-263 is a request to revise Technical Specification 3/4.1.3.4, "CEA Drop Time." The proposed change will increase the specified drop time for control element assemblies from 3.0 to 3.2 seconds.

Pursuant to 10 CFR 170.12, the required amendment application fee of $150 is enclosed.

Subscribed on this 14th day of June, 1988.

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

By: ______________________

Subscribed and sworn to before me this
14th day of June 1988.

Agnes Crabtree

Notary Public in and for the County of
Los Angeles, State of California

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Notary Public-California
LOS ANGELES COUNTY
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My Commission Expires May 19, 1989

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Cathy Karpow
Notary Public in and for the County of Orange, State of California

THE CITY OF RIVERSIDE

By: s/s Bill D. Carnahan

Bill D. Carnahan
Public Utilities Director

Alan R. Watts
Rourke & Woodruff
Attorneys for the City of Riverside

By: s/s Alan R. Watts

Subscribed and sworn to before me this 14th day of June, 1988.

s/s Margaret I. Archambault
Notary Public in and for the County of Riverside, State of California

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. for a Class 103 License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 3 of the San Onofre Nuclear Generating Station	)	DOCKET NO. 50-362 Amendment Application No. 36

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. pursuant to 10 CFR 50.90, hereby submit Amendment Application No. 36.

This amendment application consists of Proposed Change NPF-15-263 to Facility Operating License No. NPF-15. Proposed Change NPF-15-263 is a request to revise Technical Specification 3/4.1.3.4, "CEA Drop Time." The proposed change will increase the specified drop time for control element assemblies from 3.0 to 3.2 seconds.

Pursuant to 10 CFR 170.12, the required amendment application fee of $150 is enclosed.

Subscribed on this 14th day of June, 1988.

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

By: Kenneth P Baskin

Subscribed and sworn to before me this 14th day of June 1988.

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DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-263

This is a request to revise Technical Specification 3/4.1.3.4 "CEA Drop Time."

Existing Specifications

Unit 2: See Attachment "A"

Unit 3: See Attachment "C"

Proposed Specifications

Unit 2: See Attachment "B"

Unit 3: See Attachment "D"

Description

The proposed change will revise Technical Specification 3/4.1.3.4 "CEA Drop Time." The purpose of Technical Specification (TS) 3/4.1.3.4 is to ensure that actual drop times for full length CEAs are consistent with the maximum drop time assumed in the accident and transient analyses. Specifically, TS 3/4.1.3.4 requires that the drop times for individual full length CEAs from the fully withdrawn position be less than or equal to 3.0 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position, when measured with the average reactor coolant temperature greater than 520°F with all reactor coolant pumps running. Measurement of drop times under these conditions ensures that the measured drop times will be representative of insertion times experience during a reactor trip at operating conditions.

CEA drop times are measured following each removal and reinstallation of the reactor vessel head, following any maintenance on, or modification to, the CEA drive system which could affect the drop time of those specific CEAs, and at least once per 18 months. Should a CEA drop time be found to exceed 3.0 seconds, the plant must be placed in hot standby.

Prior to SONGS Unit 2 Cycle 4 startup, CEA drop times were measured individually using a visicorder to simultaneously monitor CEA position from the reed switch position transmitter (RSPT) and interruption of power to the upper gripper coil. The CEA is withdrawn from the core to its full out position and dropped by opening its individual circuit breaker. Interruption of power to the gripper coil and CEA position as a function of time is recorded on the visicorder chart. From this chart the time from interruption of power to 90% CEA insertion can be determined.

Beginning with Unit 2 Cycle 4 startup a new method of measuring CEA drop times was used. This method uses special software (CEA Drop Time Test, or CDTT software) loaded on one of the Control Element Assembly Calculators (CEACs) which will initiate a Core Protection Calculator (CPC) trip and simultaneously

monitor the positions of all 91 CEAs as a function of time. The data obtained can then be analyzed to determine individual CEA drop times. The CDTT software initiates the test by transmitting a large penalty factor to each of the CPCs channels, producing a reactor trip. It is important to note that in this method, the reactor trip breakers are the point at which power is interrupted to the CEA gripper coils, rather than the individual breakers as in the "traditional" method.

The CEA drop times measured using this method during Unit 2 startup were unexpectedly longer than using the traditional method. Although no CEAs failed to meet the 3.0 second drop time requirement, some CEAs were close to the limit. Drop times for the five slowest CEAs were remeasured using the traditional method which confirmed that there was no degradation in CEA performance compared with previous tests and the differences experienced were therefore thought to be test related. Subsequently, the new test method was used recently at ANO-2, where a number of CEAs failed the drop time test. Investigation at ANO-2 has revealed that the measured drop times differ between the two methods because the circuit dissipating gripper coil stored energy has a longer time constant when tripped by the reactor trip beakers than when tripped by the individual circuit breakers. Consequently, the longer drop times are not an anomaly of the test method. Since the new method uses the reactor trip breakers, it more accurately reflects the operation of the reactor protection system as assumed in the safety analysis. Thus, the traditional method may not provide conservative results relative to the safety analysis assumptions.

SONGS Unit 3 is currently in a refueling outage. The new test method will be used for CEA drop time measurements during SONGS Unit 3 Cycle 4 startup. A recent review of past Unit 3 CEA drop time measurements revealed that there is potential for one CEA, CEA 64, to fail to meet the 3.0 second requirement using the CDTT software.

In parallel, to avoid the potential for delay of Unit 3 restart and because of the reduced margin to the 3.0 second limit which was found on Unit 2, the proposed change would increase the specified drop time to 3.2 seconds.

It is important to note that the safety analyses typically assume that all CEAs are inserted to 90% at the maximum TS limit (3.0 seconds). This assumption provides a straightforward method for verifying compliance with the TS and allows for relatively simple modeling of reactivity insertion in the safety analyses. However, this assumption is clearly conservative since the Technical Specifications ensure that the limiting (i.e., slowest) CEA will reach the 90% limit within 3.0 seconds; consequently, most CEAs are inserted sooner.

The Safety analysis currently assumes a 0.3 second delay from the time the trip breakers open until the CEAs begin to drop into the core and 2.7 seconds to reach 90% insertion. The SONGS 2 and 3 design bases accidents have been reviewed to assess the impact of a 0.6 second delay before the CEAs begin to drop into the core and 2.6 seconds to reach 90% insertion, corresponding to a total CEA drop time of 3.2 seconds. Relative to the potential effects of increased rod drop time, the accidents can be categorized as follows:

FSAR Subsection	Category	Event
		MODERATE FREQUENCY INCIDENTS
15.1.1.1	3	Decrease in feedwater temperature
15.1.1.2	3	Increase in feedwater flow
15.1.1.3	4	Increased main steam flow
15.1.1.4	2	Inadvertent opening of a steam generator atmospheric dump valve
15.2.1.1	3	Loss of external load
15.2.1.2	3	Turbine trip
15.2.1.3	4	Loss of condenser vacuum
15.2.1.4	3	Loss of normal ac power
15.3.1.1	3	Partial loss of forced reactor coolant flow
15.4.1.1	4	Uncontrolled CEA withdrawal from a subcritical or low power condition
15.4.1.2	4	Uncontrolled CEA withdrawal at power
15.4.1.3	1	CEA misoperation
15.4.1.4	2	CVCS malfunction (inadvertent boron dilution)
15.4.1.5	1	Startup of an inactive reactor coolant system pump
15.5.1.1	2	CVCS malfunction
15.5.1.2	1	Inadvertent operation of the ECCS during power operation
15.9.1.1	4	Asymmetric steam generator transient
		INFREQUENT INCIDENTS
15.1.2.1	3	Decrease in feedwater temperature[a]
15.1.2.2	3	Increase in feedwater flow[a]
15.1.2.3	4	Increased main steam flow[a]
15.1.2.4	2	Inadvertent opening of a steam generator atmospheric dump valve[a]
15.2.2.1	3	Loss of external load[a]
15.2.2.2	3	Turbine trip[a]
15.2.2.3	4	Loss of condenser vacuum[a]
15.2.2.4	3	Loss of normal ac power[a]
15.2.2.5	2	Loss of normal feedwater flow
15.3.2.1	4	Total loss of forced reactor coolant flow
15.3.2.2	3	Partial loss of forced reactor coolant flow[a]
15.5.2.1	2	CVCS malfunction[a]

LIMITING FAULTS

15.1.3.1A	4	Steam system piping failures - pre-trip power excursion analysis
15.1.3.1B	2	Steam system piping failures - post-trip return to power
15.2.3.1	4	Feedwater system pipe breaks
15.2.3.2	2	Loss of normal feedwater flow(a)
15.3.3.1	3	Single reactor coolant pump (RCP) shaft seizure
15.3.3.2	4	Single RCP sheared shaft
15.3.3.3	3	Total loss of forced reactor coolant flow(a)
15.4.3.1	1	Inadvertent loading of a fuel assembly into the improper position
15.4.3.2	4	CEA ejection
15.6.3.1	2	Primary Sample or Instrument Line Break
15.6.3.2	2	Steam generator tube rupture
15.6.3.3	1 & 2	Loss of coolant accident (for large and small breaks, respectively)
15.6.3.4	2	Inadvertent opening of a pressurizer safety valve
15.7.3.1	1	Radioactive waste gas system leak or failure
15.7.3.2	1	Radioactive liquid waste system leak or failure (release to atmosphere)
15.7.3.3	1	Postulated radioactive releases due to liquid containing tank failures
15.7.3.4	1	Design basis fuel handling accidents
15.7.3.5	1	Spent fuel cask drop accidents
15.8	1	Anticipated transients without scram

(a) These incidents involve the same initiating event as the corresponding moderate frequency incidents but include either a concurrent single active component failure or a single operator error.

<u>Key to Categories</u>:

1 Reactor trip does not occur or, in the case of LBLOCA, is not credited.

2 Consequences are not sensitive to 0.3 second delay of reactor trip, because of the slow rate of margin degradation through the time of trip, or due to the obvious insensitivity of accident consequences as a function of the time of trip.

3 This event is bounded by another event that is presented in Chapter 15.

4 This event is potentially impacted by a 0.3 second delay of reactor trip.

Generally, the effects of a 0.3 second delay are accommodated by existing conservatism in the analyses. Where existing conservatism are insufficient to accommodate a 0.3 second trip delay, appropriate CPC penalties will be applied via addressable constants made in accordance with TS 6.8.1. Each of the category 4 events identified above are discussed below.

One existing conservatism which previously has not been credited for several analyses is the application of space-time scram curves. 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. For events whose analysis of record already applies a space time methodology, other conservatism are identified or CPC penalties will be applied.

Increased Main Steam Flow

The increase in heat removal by the steam generators as a result of increased main steam flow is defined as any rapid increase in steam generator steam flow, other than a steam line rupture, without a turbine trip. An increase in main steam flow may be caused by any one of the following incidents of moderate frequency:

a) An inadvertent increase in the opening of the turbine admission valves caused by operator error or turbine load limit malfunction.

b) Failure in the Steam Bypass System which could result in an opening of one or more of the turbine bypass valves.

c) An inadvertent opening of an atmospheric dump valve or steam generator safety valve caused by operator error or failure within the valve itself.

The most severe of these incidents is the inadvertent opening of all of the turbine bypass valves at full power. This event results in the closest approach to the SAFDL since, initially, steam flow will increase to approximately 145% of normal full main steam flow. The steam flow during the transient is proportional to the steam generator pressure until the main steam isolation signal (MSIS) results in the isolation of the steam generators and termination of the flow out of the turbine bypass valves and to the turbine. This case results in the most rapid cooldown and, consequently, largest power increase.

As shown in FSAR Table 15.1-1, a CPC low DNBR trip is generated during the event to terminate the margin degradation. The additional 0.3 second delay before CEA motion occurs results in additional margin degradation. Thus, a power penalty of 1.005 will be applied to the CPC addressable constant, BERR1 to ensure that a CPC low DNBR trip will be generated sufficiently early to compensate for the increased holding coil decay time. Therefore, the conclusions of this analysis are unaffected.

Loss of Condenser Vacuum (LOCV)

This event is initiated by a turbine trip due to a loss of condenser vacuum without a simultaneous reactor trip. The loss of load causes the 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 a corresponding increase in RCS pressure. This RCS pressure increase results in a high pressurizer pressure trip. As shown in FSAR Table 15.2-1, RCS pressure reaches the high pressurizer pressure trip setpoint at 8.5 seconds into the event. CEAs are assumed to begin to drop into the core 1.8 seconds later at 10.3 seconds into the event. Technical Specifications require a 0.9 second response time (time interval from when the monitored parameter exceeds its trip setpoint at the sensor until electrical power is interrupted to the CEA drive mechanism) for the high pressurizer pressure trip. Thus, the existing analysis can accommodate a 0.9 second holding coil decay time and the conclusions are unaffected by the proposed.

Uncontrolled CEA Withdrawal from Subcritical

The withdrawal of CEAS from subcritical conditions adds reactivity to the reactor core, causing both the core power level and the core heat flux to increase with corresponding increases in reactor coolant temperatures and Reactor Coolant System (RCS) pressure. The withdrawal of CEAs also produces a time dependent redistribution of core power. These transient variations in core thermal parameters result in the approach to specified fuel design limits and to RCS and secondary system pressure limits, thereby requiring the protective action of the Reactor Protection System (RPS). The event is terminated by the High Logarithmic Power trip. The limiting case presented in FSAR Section 15.4.1 was reanalyzed to assess the impact of an additional 0.3 second holding coil decay time. Except for the increased delay, the reanalyzed case is identical to that reported in the FSAR. When compared to the FSAR case, the 0.3 second delay results in increases in peak core power from 65% to 106% and peak heat flux from 16% to 33%. However, the acceptance criteria for this event continue to be satisfied since minimum DNBR remains greater than 1.31 and fuel centerline temperature is less than 4900°F and thus, the fuel is not predicted to melt. Although the 0.3 second delay results in more adverse results for this event, the conclusions of the FSAR analysis remain valid.

Uncontrolled CEA Withdrawal at Power

An uncontrolled CEA withdrawal results in an increase in core power and corresponding increase in reactor coolant temperature and pressure; further,

the withdrawal of CEAs produces a time dependent redistribution of core power. These transient variations in core thermal parameters may result in a rapid approach to the fuel design limits on DNBR and fuel centerline temperature, thereby requiring the protective action of the RPS.

FSAR Table 15.4-5 shows that a CPC low DNBR trip is generated at 43.45 seconds into the transient. The minimum DNBR occurs approximately 1.8 seconds after the trip breakers open. Application of the space-time scram curve is sufficient to compensate for the increased holding coil decay time and the conclusions of the analysis of this transient remain unaffected.

Uncontrolled CEA Withdrawal from Low Power Conditions

The withdrawal of CEAs from low power conditions adds reactivity to the reactor core causing the increase of power, heat flux, temperature, and pressure. FSAR Table 15.4-2 shows that the High Pressurizer Pressure Trip (HPPT) and the CPC Variable Overpower Trip (VOPT) occurs simultaneously at 151.4 seconds into the transient. The analysis assumed a conservatively high VOPT setpoint to produce results which would bound those for future cycles. Sufficient conservatism exists to compensate for the additional 0.3 second holding coil delay when the current lower VOPT setpoint is used. Therefore, the conclusions of this event are unchanged.

Asymmetric Steam Generator Events (ASGT)

The asymmetric steam generator events are examined to determine degradation in thermal margin. The most severe of these events was the postulated instantaneous closure of a single Main Steam Isolation Valve. With a 20°F differential cold leg temperature analysis trip setpoint, the ASGT event requires a 112.5% margin based on a 2 dimensional Hermite Space time calculation. The increased delay of 0.3 seconds before CEA movement would result in less than a 1.5% increase in required thermal margin. Cycle 4 COLSS margins accommodate this additional margin requirement without change. Therefore, the conclusions of this event are unchanged.

Loss of Condenser Vacuum with Single Failure

The single failures considered which have an effect on this transient are:

a) A loss of all ac power on turbine trip

b) Failure of the pressurizer level measurement channel.

The failure of the pressurizer level measurement channel produces the most adverse effect following a loss of condenser vacuum. This failure would produce a false low level signal, resulting in activation of both standby charging pumps and the closing of the letdown control valve to its minimum flow area.

The results of this analysis are only slightly more adverse than the analyses of the loss of condenser vacuum analysis presented in section 15.2.1.3. The same compensating conservatisms are present in both analyses.

Total Loss of Forced Reactor Coolant Flow

This event is initiated by the simultaneous loss of power to all four reactor coolant pumps resulting in the coastdown of the forced reactor coolant flow. The analysis determines the degradation in the thermal margin between the event initial conditions and the point of minimum transient DNBR. This required margin is preserved by COLSS. An increase in CEA drop time results in an increase of required margin of less than 2 percent. Cycle 4 COLSS margins will be changed to accommodate this additional Loss of Flow margin. Therefore, the conclusions of this event are unchanged.

Steam System Piping Failures: Pre-Trip Power Excursions

A rupture in the main Steam System piping increases steam flow from the steam generators. This increase in steam flow increases the rate of RCS heat removal by the steam generators and causes a decrease in core coolant inlet temperature. In the presence of a negative moderator temperature coefficient (MTC), this decrease in temperature causes core power to increase. A Loss of Offsite AC Power (LOAC) during the transient can contribute to an additional reduction of thermal margin due to the associated loss of power to the reactor coolant pumps.

In terms of radiological consequences, the limiting Pre-trip power excursion was the outside containment break location. The minimum DNBR occurs approximately 1.5 seconds after the trip breakers open. The transient was modeled using static scram curves. Figure 1 shows a comparison of the negative reactivity added by the static scram curve used and a space time scram curve with CEA motion delayed by 0.3 seconds. A slight non-conservatism exists prior to 1.0 seconds, but never exceeds 0.003%. By 1.2 seconds, the space time curve is conservative and at the time of minimum DNBR, the delayed space time curve added 0.37% more negative reactivity than the static curve inserted. Near full insertion the revised curve again predicts less negative reactivity insertion but this is well past the point of minimum DNBR. Therefore, the conclusions of this event are unchanged.

Feedwater System Pipe Break

This event approaches the upset pressure limit (i.e., the RCS pressure safety limit). FSAR Table 15.2-8 presents the sequence of events of the main Feedwater System Pipe Break with a loss of offsite power coincident with the reactor trip. At 35.0 seconds, a High Pressurizer Pressure Trip Condition exists. A trip signal is generated and the trip breakers are open at 35.9 seconds into the event. Although Table 15.2-8 shows that CEA motion begins 0.3 seconds later at 36.2 seconds, the analysis actually assumes a 0.84 second delay. This conservatism is more than sufficient to compensate the proposed increased CEA drop time. Therefore, the conclusion of this event remains unchanged.

Single Reactor Coolant Pump Sheared Shaft

This event is initiated by the seizure of one reactor coolant pump rotor and results in the rapid coastdown of core flow to the asymptotic 3-pump flow rate. A reactor trip is generated when the rapid flow reduction across the steam generator in the affected loop decreases the delta-P below the trip setpoint. The minimum DNBR was calculated by the TORC code using the asymptotic 3-pump flow rate of 75% of full flow. This simplified method does not credit the power reduction transient during scram which more than compensates for the 0.3 second holding coil delay. Therefore, the conclusions of this event are unchanged.

CEA Ejection

For the analysis, it is assumed that a complete and instantaneous circumferential rupture of the CEDM housing or of the CEDM nozzle results in the ejection of a CEA.

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 2.4 seconds after the trip breakers open. The revised scram reactivity data, 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 2.4 seconds after the trip breakers open. The revised scram reactivity data, at the time of interest is more conservative than the design data. Therefore, the conclusions of this event are unchanged.

In addition to the Chapter 15 Design Basis Events discussed above, the Containment Pressure Analyses described in SAR Chapter 6 was also reviewed.

Containment Pressure Analysis (SAR Section 6.2)

The peak pressure analyses address the response of the containment to LOCAs and Main Steam Line Breaks. The calculated peak containment pressure of 55.7 psig presented in the FSAR results from a 102% power main steam line break. This event was reanalyzed (Reference PCN-207 approved by Amendments 60 and 49 dated August 14, 1987) using the more realistic and detailed modeling in order to compensate for an increased MSIV closure time. The total mass/energy release to containment calculated in the reanalysis is 1.74% less than that reported in the original FSAR analysis. The additional 0.3 second holding coil delay

will increase the mass/energy release by less than 0.6%. This is well within the conservatism calculated in the reanalysis. Ignoring the margin in mass/energy release in the reanalysis, the increase in peak containment pressure due to a 0.6% increase in mass/energy is much less than the 60 psig containment design pressure. Therefore, the conclusions presented in the FSAR remain valid.

Safety Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following previously evaluated?

1. Will operation of the facility in accordance with this proposed change significantly increase the probability or consequences of any accident previously evaluated?

   Response: No

   The proposed technical specification merely changes the time requirements for insertion of CEAs 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. Because the change affects only an acceptance criterion 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.

   Each potentially impacted analysis was reviewed as discussed above. As noted, significant conservatism exist in the analyses which offset the effects of the increased CEA drop time on the consequences of previously analyzed accidents. Where such conservatism do not exist, an increase of the CPC DNBR power uncertainty multiplier (BERR1) and COLSS loss of flow margin, will offset the longer CEA drop times. Consequently, the proposed change does not involve a significant increase in the consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change does not involve any new or modified structures, systems, or components; rather, it affects only an acceptance criterion for confirming the required performance of the existing CEA hardware. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

   Response: No

   The margins of safety related to CEA insertion are defined by the analyzed events in the Safety Analysis Report which credit their insertion. As stated in response to first question above, the results of previously analyzed accidents are not significantly affected by the proposed change. Therefore, the margins of safety reflected in the analytical conclusions are not significantly reduced.

Safety and Significant Hazards Determination

Based on the above Safety Analysis it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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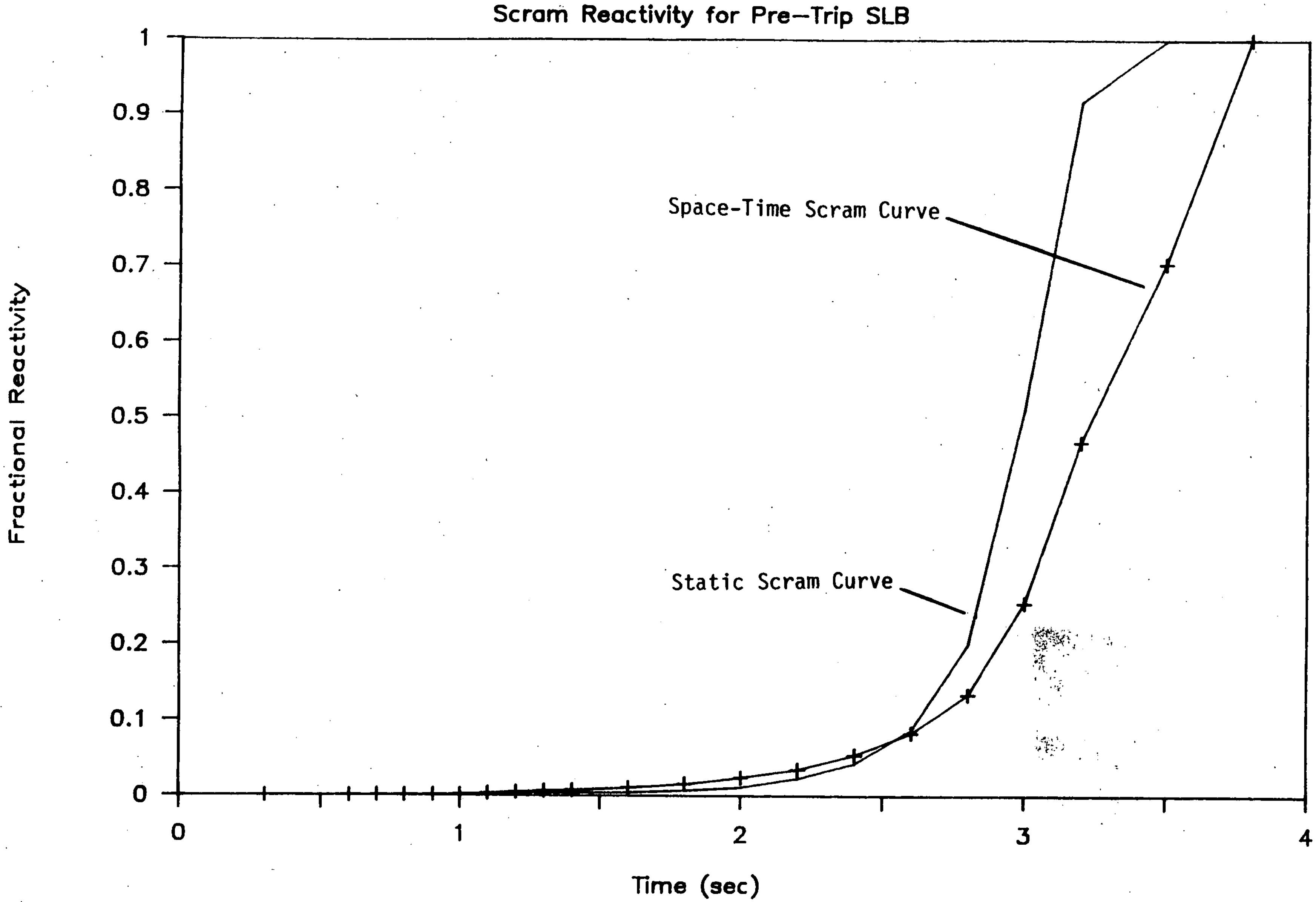
FIGURE 1



Scram Reactivity for Pre–Trip SLB
Fractional Reactivity
Time (sec)
Space-Time Scram Curve
Static Scram Curve
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
0
1
2
3
4

ATTACHMENT "A"

UNIT 2 EXISTING SPECIFICATIONS